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RECIP. NAME RECIPIENT AFFILIATION
DENTON, H. R. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Forwards updated FSAR for DC Cook Units 1 & 2, changing date to Jul 1986 to maintain ref point for changes.

Revised 08.05.86 DEH.

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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
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July 22, 1986
AEP:NRC:0509H

Donald C. Cook Nuclear Plant
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
FSAR Update - Compliance with 10 CFR 59.71(e)

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

We are transmitting to you under separate cover thirteen (13) copies of the changed pages for the 1986 version of the Final Safety Analysis Report (FSAR) for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. These pages are being transmitted to you according to the provisions of 10 CFR 50.71(e). A list of replacement pages is included with each copy.

Changed pages have been dated "July, 1986" in the lower right corner in order to maintain a reference point for changed pages in addition to vertically barring the specific change.

We hereby certify that the information contained in this update to the FSAR, to the best of our knowledge, accurately presents changes made since the previous submittal.

Very truly yours,

M. P. Alexich
Vice President

RBK
7/22/86

MPA/rjn

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

A053 1/13

SUPERSEDED PAGES PER REV TO UPDATED FSAR 50-315 LTR DTD 07-22-86 # 8607310222

CHAPTER 4

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198. NUREG/CR 3988, "MARCH-2, Meltdown Accident Response Characteristic Code Description and Users Manual", BMI-2115, Battelle Columbus Laboratories, September 1984, R. O. Wooten, et. al.

companies. The responsibility for the functional management of the major operating companies is vested in the President of each operating company reporting to the AEPSC President and Chief Operating Officer who reports to the AEPSC Chairman of the Board and Chief Executive Officer.

American Electric Power Service Corporation

The responsibility for administrative and technical direction of the AEP System and its facilities is delegated to the American Electric Power Service Corporation (AEPSC). AEPSC provides management and technological services to the various AEP System Companies.

Operating Companies

The operating facilities of the AEP System are owned and operated by the respective operating companies. The responsibility for executing the engineering, design, construction, specialized technical training, and certain operations supervision is vested in AEPSC while all or part of the administrative function responsibility is assigned to the operating companies. In the case of Cook Plant, I&MECo provides only public affairs, accounting and industrial safety direction.

The Donald C. Cook Nuclear Plant is owned and operated by Indiana & Michigan Electric Company (I&MECo) which is part of the AEP system.

1.7.1.2.4 Quality Assurance Responsibility of AEPSC

1) AEPSC provides the technical direction of the Cook Plant, and as such makes the final decisions pertinent to safety-related changes in plant design. Further, AEPSC reviews NRC letters, bulletins, notices, etc., for impact on plant design, and the need for design changes or modifications.

2) AEPSC furnishes licensing, NRC correspondence, fuel management and radiological support activities.

3) AEPSC provides additional service in matters such as supplier qualification, and spare and replacement part procurement, to the extent established by AEPSC and plant procedures.

4) The AEPSC QA Department provides technical direction in quality assurance matters to AEPSC and the Cook Plant, and oversees the adequacy and implementation of the QA Programs through review and audit activities.

Quality Assurance Responsibility of I&MECo - D.C. Cook Plant

As owner and operator, I&MECo operates the Cook Plant per licensing requirements, including the Technical Specifications and such other commitments as established by the operating licenses. The Plant Manager Instruction (PMI) system and subtier instructions and procedures describe the means by which compliance is achieved and responsibilities are assigned, including interfaces with AEPSC. Figure 1.7-3 indicates the organizational relationships within the AEP System pertaining to the operation and support of the Cook Plant.

1.7.1.2.5 Organization (AEPSC)

The Chairman of the Board and Chief Executive Officer is ultimately responsible for the Quality Assurance Program associated with the Cook Plant. This responsibility has been functionally delegated to the AEPSC Vice Chairman - Engineering and Construction. The AEPSC Vice Chairman - Engineering and Construction has further delegated responsibilities which are administered through the following division and department management personnel:

- AEPSC Manager of Quality Assurance
- AEPSC Vice President - Nuclear Operations
- AEPSC Executive Vice President and Chief Engineer

Quality Assurance Department

The AEPSC Manager of Quality Assurance reports to the AEPSC Vice Chairman - Engineering and Construction and is responsible for the Quality Assurance Department. The Quality Assurance Department consists of the following positions and sections (Figure 1.7-4):

- Quality Assurance Engineering Section
- Audits and Procurement Section
- Training and Procedures Specialist
- Quality Assurance Staff Specialist
- D.C. Cook Plant Site Quality Assurance Section

The Quality Assurance Department is organizationally independent and is responsible to perform the following:

- Identify quality problems.
- Initiate, recommend, or provide solutions through designated channels.
- Verify implementation of solutions.
- Prepare issue and maintain Quality Assurance Program documents, as required.
- Verify the implementation of the Quality Assurance Program through scheduled audits and surveillances.
- Review engineering, design, procurement, construction and operational documents for incorporation of, and compliance with applicable quality assurance requirements to the extent specified by the AEPSC management approved QA Program.
- Organize and conduct the QA orientation, training, certification and qualification of AEPSC personnel.
- Provide general guidance, when requested, for the collection, storage, maintenance, and retention of quality assurance records.
- Establish and maintain a Qualified Suppliers List (QSL) of nuclear (N) items and services.
- Identify noncompliances of the established QA Program to the responsible organizations for corrective actions and report significant occurrences that jeopardize quality to senior AEPSC management .

- Follow up on corrective actions identified by QA during and after disposition implementation.
- Assure that conditions adverse to quality are dispositioned to preclude recurrence.
- Conduct in-process QA surveillance at supplier's facilities, as required.
- Assist and advise other AEP/AEPSC groups in matters related to the Quality Assurance Program.
- Maintain a list of nuclear grade items (N-List) for the D.C. Cook Plant.
- Establish a mechanism for identifying, tracking and closing out quality-related commitments.
- Conduct audits as directed by the Nuclear Safety and Design Review Committee (NSDRC).
- Review AEPSC originated nonconformances, noncompliances and associated corrective action recommendations.
- Maintain cognizance of industry and governmental quality assurance requirements such that the Quality Assurance Program is compatible with requirements, as necessary.
- Recommend for revision to, or improvements in the established QA Program to senior AEPSC management.
- Issue "Stop Work" orders when significant conditions adverse to quality are identified to prevent unsafe conditions from occurring and/or continuing.
- Provide AEPSC management with periodic reports concerning the status, adequacy and implementation of the QA Program.
- Prepare and conduct special verification and/or surveillance programs on in-house activities, as required or requested.
- Routine attendance and participation in daily plant work schedule and status meetings.
- Provide adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments.

Amplification of Specific Responsibilities

- Qualification of the AEPSC Manager of Quality Assurance

  The AEPSC Manager of Quality Assurance shall possess the following position requirements:

  - Bachelor's degree in engineering, scientific or related discipline.
  - Ten (10) years experience in one or a combination of the following areas: engineering, design, construction, operations, maintenance of fossil or nuclear power generation facilities or utility facilities; Quality Assurance; of which at least four (4) years must be experience in quality assurance related activities.
  - Knowledge of QA regulations, policies, practices and standards.
  - The same or higher organization reporting level as the highest line manager directly responsible for performing activities affecting quality such as engineering, procurement, construction and operation, and is sufficiently independent from cost and schedule.
  - Effective communication channels with other senior management positions.
  - Responsibility for approval of QA Manual(s).
  - Performance of no other duties or responsibilities unrelated to QA that would prevent full attention to QA matters.

- Stop Work Orders

  The AEPSC Quality Assurance Department is responsible for ensuring that quality related activities are performed in a manner that meets applicable administrative, technical, and regulatory requirements. In order to carry out this responsibility, the AEPSC Vice Chairman - Engineering and Construction has given the AEPSC Manager of Quality Assurance, the authority to stop work on any quality related activity that

does not meet the aforementioned requirements. Stop work authority has been further delegated by the AEPSC Manager of Quality Assurance to the Supervisor - Quality Assurance (site).

The AEPSC Manager of Quality Assurance and the Supervisor - Quality Assurance do not have the authority to stop unit operations, but will notify appropriate plant and/or corporate management of conditions which do not meet the aforementioned criteria, and recommend that unit operations be terminated.

- <u>QA Orientation, Training, Qualification and Certification Program</u>

  a) AEPSC QA shall, if directed by AEPSC management, be responsible for establishing, maintaining and conducting a general QA orientation and training program for AEPSC personnel engaged in safety-related activities. This program includes the AEPSC QA philosophy and such facility specific programs as may be required by facility or regulatory requirements.

  b) AEPSC has established and maintains a QA Auditor training and certification program for all AEPSC QA Auditors.

- <u>Problem Identification, Reporting and Escalation</u>

  - AEPSC QA has established mechanisms for the identification and reporting and escalating safety-related problems to a level of management whereby satisfactory resolutions can be obtained.

<u>Nuclear Operations Division</u>

The AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) reports to the AEPSC Vice Chairman - Engineering and Construction and is responsible for the Nuclear Operations Division. Reporting to the AEPSC Vice President - Nuclear Operations are the following:

- Donald C. Cook Plant Manager
- Assistant Division Manager - Nuclear Engineering (not charted)
- Assistant Division Manager - Nuclear Operations (not charted)
- Consulting Nuclear Engineer - Nuclear Operations (not charted)
- Staff Engineer - Nuclear Operations (not charted).

The organization and responsibilities of the Donald C. Cook Plant Manager are defined further within this section under 1.7.1.2.6 Organization (Cook Plant).

The AEPSC Assistant Division Manager - Nuclear Engineering is responsible for two of the four sections within the Nuclear Operations Division, as follows (not charted):

- Nuclear Safety and Licensing (NS&L) Section
- Nuclear Material and Fuels Management (NMFM) Section

The AEPSC Assistant Division Manager - Nuclear Operations is responsible for the remaining two sections, as follows (not charted):

- Nuclear Operations Support (NOS) Section
- Radiological Support (RS) Section

The Nuclear Operations Division is responsible for the following:

- Assist and make recommendations on the formulation of policies and practices relative to safety and licensing, operation, maintenance, fuel management and radiological support activities for the Cook Plant.
- Provide the D.C. Cook Plant Manager with technical and managerial guidance, direction, and support to ensure safe operation of the plant.
- Coordinate and correlate division activities between AEPSC and I&MECo organizations with respect to nuclear projects.
- Provide direction to all other AEPSC engineering divisions on engineering matters pertaining to the Cook Plant.
- Provide special nuclear training for American Electric Power System personnel and maintain contacts with facilities where specialized training is available.

- Maintain liaison with the AEPSC Manager of Quality Assurance and other sources to become apprised of the latest safety, licensing and regulatory requirements, codes, standards and federal regulations applicable to the operation of the Cook Plant.
- Ensure that areas of responsibility are performed in accordance with the requirements of the QA Program.
- Provide membership on the Nuclear Safety and Design Review Committee (NSDRC).
- Provide membership on the Change Control Board (CCB) for Cook Plant.
- Provide training to the AEPSC members of the emergency response organization.
- Participate in emergency response organization activities for Cook Plant.
- Ensure that nuclear fuel material and associated nuclear material specifications, fabrication, procurement and delivery comply with established criteria, designs, and program.
- Negotiate contracts with suppliers of nuclear fuel materials and associated nuclear fuel services and provide technical support on contract.
- Ensure that the operating license for the Cook Plant is maintained.
- Supervise the review and evaluation of responses to NRC requests for information, notifications, or other NRC correspondence.
- Coordinate all activities related to public hearings and meeting with the NRC and the Advisory Committee on Reactor Safeguards (ACRS).
- Participate in, and the overall coordination of the annual FSAR update and submittal.

The basic functional areas for which each section responsibilities are as follows:

A. Nuclear Materials and Fuel Management:

  a) Reactor Physics
  b) Reload Engineering
  c) Reload Supplier Evaluation
  d) Supply of Enrichment Services

e) Technical Specifications (Core-related)
f) Liaison with Plant Nuclear Engineer
g) Core Physics Documents
h) Nuclear Fuel Supply Contracts
i) Radwaste Handling
j) Special Nuclear Material
k) Nuclear Fuel Cash Flow, Budgets
l) Nuclear Fuel Quality Assurance
m) Radwaste Disposal
n) Supply of Conversion Services
o) Fuel Cycle Cost Analysis
p) Purchase of Uranium
q) Spent Nuclear Fuel Transportation
r) Fuel Handling Cognizant Engineer

B. Nuclear Safety and Licensing:

a) Safety Reviews
b) Reviews of Proposed Changes
c) Accident Analysis
d) Seismic Coordination
e) Plant Transient Analysis
f) NRC Licensing Letters
g) Generic Licensing Issuers
h) Structural Mechanics
i) Probabilistic Risk Analysis
j) Technical Specifications
k) Failure Modes Analysis
l) Participation in the NSDRC

C. Nuclear Operations:

a) Five-year Planning Program
b) Outage Planning
c) Shift Technical Advisor Program
d) Reliability Assessment and Nuclear Plant Reliability Data System (NPRDS)
e) Availability Improvements

f) Operator Training
g) Annual Operating Report
h) Licensee Event Report Evaluation
i) Communications
j) Industry Surveys
k) Nuclear Network
l) Paperwork Reduction
m) Human Factors Engineering
n) Simulator
o) O&M Budgets
p) Design Changes and Plant Modifications
q) Regulatory Performance Improvement Program (RPIP)

D. Radiological Support:

a) Emergency Plan
b) REM System
c) ALARA
d) Meteorology
e) Environmental Radiation Monitoring
f) Radiation Monitoring
g) Health Physics Support
h) Generic Radiation Issues
i) Decommissioning
j) Respiratory Protection Program
k) Environmental Technical Specifications
l) Radiochemistry
m) Shielding
n) Radiation Dose Calculations
o) Personnel Radiation Exposure - Plant and Contractors
p) Liaison with Plant Radiation Protection
q) Radioactive Sampling
r) Plant Security
s) Radiation Training

Environmental Engineering Division

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for the Environmental Engineering Division through the AEPSC Assistant Vice President - Environmental Engineering. The Environmental Engineering Division provides a nonsafety-related function for the Cook Plant with exception of its participation on the Nuclear Safety and Design Review Committee (NSDRC).

Engineering Administration

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for Engineering Administration through the AEPSC Vice President - Engineering Administration. The AEPSC Vice President - Engineering Administration is responsible for the following divisions:

- Civil Engineering Division
- Design Division
- Materials Handling Division

Civil Engineering Division

The AEPSC Division Manager - Civil Engineering, reporting to the AEPSC Vice President - Engineering Administration, is responsible for the Civil Engineering Division. The Civil Engineering Division consists of the following sections (not charted):

- Structural Engineering Section
- Civil Engineering Laboratory Section
- Soils, Foundation and Hydro Section
- Survey and Mapping Section

The Civil Engineering Division is responsible for the following:

- Make recommendations and assist in the formulation of policies and practices relating to the structural design and engineering of

nuclear power plants, office and service buildings, and miscellaneous structures, and provide the general supervision of the structural engineering of such facilities and structures.

- Establish and maintain files of design, testing and construction documents for record purposes.
- Arrange for outside engineering and consulting assistance as required.
- Approve improvement requisitions for capital expenditures.
- Approve invoices for outside services.
- Approve purchase requisitions and contracts as authorized.
- Approve Requests for Change (RFC) pertaining to nuclear generating plants.
- Assist in overall coordination of the work of the division with other engineering divisions and interfacing organizations.
- Initiate and maintain a program of development and training for personnel in the division.
- Prepare specifications, procurement of civil/structural items and modifications to same relative to the Civil Engineering Division.
- Direct and coordinate the preparation of specifications and instructions to bidders for general construction and structural features of power plants and buildings and evaluate proposals received; make recommendations for the award of contracts.
- Direct and coordinate the preparation of contracts for the structural phases of power plant and building design and construction.
- Direct and coordinate the preparation of specifications and instructions to bidders for general construction and structural features of power plants and buildings and evaluate proposals received; make recommendations for the award of contracts.
- Provide services to the field organizations, including the assignment of personnel to the field during construction, normal or emergency outages, or as requested.
- Assist in planning and execution of maintenance work on buildings and other structures.
- Prepare site studies.

- Arbitrate disputes which arise between construction forces and outside suppliers of materials and services.
- Coordinate structural consultant's reports with design.
- Participate in periodic inspections of contractors' work.
- Check of structural drawings submitted for review.
- Direct the inspection of coating (painting) operations performed by contractors.
- Review and recommend concrete mix formulations for all new construction.
- Supervise maintenance and repairs of all masonry and concrete work in the AEP System, including supplying trained inspection personnel.
- Direct testing of materials used in concrete and testing of soils to be used in work throughout the AEP System.

Design Division

The AEPSC Division Manager - Design, reporting to the AEPSC Vice President - Engineering Administration, is responsible for the Design Division. There are two Assistant Division Managers reporting to the AEPSC Design Division Manager who are responsible for various sections as follows (not charted):

Assistant Division Manager
- Architectural Design Section
- Mechanical Design Section
- Structural Design Section

Assistant Division Manager
- Electrical Plant Design Section
- Control Services Group

The Design Division is responsible for the following:
- Formulate, administer, and implement policies and practices relating to the design of power plants and miscellaneous structures.

- Direct the development, maintenance, procedural review and implementation by which the Design Division adheres to the QA Program elements as established by the AEPSC General Procedures Manual.
- Conduct periodic management reviews and surveillances of division activities to ensure compliance with QA Program objectives, and external surveillances as necessary, of consultants outside organizations and vendors for which the division is cognizant.
- Conduct functions of the division so as to be in conformance with the operating licenses of the Cook Plant.
- Coordinate the review and/or answering of corrective actions issued and assigned to the Design Division.
- Coordinate special projects and studies, as required.
- Establish and maintain files of design documents for record purposes.
- Initiate and/or implement and control design changes and modifications.
- Coordinate the development and maintenance of the computerized Design Drawing Control (DDC) and the Vendor Drawing Control (VDC) programs which include coordinating the programs with interfacing divisions/departments.
- Control the issuance and distribution of drawings for the Cook Plant including monitoring of the Aperture Card Microfilm Program.
- Maintain the temporary QA record storage facility.
- Supervise and control the work of consultants, Architect/Engineers and outside design agencies supplying services to AEP in their discipline and process notification of defects in accordance with company requirements. Also perform detailed reviews of design work submitted by outside agencies.
- Supervise the identification of critical design decisions and ensure appropriate analyses and reviews are provided. Review, approve and/or sign off all design drawings prior to issuance.
- Provide to the field organizations such services as required during construction, normal or emergency outages or as requested, including assigning design personnel to the field.
- Maintain an up-to-date list of all major approved materials and specifications used within the division's scope of responsibility.

- Initiate and/or aid in the responses of reportable items as described in the AEPSC General Procedures and division procedures.
- Schedule, develop, coordinate and control design studies calculations/analysis, drawings, purchase documents, specifications and other design activities, as assigned for system, components or structures within the division's responsibility.
- Review and update, as required, the Cook Plant Final Safety Analysis Report (FSAR).
- Perform functions related to the Cook Plant as required in response to NRC requirements.
- Participate on committees that review nuclear activities as appointed or assigned.
- Coordinate and resolve design comments made by interfacing departments/divisions.
- Prepare, review approve and administer design specifications and purchase documents for design services and/or materials.
- Initiate and/or aid in the responses of reportable items as described in the AEPSC General Procedures and division procedures.
- Participate in the Initial Assessment Group (IAG) and provide assistance to on-site personnel and other divisions.
- Identify and report deficiencies in the division's functions, duties, and responsibilities.
- Coordinate the implementation of division commitments.

Materials Handling Division

The AEPSC Division Manager - Materials Handling, reporting to the AEPSC Vice President - Engineering Administration, is responsible for the Materials Handling Division. The Materials Handling Division contains one section that performs safety-related work as follows (not charted):

- Coal and Materials Handling Section

The Material Handling Division is responsible for the following:

- Develop policies and practices relating to the engineering of materials handling installations for Donald C. Cook Nuclear Plant.
- Review the activities of materials handling systems for the Cook Plant and approve, as required, all design changes and modifications

including the preparation of specifications, procurement of equipment and modifications to equipment.

- Arrange for outside engineering and consulting services, as required.
- Provide training and development programs necessary for personnel of the division (including the company's safety and health program), which are consistent with the written policy of American Electric Power company and American Electric Power Service Corporation.
- Prepare and administer erection and service contracts.
- Review and evaluate proposals and make recommendations for awards of purchase orders and contracts.
- Prepare, review and approve specifications, purchase and change documents, sketches, drawings, design input, design verifications and calculations, as required.
- Initiate and/or review approval and control of laboratory and field investigations, feasibility studies, improvement requisitions, reports and cost estimates pertaining to the Cook Plant.
- Provide field services to the Cook Plant including the assigning of personnel as are required during construction, normal or emergency outages, or as requested.
- Direct the review of, and response to corrective actions assigned to the Material Handling Division.
- Identify critical engineering and design input and ensure that appropriate analysis and reviews are conducted.
- Implement a corrective action system with regard to all safety-related activities of the division that will control and document all items, services, or activities which do not conform to requirements.
- Maintain a surveillance program in support of the Quality Assurance Program and review and approve the activities of this program which can be separated into the following two areas:
    - Internal management review of the Materials Handling Division.
    - External technical surveillance of consultants, outside materials handling organizations and vendors over which the division is cognizant.

- Assist in planning and execution of maintenance work on equipment and facilities.
- Review and approve manufacturer's equipment drawings prior to fabrication.
- Prepare design criteria, engineering standards conceptual layouts, studies and procedures in conjunction with materials handling equipment at the Cook Plant.
- Assist in the preparation of applications for federal, state and local permits relative to installations being made which require such permits.
- Perform shop and field inspections on equipment being fabricated or installed which is within the scope of the division's responsibility.
- Provide input for special studies and reports which may be requested by other divisions or governmental agencies such as the Nuclear Regulatory Commission.
- Provide technical guidance when requested in support of maintenance and operations activities at the Cook Plant.
- Conduct periodic management reviews of the activities of the division to ensure compliance with the objectives of the Quality Assurance Program, and external technical surveillance, as necessary, of consultants, outside materials handling organizations and vendors over which the division is cognizant.
- Establish and maintain a permanent file for QA records.
- Process RFCs in accordance with AEPSC General Procedures and division procedures.

Electrical Engineering Department

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for the Electrical Engineering Department through the AEPSC Senior Vice President - Electrical Engineering and Deputy Chief Engineer. Reporting to AEPSC Senior Vice President - Electrical Engineering and Deputy Chief Engineer is the AEPSC Manager - Generation and Telecommunications Engineering Division. The Generation and Telecommunications Engineering

Division (not charted) is the only division within the Electrical Engineering Department that is responsible for performance of electrical oriented safety-related activities. The AEPSC Assistant Manager -

Generation and Telecommunications Engineering Division reports to the AEPSC Manager - Generation and Telecommunications Engineering Division and is responsible for the section within the Electrical Engineering Department that is responsible for safety-related activities as follows (not charted):

- Electrical Generation Section

The Electrical Generation Section is responsible for the following:

- Review and recommend for approval, improvement requisitions for all capital expenditures pertaining to electrical facilities.
- Approve purchase requisitions and contract vouchers for electrical facilities.
- Perform and evaluate economic studies, investigations, analysis and reports for electrical facilities pertaining to the design, operation and maintenance of the generating plants.
- Provide guidance and advice to AEP System Companies on engineering matters.
- Maintain a constant awareness for improvements and more economic design of equipment, electric facilities, maintenance and operating methods or procedures.
- Assign membership to the Nuclear Safety and Design Review Committee (NSDRC) audit subcommittees, participating in matters covered in the committee's charter.
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization.
- Determine general layout and design of electrical facilities.
- Prepare, review and approve one-line and elementary diagrams.
- Coordinate plant and system electrical facilities.
- Prepare and/or approve specifications and purchase requisitions, and perform drawing review of electrical equipment, including control and protective relays.

- Assist field personnel in installation, start-up and the subsequent locating of problems in protective, control, or electrical equipment and in determining proper operation of equipment during normal or after emergency operations.
- Establish relay and control standards.
- Assist field personnel in making changes in equipment or control to improve operating conditions or to reduce maintenance or operating staff.
- Supervise the Electrical Engineering Cook Plant Support Team to ensure that all corporate policies and procedures are implemented.
- Maintain a constant awareness of the activities of the Cook Plant Support Team to insure compliance with all applicable procedures initiating, when required, training or retraining programs.
- Review and approve all procedures, purchase requisitions, correspondence, elementary diagrams, requests for design changes or modifications as appropriate.
- Review and approve responses to NRC correspondence as required.
- Closely follow manufacturers' engineering and designs to assure provision of adequate and reliable equipment and circuitry in the areas of turbine-generator protective controls, switchgear, electrical auxiliaries, and protective devices upon which depend the safety, reliability, economy and performance of the unit and plant.
- Coordinate work with plant resident electrical engineer in scheduling electrical engineering, design, purchase and construction activities.
- Direct calculations for proper application and settings of protective relays.
- Coordinate with the Mechanical Engineering Division to insure that all electrical devices purchased with mechanical equipment conform to accepted standards and fulfill the desired function.
- Prepare cable specifications, develop application criteria, establish and maintain system cable stock and write purchase orders for system and plant requirements for control, communication and instrumentation cables and plant requirements for power cables.

Mechanical Engineering Division

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for

the Mechanical Engineering Division through the AEPSC Assistant Vice President - Mechanical Engineering. Reporting to the AEPSC Assistant Vice President - Mechanical Engineering, are the following (not charted):

- AEPSC Assistant Division Manager(s)
- Consulting Mechanical Engineer - Nuclear
- Staff Engineer - Chief Metallurgist

Further, the AEPSC Assistant Division Manager - Nuclear is responsible for the following positions and sections (not charted):

- Nuclear Project Engineer(s)
- Turbine and Cycle Evaluation Section
- Chemical Engineering Section
- Heat Exchangers and Pumps Section
- Piping and Valves Section
- Instrumentation and Control Section
- Fire Protection and HVAC Section
- Analytical and R&D Section

The Mechanical Engineering Division is responsible for the following:

- Provide technical engineering support in areas of operation and maintenance, including: the Inservice Inspection (ISI) Program; the Quality Assurance Program; the AEP ALARA Program covering radiation protection, and; the corporate and plant Industrial Safety Program.
- Provide engineering support for the other AEPSC engineering divisions, as well as for the manufacturers, suppliers, or constructors of equipment and systems.
- Provide engineering support to the AEPSC Nuclear Operations Division.
- Preparation of equipment specifications and purchase requisitions for plant equipment, major spare parts and services related to specific areas of responsibility of MED.

- Provide technical direction and assistance to the AEPSC Design Division in the layout and arrangement of equipment, piping, systems, controls, etc., for the development of drawings.
- Develop system flow diagrams and progressive reviews to determine the adequacy of system designs.
- Provide technical assistance to the Cook Plant for use and control of special processes, including welding, heat treating, nondestructive examination, etc.
- Initiate and develop design changes in areas of responsibility of the Mechanical Engineering Division.
- Develop System Descriptions and Descriptive Articles.
- Provide support personnel for the emergency response organization.
- Provide analytical support in engineering disciplines (e.g., heat transfer, thermodynamics, fluid dynamics).
- Review and approval of mechanical design drawings.
- Provide Engineering evaluations for Condition Reports, LERs, INPO-SOERs and NRC Bulletins.

Plant Construction Division

The AEPSC Assistant Vice President - Plant Construction Division reports to the AEPSC Vice Chairman - Engineering and Construction, and is responsible for the Plant Construction Division. The Plant Construction Division consists of the following sections (not charted):

- Administrative Section
- Construction Contracts Section

The Plant Construction Division is responsible for the following:

- Provide a Construction Manager, reporting administratively to the AEPSC Assistant Vice President - Plant Construction Division and functionally to the Cook Plant Manager, to perform major modifications and maintenance work.
- Scope, bid and make recommendations relative to construction contracts.
- Administer contracts throughout the construction period.

Purchasing and Stores Department (not charted)

The AEPSC Executive Vice President - Operations reporting to the AEPSC President and Chief Operating Officer is responsible for the Purchasing and Stores Department through the AEPSC Vice President - Purchasing and Stores.

The Purchasing and Stores Department is responsible for the following:

- Purchasing "N" items only from suppliers appearing on the Qualified Suppliers List (QSL).
- Coordinate procurement activities with AEPSC Nuclear Operations and Engineering Divisions, the AEPSC Quality Assurance Department and Cook Plant personnel.
- Prepare and issue requests for quotations, contracts, service orders and purchase orders for "N" items.
- Establish a system to implement corrective action as described in the AEPSC General Procedures for the Cook Plant.
- Establish a system of records keeping, document transmittal, and document retention. QA records are maintained in accordance with the requirements of ANSI N45.2.9-1974 for the life of the plant.
- Establish a system of document control for controlled procedures, instructions, and purchasing documents for "N" items.
- Conduct training sessions involving purchasing personnel and others on an annual basis or more frequently, as required, and ascertain that training sessions include complete responsibilities associated with the purchase of safety-related items.
- Notify suppliers of their status regarding the QSL, e.g., inclusion, exclusion, conditional approval, etc.
- Notify the Indiana & Michigan Electric company Purchasing Department and the Cook Plant Stores of changes in the QSL.
- Receipt inspection, handling, storage and control of stores items.

1.7.1.2.6 Organization (Cook Plant)

The Plant Manager reports functionally and administratively to the AEPSC Vice President - Nuclear Operations Division (Manager of Nuclear Opera-

tions) and is responsible for the Cook Plant activities. Reporting to the Plant Manager are the following (Figure 1.7-5):

- Assistant Plant Manager - Maintenance
- Assistant Plant Manager - Operations
- Administrative Superintendent
- Quality Control Superintendent (reports functionally to the Plant Manager)

The Cook Plant organization, under the Plant Manager is responsible for the following:

- Ensure the safety of all facility employees and the general public relative to general plant safety, as well as radiological safety by maintaining strict compliance with plant Technical Specifications, procedures and instructions.
- Recommend facility engineering modification and initiate and approve plant improvement requisitions.
- Ensure that work practices in all plant departments are consistent with regulatory standards, safety, approved procedures, and plant Technical Specifications.
- Provide membership, as required, on the Plant Nuclear Safety Review Committee.
- Maintain close working relationships with the NRC as well as local, state, and federal government regulating officials regarding conditions which could affect, or are affected by Cook Plant activities.
- Set up plant load schedules and arrange for equipment outages.
- Develop and efficiently implement all site centralized training activities.
- Direct all facility personnel and safety programs.
- Administer the centralized facility training complex, simulator, and programs ensuring that program development is consistent with the systematic approach to training, INPO, regulatory and corporate requirements.
- Ensure that human resource activities include employee support programs consistent with INPO/NUMARC guidelines, company policies, and regulatory requirements and standards.

- Administer the NRC approved physical Security Program in compliance with regulatory standards, Modified Amended Security Plan, and company policy.
- Supervise, plan, and direct the activities related to the maintenance and installation of all power plant equipment, structures, grounds, and yards.
- Prepare plant maintenance budgets, construction budgets, improvement requisitions, and work orders.
- Prepare and maintain records and reports pertinent to equipment maintenance, cost histories, regulatory agency requirements.
- Administer contracts and schedule outside contractors' work forces.
- Enforce and coordinate plant regulations, procedures, policies, and objectives to assure safety, efficiency, and continuity in the operation of the Cook Plant within the limits of the operating license and the Technical Specifications and formulation of related policies and procedures.
- Plan, schedule, and direct the activities relating to the operation of the Cook Plant and associated switchyards; cooperate in planning and scheduling of work and procedures for refueling and maintenance of the Cook Plant; direct and coordinate fuel loading operations.
- Review reports and records and direct general inspection of operating conditions of plant equipment and investigate any abnormal conditions, making recommendations for repairs. Establish and administer equipment clearance procedures consistent with company, plant, and radiation protection standards; authorize and arrange for equipment outages to meet normal or emergency conditions.. Provide the shift operating crews with appropriate procedures and instructions to assist them in operating the plant safely and efficiently.
- Approve operator training programs administered by the Cook Plant Training Department designed to provide operating personnel with the knowledge and skill required for safe operation of the facility and for obtaining and holding NRC operator licenses. Coordinate training programs in plant safety and emergency procedures for Cook Plant Operating Department personnel to ensure that each shift group

will function properly in the event of injury of personnel, fire, nuclear incident, or civil disorder.

- Advance planning and overall conduct of scheduled and forced outages, including the scheduling and coordination of all plant activities associated with refueling, preventive maintenance, corrective maintenance, equipment overhaul, Technical Specification surveillances, and design change installations.
- Coordinate all plant activities associated with the initiation, review, approval, engineering, design, production, examination, inspection, test, turnover, and close out of design changes.
- Develop and implement an effective Quality Control Program. This encompasses, but is not limited to, the planning and directing of quality control activities to assure that industry codes, Nuclear Regulatory regulations, and company instructions and policies regarding quality control for the nuclear generating station are enforced, and that these activities are properly documented.
- Prepare reports of reportable occurrences which are mandated by the NRC and the Technical Specifications.
- Direct the activities of contractor QC/NDE personnel assigned to the QC Department and provide inspections of work performed.
- Prepare statistical reports utilized in Nuclear Regulatory Appraisal Meetings and Enforcement Conference.
- Coordinate the efforts of outside agencies such as American Nuclear Insurers (ANI), Institute of Nuclear Power Operations (INPO), and Third Party Inspector Programs.
- Maintain knowledge of developments and changes in NRC requirements, industry standards and codes, regulatory compliance activities, and quality control disciplines and techniques.
- Stop plant operation in the event that conditions are found which are in violation of the Technical Specifications or adverse to quality.
- Qualification and certification of Quality Control ensuring compliance to ANSI N45.2.6 and SNT-TC-1A criteria, as applicable.

- Conduct of the Quality Control Program, including recommendations for improvement.
- Procurement, receiving, quality control receipt inspection, storage, handling, issue, stock level maintenance, sale, and overall control of stores nuclear and standard grade material, components, and equipment.
- Provide material service and support in accordance with policies and procedures required by AEP Purchasing and Stores, AEPSC Quality Assurance, and the Nuclear Regulatory Commission (NRC), which are administered and enforced in a total effort to ensure safety and plant reliability.
- Plan and direct engineering and technical studies, nuclear fuel management, equipment performance, instrument and control maintenance, on-site computer systems, Shift Technical Advisors, and emergency planning for the Cook Plant. These activities support daily on-site operations in a safe, reliable, and efficient manner in accordance with all corporate policies, applicable laws, regulations, licenses, and Technical Specification requirements.
- Implement station performance testing and monitor programs to ensure optimum plant efficiency.
- Direct programs related to on-site fuel management and reactor core physics testing and ensure satisfactory completion.
- Establish testing and preventive maintenance programs related to station instrumentation, electrical systems, and computers.
- Recommend alternatives to plant operation, technical or emergency procedures, and design of equipment to improve safety of operations and overall plant efficiency.
- Implement the corporate Emergency Plan as it pertains to the D.C. Cook Plant site.
- Provide technical and engineering services in the fields of chemistry, radiation protection, ALARA, and environmental in support of the safe operation of the plant and the health and safety of the employees and the public.
- Plan and schedule the activities of the Physical Sciences Sections of the plant in support of operations and maintenance.

- Establish chemistry, radiochemistry, and health physics criteria which ensure maximum equipment life and the protection of the health and safety of the workers and the public.
- Establish sampling and analysis programs which ensure the chemistry, radiochemistry, and health physics criteria are within the established criteria.
- Establish and direct investigations, responses, and corrective actions when outside the established criteria.
- Administer and direct the plant's radioactive waste programs, including volume reduction, packaging and shipping.

1.7.2 QUALITY ASSURANCE PROGRAM

1.7.2.1 SCOPE

Policies that define and establish the D.C. Cook Nuclear Plant Quality Assurance Program are summarized in the individual sections of this document. The program is implemented through procedures and instructions responsive to provisions of the QAPD, and will be carried out for the life of the plant.

Quality assurance controls apply to activities affecting the quality of safety-related structures, systems and components, to an extent based on the importance of those structures, systems, or components to safety. Such activities are performed under controlled conditions, including the use of appropriate equipment, environmental conditions, assignment of qualified personnel, and assurance that all applicable prerequisites have been met.

Safety related structures, systems or components are defined as items:

- which are associated with the safe shutdown (hot) of the reactor; or isolation of the reactor; or maintenance of the integrity of the reactor coolant system pressure boundary.

or

- whose failure might cause or increase the severity of a design basis accident as described in the FSAR; or lead to a release of radioactivity in excess of 10 CFR100 limits.

In general, items are safety related if they are: classified as Seismic Class I, or Electrical Class IE; or associated with the Engineered Safety Features Actuation System; or associated with the Reactor Protection System.

A special QA program has been implemented for Fire Protection items (Section 1.7.19 herein).

Quality Assurance Program status, scope, adequacy, and compliance with 10CFR50, Appendix B, are regularly reviewed by AEPSC management through reports, meetings, and review of audit results.

The implementation of the Quality Assurance Program may be accomplished by AEPSC and/or Indiana & Michigan Electric Company or delegated in whole or in part to other AEP System companies or outside parties. However, AEPSC and/or Indiana & Michigan Electric Company retain full responsibility for all quality-related activities. The performance of the delegated organization is evaluated by audit or surveillances on a frequency commensurate with their scope and importance of assigned work.

1.7.2.2 IMPLEMENTATION

1.7.2.2.1

The Chairman of the Board of AEPSC, as Chief Executive Officer, has stated in a formal "Statement of Policy", signed by him, that it is corporate policy to comply with the provisions of applicable codes, standards and regulations pertaining to quality assurance for nuclear power plants as required by the Donald C. Cook Nuclear Plant operating licenses. The statement makes this QAPD and the associated implementing procedures and instructions mandatory, and requires compliance by all responsible organizations and individuals. It identifies the management

positions within the companies vested with responsibility and authority for implementing the program and assuring its effectiveness.

1.7.2.2.2

The Quality Assurance Program at AEPSC and the plant consist of controls exercised by organizations responsible for attaining quality objectives, and by organizations responsible for assurance functions.

The QA Program effectiveness is continually assessed through management review of various reports, NSDRC review of the QA audit program and periodically by independent outside parties.

The QA program described in this QAPD is intended to apply for the life of the D.C. Cook Nuclear Plant.

The QA program applies to activities affecting the quality of safety-related structures, systems, components, and related consumables during plant operation, maintenance, testing, and all modifications. Safety-related structures, systems and components are identified in Nuclear (N) Lists and other documents which are developed and maintained for the plant.

1.7.2.2.3

This QAPD, organized to present the Quality Assurance Program for the D.C. Cook Nuclear Plant in the order of the 18 criteria of 10CFR50, Appendix B, states AEPSC policy for each of the criteria, and describes how the controls pertinent to each are carried out. Any changes made to this QAPD that do not reduce the commitments previously accepted by the NRC must be submitted to the NRC at least annually. Any changes made to this QAPD that do reduce the commitments previously accepted by the NRC must be submitted to the NRC and receive NRC approval prior to implementation. The submittal of the changes described above shall be made in accordance with the requirements of 10CFR50.54.

The program described in this QAPD will not be changed in any way that would prevent it from meeting the criteria of 10CFR50, Appendix B and other applicable operating license requirements.

1.7.2.2.4

Documents used for implementing the provisions of this QAPD include the following:

Plant Manager Instructions (PMIs) establish the policy for compliance with quality-related criteria, and assign responsibility to the various departments, as required, for implementation. Department Head Instructions (DHIs) have been prepared, when required, to implement those activities for each department. Department Head Procedures (DHPs) have been prepared to describe the detailed activities required to support safe and effective plant operation.

The PMIs are reviewed by the AEPSC Site Quality Assurance Supervisor for concurrence that they will satisfactorily implement regulatory requirements and commitments. They are then reviewed by the Plant Nuclear Safety Review Committee (PNSRC) prior to approval by the Plant Manager.

Safety related DHIs and DHPs are reviewed by the department head of origination, AEPSC Site Quality Assurance Supervisor, PNSRC and Plant Manager prior to use.

AEPSC General Procedures (GPs) are utilized to define corporate policies and requirements for quality assurance, and to implement applicable quality assurance requirements within AEPSC.

GPs may also be used to define policies which are nonprocedural in nature.

When contractors perform work on-site under their own quality assurance programs, the programs are reviewed for compliance and consistency with the applicable requirements of the Plant's Quality Assurance Program and

the contract, and are approved by the AEPSC Site Quality Assurance Supervisor, PNSRC and Plant Manager prior to the start of work.

1.7.2.2.5

Provisions of the Quality Assurance Program for the D.C. Cook Nuclear Plant apply to activities affecting the quality of safety-related structures, systems, and components. Appendix A to this QAPD lists the ANSI Standards and Regulatory Guides that identify AEPSC's commitment. Appendix B describes necessary exceptions and clarifications to the requirements of those documents. The scope of the program and the extent to which its controls are applied, are established as follows:

a) AEPSC uses the criteria specified in the D.C. Cook Plant Final Safety Analysis Report (FSAR) for identifying structures, systems and components to which the Quality Assurance Program applies.

b) This identification process results in the N-List for the D.C. Cook Nuclear Plant. This N-List is a controlled document, issued to designated personnel. N-List items are determined by engineering analysis of the function(s) of plant structures, systems and components in relation to safe operation and shutdown.

c) The extent to which controls specified in the Quality Assurance Program are applied to N-list items is determined for each item considering its relative importance to safety. Such determinations are based on data in such documents as the plant Technical Specifications and the FSAR.

1.7.2.2.6

Activities affecting safety are accomplished under controlled conditions. Preparations for such activities include consideration of the following:

a) Assigned personnel are qualified.

b) Work has been planned to applicable engineering and/or Technical Specifications.

c) Specified equipment and/or tools are available.

d) Materials and items are in an acceptable status.

e) Systems or structures on which work is to be performed are in the proper condition for the task.

f) Proper instructions/procedures for the work are available for use.

g) Items and facilities that could be damaged by the work have been protected, as required.

h) Provisions have been made for special controls, processes, tests and verification methods.

1.7.2.2.7

Responsibility and authority for planning and implementing indoctrination and training are specifically designated, as follows:

a) The Training and Indoctrination Program provides for on-going training and periodic refamiliarization with the Quality Assurance Program for the D.C. Cook Nuclear Plant.

b) Personnel who perform inspection and examination functions are qualified in accordance with requirements of Regulatory Guide 1.58, SNT TC-1A, or the ASME Code, as applicable and with exceptions as noted in Appendix B hereto.

c) Personnel who participate in Quality Assurance Audits are qualified in accordance with Regulatory Guide 1.146.

d) Personnel assigned duties such as special cleaning processes, welding, etc., are qualified in accordance with applicable codes, standards and regulatory guides.

e) The Training/Qualification Program includes, as applicable, provisions for retraining, reexamination and recertification to ensure that proficiency is maintained.

f) Training and qualification records including documentation of objectives, content of program, attendees and dates of attendance are maintained at least as long as the personnel involved are performing activities to which the training/qualification is relevant.

g) Personnel responsible for performing activities that affect quality are instructed as to the purpose, scope and implementation of the applicable quality related manuals, instructions and procedures.

Management/supervisory personnel receive functional training to the level necessary to plan, coordinate and administer the day-to-day verification activities of the QA program for which they are responsible.

Training of AEPSC and plant personnel is performed employing two techniques, as applicable: 1) on the job and formal training administered by the department or section the individual works for; and 2) formal training conducted by NRC licensed instructors from the Training Department or other entities (internal and external to the AEP System). Records of training sessions for such training are maintained. Where personnel qualifications or certifications are required, these certifications are performed on a scheduled basis (consistent with the appropriate code or standard).

Plant employees receive introductory training in quality assurance usually within the first two weeks of employment. In addition, AEPSC personnel receive training prior to being allowed unescorted access to the plant. This training includes management's policy for implementation of the Quality Assurance Program through Plant Manager and Department Head Instructions and Procedures. These instructions also include a description of the Quality Assurance Program, the use of instructions and

procedures, personnel requirements for procedure compliance and the systems and components controlled by the Quality Assurance Program.

1.7.2.2.8

The AEPSC Information System Department (not charted) has established a Computer Software Quality Assurance Section. Procedures are being developed to establish QA requirements for safety-related computer software. The Computer Software QA Section will be subject to periodic audit by the AEPSC QA Department.

1.7.3 DESIGN CONTROL

1.7.3.1 SCOPE

Modifications to structures, systems and components are accomplished in accordance with approved design. Activities to develop such designs are controlled. Depending on the type of modification, these activities include design and field engineering; the performance of physics, seismic, stress, thermal, hydraulic, radiation and Safety Analysis Report (SAR); accident analyses; the development and control of associated computer programs; studies of material compatibility; accessibility for inservice inspection and maintenance; and determination of quality standards. The controls apply to preparation and review of design documents, including the correct translation of applicable regulatory requirements and design bases into design, procurement and procedural documents.

1.7.3.2 IMPLEMENTATION

1.7.3.2.1

Modifications to the plant are controlled by instructions and procedures. All modifications are reviewed as required by 10CFR50.59.

differences from the proven design are documented and evaluated for the intended application.

Qualification testing of prototypes, components, or features is used when the ability of an item to perform an essential safety function cannot otherwise be adequately substantiated. This testing is performed before plant equipment installation where possible, but always before reliance upon the item to perform a safety-related function. Qualification testing is performed under conditions that simulate the most adverse design conditions, considering all relevant operating modes. Test requirements, procedures and results are documented. Results are evaluated to assure that test requirements have been satisfied. Modifications shown to be necessary through testing are made, and any necessary retesting or other verification is performed. Test configurations are clearly documented.

Design reviews are performed by multi-organizational or interdisciplinary groups, or by single individuals. Criteria are established to determine when a formal group review is required, and when review by an individual is sufficient.

1.7.3.2.9

Persons representing applicable technical disciplines are assigned to perform design verifications. These persons are qualified by appropriate education or experience but are not directly responsible for the design. The designer's immediate supervisor may perform the verification, provided that:

1) The supervisor is the only technically qualified individual.

2) The supervisor has not specified a singular design approach, ruled out design considerations, nor established the design inputs.

3) The need is individually documented and approved in advance by the supervisor's management.

4) Regularly scheduled QA audits verify conformance to items 1 through 3 above.

Design verification on safety-related design verification shall be completed prior to declaring a design change operational.

1.7.3.2.10

Plant implementation of the RFC is accomplished by the Plant Manager assigning a specific plant department the responsibility for coordinating the design change. Material to perform the design change must meet the specifications established for the original system or as specified by the lead engineer. For those design changes where testing after completion is required, the testing documentation is reviewed by the organization performing the test and, when specified, by the AEPSC lead engineer or cognizant engineer. Further, completed RFCs are reviewed by AEPSC QA (Site) following installation and testing.

1.7.3.2.11

Changes to design documents, including field changes, are reviewed, approved and controlled in a manner commensurate with that used for the original design. Such changes are evaluated for impact. Information on approved changes is transmitted to all affected organizations.

1.7.3.2.12

Error and deficiencies in, and deviations from approved design documents are identified and dispositioned in accordance with established design control and/or corrective action procedures.

1.7.3.2.13

This mechanism provides for: 1) controlled submission of design changes, 2) engineering evaluation, 3) review for impact on nuclear safety, 4) review by AEPSC QA, 5) design modification, 6) AEPSC managerial review, and 7) approval and record keeping for the implemented design change.

1.7.4 PROCUREMENT DOCUMENT CONTROL

1.7.4.1 SCOPE

Procurement documents define the characteristics of item(s) to be procured, identify applicable regulatory and industry codes/standards requirements and specify supplier Quality Assurance Program requirements to the extent necessary to assure adequate quality.

1.7.4.2 IMPLEMENTATION

1.7.4.2.1

Procurement documents for safety-related materials/services originating at the plant, except as denoted below, are processed through AEPSC for review and approval. The plant may request the assistance of AEPSC cognizant engineers in any procurement activity.

Procurement control is established by instructions and procedures. These documents require that purchase documents be sufficiently detailed to ensure that purchased materials, components and services associated with safety-related structures or systems are: 1) purchased to specification and code requirements equivalent to those of the original equipment or service, 2) properly documented to show compliance with the applicable specifications, codes and standards, and 3) purchased from vendors or contractors who have been evaluated and deemed qualified.

Procedures establish the review of procurement documents to determine that: quality requirements are correctly stated, inspectable and controllable; there are adequate acceptance criteria; procurement

documents have been prepared, reviewed and approved in accordance with established requirements.

Each involved manager is responsible for procurement planning, bid solicitation and bid evaluation.

1.7.4.2.2

The N-List, in conjunction with other sources, is used to determine equipment classification. Donald C. Cook Nuclear Plant Specifications (DCC Specifications) are used to determine material and documentation requirements, codes or standards that materials must fulfill, and define the documentation that must accompany the material to the plant.

Department heads cognizant of the equipment and its quality assurance requirements review all procurement documents to assure that correct classification is made; that the appropriate plant specifications which identify quality requirements, are referenced or attached; and that the documentation requirements are properly stated. Purchase requisitions for new safety-related equipment are initiated by the AEPSC cognizant engineers who establish the initial equipment quality assurance requirements. Replacement or spare equipment is procured via the original purchase requirements. In instances where these requirements have been superseded by a revised specification, the replacement/spare part is procured to the revised requirements.

1.7.4.2.3

The contents of procurement documents vary according to the item(s) being purchased and its function(s) in the plant. Provisions of this QAPD are considered for application to service contractors also. As applicable, procurement documents include:

a) Scope of work to be performed.

c) Where quality requirements for the original items cannot be determined, requirements and controls are established by engineering evaluation performed by qualified individuals. The evaluation assures there is no adverse effect on interfaces, interchangeability, safety, fit, form, function, or compliance with applicable regulatory or code requirements. Evaluation results are documented.

d) Any additional or modified design criteria, imposed after previous procurement of the item(s), are identified and incorporated.

1.7.7.2.5

Instructions and procedures address requirements for supplier selection and control as well as procurement document control. The PMI on receipt inspection of safety-related materials addresses the program for inspection of incoming materials including a review of the documentation required under the procurement. Receipt inspection provisions apply regardless of whether procurement originates at the plant or at AEPSC. Additional inspections may apply if required by the procurement document.

Where materials and/or services are safety-related and procurement is accomplished without assistance of AEPSC, supplier selection is limited to those companies identified on the Qualified Suppliers List (QSL).

1.7.7.2.6

Materials received at the site are tagged with a "Hold" tag and placed in a designated, controlled area until receipt inspected. During receipt inspection, designated material characteristics and attributes are checked, and documentation is checked against the procurement documents. If found acceptable, the "Hold" tag is removed and replaced with an "Accepted" tag and the material is placed in a designated area of the storeroom. Material traceability to procurement documents and to end use is maintained through recording of Hold Tag and Acceptance Tag number on applicable documents.

Nonconforming materials, or missing or questionable documentation results in materials being kept on hold and placed in a designated, controlled area of the storeroom. If the nonconformance cannot be cleared, the material is either scrapped, returned to manufacturer, or dispositioned through engineering analysis.

1.7.7.2.7

Contractors providing services (on-site) for safety-related components, are required to have either a formal quality assurance program and procedures, or they must abide by the plant quality assurance program and procedures. Prior to their working at the plant, contractor quality assurance programs and procedures must be reviewed and approved by the AEPSC Site Quality Assurance Supervisor, PNSRC and the Plant Manager. Further, periodic audits of site contractor activities are conducted under the direction of the AEPSC Site Quality Assurance Supervisor.

1.7.7.2.8

Suppliers are required to furnish the following records:

a) Applicable drawings and related engineering documentation that identify the purchased item and the specific procurement requirements (e.g., codes, standards and specifications) met by the item.

b) Documentation identifying any procurement requirements that have not been met.

c) A description of those nonconformances from the procurement requirements dispositioned "accept as is" or "repair".

d) Quality records as specified in the procurement requirements.

1.7.7.2.9

The validity of supplier certificates of conformance is evaluated at the time of supplier resurvey and requalification.

1.7.8 IDENTIFICATION AND CONTROL OF ITEMS

1.7.8.1 SCOPE

Materials, parts and components (items) are identified and controlled to prevent their inadvertent use. Identification of items is maintained either on the items, their storage areas or containers, or on records traceable to the items.

1.7.8.2 IMPLEMENTATION

1.7.8.2.1

Controls are established that provide for the identification and control of materials, parts and components (including partially fabricated assemblies).

1.7.8.2.2

Items are identified by physically marking the item or its container, and by maintaining records traceable to the item. The method of identification is such that the quality of the item is not degraded.

1.7.8.2.3

Items are traceable to applicable drawings, specifications or other pertinent documents to ensure that only correct and acceptable items are used. Verification of traceability is performed and documented prior to release for fabrication, assembly, or installation.

1.7.8.2.4

Requirements for the identification by use of heat number, part number, or serial number are included in the specifications and/or purchase order.

1.7.8.2.5

Separate storage is provided for incorrect or defective materials that are on hold, and material which has been accepted for use. All safety-related materials are appropriately tagged or identified (stamping, etc.) to provide easy identification as to the materials usage status. Records are maintained for the issue of materials, to provide traceability from storage to end use in the plant.

1.7.8.2.6

When materials are subdivided, appropriate identification numbers are transferred to each section of the material, or traceability is maintained through documentation.

1.7.9 CONTROL OF SPECIAL PROCESSES

1.7.9.1 SCOPE

Special processes are controlled and are accomplished by qualified personnel using approved procedures and equipment in accordance with applicable codes, standards, specifications, criteria and other special requirements.

1.7.9.2 IMPLEMENTATION

1.7.9.2.1

Processes subject to special process controls are those for which full verification or characterization by direct inspection is impossible or impractical. Such processes include welding, heat treating, chemical

cleaning, application of protective coatings, concrete placement and nondestructive examination.

1.7.9.2.2

Special process requirements for chemical cleaning, application of protective coatings and concrete placement are set forth in AEPSC Specifications and/or directives prepared by the responsible AEPSC Cognizant Engineer. These documents are reviewed and approved by other personnel with the necessary technical competence. AEPSC Specifications are reviewed by the AEPSC QA Department.

Special process requirements for welding, heat treating and nondestructive examination (NDE) are set forth in AEPSC Specifications and the AEPSC Welding and NDE Manuals. These specifications and manuals are prepared by the AEPSC Staff Metallurgist (Corporate Level III NDE Administrator) and are reviewed and approved by other personnel with the necessary technical competence. The AEPSC NDE Manual is reviewed by the AEPSC QA Department.

Special process procedures with the exception of welding and heat treating are prepared by plant personnel with technical knowledge in the discipline involved. These procedures are reviewed by other personnel with the necessary technical competence and are qualified by testing.

Welding is performed in accordance with the procedure contained in the AEPSC Welding Manual. These procedures are qualified by the plant in accordance with applicable codes and standards, and Procedures Qualification Records are prepared. The weld procedure qualification documentation is reviewed and approved by the Plant Maintenance Superintendent or a designated plant engineer, or an AEP System welding representative. This documentation is also reviewed by either the AEPSC Staff Metallurgist, Mechanical Engineering Division or Plant Engineering Division. Weld qualification documentation is retained in the AEPSC Welding Manual.

Contractor welding procedures are qualified by the contractor. These procedures and the qualification documentation is reviewed and approved by the plant and the AEPSC Staff Metallurgists, Mechanical Engineering Division. This documentation is retained by the contractor.

1.7.9.2.3

Special process personnel qualification and certification, except for welders, is by either a designated Corporate Level III NDE Administrator or by a Plant Level III Inspector who has been qualified and certified by the designated Corporate Level III NDE Administrator. Certification is based on examination results. Personnel qualification is kept current by performance of the special process(es) and/or reexamination at time intervals specified by applicable codes, specifications and standards. Unsatisfactory performance or, where applicable, failure to perform within the designated time intervals, requires recertification.

Plant welders are qualified by the maintenance and QC Departments utilizing the procedures in the AEPSC Welding Manual. Plant welder qualification records are maintained for each welder by the Maintenance Department. Contractor and craft welders are qualified by the contractor utilizing procedures approved by the plant and the AEPSC Staff Metallurgist, Mechanical Engineering Division. Contractor and craft welder qualification records are maintained by the contractor.

1.7.9.2.4

Quality Control Technicians assigned to the Quality Control Department perform nondestructive testing for work performed by plant and contractor personnel. These individuals are qualified by SNT-TC-1A and records of the qualifications are maintained at the plant.

1.7.9.2.5

For special processes that require qualified equipment, such equipment is qualified in accordance with applicable codes, standards and specifications.

1.7.9.2.6

Qualification records are maintained in accordance with Section 17, "Quality Assurance Records".

1.7.9.2.7

The documentation resulting from welding and nondestructive testing is reviewed by appropriate management personnel.

1.7.10 INSPECTION

1.7.10.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are inspected to verify their conformance with requirements. These inspections are performed by personnel other than those who perform the activity. Inspections are performed by qualified personnel utilizing written procedures which establish prerequisites and provide documentation for evaluating test and inspection results. Direct inspection, process monitoring, or both, are used as necessary. When applicable, hold points are used to ensure that inspections are accomplished at the correct points in the sequence of activities.

1.7.10.2 IMPLEMENTATION

1.7.10.2.1

Inspections are applied to appropriate activities to assure conformance to specified requirements.

Hold points are provided in the sequence of procedures to allow for the inspection, witnessing, examination, measurement, or review necessary to assure that the critical or irreversible elements of an activity are being performed as required. Note that hold points may not apply to all procedures but each must be reviewed for this attribute.

Hold points specify exactly what is to be done (e.g., type of inspection or examination, etc.), acceptance criteria, or reference to another procedure, and the individual(s) by job title who must perform or attest to the satisfactory completion of the hold point.

When included in the sequence of a procedure, the activities required by hold points are completed prior to continuing work beyond that point.

Process monitoring is used in whole or in part where direct inspection alone is impractical or inadequate.

1.7.10.2.2

Training and Qualification Programs for personnel who perform inspections are established, implemented and documented in accordance with Section 1.7.2, "Quality Assurance Program".

1.7.10.2.3

Inspection requirements are specified in procedures, instructions, drawings, or checklists as applicable. They provide for the following as appropriate:

a) Identification of applicable revisions of required instructions, drawings and specifications.

b) Identification of characteristics and activities to be inspected.

c) Inspection methods.

d) Specification of measuring and test equipment having the necessary accuracy.

e) Identification of personnel responsible for performing the inspection.

f) Acceptance and rejection criteria.

g) Recording of the inspection results and the identification of the inspector.

1.7.10.2.4

The Plant Quality Control Department has been assigned the responsibility for establishing and executing the following programs:

a) In-process verifications and inspections.
b) Inservice inspections.

To ensure the quality of the maintenance, operation, technical, administrative, planning and construction activities at the D.C. Cook Nuclear Plant, the Plant Quality Control Department will inspect, monitor and verify key attributes that have been deemed necessary to assure the acceptability of:

a) Equipment
b) Tests
c) Processes
d) Materials
e) Parts
f) Components
g) System checks

The performance of these inspections, verifications and monitoring will be defined by instructions/procedures written by the responsible plant departments.

1.7.10.2.5

Inspections are performed, documented, and the results evaluated by designated personnel in order to ensure that the results substantiate the acceptability of the item or work. Evaluation and review results are documented.

1.7.10.2.6

Inspection of work associated with normal operation of the plant, such as surveillance tests and verification of routine maintenance, may be performed by individuals in the same group as that which performed the work, but not by personnel who directly performed or supervised the work. The qualification of these personnel is described in Appendix B hereto, item no. 9.

1.7.11 TEST CONTROL

1.7.11.1 SCOPE

Testing is performed in accordance with established programs to demonstrate that structures, systems and components will perform satisfactorily in service. The testing is performed by qualified personnel in accordance with written procedures that incorporate specified requirements and acceptance criteria. Types of tests are:

<u>Scheduled</u>

Surveillance, preventive maintenance, post-design, qualification.

<u>Unscheduled</u>

Pre- and post-maintenance.

Test parameters, including any prerequisites, instrumentation requirements and environmental conditions, are specified in test procedures. Test results are documented and evaluated.

1.7.11.2 IMPLEMENTATION

1.7.11.2.1

Tests are performed in accordance with programs, procedures and criteria that designate when tests are required and how they are to be performed. Such testing includes the following:

a) Qualification tests, as applicable, to verify design adequacy.

b) Acceptance tests of equipment and components to assure their operation prior to delivery or installation.

c) Post-design tests to assure proper and safe operation of systems and equipment prior to unrestricted operation.

d) Surveillance tests to assure continuing proper and safe operation of systems and equipment. The PMI on surveillance testing controls the periodic testing of equipment and systems to fulfill the surveillance requirements established by the Technical Specifications. The scheduling of these activities is reviewed by an Assistant Plant Manager. Controls have been established to identify uncompleted surveillance testing to assure it is rescheduled for completion to meet Technical Specification frequency requirements. Data taken during surveillance testing is reviewed by appropriate management personnel to assure that acceptance criteria is fulfilled, or corrective action is taken to correct deficiencies.

e) Maintenance tests after preventive or corrective maintenance.

1.7.11.2.2

Test procedures, as required, provide mandatory hold points for witness, or review.

1.7.11.2.3

Testing is accomplished after installation, maintenance, or repair, by surveillance test procedures or performance tests which must be satisfactorily completed prior to determining the equipment is in an operable status. All data resulting from these tests is retained at the plant after review by appropriate management personnel.

1.7.12 CONTROL OF MEASURING AND TEST EQUIPMENT

1.7.12.1 SCOPE

Measuring and testing equipment used in activities affecting the quality of safety-related systems, components and structures are properly identified, controlled, calibrated and adjusted at specified intervals to maintain accuracy within necessary limits.

1.7.12.2 IMPLEMENTATION

1.7.12.2.1

Each involved plant department has established procedures for calibration and control of measuring and test equipment utilized in the measurement, inspection and monitoring of structures, systems and components. These procedures describe calibration techniques and frequencies, and maintenance and control of the equipment.

The AEPSC Site Quality Assurance Section periodically assesses the effectiveness of the calibration program via the QA audit program.

1.7.12.2.2

Measuring and test equipment is uniquely identified and is traceable to its calibration source.

1.7.12.2.3

A system has been established utilizing labels which are to be attached to measuring and test equipment to display the date calibrated and the next calibration due date. Where labels cannot be attached, a control system is used that identifies to potential users any equipment beyond the calibration due date.

1.7.12.2.4

Measuring and test equipment is calibrated at specified intervals. These intervals are based on the frequency of use, stability characteristics and other conditions that could adversely affect the required measurement accuracy. Calibration standards are traceable to nationally recognized standards where they exist. Where national standards do not exist, provisions are established to document the basis for calibration.

The primary standards used to calibrate secondary standards have, except in certain instances, an accuracy of at least four (4) times the required accuracy of the secondary standard. In those cases where the four (4) times accuracy cannot be achieved, the basis for acceptance is documented and is authorized by the responsible manager. The secondary standards have an accuracy that assures that the equipment being calibrated will be within the required tolerances and the basis for acceptance is documented and authorized by the responsible manager.

1.7.12.2.5

A series of PMIs define the requirements for the control of standards, test equipment and process equipment.

1.7.12.2.6

When measuring and testing equipment used for inspection and testing is found to be outside of required accuracy limits at the time of calibration, evaluations are conducted to determine the validity of the results obtained since the most recent calibration. Retests or reinspections are performed on suspect items. The results of evaluations are documented.

1.7.13 HANDLING, STORAGE, AND SHIPPING

1.7.13.1 SCOPE

Activities with the potential for causing contamination or deterioration, by environmental conditions such as temperature or humidity that could adversely affect the ability of an item to perform its safety-related functions and activities necessary to prevent damage or loss are identified and controlled. These activities are cleaning, packaging, preserving, handling, shipping and storing. Controls are effected through the use of appropriate procedures and instructions.

1.7.13.2 IMPLEMENTATION

1.7.13.2.1

Procedures are used to control the cleaning, handling, storing, packaging, preserving and shipping of materials, components and systems in accordance with designated procurement requirements. These procedures include, but are not limited to, the following functions:

a) Cleaning - to assure that required cleanliness levels are achieved and maintained.

b) Packaging and preservation - to provide adequate protection against damage or deterioration. When necessary, these procedures provide for special environments such as inert gas atmosphere, specific moisture content levels and temperature levels.

c) Handling - to preclude damage or safety hazards.

d) Storing - to minimize the possibility of loss, damage, or deterioration of items in storage, including consumables such as chemicals, reagents and lubricants. Storage procedures also provide methods to assure that specified shelf lives are not exceeded.

1.7.13.2.2

Controls have been established for limited shelf life items such as "O" rings, epoxy, lubricants, solvents and chemicals to assure they are correctly identified, stored and controlled to prevent shelf life expired materials from being used in the plant. Controls are established in PMIs.

1.7.13.2.3

Packaging and shipping requirements are provided to vendors with the DCC Specifications which are a part of the purchase order. Controls for receipt inspection, damaged items and special handling requirements at the plant are established by a PMI. Special controls are provided to assure that stainless steel components and materials are handled with approved lifting slings.

1.7.13.2.4

Storage and surveillance requirements have been established to assure segregation of storage. Special controls have been implemented for critical, high value, or perishable items. Routine surveillance is conducted on stored material to provide inspection for damage, rotation of stored pumps and motors, inspection for protection of exposed surfaces and cleanliness of the storage area.

1.7.13.2.5

Special handling procedures have been implemented for the processing of nuclear fuel during refueling outages. These procedures minimize the risk of damage to the new and spent fuel and the possible release of radioactive material when placing the spent fuel into the spent fuel pool.

1.7.14 INSPECTION, TEST, AND OPERATING STATUS

1.7.14.1 SCOPE

Operating status of structures, systems and components is indicated by tagging of valves and switches, or by other specified means, in such a manner as to prevent inadvertent operation. The status of inspections and tests performed on individual items is clearly indicated by markings and/or logging under strict procedural controls to prevent inadvertent bypassing of such inspections and tests.

1.7.14.2 IMPLEMENTATION

1.7.14.2.1

For RFC (Design Change) activities, including item fabrication, installation and test, a PMI exists which specifies the degree of control required for the identification of inspection and test status of structures, systems and components.

Physical identification is used to the extent practical to indicate the status of items requiring inspections, tests, or examinations. Procedures exist which provide for the use of calibration and rejection stickers, tags, stamps and other forms of identification to indicate test and inspection status. The Clearance Permit System uses various tags to identify equipment and system operability status. Another PMI establishes a tagging system for bypassed safety functions. For those items requiring calibration, a PMI exists which requires physical indication of calibration status by calibration stickers.

1.7.14.2.2

Application and removal of inspection and welding stamps, and of such status indicators as tags, marking, labels, etc., are controlled by plant procedures.

The inspection status of materials received at the plant is identified in accordance with instructions established in a PMI. The status is identified as Hold, Hold for Quality Control Clearance, Reject, or Accept.

The inspection status of work in progress is controlled by the use of hold points in procedures. Plant Quality Control or departmental supervisory personnel inspect an activity at various stages and sign off the procedural steps covered by the inspection.

The status of welding is controlled through the use of a weld data block which identifies the inspection and nondestructive test status of each weld.

1.7.14.2.3

Required surveillance test procedures are defined in a PMI. This instruction provides for documenting bypassed tests, and for rescheduling of the test. An Assistant Plant Manager reviews the completed and signed off Weekly Surveillance Test Schedule to assure compliance.

The status of testing after minor maintenance is recorded as part of the job order. The status of testing after major maintenance is included as part of the procedure, and includes the performance of functional testing and approval of data by supervisory personnel.

Testing, inspection and other operations important to safety are conducted in accordance with properly reviewed and approved procedures. The PMI for plant procedures requires that procedures be followed as written. Alteration to the sequence of a procedure can only be

accomplished by a procedure change which is subject to the same controls as the original review and approval.

1.7.14.2.4

Nonconforming, inoperable, or malfunctioning structures, systems and components are clearly identified by tags, stickers, stamps, etc., and documented to prevent inadvertent use.

1.7.15 NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

1.7.15.2 SCOPE

Materials, parts, or components that do not conform to requirements are controlled in order to prevent their inadvertent use.. Nonconforming items are identified, documented, segregated when practical and dispositioned. Affected organizations are notified of nonconformances.

1.7.15.2 IMPLEMENTATION

1.7.15.2.1

Items, services, or activities that are deficient in characteristic, documentation, or procedure, which render the quality unacceptable or indeterminate, are identified as nonconforming, and any further use is controlled. Nonconformances are documented and dispositioned, and notification is made to affected organizations. Personnel authorized to disposition, conditionally release and close out nonconformances are designated.

The Job Order System and/or the Condition Report System (refer to Section 16.0) are used at D.C. Cook Nuclear Plant to identify nonconforming items and initiate corrective action. Systems, components, or materials which require repair or inspection are controlled under the Job Order System. In addition, the various procedures identified in Section 14 provide for identification, segregation and documentation of nonconforming items.

1.7.15.2.2

Nonconforming items are identified by marking, tagging, segregating, or by documented administrative controls. Documentation describes the nonconformance, the disposition of the nonconformance and the inspection requirements. It also includes signature approval of the disposition.

Completed Job Orders are reviewed by the supervisor responsible for accomplishing the work and the supervisor of the department/section that originated the Job Order. The QA Department periodically audits the Job Order System, and on a sample basis, Job Orders.

1.7.15.2.3

Items that have been repaired or reworked are inspected and tested in accordance with the original inspection and test requirements or alternatives that have been documented.

Items that have the disposition of "repair" or "use as is" require documentation justifying acceptability. The changes are recorded to denote the as-built condition.

When required by established procedures, surveillance or operability tests are conducted on an item after rework, repair or replacement.

1.7.15.2.4

Disposition of conditionally released items are closed out before the items are relied upon to perform safety-related functions.

1.7.16 CORRECTIVE ACTION

1.7.16.1 SCOPE

Conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances, are identified promptly and corrected as soon as practical.

For significant conditions adverse to quality, the cause of the condition is determined, and corrective action is taken to preclude repetition. In these cases, the condition, cause and corrective action taken is documented and reported to appropriate levels of management.

1.7.16.2 IMPLEMENTATION

1.7.16.2.1

Procedures are established that describe the plant and AEPSC corrective action programs. These procedures are reviewed and concurred with by the AEPSC QA Department.

AEPSC accomplishes corrective action in the following manner:

a) Audit reports which require action as a result of a corrective action request.

b) In accordance with established procedures for Condition Reports, Nonconformance Reports, Inspection Reports and Audit Reports.

c) As required by NRC Letters, I.E. Bulletins and Inspection Reports.

d) As required by 10CFR, Part 21 identified deficiencies.

1.7.16.2.2

Condition Reports provide the mechanism for plant personnel to notify management of conditions adverse to quality. Investigations of reported conditions adverse to quality are assigned by management. The investigation report is used to identify the need for changes to instructions or procedures, the initiation of a design change to correct system or equipment deficiencies, or the initiation of job orders to correct minor

deficiencies. Further, Condition Reports are used to identify those actions necessary to prevent recurrence of the reported condition. Condition Reports are also used to report violations to codes, regulations and the Technical Specifications. Condition Reports are reviewed by the PNSRC for evaluation of actions taken to correct the deficiency and prevent recurrence.

Noncompliance Reports (NCRs) provide the mechanism for AEPSC personnel to identify noncompliances. Investigation of reported conditions are assigned to the responsible individual. NCR investigation requires the determination of the cause of the condition and identification of immediate action and action taken to prevent recurrence.

The AEPSC Nuclear Operations Division receives copies of Condition Reports for distribution, on a selected basis, to cognizant engineering departments for review.

The AEPSC Nuclear Safety and Design Review Committee reviews Condition Reports, NCRs, NRC Inspection Report Responses, 10CFR21 items and QA and NSDRC audits for independent evaluation of the reported conditions and corrective actions.

The QA Department periodically audits the corrective action systems for compliance and effectiveness.

1.7.17 QUALITY ASSURANCE RECORDS

1.7.17.1 SCOPE

Records that furnish evidence of activities affecting the quality of safety-related structures, systems and components are maintained. They are accurate, complete, legible and are protected against damage, deterioration, or loss. They are identifiable and retrievable.

1.7.17.2 IMPLEMENTATION

1.7.17.2.1

Documents that furnish evidence of activities affecting quality are generated and controlled in accordance with the procedure that governs those activities. Upon completion, these documents are considered records. These records include:

a) Results of reviews, inspections, surveillances, tests, audits and material analyses.
b) Qualification of personnel, procedures and equipment.
c) Operation logs.
d) Maintenance and modification procedures and related inspection results.
e) Reportable occurrences.
f) Records required by the plant Technical Specifications.
g) Nonconformance reports.
h) Corrective action reports.
i) Other documentation such as drawings, specifications, procurement documents, calibration procedures and reports.

1.7.17.2.2

Instructions and procedures establish the requirements for the identification and preparation of records for systems and equipment under the Quality Assurance Program, and provides the controls for retention of these records.

Criteria for the storage location of quality related records and a retention schedule for these records has been established.

File Indexes have been established to provide direction for filing and to provide for the retrievability of the records.

Controls have been established for limiting access to the Plant Master File to prevent unauthorized entry, unauthorized removal and for use of

the records under emergency conditions. The Accounting Supervisor is responsible for the control and operation of the plant master file room.

1.7.17.2.3

Within AEPSC, each department/division manager is responsible for establishing procedures for the identification, collection, maintenance and storage of records generated by his department/division. These procedures shall ensure the maintenance of records sufficient to furnish objective evidence that activities affecting quality are in compliance with the established QA Program.

1.7.17.2.4

When a document becomes a record, it is designated as permanent or nonpermanent and then transmitted to file. Nonpermanent records have specified retention times. Permanent records are maintained for the life of the plant.

1.7.17.2.5

Only authorized personnel may issue corrections or supplements to records.

1.7.17.2.6

Traceability between the record and the item or activity to which it applies is provided.

1.7.17.2.7

Except for records that can only be stored as originals, such as radiographs and some strip charts, records are stored in remote, dual facilities to prevent damage, deterioration, or loss due to natural or unnatural causes. When only the single original can be retained, special fire-rated facilities are used.

1.7.18 AUDITS

1.7.18.1 SCOPE

A comprehensive system of audits is carried out to provide independent evaluation of compliance with, and the effectiveness of the Quality Assurance Program, including those elements of the program implemented by suppliers and contractors. Audits are performed in accordance with written procedures or checklists by qualified personnel not having direct responsibility in the areas audited. Audit results are documented and are reviewed by management. Follow-up action is taken where indicated.

1.7.18.2 IMPLEMENTATION

1.7.18.2.1 AEPSC QA Department Responsibilities

The basic responsibility for the assessment of the Quality Assurance Programs is vested in the AEPSC QA Department. They are primarily responsible for ensuring that proper QA programs are established and implemented. These responsibilities are discharged in cooperation with the AEPSC and plant management, and their staffs.

Stop Work Authority - Refer to Section 1.7.1.2.5 herein.

1.7.18.2.2

Internal audits are performed in accordance with established schedules that reflect the status and importance of safety to the activities being performed. All areas where the requirements of 10CFR50, Appendix B apply are audited within a period of two years.

1.7.18.2.3

The AEPSC Quality Assurance Department conducts audits to verify the adequacy and implementation of the QA Program at the plant and within AEPSC. QA audit reports are distributed to the Plant Manager and PNSRC (site audits) and the NSDRC (all audits).

1.7.18.2.4

The independent off-site review and audit organization is the AEPSC Nuclear Safety and Design Review Committee (NSDRC). This committee is composed of AEPSC, I&M and plant management members. A Charter and Procedures Manual has been developed for this committee. The NSDRC conducts periodic audits of plant operations pursuant to established criteria (Technical Specifications, etc.).

NSDRC Audit Reports are submitted for review to the Chairman of the NSDRC and to the Vice Chairman Engineering and Construction. Corrective Action Requests provide for the recording of actions taken to correct deficiencies found during these audits.

1.7.18.2.5

The plant on-site review group is the Plant Nuclear Safety Review Committee (PNSRC). This committee reviews plant operations as a routine evaluation and serves to advise the Plant Manager on matters related to nuclear safety. The composition of the committee is defined in the Technical Specifications.

The PNSRC also reviews instructions and procedures for safety-related systems prior to approval by the Plant Manager. In addition, this committee serves to conduct investigations of violations to Technical Specifications, reviews Condition Reports to determine if appropriate action has been taken and reviews all design changes.

1.7.18.2.6

Audits of suppliers and contractors are scheduled based on the status of safety importance of the activities being performed, and are initiated early enough to assure effective quality assurance during design, procurement, manufacturing, construction, installation, inspection and testing.

Principal contractors are required to audit their suppliers systematically in accordance with the foregoing scheduling criteria.

1.7.18.2.7

Regularly scheduled audits are supplemented by special audits when significant changes are made in the Quality Assurance Program, when it is suspected that quality is in jeopardy, or when an independent assessment of program effectiveness is considered necessary.

1.7.18.2.8

Audits include an objective evaluation of quality related practices, procedures, instructions, activities and items; and review of documents and records to confirm that the QA program is effective and properly implemented.

1.7.18.2.9

Audit procedures and the scope, plans, checklists and results of individual audits are documented.

1.7.18.2.10

Personnel selected for auditing assignments have experience or are given training commensurate with the needs of the audit and have no direct responsibilities in the areas audited.

1.7.18.2.11

Management of the audited organization identifies and takes appropriate action to correct observed deficiencies and to prevent recurrence. Follow-up is performed by the auditing organization to ensure that the appropriate actions were taken. Such follow-up includes reaudits when necessary.

1.7.18.2.12

The adequacy of the Quality Assurance Program is regularly assessed by AEPSC management. The following activities constitute formal elements of that assessment:

a) Audit reports, including follow-up on corrective action accomplishment and effectiveness, are distributed to appropriate levels of management.

b) Individuals independent from the Quality Assurance Organization, but knowledgeable in auditing and quality assurance, periodically review the effectiveness of the Quality Assurance Programs. Conclusions and recommendations are reported to the AEPSC Vice President - Nuclear Operations.

1.7.19 FIRE PROTECTION QA PROGRAM

1.7.19.1 Introduction

The D.C. Cook Nuclear Plant Fire Protection QA Program has been developed using the guidance of the NRC Branch Technical Position 9.5-1, Appendix "A".

This QA Program is applicable to:

1) Fire protection areas and equipment designed and/or procured after January 31, 1977 that protects safety-related items which appear in the Fire Protection Technical Specifications; and,

2) The balance of plant fire protection areas and equipment designed and/or procured after January 31, 1977.

Implementation of the Fire Protection QA Program is the responsibility of each involved AEP organization.

The QA Program for the Fire Protection Program at D.C. Cook Plant applies to the following activities: design, procurement, fabrication, construction, operation, maintenance and modification.

1.7.19.2 <u>Organization</u>

The QA program for fire protection is under the management control of AEPSC. This control consists of:

1) Formulating and verifying that the Fire Protection QA Program incorporates suitable requirements and is acceptable to the management responsible for fire protection; and,

2) Verifying the effectiveness of the QA program for fire protection through review, surveillance and audits. The QA program for fire protection is part of the overall plant QA program. These QA criteria apply to those items within the scope of the Fire Protection Program, such as fire protection systems, emergency lighting, communication and emergency breathing apparatus, as well as the fire protection requirements of applicable safety-related equipment.

AEPSC and plant management has direct functional responsibility for the formulation, implementation and assessment of the D.C. Cook Fire Protection Program.

The AEPSC Fire Protection Supervisor is responsible to the Manager - Plant Engineering Division, for aspects of the Fire Protection Program at the D.C. Cook Plant. These responsibilities provide for planning annual inspection schedules for fire and explosion hazards and training, including annual fire fighting instruction to plant personnel, fire brigades and responding fire departments.

The Fire Protection/HVAC Section Manager and the Fire Protection Engineer have coordinated the building layout, the fire suppression and fire detection systems, commensurate with fire areas within the plant. They

have established the design of the overall fire detection/ suppression system and the incremental parts of the system. Maintenance information has been provided to the plant in the form of system descriptions and equipment supplier instruction material.

The Plant Manager has delegated responsibility to various plant departments for the following fire protection activities:

a) Maintenance of fire protection system,
b) Testing of fire protection equipment,
c) Fire safety inspections,
d) Fire fighting procedures, and
e) Fire drills.

The Shift Supervisor on duty is designated as the Fire Chief and coordinates the fire fighting efforts of shift personnel and the fire brigade.

1.7.19.3 Design Control and Procurement Document Control

Quality standards are specified in the design documents such as appropriate fire protection codes and standards, and deviations and changes from these quality standards are controlled.

The plant design was reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements. These reviews include items such as:

1) Reviews to verify adequacy of wiring isolation and cable separation criteria.

2) Reviews to verify appropriate requirements for room isolation (sealing penetrations, floors and other fire barriers).

3) Reviews to determine increase in fire loadings.

4) Reviews to determine the need for additional fire detection and suppression equipment.

A review and concurrence of the adequacy of fire protection requirements and quality requirements stated in procurement documents is performed.

This review determines that fire protection requirements and quality requirements are correctly stated, verifiable and. controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed and approved in accordance with QA program requirements.

Design and procurement document changes, including field changes and design deviations are subject to the same level of controls, reviews and approvals that were applicable to the original document.

1.7.19.4 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that govern the Fire Protection Program are prescribed by documented instructions, procedures, or drawings, and are accomplished in accordance with these documents.

Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with documented procedures. Activities of the fire protection system are prescribed and accomplished in accordance with documented instructions, procedures and drawings.

Instructions and procedures for design installation, inspection, test, maintenance, modification and administrative controls are reviewed to assure that proper fire protection requirements are included.

1.7.19.5 Control of Purchased Material, Equipment and Services

Measures are established to assure that purchased material, equipment and services conform to the procurement documents. These measures include

provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor, inspections at suppliers, or receiving inspections.

Source or receiving inspection is provided, as a minimum, for those items whose quality cannot be verified after installation.

1.7.19.6 Inspection

A program for independent inspection of the fire protection activities has been established and implemented.

These inspections are performed by personnel other than those responsible for implementation of the activity.

The inspections include:

a) Inspection of: 1) installation, maintenance and modification of fire protection systems; and, 2) emergency lighting and communication equipment.

b) Inspections of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed.

c) Inspections of cable routing to verify conformance with design requirements.

d) Inspections to verify that appropriate requirements for room isolation are accomplished following construction or modification activities.

e) Measures to assure that inspection personnel are independent from the individuals performing the activity being inspected, and are knowledgeable in the design and installation requirements for fire protection.

f) Inspection procedures, instructions and/or check lists are provided for inspections.

g) Periodic inspections of fire protection systems, emergency breathing and auxiliary equipment, emergency lighting and communication equipment.

h) Periodic inspections of materials subject to degradation such as fire stops, seals and fire retardant coating.

1.7.19.7 Test and Test Control

a) Installation testing - Following installation, modification, repair, or replacement, sufficient testing is performed to demonstrate that the fire protection systems, emergency lighting and communication equipment will perform satisfactorily. Written test procedures for installation tests incorporate the requirements and acceptance limits contained in applicable design documents.

b) Periodic testing - Periodic testing schedules and methods have been implemented and the results documented. Fire protection equipment, emergency lighting and communication equipment are tested periodically to assure that the equipment functions properly.

c) Programs have been established to verify the testing of fire protection systems and to verify that test personnel are effectively trained.

d) Test results are documented, evaluated, and their acceptability determined by a qualified responsible individual or group.

1.7.19.8 Inspection, Test and Operating Status

The inspection, test and operating status for the Fire Protection System are performed as described in Section 1.7.14.

1.7.19.9 Nonconforming Items

Nonconforming items for the fire protection components are identified and dispositioned as described in Section 1.7.15.

1.7.19.10 Corrective Action

The corrective action mechanism described in Section 1.7.16 applies to the fire protection system.

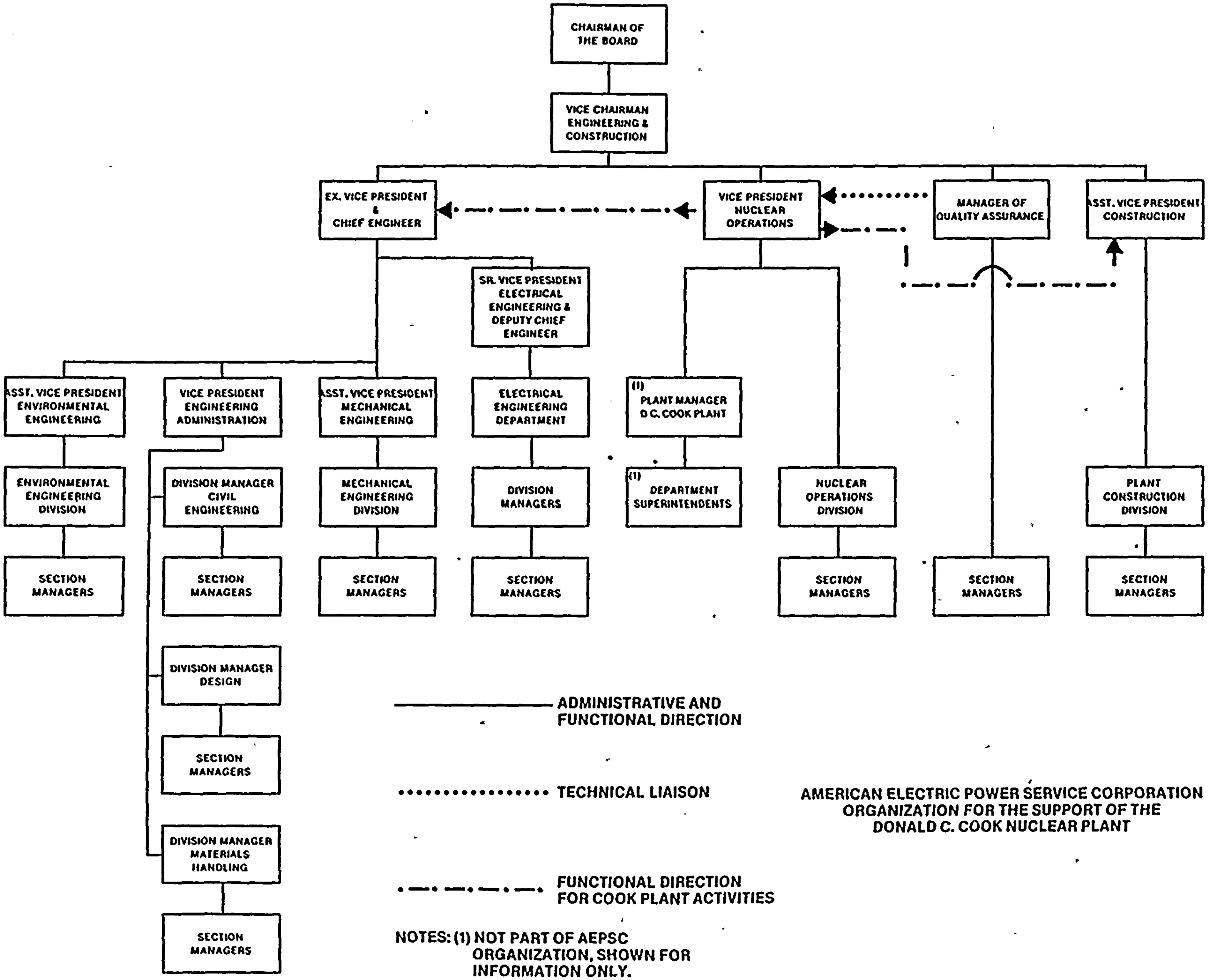
1.7.19.11 Records

Records generated to support the fire protection system and its components are controlled as described in Section 1.7.17.

1.7.19.12 Audits

Audits are conducted and documented to verify compliance with the Fire Protection Program as described in Section 1.7.18.

Audits are periodically performed to verify compliance with the administrative controls and implementation of quality assurance criteria. The audits are performed in accordance with preestablished written procedures or check lists. Audit results are documented and reviewed by management having responsibility in the area audited. Follow-up action is taken by responsible management to correct the deficiencies revealed by the audit.



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

Figure 1.7-1

QA ORGANIZATION

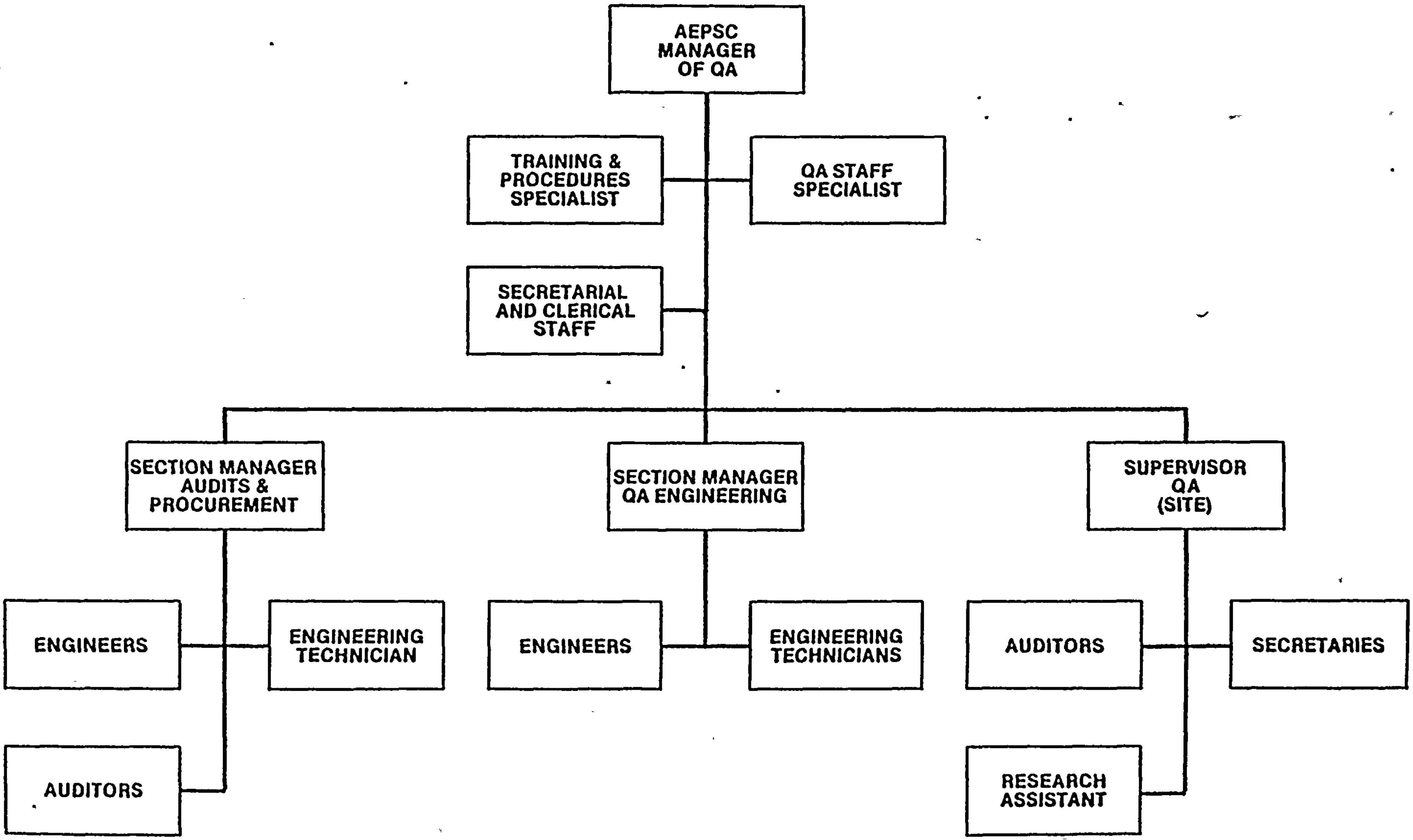
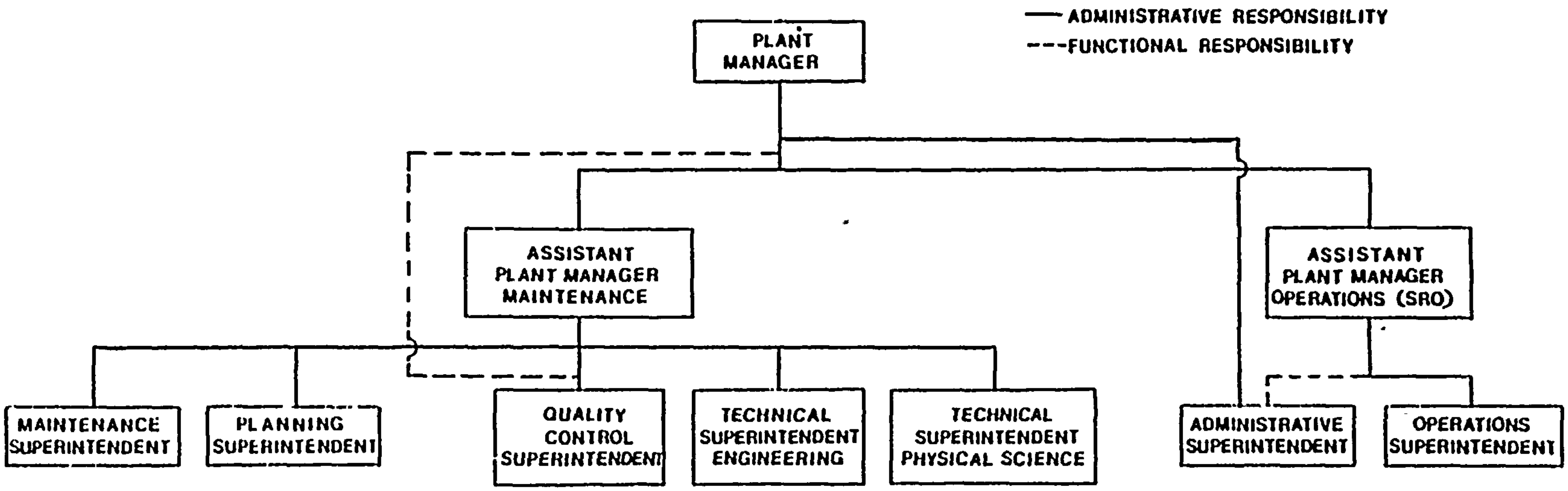


Figure 1.7-4



INDIANA & MICHIGAN ELECTRIC COMPANY
ORGANIZATION
FOR THE
DONALD C. COOK NUCLEAR PLANT

Figure 1.7-5

AEPSC/I&MECO EXCEPTIONS TO OPERATING PHASE STANDARDS AND REGULATORY GUIDES

1. GENERAL

Requirement

Certain Regulatory Guides invoke or imply Regulatory Guides and standards in addition to the standard each primarily endorses.

Certain ANSI Standards invoke or imply additional standards.

Exception/Interpretation

The AEPSC/I&MECo commitment refers to the Regulatory Guides and ANSI Standards specifically identified in Appendix A. Additional Regulatory Guides, ANSI Standards and similar documents implied or referenced in those specifically identified are not part of this commitment.

2. N18.7, General

Exception/Interpretation

AEPSC and I&MECo have established both an on-site and off-site standing committee for independent review activities. Together they form the independent review body.

The standard numeric and qualification requirement may not be met by each group individually. Procedures will be established to specify how each group will be involved in review activities. This exception/interpretation is consistent with the plant's Technical Specifications.

2a. Sec. 4.3.1

Requirement

"Personnel assigned responsibility for independent reviews shall be specified in both number and technical disciplines, and shall collectively have the experience and competence required to review problems in the following areas: . . . ."

Exception/Interpretation

AEPSC Nuclear Safety and Design Review Committee (NSDRC) and Plant Nuclear Safety Review Committee (PNSRC) will not have members specified by number nor by technical disciplines, and its members may not have the experience and competence required to review problems in all areas listed in this section. This exception/interpretation is consistent with the plant's Technical Specifications.

The NSDRC and PNSRC will not specifically include a member qualified in nondestructive testing but will use qualified technical consultants to perform this and other functions as determined necessary by the respective committee chairman.

2b. Sec. 4.3.2.1

Requirement

"When a standing committee is responsible for the independent review program, it shall be composed of no less than five persons of whom no more than a minority are members of the on-site operating organization. Competent alternatives are permitted if designated in advance. The use of alternates shall be restricted to legitimate absences of principals."

Exception/Interpretation

See Item 2a.

2c. Sec. 4.3.3.1

Requirement

". . . recommendations . . . shall be disseminated promptly to appropriate members of management having responsibility in the area reviewed."

Exception/Interpretation

Recommendations made as a result of review will generally be conveyed to the on-site or off-site standing committee. Procedures will be maintained specifying how recommendations are to be considered.

2d. Sec. 4.3.4

Requirement

"The following subjects shall be reviewed by the independent review body: . . . ."

Exception/Interpretation

Subjects requiring review will be as specified in the plant Technical Specifications.

2e. Sec. 4.3.4(3)

Requirement

"Changes in the Technical Specifications or License Amendments relating to nuclear safety are to be reviewed by the independent review body prior to implementation, except in those cases where the change is identical to a previously reviewed proposed change."

Exception/Interpretation

The NSDRC and PNSRC will not review Technical Specification changes after NRC approval prior to implementation. The basis for this position is the NSDRC and PNSRC review Technical Specification changes prior to submittal to the NRC.

2f. Sec. 4.4

Requirement

"The on-site operating organization shall provide, as part of the normal duties of plant supervisory personnel . . . ."

Exception/Interpretation

Some of the responsibilities of the on-site operating organization described in Section 4.4 may be carried out by the PNSRC and/or NSDRC as described in plant Technical Specifications.

2g. Sec. 5.2.2

Requirement

"Temporary changes, which clearly do not change the intent of the approved procedure, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedures. At least one of these individuals shall be the supervisor in charge of the shift and hold a senior operator's license on the unit affected."

Exception/Interpretation

I&MECo considers that this requirement applies only to procedures identified in plant Technical Specifications. Temporary changes to these procedures shall be approved as described in plant Technical Specifications.

2h. Sec. 5.2.6

Requirement

"In cases where required documentary evidence is not available, the associated equipment or materials must be considered nonconforming in accordance with Section 5.2.14. Until suitable documentary evidence is available to show the equipment or material is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions."

Exception/Interpretation

I&MECo initiates appropriate corrective action when it is discovered that documentary evidence does not exist for a test or inspection which is a requirement to verify equipment acceptability. This action includes a technical evaluation of the equipment's operability status.

2i. Sec. 5.2.8

Requirement

"A surveillance testing and inspection program . . . shall include the establishment of a master surveillance schedule reflecting the status of all planned in-plant surveillances tests and inspections."

Exception/Interpretation

Separate master schedules may exist for different programs such as ISI, pump and valve testing and Technical Specification surveillance testing.

2j. Sec. 5.2.13.1

Requirement

"To the extent necessary, procurement documents shall require suppliers to provide a Quality Assurance Program consistent with the pertinent requirements of ANSI N45.2 - 1971."

Exception/Interpretation

To the extent necessary, procurement documents require that the supplier has a documented Quality Assurance Program consistent with the pertinent requirements of 10CFR50, Appendix B; ANSI N45.2; or other nationally recognized codes and standards.

2k. Sec. 5.2.13.2

Requirement

ANSI N18.7 and N45.2.13 specify that where required by code, regulation, or contract, documentary evidence that items conform to procurement requirements shall be available at the nuclear power plant site prior to installation or use of such items.

Exception/Interpretation

The required documentary evidence is available at the site prior to use, but not necessarily prior to installation. This allows installation to proceed while any missing documents are being obtained, but precludes dependence on the item for safety purposes.

21. Sec. 5.2.16

Requirement

Records shall be made and equipment suitably marked to indicate calibration status.

Exception/Interpretation

See Item 6b.

2m. Sec. 5.3.5(4)

Requirement

This section requires that where sections of documents such as vendor manuals, operating and maintenance instructions or drawings are incorporated directly or by reference into a maintenance procedure, they shall receive the same level of review and approval as operating procedures.

Exception/Interpretation

Such documents are reviewed by appropriately qualified personnel prior to use to ensure that, when used as instructions, they provide proper and adequate information to ensure the required quality of work. Maintenance procedures which reference these documents receive the same level of review and approval as operating procedures.

3. N45.2.1,

3a. Sec. 2

Requirement

N45.2.1 establishes criteria for classifying items into "cleanness levels", and requires that items be so classified.

Exception/Interpretation

Instead of using the cleanness level classification system of N45.2.1, the required cleanness for specific items and activities is addressed on a case-by-case basis.

Cleanness is maintained, consistent with the work being performed, so as to prevent the introduction of foreign material. As a minimum, cleanness inspections are performed prior to closure of "nuclear" systems and equipment. Such inspections are documented.

3b. Sec. 5

Requirement

"Fitting and tack-welded joints (which will not be immediately sealed by welding) shall be wrapped with polyethylene or other nonhalogenated plastic film until the welds can be completed."

Exception/Interpretation

I&MECo sometimes uses other nonhalogenated material, compatible with the parent material, since plastic film is subject to damage and does not always provide adequate protection.

4. N45.2.2, General

Requirement

N45.2.2 establishes requirements and criteria for classifying safety related items into protection levels.

Exception/Interpretation

Instead of classifying safety related items into protection levels, controls over the packaging, shipping, handling and storage of such items

are established on a case-by-case basis with due regard for the item's complexity, use and sensitivity to damage. Prior to installation or use, the items are inspected and serviced as necessary to assure that no damage or deterioration exists which could affect their function.

4a. Sec. 3.9 and Appendix A3.9

Requirement

"The item and the outside of containers shall be marked."

(Further criteria for marking and tagging are given in the Appendix.)

Exception/Interpretation

These requirements were originally written for items packaged and shipped to construction projects. Full compliance is not always necessary in the case of items shipped to operating plants and may, in some cases, increase the probability of damage to the item. The requirements are implemented to the extent necessary to assure traceability and integrity of the item.

4b. Sec. 5.2.2

Requirement

"Receiving inspections shall be performed in an area equivalent to the level of storage."

Exception/Interpretation

Receiving inspection area environmental controls may be less stringent than storage environmental requirements for an item. However, such inspections are performed in a manner and in an environment which do not endanger the required quality of the item.

4c. Sec. 6.2.4

Requirement

"The use or storage of food, drinks and salt tablet dispensers in any storage area shall not be permitted."

Exception/Interpretation

Packaged food for emergency or extended overtime use may be stored in material stock rooms. The packaging assures that materials are not contaminated. Food will not be "used" in these areas.

4d. Sec. 6.3.4

Requirement

"All items and their containers shall be plainly marked so that they are easily identified without excessive handling or unnecessary opening of crates and boxes."

Exception/Interpretation

See N45.2.2, Section 3.9 (Exception 4b.).

4e. Sec. 6.4.1

Requirement

"Inspections and examinations shall be performed and documented on a periodic basis to assure that the integrity of the item and its container . . . is being maintained."

Exception/Interpretation

The requirement implies that all inspections and examinations of items in storage are to be performed on the same schedule. Instead, the inspections and examinations are performed in accordance with material storage procedures which identify the characteristics to be inspected an include the required frequencies. These procedures are based on technical considerations which recognize that inspections and frequencies needed vary from item to item.

5. N45.2.3,

5a. Sec. 2.1

Requirement

Cleanness requirements for housekeeping activities shall be established on the basis of five zone designations.

Exception/Interpretation

Instead of the five-level zone designation system referenced in ANSI N45.2.3, I&MECo bases its controls over housekeeping activities on a consideration of what is necessary and appropriate for the activity involved. The controls are effected through procedures or instructions. Factors considered in developing the procedures and instructions include cleanliness control, personnel safety, fire prevention and protection, radiation control and security. The procedures and instructions make use of standard janitorial and work practices to the extent possible. However, in preparing these procedures, consideration is also given to the recommendations of Section 2.1 of ANSI N45.2.3.

6. N45.2.4,

6a. Sec. 2.2

Requirement

Section 2.2 establishes prerequisites which must be met before the installation, inspections and testing of instrumentation and electrical equipment may proceed. These prerequisites include personnel qualification, control of design, conforming and protected materials and availability of specified documents.

Exception/Interpretation

During the operations phase, this requirement is considered to be applicable to modifications and initial start-up of electrical equipment. For routine or periodic inspection and testing, the prerequisite conditions will be achieved as necessary.

6b. Sec. 6.2.1

Requirement

"Items requiring calibration shall be tagged or labeled on completion, indicating date of calibration and identity of person that performed calibration."

Exception/Interpretation

Frequently, physical size and/or location of installed plant instrumentation precludes attachment of calibration labels or tags. Instead, each instrument is uniquely identified and is traceable to its calibration record.

A scheduled calibration program assures that each instrument's calibration is current.

7. N45.2.5,

7a. Sec. 2.5.2

Requirement

"When discrepancies, malfunctions or inaccuracies in inspection and testing equipment are found during calibration, all items inspected with that equipment since the last previous calibration shall be considered unacceptable until an evaluation has been made by the responsible authority and appropriate action taken.

Exception/Interpretation

I&MECo uses the requirements of N18.7, Section 5.2.16, rather than N45.2.5, Section 2.5.2. The N18.7 requirements are more applicable to an operating plant.

7b. Sec. 5.4

Requirement

"Hand torque wrenches used for inspection shall be controlled and must be calibrated at least weekly and more often if deemed necessary. Impact torque wrenches used for inspection must be calibrated at least twice daily."

Exception/Interpretation

Torque wrenches are controlled as measuring and test equipment in accordance with ANSI N18.7, Section 5.2.16. Calibration intervals are based on use and calibration history rather than as per N45.2.5.

8. N45.2.6, Sec. 1.2

Requirement

"The requirements of this standard apply to personnel who perform inspections, examinations and tests during fabrication prior to or during receipt of items at the construction site, during construction, during preoperational and start-up testing and during operational phases of nuclear power plants."

Exception/Interpretation

Personnel participating in testing who take data or make observations, where special training is not required to perform this function, need not be qualified in accordance with ANSI N45.2.6 but need only be trained to the extent necessary to perform the assigned function.

9. Reg. Guide 1.58 - General

Requirement

Qualification of nuclear power plant inspection, examination and testing personnel.

9a. C.2.6

Requirement

Regulatory Guide 1.58 endorses the guidelines of SNT-TC-1A as an acceptable method of training and certifying personnel conducting leak tests.

Exception/Interpretation

I&MECo takes the position that the "Level" designation guidelines as recommended in SNT-TC-1A, paragraph 4 do not necessarily assure adequate leak test capability. I&MECo maintains that departmental supervisors are

best able to judge whether engineers and other personnel are qualified to direct and/or perform leak tests. Therefore, I&MECo does not implement the recommended "Level" designation guidelines.

It is I&MECo's opinion that the training guidelines of SNT-TC-1A (1975), Table I-G, paragraph 5.2 specifically are oriented towards the basic physics involved in leak testing, and further, towards individuals who are not graduate engineers. I&MECo maintains that it meets the essence of these training guidelines. The preparation of leak test procedures and the conduct of leak tests at Cook Plant is under the direct supervisor of Performance Engineers who hold engineering degrees from accredited engineering schools. The basic physics of leak testing have been incorporated into the applicable test procedures. The review and approval of the data obtained from leak tests is performed by department supervisors who are also graduate engineers.

I&MECo does recognize the need to assure that individuals involved in leak tests are fully cognizant of leak test procedural requirements and thoroughly familiar with the test equipment involved. Plant performance engineers receive routing, informal orientation on testing programs, to ensure that these individuals fully understand the requirements of performing a leak test.

9b. C5, C6, C7, C8, C10

Exception/Interpretation

I&MECo takes the position that the classification of inspection, examination and test personnel (inspection personnel) into "Levels" based on the requirements stated in Section 3.0 of ANSI N45.2.6 does not necessarily assure adequate inspection capability. I&MECo maintains that departmental and first line supervisors are best able to judge the inspection capability of the personnel under their supervision, and that "level" classification would require an overly burdensome administrative work load, could inhibit inspection activities and provides no assurance of inspection capabilities. Therefore, I&MECo does not implement the

"level classification" concept for inspection, examination and test personnel.

The methodology under which inspections, examinations and tests are conducted at the Donald C. Cook Nuclear Plant requires the involvement of first line supervisors, engineering personnel, departmental supervisors and plant management. In essence, the last seven (7) project functions shown in Table 1 to ANSI N45.2.6 are assigned to supervisory and engineering personnel and not to personnel of the inspector category. These management supervisory and engineering personnel, as a minimum, meet the educational and experience requirements of "Level II and Level III" personnel, as required, to meet the criteria of ANSI 18.1 which exceeds those of ANSI N45.2.6. In I&MECo's opinion, no useful purpose is served by classification of management, supervisory and engineering personnel into "Levels."

Therefore, I&MECo takes the following positions relative to regulatory positions C5, 6, 7, 8 and 10 of Regulatory Guide 1.58.

C-5 Based on the discussion in B.1 above, this position is not applicable to the Donald C. Cook Nuclear Plant.

C-6 Replacement personnel for Donald C. Cook Nuclear Plant management, supervisory and engineering positions subject to ANSI 18.1 will meet the educational and experience requirements of ANSI 18.1 and therefore those of ANSI N45.2.6.

Replacement inspection personnel will, as a minimum, meet the educational and experience requirements of ANSI N45.2.6, Section 3.5.1 - "Level I".

C-7 I&MECo, as a general practice, complies with the training recommendations as set forth in this regulatory position.

C-8 All I&MECo inspection, examination and test personnel are instructed in the normal course of employee training in radiation protection and the means to minimize radiation dose exposure.

C-10 I&MECo maintains documentation to show that inspection personnel meet the minimum requirements of "Level I" and that management, supervisory and engineering personnel meet the minimum requirements of ANSI 18.1.

10. General

Imposition of these Regulatory Guides on AEPSC/I&MECo suppliers and subtier suppliers will be on a case-by-case basis depending upon the item or service to be procured.

11. N45.2.8,

11a. Sec. 2.9e

Requirement

Section 2.9e of N45.2.8 lists documents relating to the specific stage of installation activity which are to be available at the construction site.

Exception/Interpretation

All of the documents listed are not necessarily required at the construction site for installation and testing. AEPSC and I&MECo assure that they are available to the site as necessary.

11b. Sec. 2.9e

Requirement

Evidence that engineering or design changes are documented and approved shall be available at the construction site prior to installation.

Exception/Interpretation

Equipment may be installed before final approval of engineering or design changes. However, the system is not placed into service until such changes are documented and approved.

11c. Sec. 4.5.1

Requirement

"Installed systems and components shall be cleaned, flushed and conditioned according to the requirements of ANSI N45.2.1. Special consideration shall be given to the following requirements: . . . ."
(Requirements are given for chemical conditioning, flushing and process controls.)

Exception/Interpretation

Systems and components are cleaned, flushed and conditioned as determined on a case-by-case basis. Measures are taken to help preclude the need for cleaning, flushing and conditioning through good practices during maintenance or modification activities.

12. <u>N45.2.9</u>

12a. Sec. 5.4, Item 2

<u>Requirement</u>

Records shall not be stored loosely. "They shall be firmly attached in binders or placed in folders or envelopes for storage on shelving in containers." Steel file cabinets are preferred.

<u>Exception/Interpretation</u>

Records are suitably stored in steel file cabinets or on shelving in containers. Methods other than binders, folders, or envelopes (for example, dividers) may be used to organize the records for storage.

12b. Sec. 6.2

<u>Requirement</u>

"A list shall be maintained designating those personnel who shall have access to the files".

<u>Exception/Interpretation</u>

Rules are established governing access to and control of files as provided for in ANSI N45.2.9, Section 5.3, Item 5. These rules do not always include a requirement for a list of personnel who are authorized access. It should be noted that duplicate files and/or microforms may exist for general use.

12c. Sec. 5.6

Requirement

When a single records storage facility is maintained, at least the following features should be considered in its construction: etc.

Exception/Interpretation

The Donald C. Cook Nuclear Plant Master File Room complies with the requirements of NUREG-0800 (7/81), Section 17.1.17.4.

13. Reg. Guide 1.144,

13a. Sec C3a(2)

Requirement

Applicable elements of an organization's Quality Assurance Program for "design and construction phase activities should be audited at least annually or at least once within the life of the activity, whichever is shorter."

Exception/Interpretation

Since most modifications are straight forward, they are not audited individually. Instead, selected controls over modifications are audited periodically.

13b. Sec. C3b(1)

Requirement

This section identifies procurement contracts which are exempted from being audited.

Exception/Interpretation

In addition to the exemptions of Reg. Guide 1.144, AEPSC/I&MECo considers that the National Bureau of Standards or other State and Federal Agencies which may provide services to AEPSC/I&MECo are not required to be audited.

14. N45.2.13,

14a. Sec. 3.2.2

Requirement

N45.2.13 requires that technical requirements be specified in procurement documents by reference to technical requirement documents. Technical requirement documents are to be prepared, reviewed and released under the requirements established by ANSI N45.2.11.

Exception/Interpretation

For replacement parts and materials, AEPSC/I&MECo follow ANSI N18.7, Section 5.2.13, Subitem 1, which states: "Where the original item or part is found to be commercially 'off the shelf' or without specifically identified QA requirements, spare and replacement parts may be similarly procured, but care shall be exercised to ensure at least equivalent performance."

14b. Sec. 3.3.2

Requirement

"Procurement documents shall require that the supplier have a documented Quality Assurance Program that implements parts or all of ANSI N45.2 as well as applicable Quality Assurance Program requirements of other nationally recognized codes and standards."

Exception/Interpretation

Refer to Item 2j.

14c. Sec. 3.3(a)

Requirement

Reviews of procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation

Documents may be released for bid or contract award before completing the necessary reviews. However, these reviews are completed before the item or service is put into service, or before work has progressed beyond the point where it would be impractical to reverse the action taken.

14d. Sec. 3.3(b)

Requirement

Review of changes to procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation

This requirement applies only to quality related changes (i.e., changes to the procurement document provisions identified in ANSI N18.7, Section 5.2.13.1, Subitems 1 through 5). The timing of reviews will be the same as for review of the original procurement documents.

14e. Sec. 10.1

Requirement

"Where required by code, regulation, or contract requirement, documentary evidence that items conform to procurement documents shall be available at the nuclear power plant site prior to installation or use of such items, regardless of acceptance methods."

Exception/Interpretation

Refer to Item 2j.

Requirement

"Post-installation test requirements and acceptance documentation shall be mutually established by the purchaser and supplier."

Exception/Interpretation

In exercising its ultimate responsibility for its Quality Assurance Program, AEPSC/I&MECo establishes post-installation test requirements giving due consideration to supplier recommendations.

15. RG 1.58/ANSI N45.2.23 and ANSI N45.2.2.12

15a. ANSI N45.2.23, Sec. 1.1

Requirement

This standard provides requirements and guidance for the qualification of audit team leaders, henceforth identified as "Lead Auditors".

15b. ANSI N45.2.12, Sec. 4.2.2

Requirement

A Lead Auditor shall be appointed team leader.

Exception/Interpretation

The AEPSC audit program is directed by the AEPSC Manager of QA who is a qualified lead auditor; and is administered by designated QA Department Section Managers who are also qualified lead auditors.

Audits are, in most cases, conducted by individual auditors, not by "audit teams". These auditors are qualified by established procedures and are assigned by the responsible QA Section Manager based on their demonstrated audit capability and general knowledge of the audit subject. In certain cases, this results in an individual other than a "lead auditor" conducting the actual audit function.

Established AEPSC audit procedures require that, in all cases, the audit functions of preparation/organization, reporting of audit findings and evaluation of corrective actions be reviewed by QA Department Section Managers; thereby meeting the requirements of ANSI N45.2.23 relative to "Lead Auditors", and "Audit Team Leaders".

2.2 METEOROLOGY

Due to the extreme importance of site meteorology, particularly with regard to safety considerations, an extensive meteorological study program was initiated at the site during the summer of 1966.

The meteorological features of the plant site have been evaluated primarily on the basis of three years data obtained from the 200 foot tower which was installed on the site in 1966. Satellite aerovane stations at inland and on-site locations were used to complement the main tower data.

In most respects, the meteorological patterns are those of a typical open mid-latitude exposure. The wind speeds are strong, variations in direction are frequent and the overall wind rose shows no marked favoritism for any particular direction. The only unusual feature is the low frequency of stable conditions. Both the lapse rate and turbulence class analyses indicate far fewer stable cases than originally anticipated, reaching only 7% over the three-year period. Even in the late spring and early summer when the lake is relatively cold, the frequency of stable cases reaches only 20 to 25%. Even more favorable is the very low frequency of the combination of light winds and stable, on-shore flow. Less than 1% of the 200-foot data and only 2.5% of the satellite data are in this category.

The only major meteorological hazard expected in the site area is the tornado which has recurrence frequency of over 1000 years at the site itself. Ice storms, which would be expected with greater frequency, are not likely to damage essential facilities, but have been considered in developing certain criteria.

2.2.1 SOURCES OF DATA

<u>Site Meteorological Tower</u>

The main source for the data shown herein was a 200-foot meteorological tower which was erected at the site during the summer of 1966 and equipped with meteorological instrumentation (Fig. 2.2-1). This tower remained in continuous operation from October, 1966 until 1978. The tower instruments consisted of the following:

200 ft. level - Aerovane and aspirated resistance thermometer.

150 ft. level - Climet Bivane (the extremely strong winds at the site have damaged the Bivane, but some data has been obtained).

50 ft. level - A Aerovane and aspirated resistance thermometer.

Ground-level - Resistance thermometer, Dewcell, recording rain gage and recording barometer.

In the Unit 1 Control Room there is instrumentation and a recorder for wind speed, direction and temperature.

The elevations of the aerovane detectors are listed below.

	Elevation USGS
200 feet level on main tower	892'
50 feet level on main tower	742'
Inland satellite	701'
Beach satellite (previous location)	696.3'
Beach satellite (present location until 1978)	656'
Plant grade	608'
Average lake water level	580.4'

The inland satellite and the beach satellite are no longer in operation.

New Meteorological Instruments

The current meteorological instrumentation is located east of the plant on the micro wave tower. The tower contains the following redundant instrumentation:

180 ft. level	-	Temperature Sensors
150 ft. level	-	Wind Speed and Direction Sensors
50 ft. level	-	Wind Speed and Direction Sensors
30 ft. level	-	Temperature Sensors

Additionally a Dew Point and Precipitation Sensor are provided. Readouts from this instrumentation are provided in both control rooms and recorded in Unit 1 control room.

The base USGS elevation of the microwave tower is 735'-0".

Special Studies

Phenomena having relatively long recurrence intervals, such as tornadoes and ice storms, in the area cannot be studied directly from site observations and estimates have been derived from special reports.[1, 2, 3, 4]

Analysis

The meteorological data from the Donald C. Cook Nuclear Plant site have been abstracted, processed and analyzed on a monthly basis by Smith-Singer, Meteorologists, Inc. The computer output from which the analysis is made is too extensive to include as a part of this report. The summaries given here are derived from it. Table 2.2-1 is a sample of the original hourly records in the computer data file.

2.2.2 GENERAL METEOROLOGY

Southwestern Michigan is typical of the northern lake regions of the United States in most respects. The flat terrain and the frequent passage of well-developed extra-tropical storms create a consistently strong wind flow, as well as rapid changes in both dispersion conditions and wind direction. Some of the meteorological statistics are useful primarily for general planning of the facilities and are therefore reported with a minimum of description. Other data are important in the assessment of safety and these are discussed fully.

Temperatures, Precipitation, Humidity and Barometric Pressure

These elements are largely of value in the general engineering design. The temperature and precipitation data reported in Tables 2.2-2 and 3 have been obtained from the plant site.

High Winds

Strong winds are the most important meteorological hazard to the facilities. The region is frequented by relatively strong, gusty winds, usually accompanying the passage of squall lines or thunderstorms and the maximum wind associated with these phenomena is 90 mph on a 100 year recurrency interval.

The tornado presents a very specialized type of hazard involving both violent winds and extremely large, rapid changes in barometric pressure.

The storms are small, unpredictable in detail and rather infrequent, but they undoubtedly represent one of the few environmental factors that could, if ignored in plant design, inflict direct major damage on the facility. Typically, the tornado is a narrow funnel, often only a few hundred yards wide, in which winds may briefly reach 300 mph. Almost instantaneous changes in barometric pressure occur, reaching

2.6 LIMNOLOGY AND ECOLOGY

Lake Michigan is utilized as the source of condenser cooling water for the plant. Radioactive liquid wastes generated by the plant are processed by the Waste Disposal System and discharged into the cooling water outlet streams. All such discharges are carefully controlled and monitored prior to such releases in accordance with the provisions of 10 CFR 20.

The lake provides additional dilution capacity as well as a vast, dependable source of cooling water for the plant.

Provision is made to protect safety-related plant structures and equipment from flooding, waves, storms, and other phenomena generated in the lake.

2.6.1 LIMNOLOGY AND ECOLOGY STUDY PROGRAM

This section, for the most part, describes studies conducted before the Plant was placed into operation. Later progress reports and final summary reports are referenced in the Annual Environmental Operating Reports for the Donald C. Cook Nuclear Plant Units 1 and 2.

An extensive program of study of the limnology and ecology of Lake Michigan, with emphasis on the region adjacent to the plant site, was begun in the summer of 1966.

The initial studies[1,2] consisted of:

1. Bathymetric survey off the site, including consideration of bottom stability;

2. Bottom-type survey off the site, sediment types.

3. Determination of along shore current direction under various wind directions.

4. Determination, by dye dilution experiments, of diluting capacity of along shore currents at the site.

5. Determinations of the locations of local potable water intakes and of the possibility of plant effluent reaching them.

6. Numbers and distributions of bottom-living organisms.

7. Estimates of thermal effluent dispersion.

8. Studies of extraordinary seiches.

Subsequent studies[3] were made (1968) of the effects of power plant waste heat discharge on the ecology of Lake Michigan. Off shore waters of four power plants situated on the Lake were studied.

Measurement of temperatures were taken in actual discharge plumes and related to the benthos, zooplankton and phytoplankton samples taken in the area of the plume. No adverse trends, except for a slight decrease in the numbers of organisms found in the actual outfalls, were noted in the data collected.

Additional studies[4] were made in 1969 of thermal plume characteristics. These were made in plumes of plants operating on the Lake. The temperature, plankton and benthos were analyzed in the plumes. Again, minimal effects were recorded.

A grid of sampling stations was established in the Lake, off the plant site. Phytoplankton, zooplankton, and benthos were sampled. The temperatures were recorded to provide calibration data for the multi-spectral remote sensing over-flights by the Willow Run Laboratories. Water color and depth also were recorded to aid calibration of the over-flights.

Laboratory experiments were also conducted[4] to determine the uptake of radioactivity by amphipods. Tests were conducted in the presence and absence of sediment from which they are known to obtain most of their food. By comparing results isotopes in metabolic processes were identified. Results indicated that amphipods had a greater affinity for zinc than for the other isotopes (cerium, manganese, cesium, zirconium, ruthenium, and strontium). Also concluded was that the accumulation of strontium and zinc are enhanced by their availability in the sediment and that their accumulation involves metabolic processes.

In the winter of 1969-1970, operations were carried on to study the movements and effects of ice[5] on the shore line. Observations were made around the outfalls of operational plants as well as at the Cook Plant site. Even in the vicinity of shore line outfalls from operating plants, little melting of shore line ice occurred, and no shore line erosion could be attributed to melting of shore line ice.

An underwater gamma spectrometer was developed under a grant to the University of Michigan.[6] Using this underwater probe, activity of the bottom sediments off the plant site was mapped. These surveys of sediment radioactivity were repeated after the plant began operation to measure any change which would be attributed to the plant.

In conjunction with other utilities operating on Lake Michigan, an extensive survey of Lake Michigan was conducted. Initial results of the survey report sampling of 85 points in the Lake are indicated in Reference 7. These samplings including water samples, sediment, benthos, zooplankton, phytoplankton, fish, and biota were analyzed for radioactivity and chemistry (35 elements, using neutron activation and automatic absorption techniques).

The purpose of the survey was to:

1. Inventory the radioactive material in Lake Michigan, considering its natural radioactivity, fallout from nuclear detonations, and input from operating reactors. Also, to attempt to separate activities into biologically available an unavailable forms.

2. Establish the concentration of stable elements in Lake Michigan. This is necessary to document the upper limit for the reconcentration of radioactive waste materials along food pathways leading to man; presuming that radioactive elements can be biologically reconcentrated only to the same degree as are their stable forms.

3. Estimate the radiological and chemical wastes to be released to the Lake; - the sources include power plants, industrial plants, sewage plants, agricultural runoff and others for which there may be significant data.

4. Make a provisional forecast of the Lake five years hence. This is intended to strike a trial balance for the condition of the Lake 5 years from the end of the study.

Research was also conducted on:

1. Water temperature observations to determine temperature variations near shore, and to determine the size and shape of the plume of water from the plant discharge. These were done by recording equipment installed in place and also by boat survey and aerial multi-spectral scannings.

2. Biological Change Studies

Benthonic organisms, attached algae, floating algae, and bacteria, rooted vegetables and cladophera were sampled. These studies were conducted to determine the biological availability of food to organisms from sediments and plants. Thus, a most carefully determined reconcentration process in organisms was established.

2.6.2 REGIONAL FEATURES

Bathymetry

The basin of Lake Michigan is divided into northern and southern sub-basins by an incomplete sill of resistant materials extending from the region of Milwaukee, Wisconsin toward Muskegon, Michigan. The southern sub-basin, on which the plant is located, is shallower, rounder, and of more regular bathymetry than its northern counterpart. Figure 2.6-1 depicts the bathymetry of Lake Michigan.

The low water datum of Lake Michigan is 578.4 feet above mean sea level, according to U. S. Geological Survey figures. The lowest recorded level of the lake was 576.9 feet MSL during the 1964-65 winter; the highest recorded level was 583.5 feet MSL during the summer of 1886.

2.6.3 LOCAL FEATURES

Bathymetry

Figure 2.6-3 is a plot of the bottom of the lake adjacent to the site. It is characterized by gentle and regular topography. The 100 foot depth isopleth lies about six miles from shore. Isobaths are generally regular and parallel to the shoreline. Two sand bars lie close to

shore along the entire length of the site property. The inner bar averages about 500 feet from the shoreline while the outer bar runs approximately 1000 feet from the shoreline. Maximum water depth of five to six feet is present between the inner bar and the shore. Twelve to thirteen feet of depth is the greatest measured between the bars. The depth over the crest of the inner bar is about four feet, while the outer bar peaks at eight to nine feet beneath the surface.

Bottom Stability

A number of studies of bottom stability along the east shore of Lake Michigan have been made in the past decade or two. All have what appears to be very stable conditions near shore despite severe storms and winter icing. Present evidence indicates that the nearshore sandbars (particularly) do fluctuate in position but maintain a fairly consistent average position, with fairly consistent water depths over their crests.

Gross Currents

Although all of the currents of Lake Michigan are not thoroughly understood, certain of the larger features have been found with a surprising degree of constancy. The two most firmly established of these features are a general outflow current along the Michigan shore from Little Sable Point northward toward the Straits of Mackinac, and the presence of a large eddy near the eastern shore near Benton Harbor, Michigan. Figure 2.6-2 indicates the results of several studies made of lake currents.

Local Currents

In addition to the current major features indicated above, there appears to be a thin, elongate, counterclockwise eddy close against shore between Michigan City, Indiana and Benton Harbor (indicated by

X on Figure 2.6-2). This eddy may be controlling on alongshore currents to some degree.

The speed and direction of local water currents in the site vicinity control the movement and dispersal of plant effluent. Studies[2] indicated that along shore currents are established and controlled by interactions between local winds and any significant regional current pattern, with the winds dominant.

Eddies

Though eddies of circulation have been found, these are not "closed" in the sense that the same water recycles indefinitely. Instead, the continued pressure of wind set-up pushing new water against the east shore requires that equal volumes of east shore water must escape either by sinking or by northward flow along the Michigan shore. Thus, continuity requires that new water enter and old water be discharged from any cells of circulation existing along the Michigan shore under these winds.

Local Temperature Cycles

Figure 2.6-4 is a plot of surface water temperatures in Lake Michigan during the relatively cool year of 1965 and the relatively warm year of 1966. It can be noted that temperatures rise abruptly from a 32° icing condition in winter to a peak in July and August and then decrease linearly to ice-water temperatures by late December. Conditions in the waters directly off the plant site can, perhaps, be better represented by shifting the curve slightly to the right to conform with a reading of 73°F recorded on September 13, 1966.

Local Potable Water Intakes

A number of municipalities in southwestern Michigan utilize the waters of Lake Michigan as their source of potable water. These intakes are listed with their approximate distances from the plant discharge:

Northward

South Haven	32 miles
Benton Harbor	11 miles
St. Joseph	9 miles

Southward

Lake Township	0.25 miles
Bridgman	2.5 miles
Orchard Beach	7 miles
New Buffalo	16 miles
Grand Beach	18 miles
Michigan	19 miles
Unknown	22 miles
Michigan City, Indiana	25 miles

To the north, the outflow of the St. Joseph River interposes a physical and dynamic barrier to further progress of effluent northward along the shore. It is possible, however, that under light wind conditions when plumes are more coherent and less diluted, effluent could reach the water intakes at Lake Township, Bridgman and Orchard Beach. These intakes are also of the infiltration type, providing added protection. However, the prevailing winds of summer, when the worst dilution conditions (minimum wind and wave section) exist, are expected to carry effluent away from these areas.

2.6.4 UNUSUAL CONDITIONS

Seiches

Seiches are oscillations in the level of lakes and similar bodies of water caused by the passage of squall lines across the body of water. In Lake Michigan, these squalls have their fronts oriented NE to SW and are accompanied by an abrupt increase in barometric pressure and local high winds. There have been a number of seiches recorded in

the Great Lakes in recent years, the great majority of which were of only a few inches amplitude and, therefore, of no consequence. A few, however, have caused considerable flooding damage, and even loss of life. The most severe of the recent large seiches occurred on June 26, 1954 and caused water level increases of up to 10 feet at North Avenue in Chicago, Illinois. The greatest level increase recorded on the lake's eastern shore was 6 feet at Michigan City, Indiana.

Seiches do not have the rapidity or damaging power of a wind-wave of equal height. Instead, the rise of water is continuous over several minutes, and damage is primarily due to flooding.

Within the bounds of seiche causing conditions, the most severe initiating meteorological condition may be assumed to be a squall line traversing the entire lake from a direction west of northwest with a progress velocity sufficient to match the mode of the lake's southern sub-basin and producing a seiche front so shaped as to trap against the shore at the plant site.

The maximum recorded amplitude of an open lake seiche produced under such conditions was 4.2 feet observed at the Wilson Avenue Crib in Chicago on July 6, 1954. A previous seiche on June 26, 1954, which resulted in a rise of 3.2 feet at Wilson Avenue Crib, caused the rise estimated at less than 6 feet in the Michigan City yacht basin, a point approximately 25 miles south of the plant site in an area where seiche effects are considered more severe than those farther to the north. Taking these values in proportion, one can postulate the maximum seiche producing a water level increase of as much as 8 feet in the Michigan City yacht basin.

The infrequency of seiches of significant size on Lake Michigan restricts to some degree the volume of recorded data from which future seiche characteristics may be predicted. The large body of information which is available, including measurements and observations of

actual seiches, the characteristics of the shoreline at the plant site, historical meteorological conditions, computations based upon mathematical models, etc. confirm that no water level increase of as much as 8 feet should ever be experienced at the plant site.

However, as an added measure of conservatism, the plant safety components are protected against a water level increase of 11 feet.

<u>Wind Waves</u>

Wind generated waves are limited in their dimensions by wind velocity, fetch (open water distances available to the wind), and by the length of time the wind has blown. The greatest fetch for the plant site over Lake Michigan is 265 statute miles (223 nautical miles) to the north. The maximum deep waterwave to be expected as incident to the plant is therefore approximately 23 feet, and would require a sustained north wind of about 26 knots for over 19 hours.

The runup of such a wave on the site shore, discounting the effects of the off-shore sandbars has been calculated as 3.7 feet. This figure is overly conservative, however since a large wave approaching the beach would be tripped by each of the sand bars.

<u>Coincidence of Maximum Wave and Maximum Seiche</u>

The maximum wind wave can occur only in a fully developed sea, for which there is a definite requirement for a long wind duration. The seiche, on the other hand, accompanies a squall-line storm that moves across the lake at a speed similar to one of the lake's natural oscillation modes.

Seiches, therefore, occur at the beginning of a storm while the maximum wind wave would not manifest itself until many hours later, and it is an impossibility for the maximum seiche to coincide with the maximum wind wave.

2.7 ENVIRONMENTAL RADIATION MONITORING

The environmental radiation monitoring program, described herein, was designed to determine the effects of routine and inadvertent radioactive releases from the plant have on the environment. Provisions are made to monitor liquid and gaseous wastes before they are released, and further, to monitor the atmosphere, lake water, well water, aquatic organisms, milk and food materials when necessary. Liquid and gaseous wastes are released in batches from time to time, after appropriate decay, processing and analysis. No foreseeable environmental conditions will restrict the release of wastes. However, if extreme conditions should indicate the desirability, wastes can be retained until dispersion conditions improve.

2.7.1 DETERMINATION OF MAXIMUM ALLOWABLE RELEASE RATES TO AIR AND WATER

The first step in the program is to determine the maximum rate at which radioactive material may be continuously discharged without exceeding the limits set by 10 CFR 20 at the site boundary. The initial estimate is based on the plant design, the anticipated composition of the radioactive material to be released, and the dilution and dispersion characteristics of the air and water into which these materials are discharged.

The unit vent, through which radioactive gas waste is routinely released, runs up the outside wall of the containment building. The vent opening is at the top of the containment building, about 160 feet above grade.

For the evaluation of releases over long time periods, the sector dispersion formula, given below, is used:

$$\frac{\chi}{Q} = \frac{360}{\phi} \cdot \frac{f}{100} \cdot \frac{1}{\pi^{3/2} \quad 2^{1/2} \quad \bar{\mu} \cdot x \cdot \sigma_z}$$

where:

χ	=	downwind concentration at ground level (micro curies/ cubic meter = μCi/cc)
Q	=	rate of gaseous emission (curies/sec)
ϕ	=	angular width of the sector in question (degrees)
f	=	frequency with which the given meteorological conditions occur in that sector (percent)
z	=	vertical standard deviation of a Gaussian plume (meters)
$\bar{\mu}$	=	mean wind speed (meters/sec)
x	=	distance in the direction of the wind to the site boundary (meters).

The meteorological data in Tables 2.2-16, 2.2-17, 2.2-18 and 2.2-32 indicate that wind from the 200° sector is the most prevalent on-shore wind and, when inserted in the formula above, give an estimated annual average value for χ/Q in this sector of 1.5×10^{-6} seconds per cubic meter. For the waste gas mixture given in Table 11.1-6, the weighted average maximum permissible concentration is 3×10^{-7} μCi/cc. The maximum rate at which this mixture can be discharged continuously is Q = 0.2 curies/sec, without exceeding the 10 CFR 20 limit at the site boundary.

If suitable for discharge, liquid radioactive wastes are released to the condenser circulating water system. The maximum discharge concentration for liquids is defined in the Plant Technical Specifications. This defined concentration ensures that the limits established in 10 CFR 20 are not violated.

2.7.2 SAMPLING STATIONS

The stations for sampling airborne dust, precipitation and external radiation are placed in two rings about the plant. The inner, or indicator ring, stations are placed where it is estimated that maximum ground concentrations of material released from the plant will occur. Figure 2.7-1 indicates the locations selected for the six indicator stations (shown as A1 through A6).

Figure 2.7-2 shows the locations which have been selected for the four background air stations in the outer ring as identified as A. These locations are all about 20 miles from the plant and thus the ground-level concentrations of radioactive material originating from the plant will be less than 1 percent of the concentrations at the indicator stations.

Locations of TLD stations are shown in Figures 2.7-1, 2.7-2 and 2.7-3. Nine indicator TLD stations (shown as A1 through A9 on Figure 2.7-1) are located on an approximate 2000 foot radius and 10 indicator TLD stations are within a 4 to 5 mile radius from the plant (shown as A1 through A10 on Figure 2.7-3). Four background TLD stations located about 20 miles from the plant are identified as A on Figure 2.7-2.

Sampling Lake Water

The locations of the sampling stations for lake water are described in Table 2.7-1. Indicator lake samples are taken along the lake front from Condenser Cooling Water intake and, 500 feet north and south of the plant.

The sampling of aquatic organisms presents a number of difficulties. Out to a depth of 20 feet or more, the lake bottom is scoured sand and is almost sterile. Attempts to find suitable organisms in sufficient quantities for routine sampling have, so far, been unsuccessful.

Benthonic organisms occur only at depths greater than twenty feet; such depths occur at 1,000 feet or more from shore. Routine sampling under such circumstances is impractical for extended periods of unfavorable weather.

Fish are collected and analyzed in the program, but fish are a poor sampling medium because they range so widely that it is never certain that they represent the area where they happen to have been caught. Shore minnows apparently have a restricted range and may be more useful for monitoring purposes than large fish.

The radioactive content of lake sediment has been assayed by a rolling underwater gamma-sensitive probe developed for this purpose. An area of the lake bottom approximately 4 miles north and south of the plant to a distance of about 4 miles from shore has been surveyed.

Sampling of Well Water

Well water is the only material in the environmental sampling program that is not likely to be affected by fallout of radioactivity. With well water, and only with well water, is the before and after principle sound. There are seven wells within the exclusion area; three are west of the plant north-south axis, and four are east as shown in Table 2.7-2. The orientation of these wells with respect to the plant was chosen as a result of groundwater movement, which was found to be east to west.

Sampling of Milk

The selection of milk sampling locations are, of course, limited to pastures where milk cows graze. The locations shown in Table 2.7-3 and Figures 2.7-2 to 2.7-4 are subject to change as the location of milk cows change.

Sampling of Food

It is now evident that milk alone provides sufficient control of terrestrial pathways. Additional human food materials is not needed in the program unless radioactive materials other than noble gases, tritium and iodine are detected in the plant discharges to the atmosphere. Nevertheless, additional human food crops will be sampled annually for the purpose of information.

The noble gases do not enter into the food chains. Tritium enters freely into all food chains; however, since almost all tritium occurs as tritiated water, it does not concentrate in food pathways as do other elements. Iodine does concentrate along food pathways and it has been shown that the air-pasture-cow-milk pathway is limiting and that milk is the best monitoring medium. Lake water is not used for irrigation in the area. There is, consequently, neither need nor justification for monitoring human foods other than milk in the terrestial environment, and fish in the aquatic environment.

All sampling points have been selected on their being representative of the area and accessible for sampling. Table 2.7-4 describes the current (1982) Environmental monitoring program, as defined in the plant Technical Specifications.

2.7.3 STABLE ELEMENT STUDIES

The pre-operational phase of the environmental program includes a study of stable element concentrations in the lake water and in selected aquatic organisms. The purposes of these measurements are (1) to put an upper limit on the degree to which radioactive material discharged from the plant into the lake could be concentrated in human food taken from the lake, (2) to find critical pathways and the means for estimating population exposure by these pathways, and (3) to determine the relationship between the concentration factors in fish (and any other human foods taken from the lake) to those in aquatic organisms selected to monitor the water environment.

The principle involved in these stable element studies is that the radioactive isotopes of an element cannot be concentrated more highly than the corresponding stable isotopes of that element by biological, chemical or physical processes in the environment. The general form of these studies is described in the next paragraph.

The radioactive isotopes anticipated in the liquid waste (Table 11.1-5) are examined, as are the data on similar operating reactors. From these one obtains a list of the elements which correspond to all the radioactive isotopes which may contribute to radioactivity in food chains. Samples of lake water, edible portions of fish, and possible monitor organisms are collected and analyzed for each of the elements in the list. The data so obtained give concentration factors from water to fish, and from water to monitor organisms for the stable elements. Radioactive isotopes of these elements cannot be concentrated to factors greater than those for the corresponding stable elements.

2.7.4 MEASUREMENT OF RADIOACTIVITY

The pre-operational phase of the environmental program included the collection and analysis of samples for radioactivity; the intensity of the post-operational phase is concerned exclusively with radioactivity released from the plant. This section describes the equipment and techniques that are used to collect and analyze environmental samples for radioactivity.

Concentrations of the radioactive noble gases in the environment is measured with thermoluminescent dosimeters. The detection limit of thermoluminescent dosimeters is 1 to 2 mR per month. This sensitivity corresponds to 2 to 4 percent of the maximum permissible public concentration of the noble radioactive gases.

The air sampling units draw about 3×10^8 cc of air per week through a filter. The detection limit of a 3 x 3 inch sodium iodine gamma spectrometer is on the order of 10^{-5} µCi for most gamma-emitting radioisotopes, which corresponds to a concentration limit on the order of 10^{-13} µCi/cc, far below the maximum permissible public concentration of any radioactive material the plant could discharge to the atmosphere.

The environmental air sampling units are fitted with charcoal cartridges to collect iodine. If the air passing through this cartridge were at the public maximum concentration of iodine 131 for the entire week, the cartridge would collect 0.03 µCi.

Tritium is measured in a liquid scintillation counter with a nominal sensitivity limit of 10^{-4} µCi/cc, which is 0.03 of the permissible drinking water concentration. Analyses for the other isotopes which emit no gamma rays will be made, when necessary, by contract with an outside laboratory.

2.7.5 OPERATION OF THE PROGRAM

The environmental radiation monitoring program was started some 12 to 18 months before fuel was loaded in Unit 1. During this period, equipment was tested, the suitability of the selected sampling media and sample points were determined, analytical procedures were tested, and some data was accumulated and examined for statistical variability. Modifications that were necessary to attain reliable and coherent data were made during this period.

Prior to each liquid release, a determination is made of the amount of radioactive material released to the lake during the preceding year; for gaseous releases, periodic checks of release totals are performed to assure the Technical Specification limits will not be exceeded.

The general principles of the program have been discussed with the appropriate officials of the State of Michigan Department of Public Health and the Michigan Water Resources Commission and they have accepted its principles.

2.7.6 SUMMARY OF PREOPERATIONAL ENVIRONMENTAL MONITORING PROGRAM

The average monthly LiF Dosimeter Loadings for the quarter of August, 1971 through December, 1971, on site, vary from 3.9 ± 1.3 to 11.7 ± 0.8 mrem and offsite 3.9 ± 1.2 to 13.3 ± 1.1 mrem.

Initial water samples taken in the Lake (similar to that shown in Figure 2.7-2) and at Bridgman, St. Joseph, Benton Harbor and New Buffalo show a tritium concentration of from 0.562 ± 0.036 to 0.583 ± .036 picocuries/liter. Gross beta at the above sampling points showed 0.0 ± 2.0 picocuries/liter to 6.8 ± 1.0 picocuries/liter.

The determination of gross beta in the air particulates on site is 0.01 ± 0.01 to 0.30 ± 0.01 picocuries/cubic meter. The same values for offsite stations are 0.01 ± 0.01 to 0.24 ± 0.1 picocuries/cubic meter.

TABLE 2.7-1

LOCATIONS OF THE LAKE WATER SAMPLING STATIONS

Indicator Stations

1. Condenser cooling water intake.
2. 500 feet southwest from point of discharge along the lake shore.
3. 500 feet northeast from point of discharge along the lake shore.
   (See Figure 2.7-1)

Background Stations

Lake Township water intake, 0.25 miles south from the plant.

St. Joseph municipal water intake, 9 miles northeast from the plant.

New Buffalo near city water intake, 15 miles southwest from the plant.
(See Figure 2.7-2)

TABLE 2.7-2

WELLS AVAILABLE FROM MONITORING PROGRAM

(Refer to Figure 2.7-1 for a map indicating the location of these sample points)

Well No.	Distance from Plant in Feet	Direction from North
1	2000	9°
2	3400	65°
3	4500	84°
4	400	345°
5	250	289°
6	500	223°
7	3000	186°

TABLE 2.7-3

LOCATIONS OF THE MILK SAMPLING STATIONS

A) INDICATOR FARMS

BRIDGMAN:	G. G. SHULER & SONS RT 1, SNOW ROAD, BARODA Figure 2.7-4, FARM #1
STEVENSVILLE:	GERALD TOTZKE RT 1, BARODA Figure 2.7-4, FARM #2
GALIEN:	CLAYTON & L. SMITH RT 1, GALIEN Figure 2.7-4, FARM #3

B) BACKGROUND FARMS

SOUTH BEND:	FARM ON SOUTH SIDE OF U.S. 20 4.5 MILES EAST OF IND. 39, Figure 2.7-2
DOWAGIAC:	FARM ON EAST SIDE OF M51 1.6 MILES NORTH OF POKAGON ROAD, Figure 2.7-2

TABLE 2.7-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
1. Airborne a. Radioiodine and and Particulates	A1-A6 (Site) New Buffalo, South Bend, Dowagiac, and Coloma are Background	Continuous operation of sampler with Sample Collection as required by Dust Loading But at Least Once Per 7 Days	Radioiodine canister Analyze: Weekly for I-131 Particulate sample Gross Beta Radio-activity following Filter Change[a], composite (by loca-tion) for gamma isotopic quarterly.
2. Direct Radiation	a) T1-T9 (Site) b) New Buffalo, South Bend, Dowagiac, Coloma c) 10 TLD Monitor Locations in the Five Mile Radius	At least once per 92 Days	Gamma Dose. At Least Once Per 92 Days.
a. Waterborne Surface	L1, L2, L3	Composite* Sample Over One-Month Period	Gamma Isotopic Analysis monthly. Composite for tritium analysis-quarterly.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 24 hours.

TABLE 2.7-4 (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
b. Ground	W1-W7	Quarterly	Gamma Isotopic and Tritium analysis quarterly.
c. Drinking	St. Joseph Lake Township New Buffalo	Composite* Sample Collected over a Period of ≤ 31 days. Composite* Sample Over a 2-week Period if I-131 Analysis is Performed.	Gross Beta and Gamma Isotopic Analysis of each composite sample. Tritium Analysis of composite Quarterly. I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.
d. Sediment from Shoreline	L2, L3	2/year	Gamma Isotopic Analysis Semi-Annually.
4. Ingestion a. Milk	Stevensville Bridgman Galien Dowagiac South Bend	At least once per 15 days when animals are on Pasture. At Least Once Per 31 Days at Other Times.	Gamma Isotopic and I-131 Analysis of Each Sample.

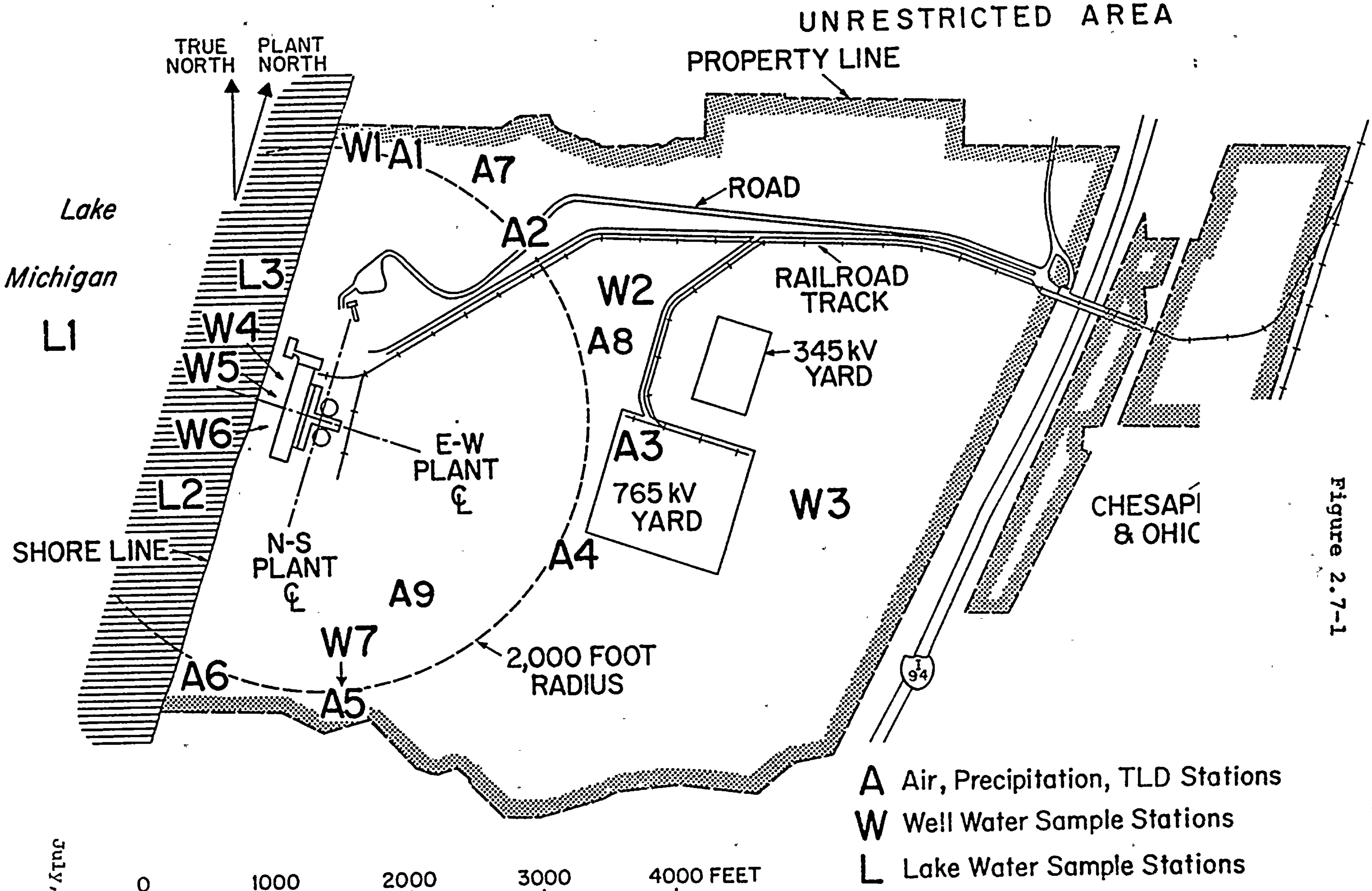
*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 24 hours.

TABLE 2.7-4 (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
b. Fish	Plant Site Off-Site	2/year	Gamma Isotopic Analysis on Edible Portion.
c. Food Products	Plant Site Off-Site (approx. 20 mi)	At time of Harvest. One Sample of Each of the Following Classes of Food Products: 1. Grapes	Gamma Isotopic Analysis on Edible Portion.
	Plant Site	At time of Harvest. One sample of Broad Leaf Vegetation	Gamma Isotopic Analysis

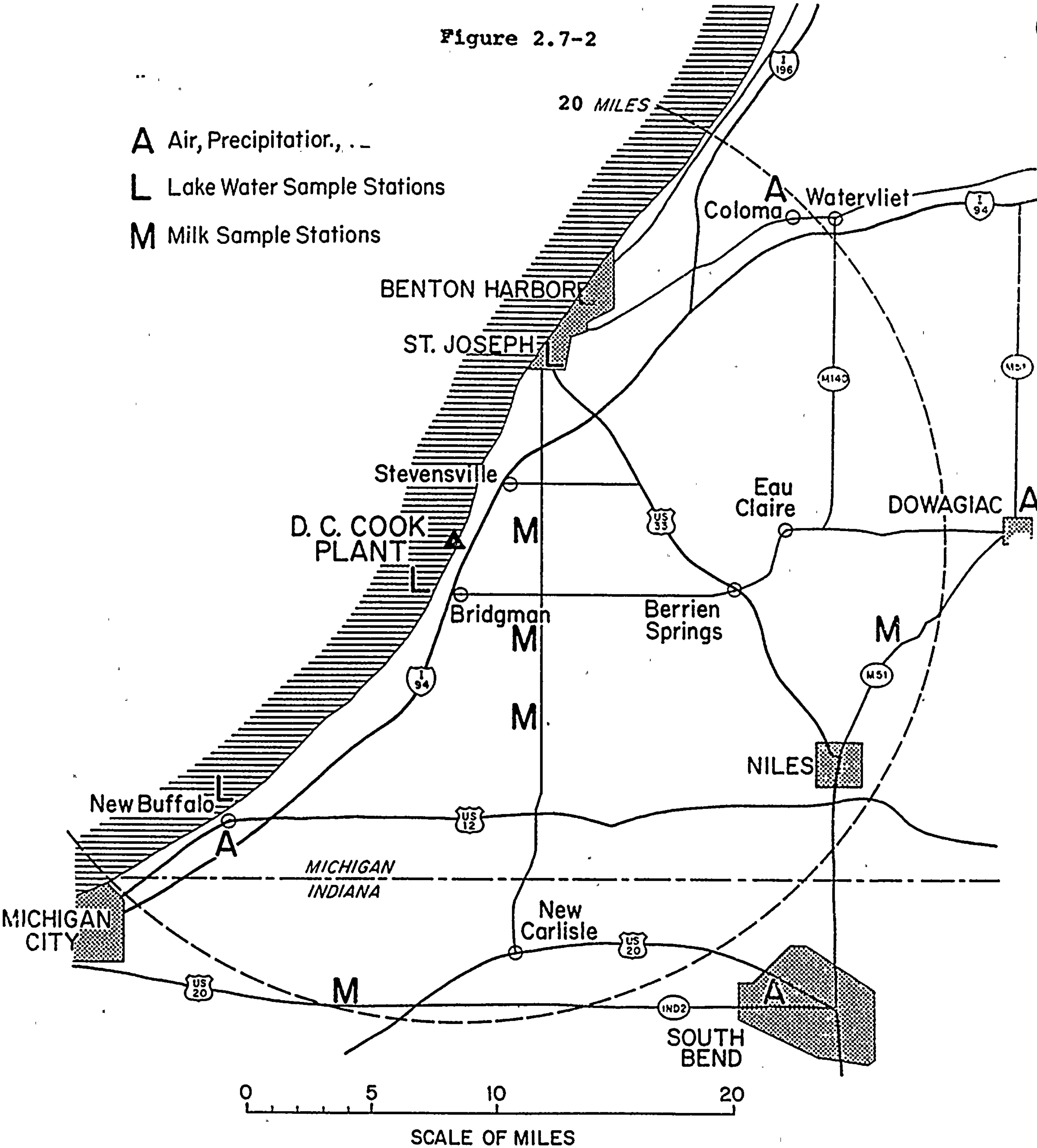
[a] Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than 10 times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.



Note
Stations A7, 8 and 9 are TLD Stations Only

Figure 2.7-1

Figure 2.7-2



A Air, Precipitatior., . _
L Lake Water Sample Stations
M Milk Sample Stations
20 MILES
Coloma
Watervliet
BENTON HARBOR
ST. JOSEPH
Stevensville
Eau Claire
DOWAGIAC
D. C. COOK PLANT
Bridgman
Berrien Springs
NILES
New Buffalo
MICHIGAN
INDIANA
MICHIGAN CITY
New Carlisle
SOUTH BEND
0
5
10
20
SCALE OF MILES

3.0 REACTOR

3.1 SUMMARY DESCRIPTION

The current Cycle 8 reactor core contains four regions of fuel in a low leakage loading pattern as described in Section 3.6.2. The fuel rods are cold worked, partially annealed Zircaloy tubes containing slightly enriched uranium dioxide fuel.

All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

The fuel assembly is a canless type with the basic assembly consisting of the RCC guide thimbles fastened to the grids, and to the top and bottom nozzles. The fuel rods are supported at several points along their length by the spring-clip grids.

Full length rod cluster control assemblies and burnable absorber (poison) rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes. For Cycle 8, the absorber material in the fixed burnable absorber rods is in the form of aluminum oxide-boron carbide pellets sealed in Zircaloy-4 tubes.

The control rod drive mechanisms for the full length RCC assemblies are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

The reactor was initially supplied with fuel from Westinghouse Electric Co. (W). Reload fuel for Cycles 2 through 7 was supplied by Exxon Nuclear Co (ENC). Cycle 8 reload fuel was supplied by Westinghouse Electric Co. The latest information regarding the current fuel cycle may be found in Sub-Chapter 3.6.

In addition to this summary description, this chapter contains: a description of the mechanical components of the reactor and reactor core, including Cycle 1 W fuel assemblies, reactor internals and control rod mechanisms (Sub-Chapter 3.2); a description of the Cycle 1 nuclear design for the W fuel (Sub-Chapter 3.3); a description of the Cycle 1 thermohydraulic design (Sub-Chapter 3.4); a description of the ENC fuel design for Cycles 2 through 7 (Sub-Chapter 3.5); and a description of the W fuel and Cycle 8 core design (Sub-Chapter 3.6).

The information contained in this chapter is principally concerned with the nuclear fuel and reactor internals design and therefore does not necessarily reflect the same information as that used in the safety analysis. For information concerning safety analysis, Chapter 14 should be consulted.

3.1.1 Performance Objectives

The current licensed thermal power limit is 3250 MWt. Calculations indicate that hot channel factors are considerably less than those used for design purposes in this application. The thermal and hydraulic design, and accident analyses (except large break LOCA) in Chapter 14, were performed at 3411 MWt for Cycle 8. These analyses identify design/safety limits for a potential uprating.

The turbine-generator and plant heat removal systems have been designed for a thermal rating of 3391 MWt. The portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power rating of 3391 MWt, as have the evaluations of activity release and radiation exposure.

The initial reactor core fuel loading was designed to yield the first cycle average burnup of 16,666 MWD/MTU, and the Cycle 2 through 7 reload designs yield an average cycle burnup of 10,000 MWD/MTU. Reload designs

for Cycle 8 and beyond yield cycle burnups of between 15,500 and 16,000 MWD/MTU. The fuel rod cladding is designed to maintain its integrity for the anticipated core life. The effects of gas release, fuel dimensional changes, and corrosion-induced or irradiation-induced changes in the mechanical properties of cladding are considered in the design of the fuel assemblies.

Rod Control Clusters are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the applicable design minimum departure from nucleate boiling (DNB) ratio (see Section 3.6.3). This is accomplished for Cycle 8 by ensuring sufficient control cluster worth to shut the reactor down by at least 1.6% in the hot condition with the most reactive control cluster stuck in the fully withdrawn position.

Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

In addition, the control rod worth, in conjunction with the boric acid injection from the boric acid injection tank, is sufficient to prevent return to criticality as a result of the maximum credible steam break (one safety valve stuck fully open), even assuming that the most reactive control rod is in the fully withdrawn position.

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. During design, nuclear parameters are calculated for various operational phases and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at 118% overpower have been conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 Principal Design Criteria

Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core

and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation, with appropriate margins for uncertainties, and for specified transient situations which can be anticipated.

The reactor core, with its related control and protection systems, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations. This includes the effects of the loss of reactor coolant flow, trip of the turbine generator, and loss of normal feedwater and loss of all off-site power.

The Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than the applicable design value for W and ENC fuel.

The integrity of fuel cladding is ensured by preventing excessive fuel swelling, excessive clad heating, and excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

a) Minimum DNB ratio equal to or greater than the applicable design values for W and ENC fuel. For Cycle 8, the design values are given in Section 3.6.3.

b) Fuel center temperature below melting point of UO_2.

c) For $\underline{W}$ fuel for the initial core and ENC reload fuel, internal gas pressure less than the nominal external pressure (2250 psia), even at the end of life. For $\underline{W}$ reload fuel in Cycle 8 and the following cycles, the rod internal gas pressure shall remain below the value which causes the fuel-cladding diametral gap to increase due to outward cladding creep during steady-state operation.

d) Clad stresses less than the Zircaloy yield strength.

e) Clad strain less than 1%.

f) Cumulative strain fatigue cycles less than 80% of design strain fatigue life for ENC fuel. Cumulative strain fatigue cycles are less than the design fatigue life for $\underline{W}$ reload fuel in Cycle 8 and the following cycles.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial core.

The rod cluster control (RCC) assemblies are divided into two categories comprising control banks and shutdown banks. The control banks used in combination with chemical shim control provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of RCC assemblies are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life, such as those due to fuel depletion and fission product buildup.

Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

The reactor core, together with the Reactor Control and Protection System, is designed so that the applicable minimum allowable DNBR value is satisfied, and that there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the Cycle 8 core at least 1.6 per cent subcritical at the hot zero power condition (k_{eff} = 0.984) following a trip from any credible operating condition, assuming the most reactive RCC assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical, assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid addition via the Emergency Core Cooling System. The design minimum shutdown margin for Cycle 8 is 1600 pcm, assuming the maximum worth control rod is in the fully withdrawn position, and allowing 10% uncertainty in the control rod calculations.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown. Redundant equipment is provided to guarantee the capability of adding boric acid to the Reactor Coolant System.

Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies, and shall be capable of limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Normal reactivity shutdown capability for Cycle 8 is provided within 2.4 seconds following a trip signal by control rods, with boric acid injection used for the long term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria, the shutdown capability prevents return to critical as a result of the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which inject

analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of $7.5 \times 10^{-4} \Delta k/k/sec$, which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per minute (~45 inches per minute).

3.1.3 SAFETY LIMITS

The reactor is capable of meeting the performance objective throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters which are pertinent to safety limits are specified below for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

Nuclear Limits

At full power, the Cycle 8 nuclear heat flux hot channel factor, F_Q did not exceed 2.10 for W fuel and 2.04 for ENC fuel. The equations and curves which show the F_Q limits as a function of power, fuel height and burnup are defined in Section 3.2.2 of the Cook Unit I Technical Specifications.

For any condition of power level, coolant temperature, and pressure which is permitted by the control and protection system during normal operation and anticipated transients, the hot channel power distribution is such that the minimum DNB ratio is greater than or equal to the applicable design value given in Section 3.6.3.

Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

a. A minimum hot shutdown margin as shown in the Technical Specifications is available, assuming a 10% uncertainty in the control rod calculation.

b. This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.

c. The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is not less than the applicable DNBR design limit. For Cycle 8 the design limit is given in Section 3.6.3.

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB), which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant, resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location, as predicted by the W-3 and WRB-1 correlations, to the existing heat flux at the same core location is the DNB ratio. The applicable design limit DNB ratio for W and ENC fuel corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

Mechanical Limits

Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod control cluster assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The internals are further designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident does not cause sufficient deformation to prevent rod cluster control assembly insertion.

Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating

3.2 MECHANICAL DESIGN

3.2.1 Mechanical Design and Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2.1-1 and in elevation in Figure 3.2.1-2. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and guide thimble plugging devices, provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters for the initial core is given in Table 3.2.1-1. Design parameters for reload fuel are presented in Tables 3.5.1-2 and 3.6.1-1.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are all identical in configuration, but contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144 inches within Zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. The enrichments of the fuel for the various regions in the first core are given in Table 3.2.1-1. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The initial core is divided into regions of three different enrichments. The loading arrangement for the initial cycle is indicated in Figure 3.2.1-3. The loading arrangement for the current Cycle 8 may be found in Figure 3.6.2-1 and Table 3.6.2-1.

The control rods, designated as Rod Cluster Control Assemblies (RCCA), consist of groups of individual absorber rods which are held together by a spider at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 3.2.1-4 shows a typical rod cluster control assembly.

As shown in Figure 3.2.1-2, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surface.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies which confines the upward flow of coolant in the core area to the fuel bearing region.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core the coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the lower core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to flat sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel inner surface. Another Inconel block is bolted to each of these blocks, and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of 1/2 inch, and there is an additional strain displacement in the energy absorbing devices of approximately 3/4 inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop.

In conclusion a set of "as built" dimensions were taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

Fuel Quality Control

Quality Control philosophy is generally based on the following inspections being performed to a 95% confidence that at least 95% of the product meets specification, unless otherwise noted, using either a hypergeometric function with zero defects for small lots or the latest revision of Mil-105D for large lots. This confidence level has been based on past experience gained during the manufacturing of over 400 metric tons of uranium cores. The following inspections are included:

1) Component Parts

   All parts received are inspected to a 95/95 confidence level. The characteristics inspected depend upon the component parts and include dimensional and visual checks, audits of test reports, material certification and non-destructive testing such as X-ray and ultrasonic. Westinghouse materials process and component specifications specify in detail the inspection to be performed.

   All material used in the manufacture of this core has been accepted and released by Westinghouse Quality Control.

2) Pellets

   Inspection is performed to a 95/95 confidence level for the dimensional characteristics such as diameter, length and squareness of ends. Additional visual inspections are performed for cracks, chips and porosity according to

standards established at the beginning of production. These standards are based upon standards used in previous cores which have in turn served as standards for over 50 million pellets manufactured and used in operating cores. Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a sample basis throughout pellet production.

3) Rod Inspection

Rod inspection consists of the following 100% non-destructive inspection and is based on the experience, specifications, procedures and standards established on previously manufactured and operating cores.

a) Leak Testing

Each rod is tested to a known leak rate using mass spectrometry with helium being the detectable gas. This is the system used previously on the leak test of over 300,000 rods.

b) X-ray

All fuel rod weld enclosures are X-rayed at 0° and 90° using weld correction forms. X-rays are taken in accordance with ASTM E-142-68, using 2-2T as the basis of acceptance.

c) Dimensional

All rods are dimensionally inspected prior to final release and upgrading. The requirements include such items as length, camber, and visual inspection.

3.3 NUCLEAR DESIGN

3.3.1 Nuclear Design and Evaluation

This section presents the nuclear characteristics of the initial core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under operational modes, including both transient and steady state, is demonstrated. Power distribution limits have been updated to the Technical Specification in effect on September 1983 which applies to the Cycle 8 core and subsequent cores with W OFA reloads. Nuclear characteristics of the ENC reload fuel and W reload fuel are discussed in Sections 3.5 and 3.6, respectively.

Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics for the initial core is presented in Table 3.3.1-1.

Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with changes in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions; (2) changes in reactivity associated with changes in the fission product poisons xenon and samarium; (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium); and (4) changes in reactivity due to burnable poison burnup.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling has been established as shown in Table 3.3.1-1, line 29. This concentration, together with the control rods, provides approximately 10 percent shutdown margin for these operations. The concentration is also sufficient to maintain the core shutdown without any RCC rods during refueling. For cold shutdown, at the beginning of core life, a concentration (shown in Table 3.3.1-1, line 37) is sufficient for one percent shutdown with all but the highest worth rod inserted. The boron concentration (Table 3.3.1-1, line 29) for refueling is equivalent to less than two percent by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial full power boron concentration with equilibrium xenon and samarium was 1152 ppm. As these fission product poisons were built up, the boron concentration was reduced to 838 ppm.

This initial boron concentration is that which permits the withdrawal of the control banks to their operational limits. The xenon-free hot, zero power shutdown (k = 0.99), with all but the highest worth rod inserted, was maintained with a boron concentration of 734 ppm. This concentration was less than the full power operating value with equilibrium xenon.

The basic features of the PDC-II procedures are as follows:

1. An $F_Q^T(z)_{Eq}$ distribution is determined along with an associated axial offset, denoted as the target axial offset (AO_T), at full power, equilibrium xenon conditions. The $F_Q^T(z)_{Eq}$ distribution is the measured $F_Q^N(z)$ distribution multiplied by the uncertainty factors 1.05 x 1.03, where 1.05 is the measurement uncertainty and 1.03 the engineering factor.

2. The $F_Q^T(z)_{Eq}$ distribution is multiplied by the V(z) factor, a cycle dependent function shown in Table 3.3.1-4 for cycle 8, to obtain the maximum anticipated $F_Q^T(z)_{Max}$ which is compared to the Technical Specification limits, $F_Q^T(z)_{TS}$. This limiting curve for $F_Q^T(z)_{TS}$ is given by the product of $F_Q^L(E)$ times K(z), which is illustrated for Cycle 8 in Figures 3.3.1-7 and 3.3.1-8, respectively. for W fuel, and in Figures 3.3.1-9 and 3.3.1-10, respectively, for ENC fuel. If $F_Q^T(z)_{Max}$ does not exceed the $F_Q^T(z)_{TS}$ limit, then operation under the PDC-II procedures will protect the $F_Q^T(Z)$ Technical Specification limits and supplemental monitoring (such as APDMS) is not required. If the product $F_Q^T(z)_{Eq} * V(z)$ exceeds the $F_Q^T(z)_{TS}$ limit, one of two alternatives is available:

   (a) Supplemental power distribution monitoring such as APDMS must be initiated above a power level equal to the minimum value of the ratio [$F_Q^T(z)_{TS}$ limit/maximum anticipated $F_Q^T(z)_{Max}$], or

   (b) Reactor core power must be reduced to a power level equal to the minimum value of the ratio [$F_Q^T(z)_{TS}$/ $F_Q^T(z)_{Max}$].

3. For each axial offset target value (AO_T), a target band (AO_{TB}) is allowed.

$$AO_{TB} = \frac{\pm 5\%}{P/P_o}$$

where P = operating reactor power (MWth)

P_o = reactor rated power (MWth)

4. Below a relative power (P/P_o) of 0.9, the axial offset is allowed to deviate from the target band for one hour out of each twenty-four consecutive hours, provided that the measured axial offset remains within a broader, but specified, axial offset band. If this requirement is violated, the core relative power must be reduced below 0.5 of rated power where no restrictions on AO are imposed. Above a relative power of 0.9, the measured AO must remain within the allowable target band at all times.

Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection System.

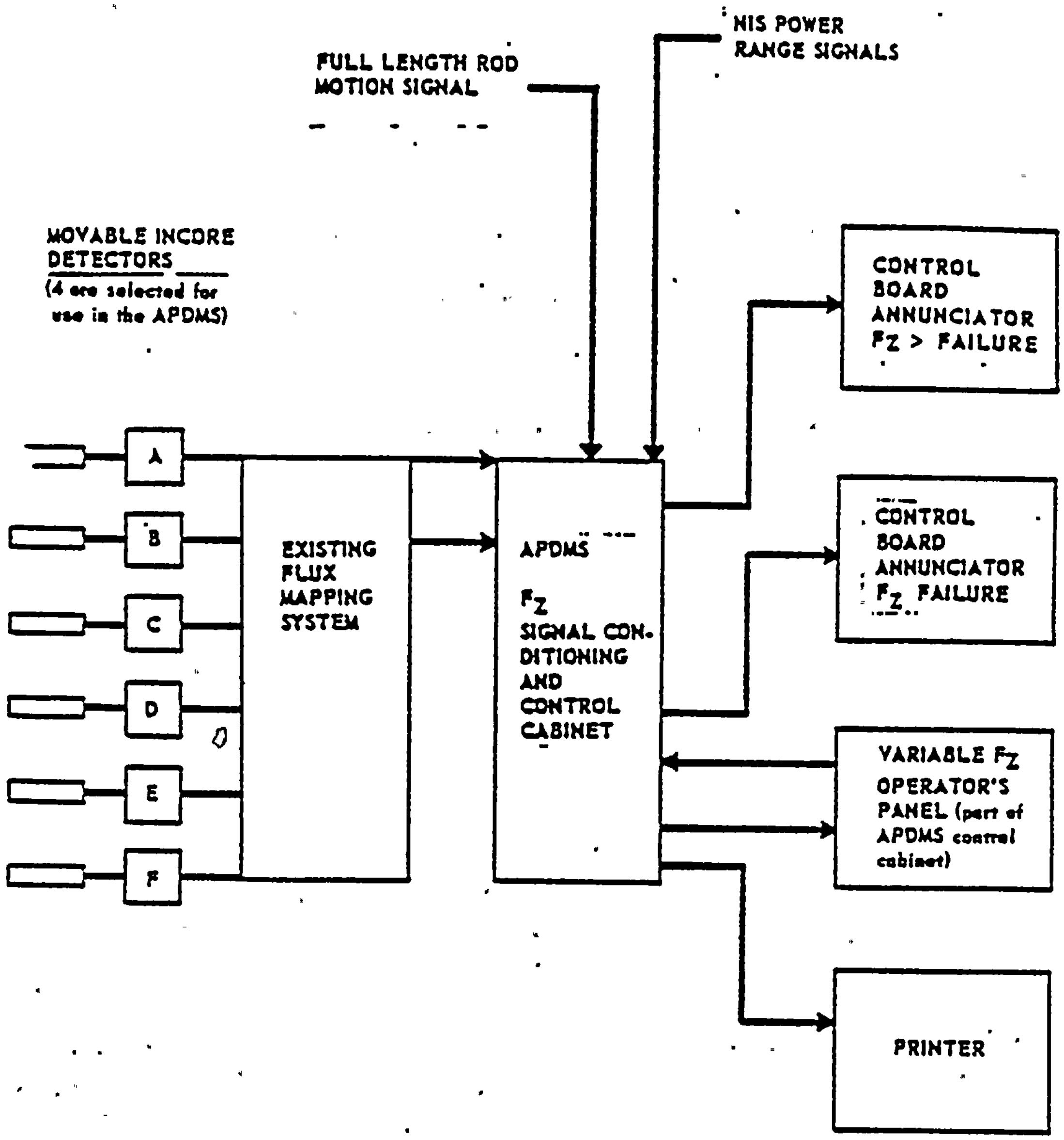


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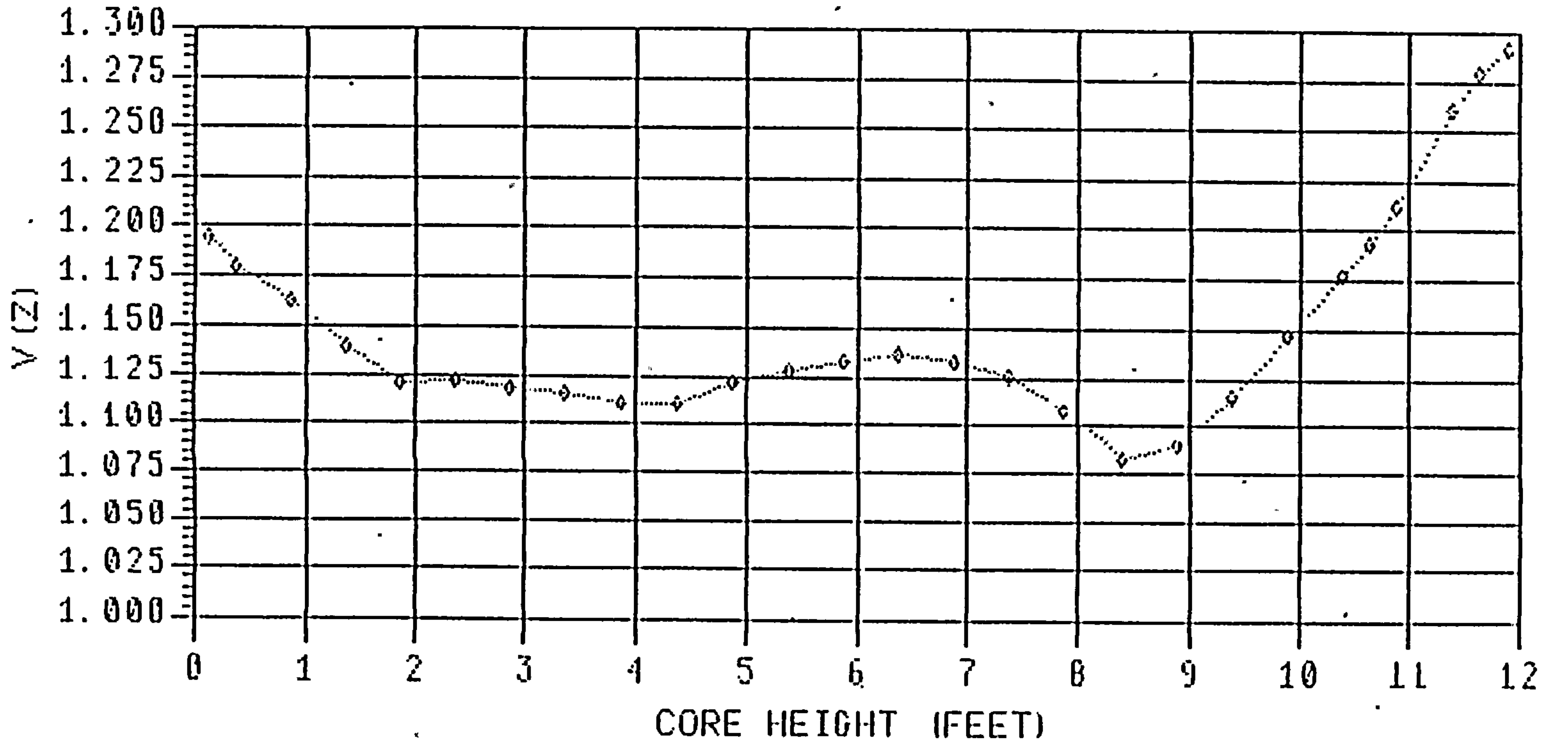
Figure 3.3.1- 5

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

Figure 3.3.1-6 Cycle dependent V(Z) Function
D.C. Cook Unit 1, Cycle 8, DeltaI=±5



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3.4 THERMAL AND HYDRAULIC DESIGN

This section describes the thermal and hydraulic design of the Unit 1 core with Westinghouse (W) fuel. The Exxon Nuclear Company thermal/hydraulic design for their fuel is discussed in Section 3.5. The thermal/hydraulic design of the Cycle 8 transition core which contains W and ENC fuel is discussed in Section 3.6.

3.4.1 Thermal and Hydraulic Evaluation for the Initial Core

Thermal and Hydraulic Characteristics of the Design

Thermal Data

Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains the maximum cladding surface temperature below about 657°F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap[1][2] and may be calculated by the following equation:

$$h = 0.6\ P + \frac{k}{f(14.4\times 10^{-6})}$$

where:

h is conductance in Btu/hr-ft^2-°F
P is contact pressure in psi
k is the thermal conductivity of the gas mixture in the rod
f is the correction factor for the accommodation coefficient

The thermal-hydraulic design assures that the temperature of the center of the hottest fuel pellet is below the melting point of the UO_2. (Melting point of 5080°F[7] unirradiated and reducing by 58°F per 10,000 MWD/MTU.) The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. The pellet surface temperature is governed by the cladding temperature and the thermal conductance of the fuel pellet-cladding gap.

The thermal conductivity of uranium dioxide was evaluated from data reported by Howard, et al.[21]; Lucks, et al.[22]; Daniel, et al. [23]; Feith[24]; Vogt, et al.[25]; Nishijima, et al.[26]; Wheeler, et al.[27]; Godfrey, et al[3]; Stora, et al.[28]; Bush[29]; Asamoto, et al.[30]; Kruger[31]; and Gyllander[32]. An examination of the UO_2 thermal conductivity data, Figure 3.4.1-1 shows that at temperatures between 0°C and ~1600°C there is little variation in the data, while above 1600°C the scatter increases considerably.

At the higher temperatures, thermal conductivity is best obtained utilizing the integral conductivity to melt, which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate for the value of $\int_0^{2800°C} kdT$ is 93 watts/cm. This conclusion is based on the integral values reported by Gyllander[32], Lyons, et al.[33], Coplin, et al.[34], Duncan[5], Bain[35], and Stora[36].

The design curve for the thermal conductivity is shown in Figure 3.4.1-1. The section of the curve at temperatures between 0°C and 1300°C is in excellent agreement with the recommendation of the IAEA panel[37]. The section of the curve above 1300°C is normalized to an integral value of 93 watts/cm[5, 32-36] from 0 to 2800°C.

In the THINC analysis, the benefit of coolant mixing in all the subchannels in the hot assembly is considered and a mixing factor of approximately 0.90 is used to evaluate the enthalpy rise to the point of minimum DNB ratio.

The above subfactors are combined to obtain the total engineering hot channel factor for an enthalpy rise of 1.01. The reduction in this subfactor at nominal operating conditions from a value of 1.075 (PSAR) was the result of the adaptation of the THINC code (multi-subchannel analyses) as a thermal and hydraulic design method. Table 3.4.1-2 is a tabulation of the design engineering hot channel factors.

Operational Limits:

The above subfactors are incoporated in THINC steady-state and transient analyses to yield operating limits for the maximum measured value of the enthalpy rise hot channel factor, $F_{\Delta H}^{N}$. For the D. C. Cook Plant Unit 1, the technical specification limit (for Cycle 7) is:

$$F_{\Delta H}^{N} = 1.51 \, [1 + 0.2 \, (1-P)]^{*} \qquad (2a)$$

where P is the ratio of operating power to rated power. The engineering subfactor $F_{\Delta H}^{E}$ is incorporated into the limiting value of 1.51*, and

$$F_{\Delta H}^{N} = 1.04 \, F_{\Delta H}^{N'} \qquad (2b)$$

where, $F_{\Delta H}^{N'}$ is the measured nuclear enthalpy rise peaking factor, and the factor of 1.04 accounts for measurement uncertainty.

* For Cycle 8, the $F_{\Delta H}^{N}$ limiting values for Westinghouse and ENC fuel at 3250 MWt rated power have been changed to the values stated in Tables 3.3.1-1 and 3.6.3-1.

The heat flux engineering subfactor of 1.03 is included in the maximum measured value of the heat flux hot channel factor,

$$F_Q^N = 1.03 \times 1.05\ F_Q^{N'} \qquad (2c)$$

where $F_Q^{N'}$ is the measured nuclear hot channel factor, and the factor of 1.05 accounts for measurement uncertainties. For Cycle 8 operations, the technical specifications require that F_Q^N not exceed the limits defined in Section 3.2.2 of the technical specifications. (See Section 3.6.2.2).

Pressure Drop and Hydraulic Forces

The total loss across the reactor vessel, including the inlet and outlet nozzles, and the pressure drop across the core are listed in Table 3.4.1-1. These values include a 10% uncertainty factor.

Thermal and Hydraulic Design Parameters

The thermal and hydraulic design parameters are given in Table 3.4.1-1.

Thermal and Hydraulic Evaluation

W-3 Equivalent Uniform Flux DNB Correlation

The equivalent uniform DNB flux $q''_{DNB,EU}$ is calculated from the W-3 equivalent uniform flux DNB correlation as follows:

$$\frac{q''_{DNB,EU}}{10^6} = [(2.022 - 0.0004302p) + (0.1722 - 0.0000984p)e^{(18.177 - 0.004129p)\chi}]$$

$$\times [1.037 + \frac{G}{10^6}(0.1484 - 1.596\chi + 0.1729\chi|\chi|)] \times [1.157 - 0.869\chi]$$

$$\times [0.2664 + 0.8357e^{-3.151D_e}] \times [0.8258 + 0.000794(H_{sat} - H_{in})] \qquad (3)$$

TABLE 3.4.1-1
THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Cycle 1 Design Parameters**
Total Heat Output, MWt	3250
Total Heat Output, Btu/Hr	$11{,}090 \times 10^6$
Heat Generated in Fuel, %	97.4
Maximum Thermal Overpower, %	112
Nominal System Pressure, psia	2250
Hot Channel Factors	
Heat Flux	
Nuclear, F_q	2.60
Engineering, F_q^E	1.03
Total	2.80
Enthalpy Rise	
Nuclear, $F_{\Delta H}^N$	1.55
Coolant Flow	
Total Flow Rate, lbs/hr	135.6×10^6
Average Velocity along Fuel Rods, ft/sec	15.5
Average Mass Velocity, lb/hr-ft^2	2.53×10^6
Coolant Temperature, °F	
Design Nominal Inlet	536.3*
Average Rise in Vessel	63.0
Average Rise in Core	65.7
Average in Core	570.3
Average in Vessel	567.8
Nominal Outlet of Hot Channel	667.5
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	52,200
Average Heat Flux, Btu/hr-ft^2	207,000
Maximum Heat Flux, Btu/hr-ft^2	579,600
Maximum Thermal Output, kw/ft	18.8
Maximum Clad Surface Temperature BOL at Nominal Pressure, °F	657
Maximum Average Clad Temperature BOL at Rated Power, °F	720
Fuel Central Temperatures (Region 3-BOL) for nominal fuel rod dimensions, °F	
Maximum at 100% Power	4250
Maximum at 112% Power	4500

* Best Estimate Nominal Inlet Temperature is 533.0°F
** See Table 3.6.3-1 for Cycle 8 design and operating values

TABLE 3.4.1-1 (cont'd.)

	Design Parameters	Current Operating Limits
DNB Ratio		
Minimum DNB Ratio at nominal operating conditions	1.97	1.30
Pressure Drop, psi		
Across Core	32	
Across Vessel, including nozzles	51	

3.5 EXXON FUEL DESIGN

The Exxon Nuclear Company (ENC) reload fuel assemblies described in this section were used in Donald C. Cook Nuclear Plant Unit 1, Cycles 2-7 and are currently being used in Cycle 8 operation (Section 3.6). Batch 4 was loaded into the core for Cycle 2, Batch 5 for Cycle 3, etc., and Batch 9 for Cycle 7. The mechanical and thermal/hydraulic designs of Batches 4 through 9 are identical, and the nuclear design is also identical except for small enrichment variations.

The design of the ENC Batch 4 fuel assemblies was identical to the design of the H. B. Robinson (HBR) Region 7 fuel assemblies, except for small differences in enrichment and holddown spring characteristics. The HBR Region 7 fuel assemblies have been approved by the NRC for operation at 2300 MWt and were loaded into the reactor core in October 1975. The Donald C. Cook Nuclear Plant Unit 1, Batch 4 fuel described in this section differs only slightly in design from the Donald C. Cook Nuclear Plant Unit 1, Batches 2 and 3 fuel assemblies. All dimensions affecting mechanical interfacing with control rods and core support structure and with Batches 2 and 3 fuel assemblies were maintained identical for Batch 4 fuel assemblies. The most significant changes include thicker cladding on the fuel rods, shorter pellets, and use of bi-metallic grid spacers. This fuel design utilized the NRC approved ENC fuel densification model for PWR's.

3.5.1 Fuel and Mechanical Design

This section describes the mechanical, chemical and thermal design for the Donald C. Cook Nuclear Plant Unit 1, Batch 4 reload fuel under normal operating conditions.

Exxon Nuclear's design configuration for the Donald C. Cook Nuclear Plant Unit 1 reload assemblies is compatible with Westinghouse fuel and Westinghouse designed reactor internals and consists of a 15 x 15 square array of 225 positions, occupied by 204 fuel rods, 20 Zircaloy-4 guide tubes and one Zircaloy-4 instrumentation tube.

The fuel consists of pressed and sintered UO_2 pellets. The nominal pellet density is 94.0% of the theoretical density, each pellet is dished on each end, and the fuel active length is nominally 144 inches. Zircaloy-4 end caps are seal welded to the Zircaloy-4 cladding. Fuel rod pitch is maintained by seven bi-metallic grid spacers constructed of Zircaloy-4 structural members with Inconel springs. The grids are equally spaced along the length of the fuel bundle and are welded to the guide tubes. The Zircaloy-4 guide tubes are mechanically attached and secured to the upper and lower tie plates. The spacers, guide tubes and tie plates form the structural skeleton of the fuel bundle. The upper tie plate is designed to be mechanically dismountable by remote handling under water.

Description of the Donald C. Cook Nuclear Plant Unit 1 Batch 4 assembly components, including purpose and rationale, is given in Table 3.5.1-1. As stated before, the values and descriptions in this table are also valid for Batches 5-9.

Mechanical Design

Fuel Assembly

Design Basis

The fuel assembly shall be dimensionally and hydraulically compatible with existing fuel and dimensionally compatible with reactor fuel handling equipment.

3.6 Westinghouse OFA Reload

For Cycle 8 operation (startup November 1983), the Donald C. Cook Nuclear Plant Unit 1 was refueled with Westinghouse (W) fuel assemblies of the 15x15 Optimized Fuel Assembly (OFA) design. This chapter evaluates the mechanical, nuclear, and thermal hydraulic design of the OFAs and justifies their compatibility with the Exxon Nuclear Company (ENC) fuel assemblies remaining in the Cycle 8 reactor core. NRC approval[1] was obtained for the mixed OFA and ENC fuel assembly core for Cycle 8 operation at 3250 MWt rated power.

The design of the W OFA (15x15 fuel rod array) is similar to the W 15x15 LOPAR (low parasitic) fuel which was used in the Cook Unit 1, Cycle 1 core. W LOPAR fuel also has had substantial operating performance in a number of nuclear plants[2]. The major difference introduced by the W 15x15 OFA design is the use of five intermediate Zircaloy grids, replacing five intermediate Inconel grids for the LOPAR fuel. The 15x15 Zircaloy grid design is similar to the W 17x17 OFA grid design. The W 17x17 OFA design has been generically approved by the NRC via their review of the W 17x17 OFA Reference Core Report.[3] Operating experience has been obtained for six demonstration 17x17 OFAs which contain Zircaloy intermediate grids.[2] Two assemblies have satisfactorily completed three cycles of irradiation to about 28,000 MWD/MTU burnup, two have completed two cycles to about 19,400 MWD/MTU, and two have completed one cycle in excess of 9,000 MWD/MTU. The demonstration OFAs have been examined and provide reason to expect good performance from the 15x15 OFA design.

Although Cook Unit 1 is licensed for a maximum power level of 3250 MWt, the thermal-hydraulic design summarized in this chapter and accident analyses (except large break LOCA) in Chapter 14 were performed at a reactor power level of 3411 MWt. These conservative design and safety analyses provide an early identification of those design/safety limits for a potential uprating. The nuclear design for Cycle 8 was performed at the approved maximum 3250 MWt.

All analyses were performed utilizing W standard methods, which are described in the W Reload Safety Evaluation Methodology Topical.[4] The approved Westinghouse Improved Thermal Design Procedure (ITDP) is used in the DNB

analyses of both W and ENC fuel. The W WRB-1 correlation is used in the OFA DNB analyses. Both the ITDP and WRB-1 correlation were previously used to license D. C. Cook Unit 2 operation. The ENC fuel is analyzed using the W-3 DNB correlation. Another feature being introduced with the Cycle 8 reload includes the Westinghouse Wet Annular Burnable Absorber (WABA) rods, which received NRC approval[1,5]. A WABA description and evaluation is summarized in Section 3.6.1.4.

3.6.1 Fuel Mechanical Design

Each W OFA consists of 204 fuel rods, 20 guide thimble tubes, and 1 instrumentation thimble tube arranged within a supporting structure. The instrumentation thimble is located in the center position and provides a channel for insertion of an incore neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a neutron source assembly, a WABA assembly, or a thimble plug assembly, depending on the position of the particular fuel assembly in the core. The fuel rod pitch is maintained by two Inconel end grids and five Zircaloy-4 intermediate grids. The Zircaloy-4 guide tubes are mechanically attached to the OFA top and bottom nozzles. The guide tubes, nozzles and grids form the structural skeleton of the fuel bundle. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzle. Figure 3.6.1-1 shows an OFA fuel length schematic view, and Table 3.6.1-1 shows OFA design values and compares them with the ENC fuel assembly design.

Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate, which is fitted with alignment pins to locate and orient the assembly. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against the holddown springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

The 15x15 OFA design meets the same basic mechanical design requirements and criteria stated for the 17x17 OFA in WCAP-9500-A[6]. Design values for the properties of materials which comprise the fuel rod, fuel assembly, and core components are given in Reference 7.

The description and design criteria of the ENC fuel assemblies for the Cycle 8 core are presented in Section 3.5.

3.6.1.1 Mechanical Compatibility of Fuel Assemblies

<u>Design Basis</u>

The OFA shall be dimensionally and hydraulically compatible with the existing ENC fuel and dimensionally compatible with other core components and fuel handling equipment.

<u>Evaluation</u>

Table 3.6.1-1 presents a comparison of the OFA and ENC fuel assemblies, and Figure 3.6.1-1 gives an illustrative dimensional comparison of the fuel assemblies. The 15x15 OFA has the same cross-sectional envelope as the 15x15 ENC fuel assembly. However, as shown in Figure 3.6.1-1, the OFA is slightly longer (55 mils). This change in overall assembly length is directly related to the increase in adapter plate thickness, which provides additional load bearing margin consistent with the 17x17 3-leaf nozzle design, while maintaining fuel rod to nozzle clearances. Mechanical interaction between fuel assemblies is confined to the grid location. As shown in Figure 3.6.1-1, the grid elevations of the 15x15 OFA match the 15x15 ENC fuel assembly, minimizing the effects of mechanical and hydraulic interaction between assemblies.

ENC, in establishing their assembly design, demonstrated compatibility with the <u>W</u> LOPAR assembly design which was the initial Cook Unit 1 fuel. <u>W</u> has designed the OFA and LOPAR assemblies to be compatible. Consequently, compatibility of OFA and ENC fuel is assured.

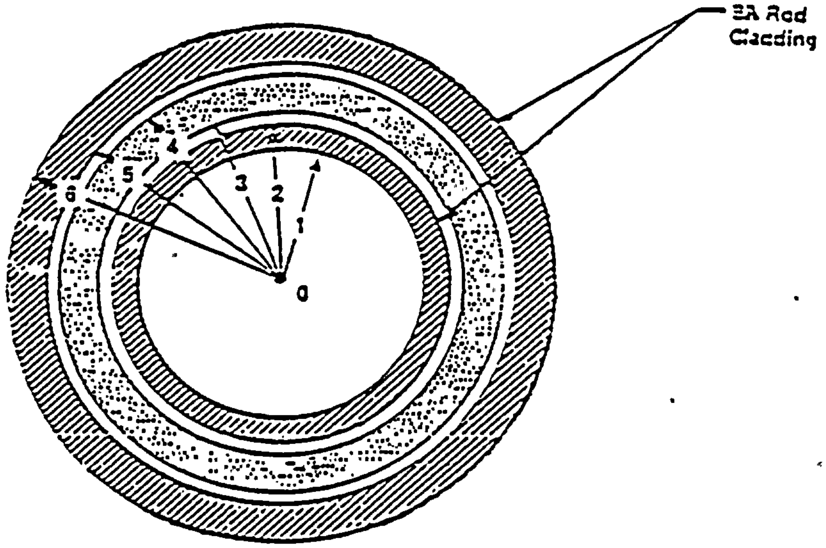
The OFA is designed to be compatible with existing fuel handling equipment. The OFA compatibility with other core components is shown, in Section 3.6.1.4, to be acceptable.

A core coolable geometry must be maintained during a seismic event. An analysis described in Section 3.5 demonstrates the adequacy of the ENC fuel assemblies when mixed with W 15x15 LOPAR fuel assemblies. This analysis bounds a mixed core consisting of W OFAs and ENC fuel assemblies, and, therefore, still shows that ENC assemblies maintain a coolable geometry. For W OFAs, a seismic and LOCA loads analysis for a mixed W and ENC fueled core showed that the OFAs maintain a coolable geometry. The NRC has also concluded that the structural integrity of W and ENC fuel assemblies is satisfied for all combinations of W and ENC assemblies in mixed cores.[1]

The analyses for the maximum OFA response during a seismic and LOCA accident are presented in Reference 8. The results of these analyses are summarized below:

a. OFA Grid Analysis During LOCA Accident

The fuel assembly response resulting from the most limiting main coolant pipe break (reactor vessel inlet) was analyzed using time history numerical techniques. The vessel motion for the LOCA accident produces substantial lateral loads in the reactor core. The maximum number of fifteen assemblies across the core diameter is factored into a reactor core finite element model which simulates fuel assembly interaction during lateral excitation. This model is consistent with the one described in Reference 9. Four fuel assembly reactor core patterns were selected to account for the reload transitions to an all-OFA core. The W/ENC fuel assembly relative locations for various core patterns are shown in Figure 3.6.1-2. These reactor core reload patterns are consistent with typical reload configurations.



Zone Number	Previous Design BA	WABA Design
0-1	Air	Water
1-2	Stainless steel	Zircaloy
2-3	Air	Helium
3-4	Borosilicate glass	$Al_2O_3 \cdot B_4C$
4-5	Air	Helium
5-6	Stainless steel	Zircaloy

Figure 3.6.1-9

Comparison of Borosilicate Glass
Absorber Rod with WABA Rod

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3.6.2 NUCLEAR DESIGN

The nuclear design of cores with W OFA and ENC fuel is accomplished by using the standard calculational methods as described in the W Reload Safety Evaluation Methodology(1). The dimensional and material differences between the W and ENC assemblies are small so that the W computer codes and methods are also valid for the ENC fuel. Dimensions and composition for each of the two fuel designs are used to establish the models. The burnup distribution of the ENC fuel assemblies remaining in Cycle 8 has been obtained by depleting the loading patterns from earlier cycles using two-dimensional and three-dimensional models of the applicable cores.

Changes in the nuclear characteristics during the transition cycles from an ENC fueled core to a W 15x15 OFA core are primarily due to fuel management considerations (number of feed assemblies, feed enrichment, cycle burnup, etc.) and not due to the differences in fuel assembly design. Each reload core design is evaluated to assure that design and safety limits for the OFA and ENC fuel are satisfied according to the W reload safety evaluation methodology. For the evaluation of the worst-case $F_Q(Z)$ envelope, axial power shapes are synthesized with the limiting F_{xy} values chosen over three overlapping burnup windows during the cycle.

In order to accommodate potential increases in future feed enrichments, a criticality analysis of the fuel storage areas was performed for nominal enrichments up to and including 4.00 wt.% U-235 in W 15x15 OFA fuel. These analyses confirm that all current safety criteria applicable to fuel storage are satisfied(2).

3.6.2.1 Computerized Methods, Codes and Cross Section Data

Westinghouse Electric Corporation's principal PWR neutronic design tools are the ARK code for generation of cross sections or the basic nuclear parameters, the depletable TURTLE code for computing reactivity and xy power distribution, PALADON for analyses requiring a three-dimensional simulation, and THURTLE and APOLLO for generation of axial powers shapes.

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APOLLO is used to calculate differential rod worth versus core height, axial nuclear hot channel factor versus control rod height, axial xenon oscillations and stability studies, and axial power distributions.

3.6.2.2 Unit 1 Cycle 8 Neutronic Design

3.6.2.2.1 Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described in References 3-7 with primary reliance upon the 3D PALADON code.

The burnup history of each of the exposed fuel assemblies was calculated by a three-dimensional, four node radially and 48 node axially per assembly, 3D-PALADON model which was utilized to simulate operation of the core for Cycles 5, 6 and 7.

Calculations for BOC 8 utilized the assembly exposures, calculated at an EOC 7 burnup of 10,446 MWD/MTU. The 3D PALADON model was verified using the 2D pin-by-pin TURTLE model. Axial effects in the 2D models are accounted for through the bucking term B_z^2.

3.6.2.2.2 Design Bases

The nuclear design bases for the Cycle 8 core are as follows:

1. The design shall permit operation within the Technical Specifications for the D.C. Cook Unit 1 nuclear plant.

2. The Cycle 8 loading pattern shall permit full power (3250 MWt total power) operation of the core throughout the Cycle 8 reactivity life time of about 15,750 MWD/MTU. Power distributions and control rod worth (both shutdown worth and the worth of a potentially ejected rod) are maintained within the ranges analyzed in the Cycle 8 safety analysis.

3. At hot full power (3250 MWt total power) the peak $F^N_{\Delta H}$ shall not exceed 1.435 in any single fuel rod throughout the cycle under nominal operating conditions.

4. The moderator temperature coefficient (MTC) is maintained less than or equal to +5 pcm/°F below 70% of rated power and less than or equal to 0 pcm/°F at or above 70% of rated power.

5. The worth of all rods minus the most reactive stuck rod shall exceed BOC and EOC shutdown requirements.

3.6.2.2.3 Design Description and Results

The Cycle 8 reactor core consists of a mixed W OFA/ENC fueled core of 193 assemblies, each having a 15x15 fuel rod array. A description of the W OFAs and ENC fuel assemblies are given in Sections 3.6.1 and 3.5.1 respectively.

The Cycle 8 loading pattern is given in Figure 3.6.2-1 which shows the region number, sources, and the burnable absorber configuration. The core consists of 44 fresh W OFAs with an average enrichment of 3.30 w/o U-235, 36 fresh OFAs with an average enrichment of 3.60 w/o and 113 exposed ENC assemblies. A low leakage loading pattern was developed which results in the scatter-loading of the OFAs throughout the interior of the core. WABA rods are inserted into a number of OFAs to control power peaking and MTC. The exposed ENC fuel is also scatter-loaded in the center in a manner to control the power peaking. The WABA rods contain 0.0153 gm/in of B-10, and 864 of these rods are distributed among 68 fresh assemblies loaded in the core interior. Pertinent fuel assembly parameters for the Cycle 8 fuel are given in Tables 3.6.1-1 and 3.6.2-1.

Physics Characteristics

The neutronics characteristics of the Cycle 8 core are compared with those Cycle 7 and are presented in Table 3.6.2-2. The reactivity coefficients of the Cycle 8 core are bounded by the coefficients used in the safety analysis.

The boron letdown curve for Cycle 8 is shown in Figure 3.6.2-2. The BOC 8 xenon free critical boron concentration is calculated to be 1,395 ppm. At 150 MWD/MTU and equilibrium xenon, the critrical boron concentration is 1,098 ppm. The Cycle 8 length is projected to be 15,750 MWD/MTU at a core power of 3,250 MWt with 10 ppm soluble boron remaining.

Power Distribution Considerations

Representative calculated power maps for Cycle 8 are shown in Figures 3.6.2-3 and 3.6.2-4 for BOC and EOC conditions, respectively. Steady state core axial power distributions are shown for BOC, MOC, and EOC in Figure 3.6.2-5. The axial power distributions as well as the radial power distributions are representative of the all rods out, equilibrium xenon configurations. A comparison between predicted power distribution (TURTLE) and measured power distribution (Flux Map 27) is shown in Figure 3.6.2-6. The Cycle 8 core loading satisfies the LOCA $F_Q(Z)XP$ envelope limits of $\leq$ 2.10 for W fuel and $\leq$ 2.04 for ENC fuel. The equations and curves which show the F_Q limits as a function of power, fuel height and fuel burnup are defined in Section 3.2.2 of the Cook Unit 1 Technical Specifications.

Control Rod Reactivity Requirements

Detailed calculations of shutdown margins for Cycle 8 are compared with Cycle 7 data in Table 3.6.2-3. The D. C. Cook Unit 1 Technical Specifications for Cycle 8 require a minimum required shutdown margin of 1,600 pcm at BOC and EOC. The Cycle 8 analysis indicates excess shutdown margin of 1,020 pcm at BOC and 880 pcm at EOC. The Cycle 7 analysis indicated an excess shutdown margin of 826 pcm at BOC and 1,044 pcm at EOC, for a minimum shutdown margin of 1750 pcm.

The control rod groups and insertion limits for Cycle 8 will remain unchanged from Cycle 7. With these limits the nominal worth of the control bank, D-Bank, inserted to the insertion limits at HFP is 165 pcm at BOC and 250 pcm at EOC. The control rod shutdown requirements in Table 3.6.2-3 allow for a HFP D-Bank insertion equivalent to 1,190 pcm and 500 pcm at BOC and EOC, respectively.

Moderator Temperature Coefficient

The Technical Specifications require that the moderator temperature coefficient be less than or equal to +5 pcm/$^{\circ}$F below 70% of rated power and less than or equal to 0 pcm/$^{\circ}$F at or above 70% power. The HZP, ARO moderator temperature coefficient is calculated to be +2.82 pcm/$^{\circ}$F and meets the Technical Specification limit below 70% power. The moderator temperature coefficient at or above 70% power is calculated to be less than 0 pcm/$^{\circ}$F and also meets the Technical Specification requirements.

REFERENCES, SECTION 3.6.2

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TABLE 3.6.2-1

FUEL ASSEMBLY DESIGN PARAMETERS
D.C. COOK UNIT 1 - CYCLE 8

Region	8	9	10A	10B
Fuel Type	ENC	ENC	W OFA	W OFA
Enrichment (w/o of U 235)	2.905	2.903	3.297	3.598
Density (percent theoretical)	94.0	94.0	95.143	95.295
Number of Assemblies	49	64	44	36
Region Fuel Loading (MTU)	20.93	27.32	20.23	16.58
Burnup (MWD/MTU)				
BOC 8	20,300	9,450	0	0
EOC 8 (Estimate)	32,600	25,600	19,000	15,500
Fueled Stack Height (inches, cold)	144	144	144	144

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Table 3.6.2-2 D.C. Cook Unit 1 Neutronics Characteristics of Cycle 8 Compared with Cycle 7 Data

	Cycle 7		Cycle 8	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, Equilibrium Xenon (ppm)	904	10	1098	10
HZP, ARO, No Xenon (ppm)	1296	---	1534	---
Moderator Temperature Coefficient				
HFP, ARO (pcm/°F)	-4.0	-25.2	-5.38	-25.65
HZP, ARO (pcm/°F)	+3.6	---	+2.82	---
Doppler Coefficient (pcm/°F)	-1.0	-2.0	-2.11	-2.23
Boron Worth, HZP (pcm/ppm)	-9.8	-10.8	-9.1	-11.4
Total Nuclear Peaking Factor				
F_Q^N, HFP, Equilibrium Xenon	1.655	1.444	1.596	1.526
Delayed Neutron Fraction	.0061	.0060	.0061	.0052
Control Rod Worth of All Rods In				
Minus Most Reactive Rod, HZP,(pcm)	5640	6404	6060	6520
Excess Shutdown Margin, (pcm)	826	1044	1020	830

Table 3.6.2-3 D.C. Cook Unit 1 Control Rod Shutdown Margins and Requirements of Cycle 8 Compared to Cycle 7

	Cycle 7		Cycle 8	
	BOC	EOC	BOC	EOC
<u>Control Rod Worth (HZP), pcm</u>				
All Rods Inserted (ARI)	---	---	7422	7849
ARI less most reactive (N-1)	5640	6404	6060	6520
N-1 less 10% allowance [(N-1)*.9)]	5076	5764	5450	5870
<u>Reactivity Insertion, pcm</u>				
Power Defect (Moderator + Doppler)	---	1820	1180	1870
Flux Redistribution	---	600	410	970
Void	---	50	50	50
Sum of the above three	---	2470	1640	2890
Rod insertion allowance	---	500	1190	500
Total Requirements	2500	2970	2830	3390
Shutdown Margin (N-1) * .9 - Total Requirements	2576	2794	2620	2480
Required Shutdown Margin	1750(a)	1750(a)	1600(b)	1600(b)
Excess Shutdown Margin	826	1044	1020	880

(a) Technical Specification Limit for Cycle 7
(b) Technical Specification Limit for Cycle 8

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	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					8	10B SS*	10B 4	10B	10B 4	10B	8				
2			8	9	10B 12	9	10A 12	8	10A 12	9	10B 12	9	8		
3		8	9	10B 12	9	10A 16	8	10A 16	8	10A 16	9	10B 12	9	8	
4		9	10B 12	9	9	9	10A 16	9	10A 16	9	9	9	10B 12	9	
5	8	10B 12	9	9	8	10A 16	8	9	8	10A 16	8	9	9	10B 12	8
6	10B	9	10A 16	9	10A 16	8	10A 12	8	10A 12	8	10A 16	9	10A 16	9	10B
7	10B 4	10A 12	8	10A 16	8	10A 12	9	9 SS	9	10A 12	8	10A 16	8	10A 12	10B 4
8	10B	8	10A 16	9	9	8	9	8	9	8	9	9	10A 16	8	10B
9	10B 4	10A 12	8	10A 16	8	10A 12	9	9	9	10A 12	8	10A 16	8	10A 12	10B 4
10	10B	9	10A 16	9	10A 16	8	10A 12	8	10A 12	8	10A 16	9	10A 16	9	10B
11	8	10B 12	9	9	8	10A 16	8	9 SS	8	10A 16	8	9	9	10B 12	8
12		9	10B 12	9	9	9	10A 16	9	10A 16	9	9	9	10B 12	9	
13		8	9	10B 12	9	10A 16	8	10A 16	8	10A 16	9	10B 12	9	8	
14			8	9	10B 12	9	10A 12	8	10A 12	9	10B 12	9	8		
15					8	10B	10B 4	10B	10B 4	10B SS*	8				

X - Region Number
YY - Number of Wet Annular Burnable Absorbers, or

- Source Rods (unirradiated source, SS; irradiated source, SS*)

Figure 3.6.2-1 D. C. Cook Unit 1, Cycle 8 Loading Pattern

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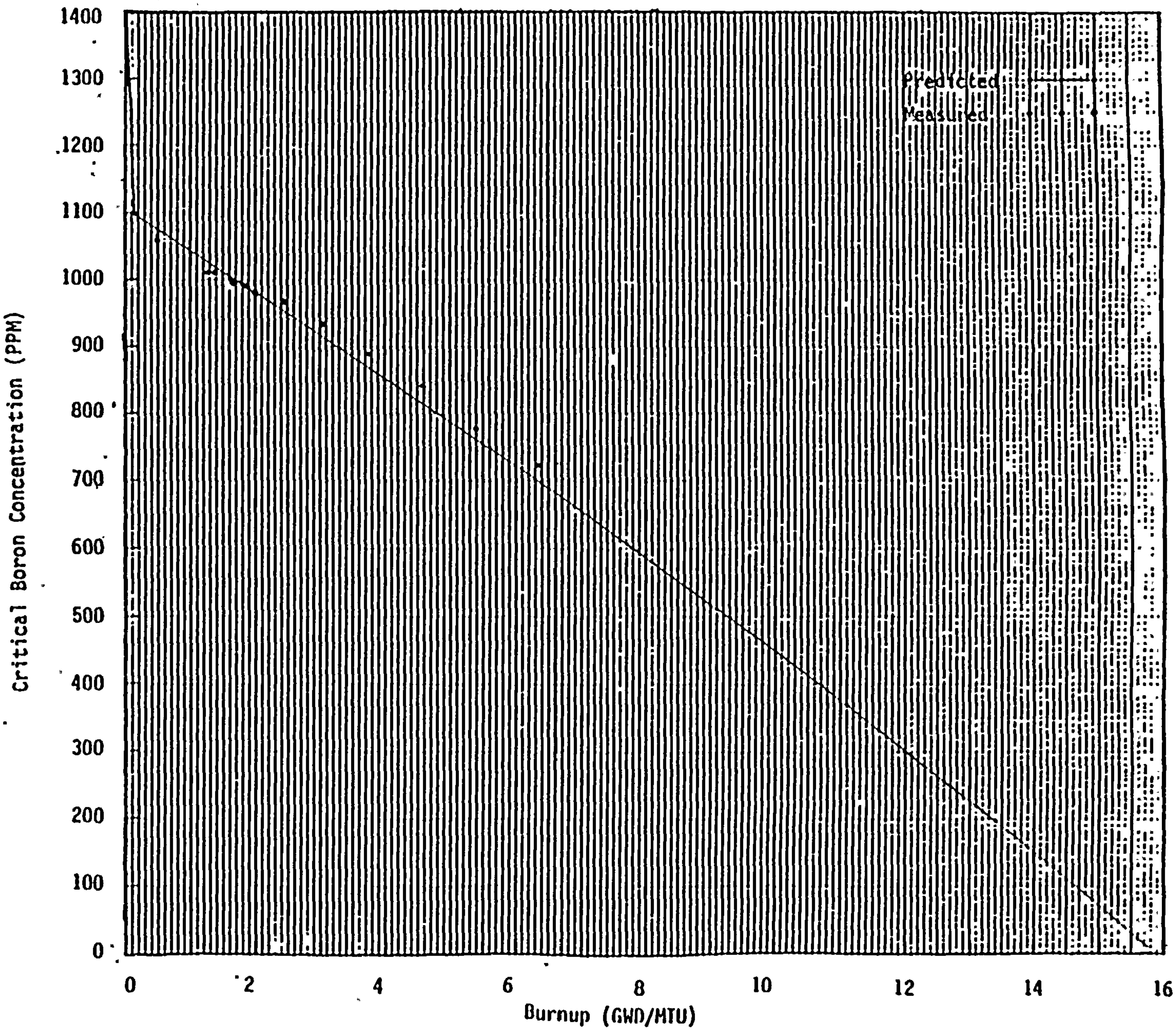

FIGURE 3.6.2-2 D.C. COOK UNIT 1, CYCLE 8, BORON LETDOWN CURVE

8 0.91 0.93					
9 1.11 1.19	9 1.13 1.22				
8 0.94 0.98	10A 1.20 1.31	8 0.96 1.02			
9 1.08 1.18	8 0.95 1.02	10A 1.13 1.25	8 0.97 1.07		
9 1.10 1.20	10A 1.14 1.23	9 1.09 1.18	9 1.23 1.33	9 1.16 1.24	
10A 1.12 1.21	8 0.94 1.03	10A 1.22 1.33	9 1.15 1.26	10B X 1.15 1.39	9 0.71 0.95
8 0.97 1.02	10A 1.22 1.34	9 1.16 1.25	10B 1.05 1.36	9 0.58 0.88	8 0.28 0.60
10B 1.05 1.33	10B 1.01 1.33	10B 0.95 1.33	8 0.40 0.79		1 AP MP X

AP Assembly Power
MP Maximum Power

1, 2, 3, ...In upper left corner indicates Region 1, 2, 3, ...

X in corner indicates assembly and quadrant of peak

REGION	POWER SHARING	ACCUMULATED BURNUP
8	0.75	20400
9	1.04	9600
10A	1.18	200
10B	1.04	200

Figure 3.6.2.3 D. C. Cook Unit 1, Cycle 8, Relative Power Distribution 150 MWD/MTU, 1098 ppm, 3250 MWt

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8 0.81 0.84					
9 0.37 1.04	9 1.01 1.11				
8 0.90 0.95	10A 1.24 1.31	8 0.98 1.03			
9 1.03 1.11	8 0.97 1.02	10A 1.27 1.33	8 0.93 0.98		
9 1.08 1.14	10A 1.29 1.35	9 1.05 1.11	9 1.06 1.11	9 1.02 1.08	
10A 1.28 1.35 X	8 0.97 1.04	10A 1.29 1.35	9 1.05 1.14	10B 1.20 1.31	9 0.79 0.96
8 0.94 1.02	10A 1.24 1.32	9 1.05 1.13	10B 1.12 1.31	9 0.67 0.90	8 0.37 0.69
10B 0.94 1.17	10B 0.84 1.20	10B 0.87 1.17	8 0.44 0.74		1 AP MP X

AP Assembly Power
MP Maximum Power

1, 2, 3, ... In upper left corner indicates Region 1, 2, 3, ...

X in corner indicates assembly and quadrant of peak

REGION	POWER SHARING	ACCUMULATED BURNUP
8	0.77	32600
9	0.98	25600
10A	1.27	19000
10B	1.02	15500

Figure 3.6.2-4 D. C. Cook Unit 1, Cycle 8, Relative Power Distribution, 15750 MWD/MTU, 10 ppm, 3250 MWt

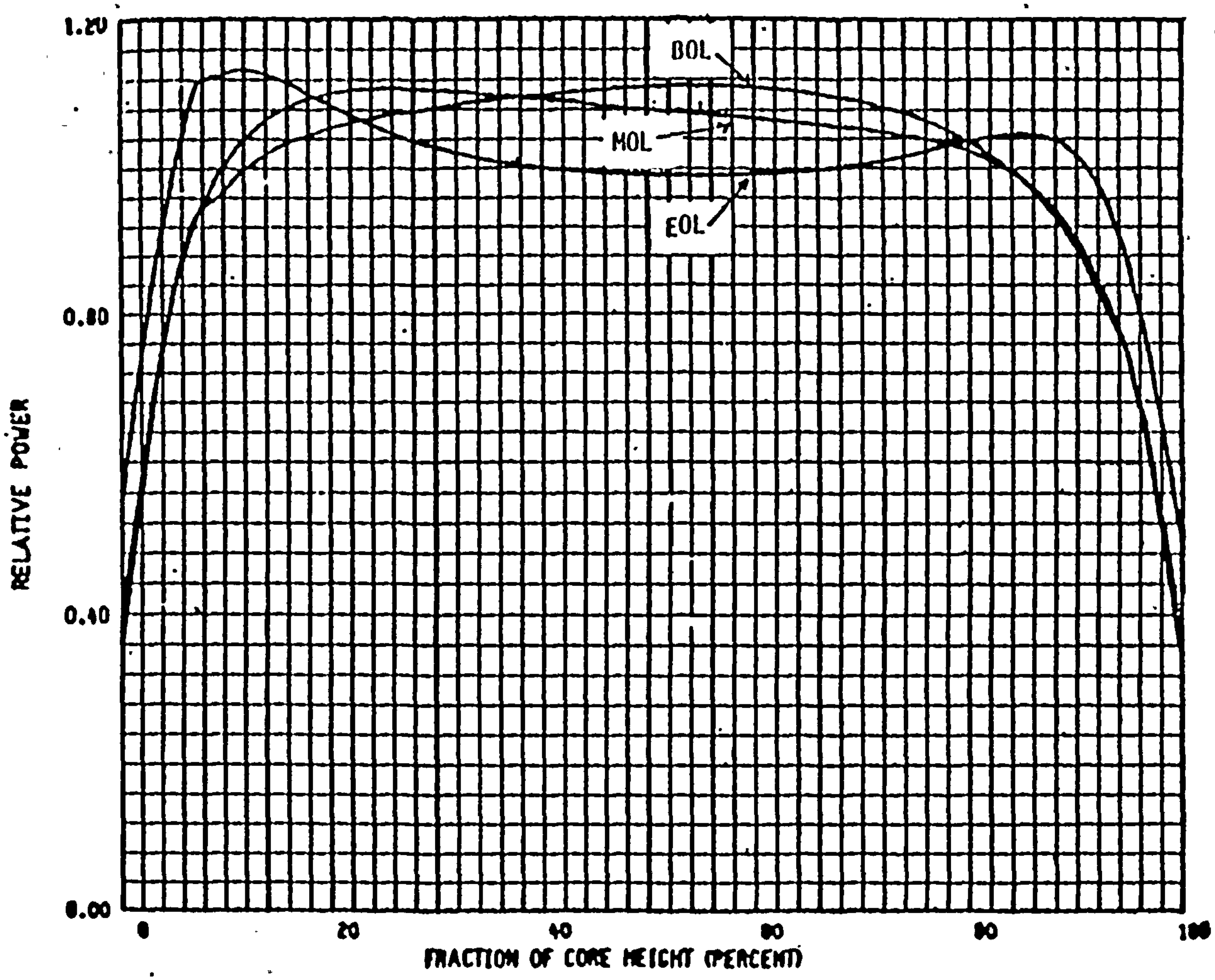

Figure 3.6.2-5 D. C. Cook Unit 1, Cycle 8, Relative Axial Power Distribution, ARO, HFP, Equilibrium Xenon

.858 .887 -3.3	1.056 1.086 -2.8	.914 .942 -3.0	1.068 1.086 -1.7	1.122 1.110 +1.1	1.183 1.158 +2.2	.994 .973 +2.2	1.057 1.037 +1.9
1.065 1.086 -1.9	1.083 1.117 -3.0	1.183 1.222 -3.2	.947 .964 -1.8	1.186 1.183 +0.3	.973 .958 +1.6	1.280 1.246 +2.7	1.025 1.006 +1.9
.929 .942 -1.4	1.205 1.222 -1.4	.931 .968 -3.8	1.120 1.158 -3.3	1.062 1.096 -3.1	1.223 1.249 -2.1	1.153 1.146 +0.6	.942 .934 +0.9
1.083 1.086 -0.3	.968 .964 +0.4	1.134 1.158 -2.1	.934 .969 -3.6	1.162 1.203 -3.4	1.102 1.136 -3.0	1.056 1.060 -0.4	.399 .401 -0.5
1.140 1.110 +2.7	1.211 1.183 +2.4	1.076 1.096 -1.8	1.153 1.203 -4.2	1.090 1.137 -4.1	1.101 1.147 -4.0	.590 .592 -0.3	
1.190 1.158 +2.8	.972 .958 +1.5	1.248 1.249 -0.1	1.112 1.136 -2.1	1.114 1.147 -2.9	.697 .718 -2.9	.277 .289 -4.2	
1.006 .973 +3.4	1.272 1.246 +2.1	1.141 1.146 -0.4	1.048 1.060 -1.1	.581 .592 -1.9	.284 .289 -1.7		
1.059 1.037 +2.1	1.015 1.006 +0.9	.932 .934 -0.2	.396 .401 -1.3				

MEAS.
PRED
% DIF.

$\left(\frac{M-P}{P}\right) \times 100$

Figure 3.6.2-6 D. C. Cook Unit 1, Cycle 8, Power Distribution Comparison of Map 27, 99.34% Power, Bank D at 222 Steps, 1699 MWD/MTU

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3.6.3 Thermal and Hydraulic Design

Introduction

This section describes the thermal and hydraulic design of the Donald C. Cook Unit 1 Cycle 8 transition core which contains both Westinghouse Optimized Fuel Assemblies (OFA) and Exxon Nuclear Company (ENC) fuel. The design bases, DNB correlations used in the design of the two fuel types, DNB performance when transitioning cores, and the hydraulic compatibility of Westinghouse OFA and ENC fuel are summarized below and discussed in the following sections.

The thermal hydraulic design of this core is conservatively analyzed at 3411 MWt core power with a 577.1°F vessel average temperature, even though the Cycle 8 core will continue to be limited to its current rated parameters of 3250 MWt core power and a 567.8°F vessel average temperature. The analyses employed the Improved Thermal Design Procedure[1] (ITDP) and THINC IV[2,3] computer code. The WRB-1[4] DNB correlation was used in the Westinghouse 15x15 OFA analyses, whereas the W-3 correlation was used to analyze the ENC fuel.

Summary

The design method employed to meet the DNB design basis is the ITDP[1]. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent probability that the minimum DNBR will be greater than or equal to the limit DNBR for the peak power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a design DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, the limit DNBR values are increased to values designated as the safety analysis limit DNBRs. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR values is not required to meet the design basis.

In this application, the WRB-1 DNB correlation[4] is employed in the thermal hydraulic design of the Westinghouse 15x15 OFA fuel. Due to an improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation compared to previous DNB correlations, a correlation limit DNBR of 1.17 is applicable. The W-3 DNBR correlation[5,6] was used in the design of the ENC fuel assembly. A W-3 correlation limit DNBR of 1.30 is applicable.

The table below indicates the relationships between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

	W 15x15 OFA		ENC 15x15	
	Typical	Thimble	Typical	Thimble
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit	1.32	1.31	1.58	1.50
Safety Analysis Limit	1.69	1.69	1.58	1.50

The margin to the safety analysis DNBR limit is more than sufficient to cover the maximum rod bow penalty at full flow conditions[7] and a 5 percent transition core penalty, both applied to the OFA only. The 5 percent transition penalty was determined by analyzing Westinghouse 15x15 OFA and ENC assembly loading patterns at various core conditions in the same manner as the Westinghouse 17x17 OFA/LOPAR fuel analysis which was reviewed and approved by the NRC[8]. The 5 percent transition penalty for OFA is due to the higher OFA mixing vane loss coefficient compared to that of the ENC fuel. This results in localized flow redistribution from the OFA to the ENC assembly near mixing vane grid positions. When the full transition is complete (all ENC assemblies removed from core), the transition core penalty will no longer apply to OFA assemblies.

TABLE 3.6.3-1

D.C.Cook Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters	Cycle 8 Operating Parameters Based Upon: All ENC Core All W Core	Cycle 8 Design Parameters (1) Based Upon: All ENC Core All W Core
Reactor Core Heat Output, MWt	3,250	3,411
Reactor Core Heat Output, 10^6 BTU/hr	11,092	11,642
Heat Generated in Fuel, %	----	97.5% (ENC) 97.4% (W)
System Pressure, Nominal, psia	2,250	2,280
System Pressure, Minimum Steady-State, psia	2,220	2,250
Minimum DNBR at Nominal Conditions		
Typical Flow Channel		2.64 (ENC) 2.47 (W)
Thimble (Cold Wall) Flow Channel		2.18 (ENC) 2.33 (W)
Design DNBR for Design Transients		
Typical Flow Channel		$\geq$ 1.58 (ENC) $\geq$ 1.69 (W)
Thimble Flow Channel		$\geq$ 1.50 (ENC) $\geq$ 1.69 (W)
DNB Correlation		W-3 (ENC) WRB-1 (W)

(1) Based Upon Improved Thermal Design Procedure (ITDP)

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TABLE 3.6.3-1 (Continued)

D.C.Cook Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters	Cycle 8 Operating Parameters Based Upon: All ENC Core All W Core	Cycle 8 [1] Design Parameters Based Upon: All ENC Core All W Core
Coolant Conditions		
Minimum Measured Flow, 10^3 gpm	-	366.4
Effective Flow Area for Heat Transfer, ft^2	50.9 ENC 51.5 (W)	50.9 (ENC) 51.5 (W)
Average Velocity Along Fuel Rods, ft/sec	-	16.4 (ENC) 16.2 (W)
Average Mass Velocity, 10^6 $lb_m/hr\text{-}ft^2$	-	2.64 (ENC) 2.61 (W)
Nominal Vessel/Core Inlet Temperature, °F	-	545.5
Vessel Average Temperature, °F	567.8	577.1
Core Average Temperature, °F	-	579.5
Vessel Outlet Temperature, °F	-	608.6
Average Temperature Rise in Vessel, °F	-	63.1
Average Temperature Rise in Core, °F	-	64.9
Average Enthalpy Rise in Core, BTU/lb_m	-	86.57
Film Coefficient at Average Conditions, BTU/hr-ft^2-°F.	-	6215 (ENC) 6131 (W)
Average Film Temperature Difference, °F	-	34.9 (ENC) 35.5 (W)

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TABLE 3.6.3-1 (Continued)

D.C.Cook Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters	Cycle 8 Operating Parameters Based Upon: All ENC Core All W Core	Cycle 8 (1) Design Parameters Based Upon: All ENC Core All W Core
Heat Transfer		
Active Heat Transfer, Surface Area, ft^2	52,400 (ENC) 52,100 (W)	52,400 (ENC) 52,100 (W)
Average Heat Flux, BTU/hr-ft^2	206,200 (ENC) 207,400 (W)	216,400 (ENC) 217,700 (W)
Maximum Heat Flux for Normal Operation, BTU/hr-ft^2 (2)	478,400 (ENC) 481,200 (W)	502,000 (ENC) 505,100 (W)
Average Linear Power, KW/ft	6.70	7.03
Peak Linear Power for Normal Operation, kW/ft (2)	15.54	16.31
Maximum Clad Surface Temperature, °F		662.3 (ENC) 662.4 (W)
Fuel Centerline Temperature		
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F		4700 (W)

(2) Based Upon 2.32 F_Q Peaking Factor

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TABLE 3.6.3-1 (Continued)

D.C.Cook Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters	Cycle 8 Operating Parameters Based Upon: All ENC Core All W Core	Cycle 8 [1] Design Parameters Based Upon: All ENC Core All W Core
Calculational Factors		
Engineering Heat Flux Factor	-	1.000 (ENC) 1.000 (W)
Fuel Densification Factor (axial)	-	1.002
Radial Peaking Factor		
Design Nuclear Enthalpy Rise Hot Channel Factor	-	1.45 [3] (ENC) 1.49 [3] (W)
Pressure Drop		
Across Core, psi (Best Estimate Flow)	-	27.1 (ENC) 27.2 (W)

(3) Does Not Include Measurement Uncertainty

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3.0 REACTOR

3.1 SUMMARY DESCRIPTION*

This chapter describes 1) the mechanical components of the reactor and reactor core including the fuel rods and fuel assemblies, reactor internals, and the control rod drive mechanism (CRDM's), 2) the nuclear design, and 3) the thermal-hydraulic design.

Information in this Chapter 3 is principally taken from the original FSAR and documentation issued as formal amendments to the original FSAR. For information pertaining to this chapter which was not previously transmitted by formal amendment, the chapter contains brief reference to the change and then the reader is referred to the appropriate document containing the information. For example, most of the nuclear design type information is based on Cycle 1 for Donald C. Cook Nuclear Plant, Unit 2. For the information concerning the nuclear design and core management changes referring to Cycle 3, the reader is referred to WCAP-9828 (Reference 33 of Section 3.3). The vessel average temperature was increased from 572.2°F to 573.8°F as per Unit 2 license amendment 19 dated may 1980.

The information contained in this chapter is principally concerned with the nuclear fuel and reactor internals design. It does not necessarily reflect the same information as that used in the safety analysis. For information concerning safety analysis, Chapter 14 should be consulted.

The reactor core is comprised of an array of fuel assemblies which are identical in mechanical design, but different in fuel enrichment. The design of the initial core employs three enrichments in a three-region core, whereas different enrichments may be employed for a particular refueling scheme.

*When fuel was entirely supplied by Westinghouse Electric Co. Section 3.5 describes the fuel supplied by Exxon Nuclear Co., the first batch (72 assemblies) of which was inserted in Cycle 4.

The core is cooled and moderated by light water at a pressure of 2250 psia in the Reactor Coolant System. The moderator coolant contains boron as a neutron poison. The concentration of boron in the coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, is employed in the first core to establish the desired initial reactivity.

Two hundred and sixty four fuel rods are mechanically joined in a square array to form a fuel assembly (see Figure 3.2-1). The fuel rods are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. The grid assembly consists of an "egg-crate" arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support as well as coolant mixing vanes. The fuel rods consist of slightly enriched uranium dioxide ceramic cylindrical pellets contained in slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses, strains, and to increase fatigue life.

The center position in the assembly is reserved for the incore instrumentation, while the remaining 24 positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. Depending upon the position of the assembly in the core, the guide thimbles are used as core locations for rod cluster control assemblies (RCCA's), neutron source assemblies, and burnable poison assemblies. Otherwise, the guide thimbles are fitted with plugging devices to limit bypass flow.

The bottom nozzle is a box-like structure which serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly.

The top nozzle assembly functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the RCCA or other components.

The RCCA's each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies contain full length absorber material to control the reactivity of the core under operating conditions.

THE CRDM's for the full length RCCA's are of the magnetic jack type. Control rods are positioned by electro-mechanical (solenoid) action utilizing gripper latches, which engage grooved drive rods which in turn are coupled to the RCCA's. The CRDM's for the full length rods are so designed that upon a loss of electrical power to the coils, the RCCA is released and falls by gravity to shutdown the reactor.

The CRDM's for the part length control rods are of a roller nut type mechanism which are inactive since the part length control rods have been removed.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDM's, direct coolant flow past the fuel elements and to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation.

The nuclear design analyses and evaluation establish physical locations for control rods and burnable poison and physical parameters such as fuel enrichments and boron concentration in the coolant. The nuclear design evaluation established that the reactor core has inherent

characteristics which, together with corrective actions of the reactor control and protective systems, provide adequate reactivity control even if the highest reactivity worth RCCA is stuck in the fully withdrawn position.

The design also provides for inherent stability against diametral and azimuthal power oscillations and for control of induced axial power oscillations through the use of the control rods.

The thermal-hydraulic design analyses and evaluation establish coolant flow parameters which assure that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design induces additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies.

Instrumentation is provided in and out of the core to monitor the nuclear, thermal-hydraulic and mechanical performance of the reactor and to provide inputs to automatic control functions.

Table 3.1-1 presents a comparison of the principal nuclear, thermal-hydraulic and mechanical design parameters between D. C. Cook Unit 2 and the Trojan Unit (Docket Number 50-344) first core.

The effects of fuel densification were evaluated with the methods described in Reference (1).

The analysis techniques employed in the core design for Cycles 1, 2 and 3 are tabulated in Table 3.1-2. The loading conditions considered in general for the core internals and components are tabulated in Table 3.1-3. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel

TABLE 3.1-2

ANALYTIC TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Section Referenced
Mechanical Design of Core Internals			
Loads, Deflections, and Stress Analysis	Static and Dynamic Modeling	Blowdown code, FORCE, Finite element structural analysis code, and others	14.3.3
Fuel Rod Design			
Fuel Performance Characteristics (temperature, internal pressure, clad stress, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel density changes, heat Transfer, fission gas release, etc.	Westinghouse fuel rod design model	3.2.1.3.1 3.3.3.1 3.4.2.2 3.4.3.4.2
Nuclear Design			
1. Cross Sections and Group Constants	Microscopic data	Modified ENDF/B library	3.3.3.2
	Macroscopic constants for homogenized core regions	LEOPARD/CINDER type	3.3.3.2

TABLE 3.1-2 (Continued)

ANALYTIC TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Section Referenced
Nuclear Design (Continued)			
	Group constants for control rods with self-shielding	HAMMER-AIM	3.3.3.2
2. X-Y Power Distributions, Fuel Depletion, Critical Boron Concentrations, X-Y Xenon Distributions, Reactivity Coefficients	2-D, 2-Group Diffusion Theory	TURTLE	3.3.3.3
3. Axial Power Distributions, Control Rod Worths, and Axial Xenon Distribution	1-D, 2-Group Diffusion Theory	PANDA	3.3.3.3
4. Fuel Rod Power	Integral Transport Theory	LASER	3.3.3.1
Effective Resonance Temperature	Monte Carlo Weighting Function	REPAD	

1. A rotating bend fatigue experiment on unirradiated Zircaloy-4 specimens at room temperature and at 725°F. Both hydrided and non-hydrided Zircaloy-4 cladding were tested.

2. A biaxial fatigue experiment in gas autoclave on unirradiated Zircaloy-4 cladding both hydrided and non-hydrided.

3. A fatigue test program on irradiated cladding from the CVS and Yankee Core V conducted at Battelle Memorial Institute.

The results of these test programs provided information on different cladding conditions including the effects of irradiation, of hydrogen level, and of temperature.

The Westinghouse design equations followed the concept for the fatigue design criterion according to the ASME Code, Section III. Namely,

1. The calculated pseudo-stress amplitude (S_a) has to be multiplied by a factor of 2 in order to obtain the allowable number of cycles (N_f).

2. The allowable cycles for a given S_a is 5 percent of N_f, or a safety factor of 20 on cycles.

The lesser of the two allowable number of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{fk}} \leq 1$$

Where: n_k = number of diurnal cycles of mode k.

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor which is subjected to daily load follow is the failure of the clad by low cycle strain fatigue. During their normal residence time in the reactor, the fuel rods may be subjected to ~1000 cycles with typical changes in power level from 50 to 100 percent of their steady-state values.

The assessment of the fatigue life of the fuel rod clad is subjected to a considerable uncertainty due to the difficulty of evaluating, the strain range which results from the cyclic interaction of the fuel pellets and clad. This difficulty arises for example from such highly unpredictable phenomena as pellet cracking, fragmentation, and relocation. Nevertheless, since early 1968, Westinghouse has been investigating this particular phenomenon both analytically and experimentally. Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 claddings were performed which permitted a definition of a conservative fatigue life limit and recommendation of a methodology to treat the strain fatigue evaluation of the Westinghouse reference fuel rod designs.

However, Westinghouse is convinced that the final proof of the adequacy of a given fuel rod design to meet the load follow requirements can only come from incore experiments performed on actual reactors. The Westinghouse experience in load follow operation dates back to early 1970 with the load follow operation of the Saxton reactor. More recently, successful load follow operation has been performed on reactor A (300 load follow cycles) and reactor B (150 load follow cycles). In both cases, there was no significant coolant activity increase that could be associated with the load follow mode of operation.

The potential effects of operation with waterlogged fuel are discussed in Section 3.4.3.6. Waterlogging is not considered to be a concern during operational transients.

3. Rod Inspection

Fuel rod, control rodlet, burnable poison and source rod inspection consists of the following nondestructive examination, techniques and methods, as applicable.

a. Leak Testing

Each rod is tested using a calibrated mass spectrometer with helium being the detectable gas.

b. Enclosure Welds

All weld enclosures are X-rayed using weld correction forms. X-rays are taken in accord and with Westinghouse specifications which meet the requirements of ASTM E-142.

c. Dimensional

All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber and visual appearance.

d. Plenum Dimensions

100 percent of the fuel rods are inspected by fluoroscope, X-ray or other approved methods as discussed in Section 3.2.1.4.3 to ensure proper plenum dimensions.

e. Pellet-to-Pellet Gaps

100 percent of the fuel rods are inspected by fluoroscope, gamma scanning or other methods as discussed in Section 3.2.1.4.3 to ensure that no significant gaps exist between pellets.

f. Enrichment

100 percent of the fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g. Traceability

Traceability of rods and associated rod components is established by Quality Control.

4. Assemblies

Each fuel, control rod, burnable poison and source rod assembly is inspected for drawing and/or specification requirements.

5. Other Inspections

The following inspections are performed as part of the routine inspection operation:

a. Tool and gage inspection and control including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and condition of tools.

b. Audits are performed of inspection activities and records to assure that prescribed methods are followed and that records are correct and properly maintained.

c. Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper internals assembly, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. Four slots are milled into the core plate at the same positions. As the upper internals structure is lowered into the barrel, the slots in the upper plate engage the flat-sided pins in the axial direction. Lateral displacement of the core plate is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper internals assembly is lowered into place. Proper alignment of the lower core support structure, the upper internals assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance features. The upper internals assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper support flange and is compressed when the reactor vessel head is installed on the pressure vessel.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support flange and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support plate and upper core plate. The upper support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom penetration tubes (see Figure 3.2-8) which admit the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal table) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal table. During normal operation, the retractable thimbles are stationary and are moved only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal table is cleared for the retraction operation.

TABLE 4.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40
Number of heat transfer loops	4
Design pressure, psig	2485
Nominal operating pressure, psig	2235
Total system volume including pressurizer and surge line (ambient conditions), ft^3 (estimated)	12,500
System liquid volume, including pressurizer and surge line (ambient conditions), ft^3	11,892
System liquid volume, including pressurizer max. guaranteed power, ft^3 (estimated)	11,780
Total Reactor heat output (100% power) Btu/hr	$11,089 \times 10^6$ (Unit 1) (3250 MWt) $11,641 \times 10^6$ (Unit 2) (3411 MWt)

	Unit 1	Unit 2
Reactor vessel coolant temperature at full power:		
Inlet, nominal, ^oF	536.3	541.27
Outlet, ^oF	599.3	606.35
Coolant temperature rise in vessel at full power, avg., ^oF	63.0	64.8
Total coolant flow rate, lb/hr	135.6×10^6	134.6×10^6
(*) Steam pressure at full power, psia	758	820
Steam Temp. @ full power, ^oF	512.1	521.1
Total Reactor Coolant Volume at ambient conditions, ft^3	12,438	12,438

(*)

Equivalent to 88,500 gpm/loop. This value is the one used in non-ITDP transients. The value used in the analysis of ITDP transients is 142.7×10^6 lb/hr. Measured values are typically 146.0×10^6 lb/hr.

TABLE 4.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Pressure, psig
Design Pressure	2485
Operating Pressure	2235
Safety Valves	2485
Power Relief Valves*	2335
Pressurizer Spray Valves (Begin to Open)	2260
Pressurizer Spray Valves (Full Open)	2310
Pressurizer Pressure High - Reactor Trip	2378
High Pressure Alarm	2310
Pressurizer Pressure Low - Reactor Trip	1872
Low Pressure Alarm	2135
Pressurizer Pressure Low - Safety Injection	1823
Hydrostatic Test Pressure	3106
Backup Heaters On	2185
Proportional Heaters (Begin to Operate)	2250
Proportional Heaters (Full Operation)	2220

*During Start-up and Shut-down when Reactor Coolant System pressure drops below 425 psig, a safeguard circuit is manually switched on which allows opening of two Power Relief Valves at 435 psig for low temperature overpressure protection of the Reactor Vessel.

TABLE 4.1-5

STEAM GENERATOR DESIGN DATA*

Number of Steam Generators	4	
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107	
Design temperature, Reactor Coolant/Steam, ^{o}F	650/600	
Reactor Coolant Flow, lb/hr	33.9×10^{6}	
Total Heat Transfer Surface Area, ft^{2}	51,500	
Primary Side:	Unit 1	Unit 2
Heat Transfer Rate (per unit), Btu/hr	2773×10^{6}	2903×10^{6}
Coolant Inlet Temperature, ^{o}F	599.3	606.35
Coolant Outlet Temperature, ^{o}F	536.3	541.27
Flow Rate, (per unit), lb/hr	33.9×10^{6}	33.7×10^{6}
Pressure loss, psi	31.4	31.4
Heat Transfer Area, ft^{2}	51,500	51,500
Fouling Factor, hr-ft^{2}-^{o}F/Btu	0.0002	0.00005
Secondary Side:		
Steam Temperature at full power, ^{o}F	512.1	521.1
Steam Flow, lb/hr	3.53×10^{6}	3.685×10^{6}
Steam Pressure at full power, psia	758	820
Maximum moisture carryover, wt %	0.25	0.25
Feedwater Temperature at No. 6 Heater Outlet	436.5	431.3
Overall Height, ft-in.	67-8	
Shell OD, upper/lower, in.	175.75/135	
Number of U-tubes	3388	
U-tube outer Diameter, in.	0.875	
Tube Wall Thickness, (minimum), in.	0.050	
Number of manways/ID, in.	4/16	
Number of handholes/ID, in.	2/6	

*Quantities are for each steam generator

TABLE 4.1-5 (cont'd.)

STEAM GENERATOR DESIGN DATA*

	Rated Load	No Load
Reactor Coolant Water Volume, ft^3	1080	1080
Primary Side Fluid Heat Content, Btu	28.7×10^6	27.7×10^6
Secondary Side Water Volume, ft^3	1837	3524
Secondary Side Steam Volume, ft^3	4030	2344
Secondary Side Fluid Heat Content, Btu	5.738×10^7	9.628×10^7

*Quantities are for each steam generator

valves, regardless of size, are provided with double-packed stuffing boxes. Leakage to the atmosphere is essentially zero for these valves.

Indication of valve position for the pressurizer safety and power-operated relief valves is provided by a four channel acoustic flow monitor. There are four accelerometers, one strapped to the discharge of each of the three pressurizer safety valves and one on the common discharge of the three power relief valves. Flow through any of these valves produces an acoustic energy input to the respective accelerometer and this is amplified on the assigned channel of the monitor which is located in the Control Room. Indication on four vertical rows of light emitting diodes represents a bar graph display of relative flow through the monitored valves.

Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and with back-pressure compensation designed to prevent system pressure from exceeding the design pressure by more than 110 percent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The set pressure of the valves is 2485 psig.

A water seal is maintained below each safety valve seat to minimize leakage. The 6" pipes connecting the pressurizer nozzles to their respective safety valves, are shaped in the form of a loop seal. Condensate, as a result of normal heat losses to the ambient, will accumulate in the loop, thus flooding the valve seat. The water will prevent any leakage of hydrogen gas or steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they will start lifting and the water from the seal will discharge during the actuation period. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

Power Relief Valves

The pressurizer is equipped with 3 power-operated relief valves which limit system pressure for a large power mismatch and thus lessen the likelihood of an actuation of the fixed high-pressure reactor trip. The relief valves operate automatically or by remote manual control. The operation of these valves also limits the undesirable operation of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves. An acoustic flow monitor and a temperature indicator on the common discharge of the relief valves alerts the operator to the passage of steam due to leakage or valve opening.

The power relief valves are designed to limit the pressurizer pressure to a value below the high-pressure trip set point for all design transients up to and including a full load reduction to auxiliary power with steam dump actuation. In addition, during startup and shutdown transient conditions, when the reactor coolant system might be in a water solid condition and the RCS pressure is under 425 psig, a safeguard circuit is energized in the control room to allow automatic opening of two power relief valves at 435 psig for low-temperature over-pressure protection of the reactor vessel.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 4.1-8.

4.2.2.9 Reactor Coolant System Supports

1. Steam Generator Support

   Each steam generator is supported by a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 46½ feet apart. The vertical columns have a ball joint connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

4.2.4 PROTECTION AGAINST PROLIFERATION OF DYNAMIC EFFECTS

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into certain areas of the containment building during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate and all essential equipment inside the containment against blowdown jet forces and pipe whip to meet the missile protection criteria of Section 1.4.1 and the following:

1. A break of a steam or feedwater pipe inside the containment must not cause a break in a steam or feedwater pipe of another loop.

2. The leak tightness of the containment liner must not be damaged by a whip or blowdown jet force of a pipe which is part of the reactor coolant pressure boundary or which is necessary to function after a LOCA.

The concrete deck over the Reactor Coolant System also provides shielding and missile damage protection.

Reactor coolant pressure boundary equipment and piping are supported and provided with restraints to resist the actions of seismic, thermal expansion and pipe rupture effects.

4.2.5 MATERIALS OF CONSTRUCTION

The materials used in the Reactor Coolant System are selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

Reactor Coolant System materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steel and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The water in the secondary side of the steam generators is held within the chemistry specification given in Table 4.2-3 to control deposits and corrosion inside the steam generators.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, stress, and time.

It is a characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of chlorides, fluorides, and free oxygen. Experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses which resulted from normal manufacturing procedures. The steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Inconel units has confirmed this conclusion.

External insulation of Reactor Coolant system components is compatible with component materials. The cylindrical shell exterior, closure flanges and bottom head of the reactor vessel are insulated with stainless steel, metallic, reflective insulation. The closure head is insulated with stainless steel, metallic, reflective insulation. Other external corrosion-resistant surfaces in the Reactor Coolant System are insulated with low or halide-free insulating material as required.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low RT_{NDT} temperature, and thereby gives assurance that the finished vessels can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the RT_{NDT} temperature.

The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample locations are described in Section 4.5.1.3. The calculation method used to obtain the maximum neutron ($E > 1$ Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectrum at the sample can be applied with confidence to the adjacent section of reactor vessel, the vessel exposure will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron ($E > 1$ Mev) exposure of the vessel after 28 EFPY of operation at 3250 MWt and 80% load factor was computed to be $1.3 \times 10^{19} n/cm^2$ at 1/4T.

The limiting material is in the intermediate shell, with an original RT_{NDT} of 45°F, and a copper content of .15%. The estimated end-of-life shift in RT_{NDT} is 120°F at 1/4 T.

To evaluate the RT_{NDT} temperature shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Sub-Section 4.5.1.3.

The methods used to measure the initial RT_{NDT} temperature of the reactor vessel base plate material are given in Sub-Section 4.5.1.3.

4.2.6 MAXIMUM HEATING AND COOLING RATES

The Reactor Coolant System operating cycle used for design purposes is given in Table 4.1-10 and described in Section 4.1.5. The normal system heating and cooling rate is 60°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

The fastest cooldown rates, which result from the hypothetical case of a break of a main steam line, are discussed in Chapter 14.

Surface thermocouples on each Steam Generator above the level of the tubesheet are provided to permit a direct measurement of Steam Generator temperature, to determine that no more than a 50°F difference exists with the Reactor Coolant System cold leg temperature prior to starting a reactor coolant pump in the inactive loop with the Reactor Coolant System in the water solid condition.

TABLE 4.2-3

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION

Cation Conductivity, max.	20 μ mhos/cm^3
Total Suspended Solids, max.	1.0 ppm
pH (normal operation), 25°C	8.3-9.5
Free Caustic max.	1.0 ppm
Dissolved Oxygen	Essentially Zero (less than 0.005 ppm)
Chlorides, Max.	0.5 ppm
Silica, Max.	5 ppm

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4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 SYSTEM HEATUP AND COOLDOWN RATES

Operating limits for the Reactor Coolant System with respect to heatup and cooldown rates are defined in the Technical Specifications.

Assurance of adequate fracture toughness of the reactor coolant system is provided by compliance with the requirements for fracture toughness testing included in Section III of the ASME Boiler and Pressure Vessel Code and the Code of Federal Regulations, 10 CFR 50, Appendices G and H.

The original heatup and cooldown curves for the plant were based on the actual measured fracture toughness properties of the vessel materials determined in accordance with the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, as implemented by the Code Case #1514. Allowable pressures as a function of the rate of temperature change and the actual temperature relative to the nil-ductility transition reference temperature (RT_{NDT}) of the reactor vessel were established according to the methods given in Appendix G 2000, "Protection Against Non-Brittle Failure", of Section III of the ASME Boiler and Pressure Vessel Code. Typical curves incorporating allowances for instrument error in measurement of temperature and pressure are given in the Technical Specifications.

The original curves are based on a temperature scale relative to the RT_{NDT} of the vessel, including appropriate estimates of ΔRT_{NDT} caused by radiation. Predicted ΔRT_{NDT} values were derived by using Figure B - 3/4.4-2 of the Technical Specifications, and the fluence of 1/4T corresponding to the maximum for the service period applicable. Initial RT_{NDT} included an assumed ΔRT_{NDT} corresponding to that predicted after 2 integrated full power years of operation.

The heatup and cooldown curves are updated based on information obtained from our radiation surveillance program in accordance with the Technical Specifications.

The use of RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for other vessel regions which are exposed to much lower radiation levels. Therefore, the flanges, nozzles, and other such regions less affected by radiation are favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}. Changes in fracture toughness of the core region plates or forgings, weldments and associated heat affected zones due to radiation damages are monitored by a surveillance program which conforms with ASTM E-185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels". The evaluation of the radiation damage in this surveillance program is based on pre-irradiation and post-irradiation testing by Charpy V-notch and tensile specimens and post-irradiation testing of wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals.

4.4.2 REACTOR COOLANT ACTIVITY LIMITS

Release of activity into the reactor coolant in itself does not constitute a hazard to the public. Activity in the coolant could constitute a hazard to the public only if the Reactor Coolant System barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems are designed for operation with activity in the Reactor Coolant System corresponding to 1 percent fuel defects. In the event of steam generator tube leakage, high activity level at the condenser air ejector exhaust will initiate an alarm to warn the operator to take corrective action. The Reactor Coolant System activity limit during operation is defined in the Technical Specifications.

4.4.3 MAXIMUM PRESSURE

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III. Reactor Coolant System pressure settings are given in Table 4.1-2.

4.4.4 SYSTEM MINIMUM OPERATING CONDITIONS

Minimum Operating Conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications.

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the adjacent vessel wall because the specimens are located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the RT_{NDT} measurements are representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The calculated maximum fast neutron exposure (nvt) at the vessel wall is 2.0×10^{19} nvt > 1 Mev. The reactor vessel surveillance capsules are located at orientations shown in Figure 4.5-2. The capsule lead factors (ratio of fast fluence at the capsule location versus that at the vessel inner wall) for Unit 1 and 2 are listed below:

Capsule Identification	Lead Factor
T,Y,Z,S	3.7
W,V,U,X	1.1

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Sub-Section 4.5.1.3 and indicate good agreement.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure will be made by use of data on withdrawn capsules.

The current schedule for removal of capsules is as follows:

Capsule	Removal Time Effective Full Power Years (EFPY) Unit 1	Unit 2
T	1.25*	1.09*
X	3.48*	3.24*
Y	5	5
U	9	9
S	32	32
V,W,Z	Standby	Standby

Quality Control Program

Table 4.5-1 summarizes the quality control program with regard to inspections performed on Reactor Coolant system components. In addition to the inspections shown in Table 4.5-1, there were those performed by the equipment supplier to confirm the adequacy of material he received, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME code requirements. The inspection procedures and acceptance standards required on piping materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the accessible external surfaces of the primary Reactor Coolant System pressure containing

*Capsules removed at time indicated.

section test facility. The program also includes inspection and testing of the installed ice condenser before and after the initial ice loading prior to initial plant startup, and inspections and testing throughout the plant operating lifetime.

Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

The ice condenser has no active components important to fulfillment of its safety function and thus is not susceptible to failure of active components and the resulting consideration of additional capability to accommodate failures.

In any case, the ice condenser does have an excess of capability for both rate and quantity of energy released from the Reactor Coolant System. The door panels located at the inlet and outlet of the ice condenser are the only elements required to move during the accident. These items are considered as passive or static elements equivalent to rupture discs rather than active components requiring an external signal and energy source to function.

Inspection of Containment Pressure Reducing Systems

Criterion: Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

The ice condenser design includes suitable provision for visual inspections of the ice beds flow channels, door panels, and cooling equipment. The discussion of inspection for other containment pressure reducing systems is presented in Chapter 6.

Testing of Containment Pressure Reducing System Components

Criterion: The containment pressure-reducing systems shall be designed to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

The ice condenser is a completely static engineered safety feature containing no active components required to function during an accident condition. However, provision is made for periodic testing of elements of the ice condenser including door panels, inspection of the ice beds, and sampling of the ice.

Containment Isolation Valves

Criterion: Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Redundant valving is provided for piping that is open to the atmosphere and connects to the Reactor Coolant System or is open to the containment atmosphere. Details of this and other requirements for valving are given in Sub-Chapter 5.4.

Provisions for Testing of Penetrations

Criterion: Provisions shall be made, to the extent practical, for periodically testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

A Containment Penetration and Weld Channel Pressurization System, described in detail in Sub-Chapter 5.6 provides a ready means for testing such penetrations.

Provisions for Testing of Isolation Valves

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus and the leakage during periods of reactor shutdown.

Initiation of the containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel places that channel in the tripped mode.

NDT Temperature Requirement for Containment Material

Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes.

The selection and use of containment materials comply with the applicable codes and standards listed in Section 5.2.2.

The concrete containment structure is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not directly exposed to the temperature of the environs. The containment ambient temperature during operation is between 60 and 120°F, and the ice bed operating temperature is approximately 10-20°F, which is above the NDTT + 30°F for the liner material. Containment penetrations which are exposed to the environment are also designed to the NDTT + 30°F criterion.

Initial Leakage Rate Testing of Containment

Criterion: The containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetration and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.

The containment was designed so that its maximum integrated leakage under accident conditions meets the site exposure criteria set forth in 10 CFR 100 guidelines. The ice condenser and the spray systems provide assurance that with a containment leak rate of 0.25 per cent by weight per day, the exposure at the minimum exclusion distance is less than 10 CFR 100 guidelines.

The preoperational leak rate tests included an integrated leak rate test of the containment and a sensitive leak rate test of the penetrations and weld channels. The leak rate test is an integrated leak test at design pressure to verify that the structure leaks less than the allowable value. The sensitive leak rate test is performed by pressurizing the double penetrations at slightly above design pressure. This test is conducted with the containment at atmospheric pressure. These tests demonstrate the integrity of the double leakage barriers provided by the penetrations and the overall integrity of the containment. The preoperational leak tests of each unit containment demonstrated that the integrated leak rates were less than 0.25 percent by weight of the containment free volume per day.

Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by tests during the plant lifetime.

The containment is provided with weld channel and containment penetration pressurization systems so that periodic integrated leak rate tests can be made of those areas of containment in which leakage is likely to be confined. The containment is designed to permit full integrated leak rate tests.

5.2.2 CONTAINMENT SYSTEM STRUCTURE DESIGN

The general arrangement of the Ice Condenser Reactor Containment is shown on Figure 5.2.2-1 and 5.2.2-1A (Elevations) and Figures 5.2.2-2 and 5.2.2-2A (Cross sections).

The containment is divided into three main compartments. These are:

a) The lower compartment.
b) The upper compartment.
c) The ice condenser compartment.

The lower compartment encloses the reactor system and associated auxiliary systems equipment. The upper compartment contains the refueling cavity, refueling equipment and polar crane used during refueling and maintenance operations. The upper and lower compartments are separated by a divider barrier. The ice condenser which contains borated ice provided to absorb the loss-of-coolant accident energy, is in the form of an enclosed and refrigerated annular compartment, located circumferentially between the crane wall and the outer wall of the containment and extends from below to above the operating deck.

The reactor containment structure is a reinforced concrete vertical right cylinder with a slab base and a hemispherical dome. A welded steel liner with a minimum thickness of 3/8" at the dome and wall, and 1/4" at the bottom is attached to the inside face of the concrete

shell, to insure a high degree of leak tightness. The containment structure is designed to contain the radioactive material which might be released following a loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.

The structure as shown in Figure 5.2.2-3 consists of side walls measuring 113 ft in height from the liner on the base to the spring line of the dome and has an inside diameter of 115 ft. The thickness of the cylinder is 3 ft - 6 in and the thickness of the dome is 3 ft - 6 in at the spring line tapering uniformly to 2 ft - 6 in at the peak of the dome. The base mat consists of a 10 ft thick structural concrete slab, increasing to 20 ft adjacent to the recirculation sump area.

Figures 5.2.2-4, 5.2.2-4A, 5.2.2-4B indicate the typical re-bar pattern for the containment building side walls and dome.

In general the reinforcing consists of meridional and hoop reinforcing in both faces of the wall and dome. Radial shear reinforcing is provided where required. At the base of the containment wall, for a distance of ten (10) ft. above the base mat, inclined radial shear bars are provided. The bars are anchored by bond in the compression face of the wall and by hooking around the wall hoop reinforcing in the tension face of the wall.

Radial shear bars are not anchored by bond in regions of bi-axial tension.

Additionally tangential reinforcing was placed, in both faces of the containment side wall, inclined at an angle of 45° to each side of a vertical such that the reinforcing runs diagonally up from the base slab to the right and to the left of the vertical in each face.

5.2.4.2 Piping Penetrations

Piping penetrations are provided for all piping passing through the containment walls. The core pipe is contained within a sleeve which is welded to the containment liner. Several core pipes may pass through the same penetration assembly to minimize the number of penetrations required. In such cases, each core pipe is welded to the end plates in the penetration assembly. A connection is provided on the penetration assembly outside of the containment to allow pressurization of the annulus formed by the pipe and sleeve. In the case of a pipe carrying a hot fluid, the core pipe may be insulated and cooling may be provided to limit the concrete temperature abutting the sleeve to 150°F.

The design ensures that, even under postulated accident conditions, potential resultant torsional, axial, bending and shear loads will not cause a breach of containment integrity. Penetrations were analyzed for the following conditions: a) Normal Operating Conditions; b) Transient Conditions; c) Seismic; d) Pipe Rupture (including consideration of the status of each pipe during the course of an accident). Loads on the penetration sleeve were combined following the principles in ASME Boiler and Pressure Vessel Code Section III. Penetrations were designed such that the rupture of connecting piping will not cause a loss of containment integrity.

Piping between the containment penetrations and the isolation valves outside the containment were designed in conformance with USAS B31.1 for design loads.

The main-steam pipe penetration assembly is similar to the hot pipe penetration illustrated in Figure 5.2-2. The core pipe within the penetration has a structural capability greater than that of the pipes welded to it. The penetration sleeve and core pipe are joined by a flued head which has a structural capability not less than the core pipe within the penetration assembly. The penetration sleeve in turn has adequate structural capability.

A complete thermal analysis was made of the penetration assembly to determine thermal insulation requirements to be used in conjunction with expanded plate-type coolers, to limit concrete temperature during normal operation to 150°F. Coolers were provided with redundant circuitry and capacity to maintain concrete temperature below 150°F with one circuit out of service. Thermal analysis to determine the time dependent limitations of penetration sleeve temperature limitations with regard to the containment liner and concrete was performed to cover conditions of loss of cooling water.

The thermal growth of the penetration sleeve and stress at the anchors and liner weld was considered in establishing temperature limitations.

The penetration assembly is anchored into the containment wall with a structural capability based upon Maximum Pipe Rupture Loads with regard to torsion, bending, shear, and jet thrust. Earthquake loads were considered.

The penetration assembly was designed to withstand any strains imposed by the liner.

The radial deformation imposed by the liner on the penetration sleeve was considered to be uniform around the circumference of the penetration sleeve and the moments and hoop stresses in the penetration sleeve then determined.

Stresses in the penetration were limited to the values stated in ASME Boiler and Pressure Vessel Code, Section III.

Sump Penetration

Two piping penetrations in the containment sump area are of the pipe and outer sleeve design. The outer sleeve is welded directly to the base of the liner. The weld to the liner is covered by a pressurization channel which is used to demonstrate liner integrity. The inner and outer pipes extend through the containment wall and are connected to an isolation valve and enclosure.

Fuel Transfer Penetration

A piping penetration, designated the fuel transfer tube penetration, is provided for fuel movement between the refueling canal in the containment and the transfer canal in the auxiliary building. The penetration consists of a stainless steel pipe installed inside a 24" pipe, as shown in detail on Figure 5.2-4. The inner pipe acts as the transfer tube and connects the containment refueling cavity with the fuel transfer canal in the auxiliary building.

The outer pipe is welded to the containment liner and provision was made for the employment of a seal ring for pressurizing welds essential to containment integrity. Bellows expansion joints were provided on the outer pipe to compensate for any differential movement between the inner and outer pipes and also between the containment and auxiliary building structures.

Specification and Tests

Piping penetrations were designed to the intent of USAS B31.1 1967 Edition and N-Cases' (1955), and ASME Boiler and Pressure Vessel Code Section III 1968 Edition.

Material specifications for the piping penetrations are as follows;

Penetration Sleeve		ASTM A - 333 Gr 6
Penetration Reinforcing Rings		ASTM A - 442 Gr 60*
Penetration Sleeve Reinforcing		ASTM A - 442 Gr 60*
Bar Anchoring Rings, End Plates or Flued Heads		ASTM A - 442 GR 60*
		ASTM A - 350 LFI
		ASTM A -182 F 316 and F 304
Rolled Shapes		ASTM A - 442 Gr 60*
Core Pipe	Carbon Steel	ASTM A - 106 Grades B and C
		ASTM A - 155 KC 70 Class I
	Stainless Steel	ASTM A - 312 TP 304
		ASTM A - 358 Class I TP 304
		ASTM A - 376 TP 304 and TP 316
		ASTM A - 213 (Type 136)
		ASTM A - 249 (Type 316)

* or ASTM A 516 Gr. 70.

NDTT has been considered where required for the materials listed above. The piping penetration assemblies were tested, prior to installation, by pressurizing the annulus between the core pipe and sleeve for 30 minutes during which time the exterior was checked for leaks using a soap bubble solution. . If any leakage was found, the assembly was repaired and the assembly retested. Following the soap bubble leakage test, the annulus was pressurized with a mixture of air and 20% by weight of freon gas. The assembly was then tested for leakage using a halogen leak detector with a sensitivity of 10^{-7} standard cc per second. A mass spectrometer examination was substituted for the halogen leak detection test where it was deemed required.

sheet is also provided on the surfaces of the panels on the crane wall and end walls, and forms the vapor barrier for the compartment on those walls.

The panels are clamped to the walls by studs, clamp washers and nuts. The weight of the panels is transmitted to the floor at the outer wall of the ice condenser and to the ice support structure at the inner wall.

The fastening of the insulated duct panels to the walls, the construction incorporating sheet metal faces, and the additional constraints provided by the configuration of the ice baskets and support structure, eliminates any mechanism which would allow the insulation material to significantly impede the performance of the ice condenser during accident conditions.

The floor of the ice condenser is cooled by embedded pipe coils through which chilled glycol is circulated.

The wall panel design is such that the structure of panels resists compression of insulation with exception to slight momentary compression of the interior insulation layer between the duct and the ice side galvanized sheet metal face. This slight compression is due to the deformation of the face and will return after the pressure is reduced. During accident conditions the slightly higher conductivity of this insulation has no effect on ice condenser performance.

An additional load is imposed on the refrigeration system due to the heat flow into the upper plenum. The walls of the plenum are also insulated by prefabricated fiberglass panels, but the air flow from the duct panel exhaust to the air handling units circulates in the plenum, picking up heat input through the insulation and top deck doors. This ensures that moisture leaking into the ice condenser plenum is picked up by the air and freezes on the cooler coils. Together with the vapor barrier on the inner face of the insulated duct panels, this minimizes the ingress of moisture into the ice bed.

Ice Condenser Doors

Inlet Doors

The inlet doors at the bottom of the ice condenser are suitably insulated panels mounted as vertically hinged pairs, on an angle section frame between the concrete pillars supporting the crane wall as shown in Figure 5.3-4.

The doors consist of a 2½ inch composite panel with steel facings and a foam core, and a 4½ inch foam insulation backing that is enclosed with a sheet metal cover.

The doors are provided with springs which produce a small force to resist door opening. The magnitude of the force produced by the springs when the doors are fully open is equivalent to a differential pressure of approximately 1 pound per square foot. The doors are normally held shut, against a seal moutned on the frame, by the static differential pressure due to the higher density air in the ice condenser compartment. With zero differential pressure across the doors (no cold air head), the neutral position of the spring is set so that the doors are slightly open (approximately ¼ inch). Thus, all doors will begin to open at the same differential pressure (cold air head) and, within the limits of spring tolerances, the doors will all open equal amounts.

Provision is made for a drain area of approximately 15 sq. ft. at the bottom of the ice condenser to permit water and air to flow from the ice condenser during and after the reactor coolant blowdown. This provision assures that, if doors reclose after a large break before all water drains out, or for small break or residual heat release cases where doors are not fully open, water from melted ice can drain from the compartment. The total drain area is provided by several individual drains.

Ice Machine

Three ice machines are installed in the auxiliary building. The machines are each capable of producing ten tons of borated ice per day, which is adequate for all recharging requirements. The ice is made in a shape and size convenient for handling, and provision is made for checking that ice loading and ice chemistry are maintained within the prescribed limits.

Instrumentation - Monitoring System

There are 96 temperature sensing elements which are distributed throughout the ice bed and are monitored and recorded in the Control Room. An annunciator panel provides an alarm for a pre-set deviation from the prescribed limits of ice bed equilibrium temperature.

Each inlet door panel operates two switches when in the closed position. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. An annunciator panel in the Control Room gives an alarm signal for the door open condition.

Similarly, ice condenser compartment equipment and personnel access doors are fitted with switches providing Control Room indication of the position of those doors. Also see Appendix M, Section 6.10.

Ice Condenser Materials

Corrosion of ice condenser components will be greatly reduced by the low temperature operation of the ice condenser. Corrosion at ice condenser operating temperature, even at saturation, is almost non-existent.

Ice bed structural steel member materials were impact tested to meet the temperature requirements of N1210 of Subsection B Section III Nuclear Vessel Code of ASME, of at least 30°F lower then than the lowest service temperature for all section thicknesses in excess of 5/8 inch. For section thickness equal to or less than 5/8 inch material was either impact tested or specified to fine grain practice.

To further inhibit corrosion, galvanizing is used for baskets and metal panels. Structural members are protected by corrosion resistant paints. At ice condenser operating temperature, the corrosion resistant paints are expected to last the life of the ice condenser without maintenance.

Any ice condenser equipment whose performance might be affected by corrosion employs corrosion resistant materials for critical components. Any corrosion that would develop over long term operation would not impair the performence of the ice condenser.

The ice condenser cooling system utilizes ethylene glycol as a coolant. According to published data[1] all ice condenser materials selected have good chemical resistance to ethylene glycol.

The insulation panels in the ice bed region are provided with a vapor barrier which would prevent moisture from reaching the containment vessel insulation interface region. A small proportion of the air flow bleeds from the ducts into the annulus between the duct panel and the containment. The inner surface temperature at the boundary of the ice condenser, in particular, the containment liner, varies with external ambient conditions from a maximum of 80°F down to 0°F. For all boundary temperatures above the duct air temperature, any moisture in the insulation diffuses to the cooling ducts and in addition is absorbed by the bleed flow of air. For containment liner temperature below duct air temperature, which correspond to near zero external ambient conditions, any moisture transferred to the annulus between the containment liner and the duct panels forms as frost, but since the temperature gradients are reversed for this condition, the frost cannot detract from the performance of the cooling system.

With the above factors considered, removal of the ice condenser wall panels for containment liner inspection should not be necessary, as is the case for the analogous situation where the steel is encased in concrete. Access to the containment liner in the ice condenser region for containment inspection is provided by three special inspection parts through duct panels located approximately at the quarter points, in the lower section of the ice bed. Access to the containment liner in the plenum region is provided by removal of a plenum insulation panel.

5.3.3 ICE CONDENSER OPERATING CONSIDERATIONS

<u>Refrigeration System</u>

The ice condenser and associated systems provide a completely reliable, static heat sink which is instantaneously available if needed during a loss-of-coolant accident. The design assures that the quantity and configuration of the ice is maintained within the limits acceptable for the accident requirements. The insulation system minimizes the total heat gain into the compartment, and air leakage flow through the compartment, the double-walled insulation adjacent to the compartment further minimizes heat flows which would otherwise tend to promote ice sublimation and mass transfer within the ice bed. Long-term ice storage tests have shown that the ice can be stored for long periods of time without significant weight loss or physical distortion.

The ice condenser refrigeration system is provided with excess capacity to assure that the ice is maintained below the freezing temperature. The capacity of the refrigeration units exceed the maximum heat gain to the ice condenser compartment, and two standby refrigeration units are available for maintenance shutdowns, emergency shutdowns, and ice machine operation. The ethylene glycol loops for each unit of the twin-unit plant are separate, but can be interconnected. The many air handling units in each ice condenser compartment also have excess capacity so that the compartment temperature can easily be maintained if fans are shut down for defrosting the coils, for maintaining of the equipment, or as a result of equipment failure. The air duct feeding the

insulated duct panels is continuous around the pheriphery of the upper plenum in the ice condenser compartment, so that air flow is maintained.

The double-walled insulation adjacent to the ice compartment provides additional protection against heat gain if the entire refrigeration system were to shut down. Because of the air gap and second insulation layer, the time required to raise the ice and internal structures to melting temperature would be about two weeks, allowing more than sufficient time to repair or restart the refrigeration equipment. An additional 2 weeks would elapse before even 10 per cent of the ice would melt if the refrigeration system were not operating.

Ice Bed Loading

Prior to the initial plant startup, the ice condenser compartment is cooled to operating temperature, 10-20°F, and loaded with ice. One or more ice making machines generates the ice in flake form for ease in handling. The ice is moved into the containment through a normally closed penetration by a pneumatic conveying system. The system feeds ice to a loading head, which is positioned by the bridge crane, through removable tubes in the plenum. This system is also used to replenish sections of the ice condenser, if required.

Ice Condenser Inspection

The design of the ice condenser and its components is such that a minimum amount of surveillance is required. However, in order to ensure satisfactory performance in the event of a loss-of-coolant accident, the following inspections and monitoring are required:

a) Total weight of ice initially installed in the condenser is determined by weighing the loads of ice being installed.

b) Flow passages and ice beds typical of all sections of the condenser are visually inspected to check that no significant changes are occurring.

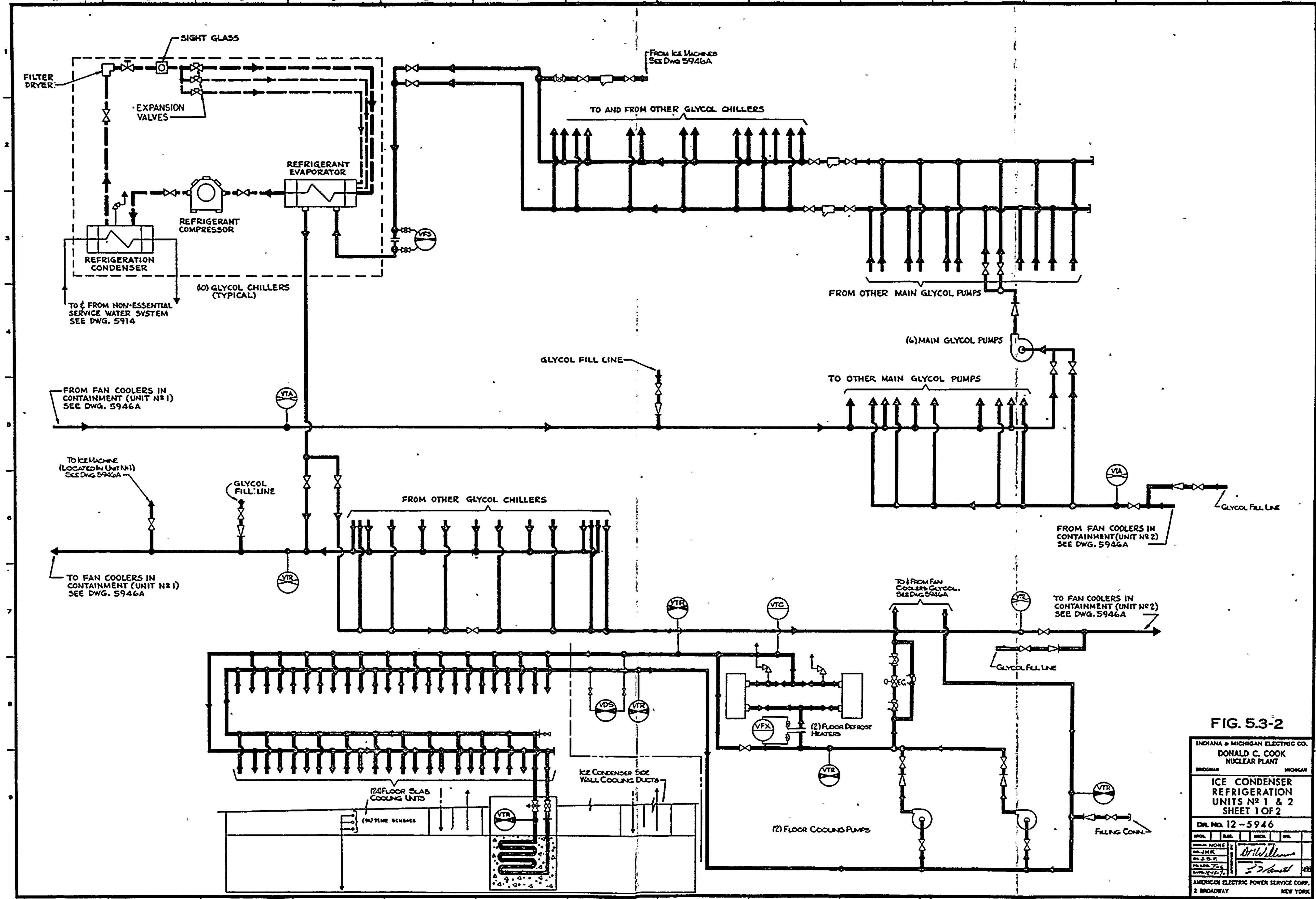

FIG. 5.3-2

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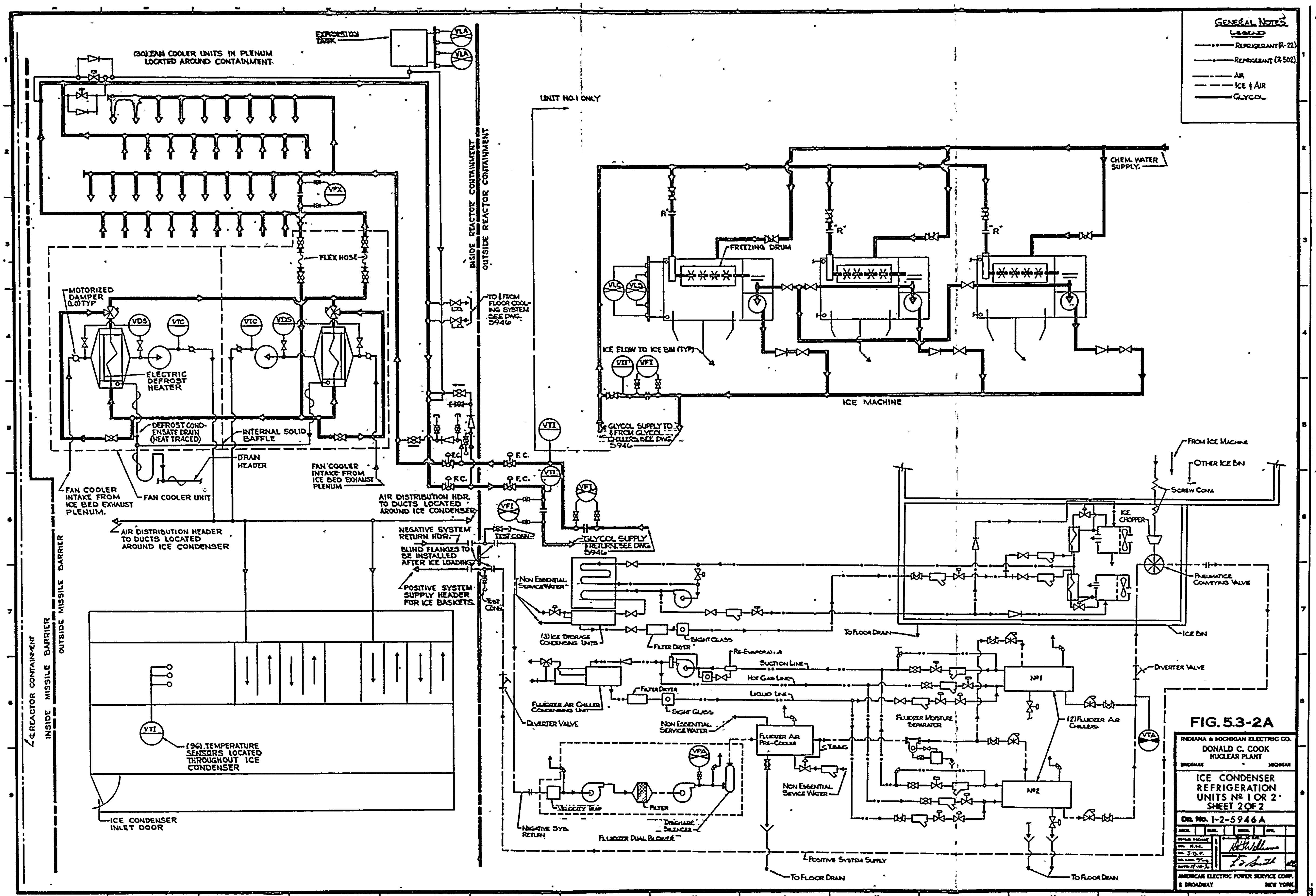

GENERAL NOTES
LEGEND
REFRIGERANT (R-22)
REFRIGERANT (R-502)
AIR
ICE & AIR
GLYCOL
EXPANSION TANK
VLA
(30) FAN COOLER UNITS IN PLENUM LOCATED AROUND CONTAINMENT.
VPX
FLEX HOSE
MOTORIZED DAMPER (L.O.) TYP
VDS
VTC
ELECTRIC DEFROST HEATER
DEFROST CONDENSATE DRAIN (HEAT TRACED)
INTERNAL SOLID BAFFLE
DRAIN HEADER
FAN COOLER INTAKE FROM ICE BED EXHAUST PLENUM
FAN COOLER UNIT
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
TO & FROM FLOOR COOLING SYSTEM SEE DWG. 5946
F.C.
VTI
VFI
AIR DISTRIBUTION HDR. TO DUCTS LOCATED AROUND ICE CONDENSER
AIR DISTRIBUTION HEADER TO DUCTS LOCATED AROUND ICE CONDENSER
NEGATIVE SYSTEM RETURN HDR.
BLIND FLANGES TO BE INSTALLED AFTER ICE LOADING
POSITIVE SYSTEM SUPPLY HEADER FOR ICE BASKETS
TEST CONN.
GLYCOL SUPPLY RETURN SEE DWG. 5946
℄ REACTOR CONTAINMENT
INSIDE MISSILE BARRIER
OUTSIDE MISSILE BARRIER
(96) TEMPERATURE SENSORS LOCATED THROUGHOUT ICE CONDENSER
ICE CONDENSER INLET DOOR
UNIT NO. 1 ONLY
CHEM. WATER SUPPLY
FREEZING DRUM
"R"
VLC
VLS
ICE FLOW TO ICE BIN (TYP)
VTI
VFI
GLYCOL SUPPLY TO & FROM GLYCOL CHILLERS SEE DWG. 5946
ICE MACHINE
FROM ICE MACHINE
OTHER ICE BIN
SCREW CONV.
ICE CHOPPER
PNEUMATIC CONVEYING VALVE
NON ESSENTIAL SERVICE WATER
(3) ICE STORAGE CONDENSING UNITS
FILTER DRYER
SIGHT GLASS
TO FLOOR DRAIN
ICE BIN
RE-EVAPORATOR
SUCTION LINE
HOT GAS LINE
LIQUID LINE
DIVERTER VALVE
No 1
No 2
FLUIDIZER AIR CHILLER CONDENSING UNIT
FLUIDIZER MOISTURE SEPARATOR
(2) FLUIDIZER AIR CHILLERS
NON ESSENTIAL SERVICE WATER
FLUIDIZER AIR PRE-COOLER
VTA
VPA
VELOCITY TRAP
FILTER
DISCHARGE SILENCER
FLUIDIZER DUAL BLOWER
NEGATIVE SYS. RETURN
POSITIVE SYSTEM SUPPLY
TO FLOOR DRAIN
INDIANA & MICHIGAN ELECTRIC CO.
DONALD C. COOK NUCLEAR PLANT
BRIDGMAN MICHIGAN
ICE CONDENSER REFRIGERATION UNITS No 1 OR 2 SHEET 2 OF 2
DR. NO. 1-2-5946A
AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY NEW YORK
JULY 1982

FIG. 5.3-2A

5.5.5 INCIDENT CONTROL

In the event of an incident the two independent Containment Air Recirculation/Hydrogen Skimmer System fans automatically start after a 10 minute time delay after initiation of 2/4 hi-hi containment pressure signals. The operation of either fan ensures the reduction of the containment pressure to the limits described in Chapter 14.

At the same time the Air Recirculation/Hydrogen Skimmer fans start, the hydrogen skimmer valves in the two Containment Air Recirculation/ Hydrogen Skimmer headers open, thus causing the Air Recirculation/ Hydrogen Skimmer System fans to continuously purge all potential hydrogen pockets in the Containment.

All other Containment Ventilation Systems are not designed for operation during a loss of coolant accident.

During refueling, the occurrence of a High Containment Radiation Signal from the upper compartment area containment particulate and radiogas monitors will automatically trip the purge fans and close all ventilation system isolation control valves, thus isolating the Containment.

5.5.6 MALFUNCTION ANALYSIS

Sufficient redundancy exists in all recirculation ventilation systems to ensure a normal operation with one active component out of service.

The two filter cleanup units provide redundancy for small leakage rates. The Containment Purge Supply and Exhaust System is fitted with dual supply and exhaust fans. Simultaneous failure of a supply and an exhaust fan would result in an 80-minute purge rate.

The Containment Air Recirculation/Hydrogen Skimmer Systems are two 100% redundant systems, therefore the loss of either system or any component of either system will not impair system operation.

5.5.7 TESTS AND INSPECTION

All systems are inspected, tested and balanced upon installation. Charcoal and particulate filters are individually tested before shipment, upon installation and periodically thereafter as required. Replacement filters will be tested in the same manner.

The Containment Air Recirculation/Hydrogen Skimmer fans were tested during installation and are tested periodically to ensure proper functioning. The initial test of these fans were conducted at both no flow and full flow, verifying the fan capability to deliver the required amount of air. The periodic fan flow tests are conducted at no flow to assure that the fan is still operable.

5.6 CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

5.6.1 DESIGN BASES

The Containment Penetration and Weld Channel Pressurization System is designed to provide a means for determining the leak-tightness of the containment in a more sensitive and accurate manner than the conventional integrated leak rate test.

In addition, the system provides means for pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel liner in the event of a loss-of-coolant accident. Although the system is not considered as an engineered safety feature and no credit is taken for its operation in calculation of off-site accident doses, it does provide assurance that the containment leak rate would be lower than that assumed in the accident analysis.

5.6.2 SYSTEM DESIGN AND OPERATION

The Containment Penetration and Weld Channel Pressurization System is shown in Figure 5.6-1. A regulated supply of air from the two Containment Penetration and Weld Channel Pressurization System air receiver tanks is supplied to containment penetrations and liner weld channels during local leak testing or in the event of a loss-of-coolant accident. The system maintains a pressure in excess of containment design pressure thereby ensuring that there will be no out-leakage of the containment atmosphere through the penetrations and liner welds during an accident.

The pressurization system air receivers are normally supplied with clean, dry air from the control air system. This air source is normally supplied by two plant air compressors which are, in turn, backed up by two control air compressors. Each of the two pressurization system air

receivers is further backed up by a bank of nitrogen bottles. These bottles automatically maintain system pressure in the unlikely event the control air system fails.

A standby source of gas pressure for each air supply system is provided by a bank of nitrogen cylinders which are installed so as to deliver nitrogen at a slightly lower pressure than the normal regulated air supply pressure. Thus, in the event of failure of the normal and backup air supply systems during periods when the system is in operation, the penetrations and weld channel pressure requirements will be automatically maintained by the nitrogen supply. This allows reliable pressurization under accident conditions.

Should one zone indicate a leak during testing, the specific penetration or weld channel containing the leak can be identified by isolating the individual air supply line to each component in the zone.

In order to provide facility for testing the larger penetrations, non-automatic pressurizing branch lines are provided to:

a) The double-gasketed space on each hatch of the personnel air lock.

b) The double-gasketed space at the equipment hatch flange.

c) The double-gasketed space of the spent fuel transfer tube flange.

The pressurization system is divided into four (4) major zones, each of which include a restriction orifice to ensure that high air consumption in any one zone will not affect the other zones. The pressurization system supply to the zones is divided, with each receiver tank servicing two zones. Each zone serves a portion of the containment weld channels or penetrations, and can be further subdivided by manual valves for testing purposes.

The normally closed pressurization valves are automatically opened on a safety injection signal to pressurize the system. Valve position is indicated in the control room.

5.6.2.1 Instrumentation

The instrumentation provided for the Containment Penetration and Weld Channel Pressurization System is described below:

A. A pressure alarm is installed immediately downstream of each pressurization system air receiver. These alarms alert the control room operator to failure of the control air feeds.

B. A pressure alarm is installed on each nitrogen gas supply manifold to warn the operator of nitrogen supply pressure failure.

C. A pressure alarm is installed downstream from the control air and nitrogen bottle regulated feeds. These alarms will alert the operator in the extremely unlikely event that the redundant supply feed to either half of the system has failed.

D. A pressure alarm downstream of each zone's power operated valve indicates when that zone is pressurized.

E. A pressure indicator is located downstream of each regulated feed. Both indicators are located in the control room and provide system surveillance.

F. Pressure gauges are installed on each of the four pressurization legs downstream of the power operated valves.

G. Local test pressure connections are provided as necessary to allow for leak testing. Test connections have normally closed globe valves, which are plugged when not in service.

H. A flow alarm is installed downstream of each of the four power operated valves. These alarms alert the operator to high flow in any of the four zones.

5.6.3 TEST DURING ERECTION

Following the successful completion of inspection of the seam welds, the channels were tested with air at a pressure of 50 psig for at least 15 minutes. Following this strength test, the channel fillet weld joints were tested using a tracer gas technique at a pressure of 14 psig for two hours. Allowable leakage did not exceed 0.025% of total containment free volume for all zones. The bottom liner weld channels were pressure tested prior to being covered with concrete.

5.6.4 DESIGN EVALUATION

The system provides an extremely efficient and accurate method of testing the leak-tightness of the containment. The use of the Containment Penetration and Weld Channel Pressurization System as a testing medium not only provides indication of leakage, if any, but allows the leak to be readily located so that corrective action can be taken if necessary.

The employment of the system during a loss-of-coolant accident, while not considered for analysis of the consequences of the accident, provides an additional conservatism in ensuring that leakage is minimized. No detrimental effect on any safeguards system will be felt should the pressurization system fail to operate under these circumstances.

TABLE 6.1-1

NET POSITIVE SUCTION HEADS FOR

POST-DBA OPERATIONAL PUMPS

PUMP	Flow and Condition (per pump)	Suction Source	$NPSH_a$ (minimum) ft_{abs}	$NPSH_r$ ft_{abs}	Water Temp. °F
1. Safety Injection	650 gpm runout flow	Refueling Water Storage Tank	35	24.4	100 max.
2. Centrifugal Charging	550 gpm runout flow	Refueling Water Storage Tank	31	22.5	100 max.
3. *Residual Heat Removal	4500 gpm runout flow	Containment Sump	31	19.0	160
4. *Containment Spray	3200 gpm rated flow	Containment Sump	30	9.0	160
5. Component Cooling	9000 gpm rated flow	Closed Loop	55	16.5	95
6. Essential Service Water	10000 gpm rated flow	Screenhouse (with 11 ft drawdown)	52	39.0	76 max.

* The $NPSH_a$ for the Unit 2 Residual Heat Removal and Containment Spray Pumps, taking suction from the containment sump, were recalculated using higher maximum water temperatures during recirculation.

3. Residual Heat Removal	5050 gpm runout flow	Containment Sump	30.5 ft.	23 ft.	190°F
4. Containment Spray	3600 gpm runout flow	Containment Sump	32.5 ft.	10 ft.	190°F