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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
1992 FINAL SAFETY ANALYSIS REPORT UPDATE

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

July 22, 1992

Dear Dr. Murley:

Enclosed please find ten copies of the changed pages for the 1992 update to the Cook Nuclear Plant Final Safety Analysis Report. These pages are being transmitted to you according to the provisions of 10 CFR 50.71(e). Instructions for incorporating the update are included with each copy.

Changed pages have been dated "July, 1992" in the lower right corner in order to identify changed pages in addition to vertically barring the specific change. Vertical change bars next to the July, 1992 date in the lower right corner indicate that the information has only shifted pages.

The containment integrity analysis presented in Unit 1 Chapter 14.3.4 represents a rearrangement of text to provide a more useable format, but the technical content remains essentially unchanged. Full-page margin bars are provided based on the total reorganization of the section.

We hereby certify that the information contained in this update to the FSAR, to our knowledge, accurately presents changes made between January 22, 1991 and January 22, 1992.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick

dag

Enclosure

A053
1/10

cc: w/o enclosure
D. H. Williams, Jr.
A. A. Blind - Bridgman
G. Charnoff
A. B. Davis
J. R. Padgett
NRC Resident Inspector
NFEM Section Chief

✓ Instructions for Updating FSAR

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design provides for periodic testing of active components for operability and required functional performance as well as incorporating provisions to facilitate physical inspection of critical components.

3. Heat removal systems are provided within the containment to cool the containment atmosphere under design basis accident conditions. Two systems of different design principles are provided, the containment spray system and the ice condenser system. These systems have the capacity to adequately cool and reduce the pressure of the containment atmosphere as well as reduce the concentration of halogen fission products.

1.4.8 FUEL AND WASTE STORAGE SYSTEMS

Fuel storage and waste handling facilities are designed such that accidental releases of radioactivity will not exceed the guidelines of 10 CFR 100.

During refueling of the reactor, operations are conducted with the spent fuel under water. This provides visual control of the operation at all times and also maintains low radiation levels. The borated refueling water assures subcriticality and also provides adequate cooling for the spent fuel during transfer. Spent fuel is taken from the reactor core, transferred to the refueling cavity, and placed in the fuel transfer canal. Rod cluster control assembly transfer from a spent fuel assembly to a new fuel assembly is accomplished prior to transferring the spent fuel to the spent fuel storage pool. The spent fuel storage pool is supplied with a cooling system for the removal of the decay heat

of the spent fuel. Racks are provided to accommodate the storage of a total of two thousand and fifty fuel assemblies. The storage pool is filled with borated water at a concentration to match that used in the reactor cavity during refueling operations. The spent fuel is stored in a vertical array with sufficient center-to-center .

distance between assemblies to assure subcriticality ($k_{eff} \leq 0.95$) even if unborated water were introduced into the pool.[3,4] The water level maintained in the pool provides sufficient shielding to permit normal occupancy of the area by operating personnel. The spent fuel pool is also provided with systems to maintain water cleanliness and to indicate pool water level. Radiation is continuously monitored and a high radiation level is annunciated in the control room.

Water removed from the spent fuel pool must be pumped out as there are no gravity drains. Spillage or leakage of any liquids from waste handling facilities within the auxiliary building go to waste drain system floor drains. These floor drains are connected to separate "contaminated" sumps in the auxiliary building.

Postulated accidents involving the release of radioactivity from the fuel and waste storage and handling facilities are shown in Sub-Chapter 14.2 to result in exposures well within the guidelines of 10 CFR 100.

The refueling cavity, the refueling canal, the fuel transfer canal, and the spent fuel storage pool are reinforced concrete structures with a corrosion resistant liner. These structures have been designed to withstand loads due to postulated earthquakes. The fuel transfer tube, which connects the refueling canal and the fuel transfer canal which forms part of the reactor containment, is provided with a valve and a blind flange which closes off the fuel transfer tube when not in use.

1.4.9 EFFLUENTS

Gaseous, liquid and solid waste disposal facilities have been designed so that the discharge of effluents and off-site shipments are in accordance with applicable governmental regulations.

from experimental and analytical development programs into the core thermal design codes used to evaluate the loss-of-coolant accident.

This program has been completed. A preliminary evaluation of the loss-of-coolant accident utilizing the results of the Flashing Heat Transfer Program in the core thermal design code has been presented in Reference 18.

8. <u>Blowdown Forces Program (Item 15 in Reference 1)</u>

The objective of the program was to develop digital computer programs for the calculation of pressure, velocity, and force transients in the Reactor core and internals during a loss-of-coolant accident, and to utilize these codes in the calculation of blowdown forces on the fuel assemblies and reactor internals to assure that the stress and deflection criteria used in the design of these components are met.

Westinghouse has completed the development of BLODWN-2, an improved digital computer program for the calculation of local fluid pressure, flow and density transients in the Reactor Coolant System.

Extensive comparisons have been made between BLODWN-2 and available test data, and the results are given in Reference 19. Agreement between code predictions and data has been good.

An analysis using the BLODWN-2 Program has been applied to this plant. It was concluded from the analysis that the design of this reactor meets the established design criteria.

9. <u>Gross Failed Fuel Detector Program</u>

Since the Donald C. Cook Nuclear Plant will not use the <u>W</u> delay neutron failed fuel monitor, the <u>W</u> R & D on this monitor is no longer applicable.

A description of the Failed Fuel Detection System to be used at the Donald C. Cook Nuclear Plant is given in response to Question 1.5 (Unit 1, Appendix Q, Original FSAR).

10. <u>Reactor Vessel Thermal Shock (Item 16 in Reference 1)</u>

The effects of safety injection water on the integrity of the reactor vessel following a postulated loss-of-coolant accident, have been analyzed using data on fracture toughness of heavy section steel both at beginning of plant life and after irradiation corresponding to approximately 40 years of equivalent plant life. The results show that under the postulated accident conditions, the integrity of the reactor vessel is maintained.

Fracture toughness data is obtained from a Westinghouse experimental program which is associated with the Heavy Section Steel Technology (HSST) Program at ORNL and Euratom programs. Since results of the analyses are dependent on the fracture toughness of irradiated steel, efforts are continuing to obtain additional fracture toughness data. Data on two-inch thick specimens is expected in 1970 from the HSST Program. The HSST is scheduled for completion by 1973.

A detailed analysis considering the linear elastic fracture mechanism method, along with various sensitivity studies was submitted to the AEC Staff and members of the ACRS enlisted: "The Effects of Safety Injection On A Reactor Vessel And Its Internals Following A Loss Of Coolant Accident" (December, 1967), (Proprietary). Revised material for this report plus additional analysis and fracture toughness data was presented at a meeting with the Containment and Component Technology Branch on August 9, 1968, and forwarded by letter for AEC review and comment on October 29, 1968.

References, Section 1.6 (Continued)

15. D. D. Malinowski, "Iodine Removal in the Ice Condenser System,_" WCAP-7426, April 1970.

16. WCAP-7411-L, "Rod Bundle Axial Non-Uniform Heat Flux DNB Tests and Data", J. O. Cermak, (WNES Proprietary Class 2).

17. WCAP-7435, "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report".

18. WCAP-7422-L, "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident," J. Dorrycott (WNES Proprietary Class 2).

19. WCAP-7401, "Loss-of-Coolant Accident: Comparison Between BLODWN-2 Code Results and Test Data", S. Fabic, (WNES Proprietary Class 3).

20. WCAP-7544, "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group II Test Report, "F. F. Cadek, P. P. Pominicis, R. H. Legse, September 1970.

21. WCAP-8185, " Reference Core Report 17 x 17 Volume 1," April 1974.

22. WCAP-8288, "Safety Analysis of the 17 x 17 Fuel Assembly for a Combined Seismic and Loss-of-Coolant Accident," December 1973.

23. WCAP-8279, "Hydraulic Flow Tests of the 17 x 17 Fuel Assembly," February 1974.

24. WCAP-8297, "Effect of 17 x 17 Fuel Assembly Geometry on DNB," March 1974.

25. WCAP-8299, "The Effect of 17 x 17 Geometry on Interchannel Thermal Mixing," March 1974.

References, Section 1.6 (Continued)

26. WCAP-8449, "17 x 17 Drive Line Components Test - Phase JB, II, III, D-Loop - Drop and Deflection," December 1974.

27. WCAP-8692, "Fuel Rod Bowing," December 1975.

28. WCAP-8768, "Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries," June, 1976.

American Electric Power
Service Corporation

AMERICAN ELECTRIC POWER

STATEMENT OF POLICY
FOR THE DONALD C. COOK NUCLEAR PLANT
QUALITY ASSURANCE PROGRAM

POLICY

American Electric Power Company, Inc., recognizes the fundamental importance of controlling the design, modification, and operation of Indiana Michigan Power Company's Donald C. Cook Nuclear Plant (Cook Nuclear Plant) by implementing a planned and documented Quality Assurance Program, including Quality Control, that complies with applicable regulations, codes, and standards.

The Quality Assurance Program has been established to control activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant. The Quality Assurance Program supports the goal of maintaining the safety and reliability of the Cook Nuclear Plant at the highest level through a systematic program designed to assure that safety-related items are conducted in compliance with the applicable regulations, codes, standards, and established corporate policies and practices.

As President and Chief Executive Officer of American Electric Power Company, Inc., I maintain the ultimate responsibility for the Quality Assurance Program associated with the Cook Nuclear Plant. I have delegated functional responsibility for the Quality Assurance Program to the American Electric Power Service Corporation (AEPSC) Senior Executive Vice President-Engineering and Construction. He has, with my approval, delegated further responsibilities as outlined in this statement.

IMPLEMENTATION

The AEPSC Director-Quality Assurance, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been assigned the overall responsibility for specifying the Quality Assurance program requirements for the Cook

Nuclear Plant and verifying their implementation. The AEPSC Senior Executive Vice President-Engineering and Construction has given the AEPSC Director-Quality Assurance authority to stop work on any activity affecting safety-related items that does not meet applicable administrative, technical, and/or regulatory requirements. The AEPSC Director-Quality Assurance does not have the authority to stop unit operations, but shall notify appropriate plant and/or corporate management of conditions not meeting the aforementioned criteria and recommend that unit operations be terminated.

The AEPSC Vice President-Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been delegated responsibility for effectively implementing the Quality Assurance Program. The AEPSC Vice President-Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, except Quality Assurance, having a supporting role for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations.

The Plant Manager, under the direction of the AEPSC Vice President-Nuclear Operations, is delegated the responsibility for establishing the Cook Nuclear Plant Quality Control Program and implementing the Quality Assurance Program at the Cook Nuclear Plant.

The AEPSC Director-Quality Assurance is responsible for providing technical direction to the Plant Manager for matters relating to the Quality Assurance Program at the Cook Nuclear Plant. The AEPSC Director-Quality Assurance is also responsible for maintaining a Quality Assurance Section at the Cook Nuclear Plant to perform required reviews, audits, and surveillances, and to provide technical liaison services to the Plant Manager.

The implementation of the Quality Assurance Program is described in the AEPSC General Procedures (GPs) and subtier department/ division procedures, Plant Manager's Instructions (PMIs), and subtier Department Head Instructions and Procedures, which in total document the requirements for implementation of the Program.

Each AEPSC and Cook Nuclear Plant organization involved in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant has the responsibility to implement the applicable policies and requirements of the Quality Assurance Program. This responsibility includes being familiar with, and complying with, the requirements of the applicable Quality Assurance Program requirements.

COMPLIANCE

The AEPSC Director-Quality Assurance shall monitor compliance with the established Quality Assurance Program. Audit programs shall be established to ensure that AEPSC and Cook Nuclear Plant activities comply with established program requirements, identify deficiencies or noncompliances and obtain effective and timely corrective actions.

Employees engaged in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant who believe that the Quality Assurance Program is not being complied with, or that a deficiency in quality exists, should notify their supervisor, the AEPSC Director-Quality Assurance, and/or the Plant Manager. If the notification does not in the employee's opinion receive prompt or appropriate attention, the employee should contact successively higher levels of management. Employees reporting such conditions shall not be discriminated against by companies of the American Electric Power System. Discrimination includes discharge or other actions relative to compensation, terms, conditions, or privileges of employment.

R. E. Disbrow
President and
Chief Executive Officer
American Electric Power Company, Inc.

1.7.1 ORGANIZATION

1.7.1.1 SCOPE

American Electric Power Service Corporation (AEPSC) is responsible for establishing and implementing the Quality Assurance (QA) Program for the operational phase of the Donald C. Cook Nuclear Plant (Cook Nuclear Plant). Although authority for development and execution of various portions of the program may be delegated to others, such as contractors, agents or consultants, AEPSC retains overall responsibility. AEPSC shall evaluate work delegated to such organizations. Evaluations shall be based on the status of safety importance of the activity being performed and shall be initiated early enough to assure effective quality assurance during the performance of the delegated activity.

This section of the Quality Assurance Program Description (QAPD) identifies the AEPSC organizational responsibilities for activities affecting the quality of safety-related nuclear power plant structures, systems, and components, and describes the authority and duties assigned to them. It addresses responsibilities for both attaining quality objectives and for the functions of establishing the QA Program, and verifying that activities affecting the quality of safety-related items are performed effectively in accordance with QA Program requirements.

1.7.1.2 IMPLEMENTATION

1.7.1.2.1 <u>Source of Authority</u>

The President and Chief Executive Officer of American Electric Power Company, Inc. (AEP) and AEPSC is responsible for safe operation of the Cook Nuclear Plant. Authority and responsibility for effectively implementing the QA Program for plant modifications, operations and maintenance are delegated through the AEPSC Senior Executive Vice President - Engineering and Construction, to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations).

In the operation of a nuclear power plant, the licensee is required to establish clear and direct lines of responsibility, authority and accountability. This requirement is applicable to the organization providing support to the plant, as well as to the plant staff. |*

The AEPSC corporate support of the Cook Nuclear Plant is the responsibility of the entire organization under the direction of the Manager of Nuclear Operations who maintains primary responsibility for the Cook Nuclear Plant within the corporate organization. The AEPSC Vice President - Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, other than the Quality Assurance Division, having a supporting role for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations (reference Figure 1.7-1).

In order to facilitate a more thorough understanding of the support functions, some of the responsibilities, authorities, and accountabilities within the organization are as follows:

1) The responsibilities of the Manager of Nuclear Operations shall be dedicated to the area of Cook Nuclear Plant operations and support.

2) The Manager of Nuclear Operations shall be responsible for, and has the authority to direct, all Cook Nuclear Plant operational |* and support matters within the corporation and shall make, or |* concur, in all final decisions regarding significant nuclear |* safety matters.

3) AEPSC organization managers responsible for Cook Nuclear Plant matters shall be familiar with activities within their scope of responsibility that affect plant safety and reliability. They shall be cognizant of, and sensitive to, internal and external |* factors that might affect the operations of Cook Nuclear Plant.

4) AEPSC organization managers responsible for Cook Nuclear Plant matters have a commitment to seek and identify problem areas and take corrective action to eliminate unsafe conditions, or to improve trends that will upgrade plant safety and reliability.

5) The Manager of Nuclear Operations shall ensure that Cook Nuclear Plant personnel are not requested to perform inappropriate work or tasks by corporate personnel, and shall control assignments and requests that have the potential for diverting the attention of the Plant Manager from the primary responsibility for safe and reliable plant operation. |*

6) AEPSC division and department managers having Cook Nuclear Plant support responsibilities, as well as the Plant Manager and plant organization managers, shall be familiar with the policy statements from higher management concerning nuclear safety and operational priorities. They shall be responsible for ensuring that activities under their direction are performed in accordance with these policies. |* |*

1.7.1.2.2 <u>Responsibility for Attaining Quality Objectives in AEPSC Nuclear Operations</u>

The AEP President and Chief Executive Officer has delegated the functional responsibility of the Quality Assurance Program to the AEPSC Senior Executive Vice President - Engineering and Construction. |

The AEPSC Director - Quality Assurance, under the direction of the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for specifying QA Program requirements and verifying their implementation.

The AEPSC Vice President - Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for effectively implementing the QA Program.

The Plant Manager, under the direction of the AEPSC Vice President - Nuclear Operations, is responsible for establishing the Cook Nuclear Plant Quality Control Program and implementing the QA Program at the Cook Nuclear Plant.

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

AEPSC has an independent off-site Nuclear Safety and Design Review Committee (NSDRC) which has been established pursuant to the requirements of the Technical Specifications for the Cook Nuclear Plant. The function of the NSDRC is to oversee the engineering, design, operation, and maintenance of the Cook Nuclear Plant by performing audits and independent reviews of activities which are specified in the Facility Operating Licenses.

The Cook Nuclear Plant on-site review group is the Indiana Michigan Power Company (I&M) Plant Nuclear Safety Review Committee (PNSRC). This committee has also been established pursuant to the requirements of the Cook Nuclear Plant Technical Specifications. The function of the PNSRC is to review plant operations on a continuing basis and advise the Plant Manager on matters related to nuclear safety.

1.7.1.2.3 Corporate Organization

American Electric Power Company

AEP, the parent holding company, wholly owns the common stock of all AEP System subsidiary (operating) companies. The major operating companies and generation subsidiaries are shown in Figure 1.7-2. The President and Chief Executive Officer of AEP is the Chief Executive Officer of AEPSC and all operating 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.

American Electric Power Service Corporation

The responsibility for administrative and technical direction of the AEP System and its facilities is delegated to 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 functional responsibility is assigned to the operating companies. In the case of Cook Nuclear Plant, I&M general office staff (headquarters) provides public affairs, accounting, industrial safety direction and procurement support. |*

The Cook Nuclear Plant is owned and operated by I&M which is part of the AEP System.

1.7.1.2.4 Quality Assurance Responsibility of AEPSC

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

2) AEPSC furnishes quality assurance, engineering, design, construction, licensing, NRC correspondence, fuel management and radiological support activities.

3) AEPSC provides additional service in matters such as supplier qualification, procurement of original equipment and replacement parts, and the process of dedicating commercial grade items or services to safety-related applications. |*

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

5) Cognizant Engineer - (e.g., System Engineer, Equipment Engineer, Lead Engineer, Responsible Engineer, etc.) - The cognizant engineer, and/or engineer with the other titles noted, is that AEPSC individual who provides the engineering/design expertise for a particular area of responsibility. This responsibility includes the implementation of the quality assurance and quality control measures for systems, equipment, structures, or functional areas included in that individual's responsibility. The various titles used for the identification of an individual's responsibility and assignment shall be understood to mean the same as cognizant engineer in the respective areas of responsibility.

Quality Assurance Responsibility of I&M - Cook Nuclear Plant

I&M's Cook Nuclear Plant staff operates the Cook Nuclear Plant in accordance with 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 Nuclear Plant.

1.7.1.2.5 Organization (AEPSC)

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

- AEPSC Director - Quality Assurance
- AEPSC Vice President - Nuclear Operations
- AEPSC Senior Vice President and Chief Engineer
- AEPSC Vice President - Project Management and Construction

Quality Assurance Division

The AEPSC Director - Quality Assurance, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Quality Assurance Division (QAD). The QAD consists of the following sections (Figure 1.7-4):

- Quality Assurance Engineering Section
- Nuclear Software Quality Assurance Section
- Audits and Procurement Section

- Quality Assurance Support Section
- Quality Assurance Section (Site)

The QAD is organizationally independent and is responsible to perform the following:

- Specify QA Program requirements.
- Identify quality problems.
- Initiate, recommend, or provide solutions through designated channels.
- Verify implementation of solutions, as appropriate.
- Prepare, issue and maintain QA Program documents, as required.
- Verify the implementation of the QA Program through scheduled audits and surveillances.
- Verify the implementation of computer software quality assurance through reviews, surveillances and audits.
- Audit 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 auditor orientation, training, certification and qualification of AEPSC audit personnel.
- Provide direction for the collection, storage, maintenance, and retention of quality assurance records.
- Maintain, on data base, a list of suppliers of nuclear (N) items and services, plus other selected categories of suppliers.
- 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.
- Review the disposition of conditions adverse to quality to assure that action taken will preclude recurrence.
- Conduct in-process QA audits or surveillances at supplier's facilities, as required. |

- Assist and advise other AEP/AEPSC groups in matters related to the QA Program.
- Conduct audits as directed by the NSDRC.
- Review AEPSC investigated Problem Reports and associated corrective and preventive action recommendations.
- Maintain cognizance of industry and governmental quality assurance requirements such that the QA Program is compatible with requirements, as necessary.
- Recommend for revision to, or improvements in, the established QA Program to senior AEPSC management. |*
- Audit dedication plans for commercial grade items and services.
- Issue "Stop Work" orders when significant conditions adverse to safety-related items 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.
- Routinely attend, and participate 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.
- Establish and maintain a central file for equipment environmental qualification documentation.
- Determine the acceptability of vendors to supply products and services for safety related applications. |

<u>Amplification of Specific Responsibilities</u>

- <u>Qualification of the AEPSC Director - Quality Assurance</u>

  The AEPSC Director - Quality Assurance shall possess the following position requirements:

  - Bachelor's degree in engineering, scientific, or related discipline. |*
  - Ten (10) years experience in one of, or a combination of, the following areas: engineering, design, |*

construction, operations, maintenance of fossil or nuclear power generation facilities' or utility facilities' QA, of which at least four (4) years must be experience in nuclear 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 the quality of safety-related items, 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.

- <u>Stop Work Orders</u>

The AEPSC QAD is responsible for ensuring that activities affecting the quality of safety-related items are performed in a manner that meets applicable administrative, technical, and regulatory requirements. In order to carry out this responsibility, the AEPSC Senior Executive Vice President - Engineering and Construction has given the AEPSC Director - Quality Assurance the authority to stop work on any activity affecting the quality of safety-related items that does not meet the aforementioned requirements. Stop work authority has been further delegated by the AEPSC Director - Quality Assurance to the AEPSC Quality Assurance Superintendent (site). |*

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

- QA Auditor, Qualification and Certification Program

- AEPSC has established and maintains a QA auditor training and certification program for all AEPSC QA auditors.

- Problem Identification, Reporting and Escalation

- AEPSC has established mechanisms for the identification, reporting and escalation of problems affecting the quality of safety-related items to a level of management whereby satisfactory resolutions can be obtained.

Nuclear Operations Division

The AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations), reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Nuclear Operations Division (NOD).

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

NOD is responsible for the following:
- Formulate policies and practices relative to safety, licensing, operation, maintenance, fuel management, and radiological support.

- Provide the Plant Manager with the technical and managerial guidance, direction and support to ensure the safe operation of the plant.
- Provide direction to all other AEPSC engineering and design organizations on engineering and design matters pertaining to the Cook Nuclear Plant.
- Maintain liaison with the AEPSC Director - Quality Assurance.
- Implement the requirements of the AEPSC QA Program.
- Maintain knowledge of the latest safety, licensing, and regulatory requirements, codes, standards, and federal regulations applicable to the operation of Cook Nuclear Plant. |*
- Accomplish the procurement, economic, technical, licensing and quality assurance activities dealing with the reactor core and its related fuel assemblies and components.
- Prepare bid specifications, evaluate bids, and negotiate and administer contracts for the procurement of all nuclear fuel and related components and services.
- Maintain a special nuclear material accountability system.
- Provide analyses to support nuclear steam supply system operation, |* including reactor physics, fuel economics, fuel mechanical behavior, core thermal hydraulic and LOCA and non-LOCA transient safety analysis and other analysis activities as requested, furnish plant Technical Specification changes and other licensing work, and participate in NRC and NSDRC meetings as required by these analyses.
- Perform reactor core operation follow-up activities and other reactor core technical support activities as requested, and arrange for support from the fuel fabricator, when needed. |*
- Contract for, and provide technical support for, disposal of both high level and low level radioactive waste.
- Coordinate the development of neutronics and thermal hydraulic safety codes and conduct safety analyses.
- Conduct studies of the Cook Nuclear Plant licensing bases to determine the optimal changes to support unit operations at a lower primary pressure and temperature.

- Coordinate NOD computer code development, and provide the interface control for NOD with the AEPSC Information System Department and Cook Nuclear Plant. |*
- Obtain and maintain the NRC Operating License and Technical Specifications for the Cook Nuclear Plant.
- Act as the communication link between the NRC, AEPSC, and the plant staff.
- Perform and coordinate efforts involved in gathering information, performing calculations and generic studies; preparing criteria, |* reports, and responses; reviewing items affecting safety; and |* interpreting regulations.
- Review, coordinate, and resolve all matters pertaining to nuclear safety between Cook Nuclear Plant and AEPSC. This includes, but is not limited to: the review of certain plant design changes to ensure that the requirements of 10CFR50.59 are met; the preparation of safety evaluations, or reviews, for any designated |* subject; the preparation of changes to, and appropriate interpretation of, the plant Technical Specification submittals of license amendments; and the analysis of plant compliance with regulatory requirements.
- Primary corporate contact for most oral and written communication with the NRC.
- Provide support in key areas of expertise, such as nuclear |* engineering, probabilistic analysis, thermohydraulic analysis, chemical engineering, mechanical engineering, electrical engineering, and technical writing.
- Interface with vendors and other outside organizations on matters connected with the nuclear steam supply system and other areas affecting the safe design and operation of nuclear plants.
- Participate, as appropriate, in the review of nuclear plant |* operating experiences, and relate those experiences to the design and safe operation of Cook Nuclear Plant.

- Review, evaluate, and respond to NRC requests for information and NRC notifications of regulatory changes resulting in plant modifications or new facilities. Such responses are generated in accordance with appropriate AEPSC Administrative Procedures.
- Develop, specify, and/or review conceptual nuclear safety criteria for Cook Nuclear Plant in accordance with established regulations. |* This includes all information contained in the FSAR, as well as specialized information such as environmental qualification and seismic criteria.
- Review and evaluate performance requirements for systems, equipment and materials for compliance with specified safety criteria.
- Review, on a conceptual basis, plant reports and proposed plant safety-related design changes, to the extent that they are related to the ultimate safe operation of the plant, for compliance with safety regulations, plant Technical Specifications, the Updated | FSAR design basis, and with any other requirements under the Operating License, to determine if there are any unreviewed safety |* questions as defined in 10CFR50.59.
- Perform reviews of Problem Reports and 10CFR21 reviews in accordance with corporate requirements.
- Operate the Action Item Tracking System (AIT) for AEPSC internal commitment tracking.
- Coordinate design changes for the Cook Nuclear Plant, acting as a |* focal point within AEPSC. This program primarily involves project management responsibilities for scheduling and implementing Request for Changes (RFCs), and includes extensive interfacing |* with engineering, design, construction, and Cook Nuclear Plant.
- Provide working-level coordination with the Institute of Nuclear |* Power Operations (INPO). This effort includes providing AEPSC access to INPO resources, such as NUCLEAR NETWORK and Nuclear |* Plant Reliability Data System (NPRDS), and effectively integrating AEPSC and Cook Nuclear Plant efforts towards utilizing INPO recommendations contained in operating experience reports to improve Cook Nuclear Plant performance.

- Coordinate daily communication with the Cook Nuclear Plant, provide AEPSC management with a daily plant status report, and make presentations to senior management at regularly scheduled construction staff meetings.
- Process incoming vendor information.
- Coordinate operations within AEPSC that support the Cook Nuclear Plant Facility Data Base (FDB).
- Contribute to the annual FSAR updates through reviews of Licensee Event Reports, design changes and the Annual Operating Report. |*
- Radiological, emergency and security planning.
- Corporate support of the Cook Nuclear Plant's radiation protection and health physics program, technical service and advice on the radiological aspects of design changes, modifications or capital improvements, the ALARA program, the radiation monitoring system, the environmental radiological monitoring and sampling program, dose and shielding analysis, radiochemistry review, and meteorological monitoring.
- Cook Nuclear Plant and corporate emergency planning, including procedure development, exercise scheduling, facility procurement and maintenance, and the liaison with off-site emergency planning groups, such as FEMA and the Michigan State Police. |*
- Review federal codes and regulations as they relate to the development, implementation, revision and distribution of the Modified Amended Security Plan (MASP).
- Interface with the plant's security department providing support for the security plan, reviewing security facilities, maintaining security document files, and developing the employee fitness for duty/background screening program.
- Provide Nuclear General Employee Training (NGET) for AEPSC personnel.
- Coordinate the development of training for AEPSC personnel who support the operation and maintenance of Cook Nuclear Plant, ensuring a unified training program meeting annual goals and objectives.
- Participate on ALARA subcommittees.

- Prepare responses to the NRC on radiological, emergency planning and security issues.
- Serve as technical advisors on plant audits.
- Remain cognizant of current decommissioning practices and developments.

AEPSC Engineering and Design

The AEPSC Senior Vice President and Chief Engineer, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for certain engineering and design functions through the AEPSC Assistant Vice President - Civil Engineering, the AEPSC Assistant Vice President - Design and the AEPSC Assistant Vice President - Nuclear Engineering.

The AEPSC Electrical Engineering, and System Planning Departments provide periodic, technical assistance for the Cook Nuclear Plant. The administrative and quality assurance controls for this assistance, along with that provided by the Civil Engineering Department, are controlled through documented interface agreements with the AEPSC Nuclear Engineering Department.

Civil Engineering Department

The AEPSC Assistant Vice President - Civil Engineering, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Civil Engineering Department.

The Civil Engineering Department (CED) is responsible for the technical aspects for the following:

- Make recommendations and assist in the formulation of policies and practices relating to the design and engineering of office and service buildings, miscellaneous structures and material handling equipment, and provide the general supervision of the engineering of such facilities, structures and equipment.

- Review the activities of equipment, facilities, buildings and other structures at the Cook Nuclear Plant and approve, as required, all design changes and modifications, including the preparation of specifications; and procurement of, and modifications to, equipment.
- Provide training and development programs necessary for personnel of the department, (including the company's safety and health program), which are consistent with the written policies of AEPSC.
- Prepare design criteria, engineering standards, conceptual layouts, studies and procedures in conjunction with equipment, facilities, buildings and other structures at the Cook Nuclear Plant.
- Identify critical engineering and design input, and ensure that appropriate analysis and reviews are conducted. |*
- Prepare, review and approve specifications, sketches, drawings, design verifications and calculations, as required.
- Provide input for special studies and reports which may be requested by other organizations or governmental agencies such as the NRC.
- Initiate and/or review, approve and control laboratory and field investigations and feasibility studies.
- Prepare and review improvement requisitions for capital and lease expenditures.
- Review and evaluate proposals and make recommendations for the award of purchase orders and contracts.
- Prepare and administer equipment, labor and service contracts.
- Provide technical guidance, when requested, in support of activities at the Cook Nuclear Plant under the department's responsibilities. |* |*
- Prepare and approve design changes pertaining to Cook Nuclear Plant in accordance with the GPs.
- Arrange for outside engineering and consulting assistance, as required. |*

- Arbitrate disputes which arise between construction forces and outside suppliers of materials and services.
- Coordinate consultant's reports with other interfacing engineering organizations.
- Perform shop and field inspections on equipment being fabricated, or installed, which is within the scope of the department's responsibility. |* |*
- Approve invoices for outside services.
- Provide field services to the Cook Nuclear Plant, including the assigning of personnel, as are required, during construction, normal or emergency outages, or as requested. |* |*
- Assist in the planning and execution of maintenance work on equipment, facilities, buildings and other structures.
- Supervise maintenance and repairs of all masonry and concrete work at Cook Nuclear Plant, including supplying trained inspection personnel.
- Direct testing of materials used in concrete and testing of soils to be used in work at the Cook Nuclear Plant.
- Review and recommend concrete mix formulations for all new construction.
- Prepare site studies.
- Implement a corrective action system, with regard to all activities of the department affecting quality of safety-related items, that will control and document all items, services or activities which do not conform to requirements.
- Direct the review of, and response to, assigned corrective actions.
- Assist in the preparation of applications for federal, state and local permits relative to installations being made which require such permits.

- Conduct periodic management reviews of the activities of the department to ensure compliance with the objectives of the QA Program, and external technical surveillance, as necessary, of consultants, outside organizations and vendors over which the department is cognizant.
- Establish and maintain a permanent file for QA records.

<u>Design Department</u>

The AEPSC Assistant Vice President - Design, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Design Department. |*

The Design Department is responsible for the following:

- Develop, review and approve designs and drawings for mechanical, electrical and structural systems, equipment and facilities of the Cook Nuclear Plant.
- Initiate, develop, approve and maintain design procedures, specifications, standards, criteria and guidelines.
- Perform required calculations and analyses, including pipe stress, pipe support design, cable sizing, conduit and cable tray support and structural steel and concrete. |*
- Initiate and develop design changes in the areas of responsibility of the Design Department.
- Provide NRC responses, as required. |*
- Assist field personnel in the resolution of problems stemming from the installation of design changes, or from as-found plant conditions, including assigning design personnel to the field. |*
- Participate, as assigned, on the NSDRC and NSDRC subcommittees, and participate in matters covered in the committee's charter.
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization, including resource allocation. |*
- Formulate, administer, and implement policies and practices relating to the design of the Cook Nuclear Plant.

- Direct the development, maintenance, procedural review and implementation by which the Design Department adheres to the QA Program elements as established by AEPSC General Procedures.
- Conduct functions of the department so as to be in conformance with the operating licenses of the Cook Nuclear Plant.
- Investigate and evaluate problems.
- Coordinate special projects and studies, as required.
- Establish and maintain files of design documents for record purposes.
- 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 Nuclear Plant, including monitoring of the Aperture Card Microfilm Program.
- Supervise and control the work of consultants, architect/engineers and outside design agencies supplying services to AEPSC in their discipline and process notification of defects in accordance with company requirements. Also perform detailed reviews of design work submitted by outside agencies. |*
- Provide input to the list of major approved materials, and maintain current specifications used within the group's scope of responsibility. |*
- Provide engineering and design support to NOD.
- Review and update applicable sections of Cook Nuclear Plant Updated FSAR as assigned. |
- Participate on committees that review nuclear activities as members, when 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.

Nuclear Engineering Department

The AEPSC Assistant Vice President - Nuclear Engineering, reporting to the AEPSC Senior Vice President and Chief Engineer, is responsible for the Nuclear Engineering Department.

The Nuclear Engineering Department (NED) is responsible for the following:

- Provide planning and engineering, in conjunction with other specialists, sections, and divisions, of the electrical facilities inside Cook Nuclear Plant up to the high voltage (HV) bushings of the main generator transformers and mechanical facilities inside Cook Nuclear Plant including:

  * determination of general layout and design;
  * selection of equipment;
  * preparation of one-line and flow diagrams; and,
  * coordination of inside and outside plant facilities.

- Provide engineering and design of all controls for operation and protection of nuclear steam supply, steam generator, turbine generator, auxiliary equipment and general plant protection, including checking and approving elementary, one-line, and flow drawings.
- Interface with other organizations to ensure that all purchased equipment conforms to accepted standards and fulfills the desired function.
- Closely follow manufacturer's engineering and design processes to assure provision of adequate and reliable equipment upon which depend the safety, reliability, economy, and performance of the unit and plant. |*
- Prepare cost estimates and improvement requisitions for plant facilities, including review of improvement requisitions and cost estimates prepared by others.

- Prepare and/or approve specifications and purchase requisitions, and perform drawing review of equipment, as appropriate. |*
- Review and approve procedures, correspondence, Requests for Design Changes or modifications, as appropriate. |*
- Obtain, review and perform engineering evaluations, including environmental equipment qualification (EQ). |*
- Perform calculations for proper application of equipment.
- Perform and evaluate economic studies, investigations, analyses and reports for facilities pertaining to the design, operation and maintenance of the Cook Nuclear Plant.
- Assist field personnel in installation, start-up, and subsequent locating of problems in equipment, and in determining proper operation of equipment during normal, or after, emergency operations. |*
- Maintain a constant awareness for improvements and more reliable design of equipment and facilities, maintenance and operating methods or procedures.
- Maintain a constant awareness of activities to ensure compliance with all applicable policies and procedures, initiating, when required, training or retraining programs. |*
- Participate, as assigned, on the NSDRC and NSDRC subcommittees, and participate in matters covered in the committee's charter. |*
- Provide responses to NRC correspondence, as required. |*
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization.
- Provide technical engineering support in areas of operation and maintenance, including: the Inservice Inspection (ISI) Program; the QA Program; the Fire Protection QA Program; the AEP ALARA Program covering radiation protection; and, the corporate and plant Industrial Safety program.
- Provide engineering support to the AEPSC NOD.
- 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.

- Initiate and develop design changes in areas of responsibility of the NED.
- Develop System Descriptions.
- Provide support personnel for the emergency response organization.
- Provide analytical support in engineering disciplines (e.g., heat transfer, thermodynamics, fluid dynamics).
- Provide engineering evaluations for PRs, LERs, INPO SOERs, and NRC Bulletins.
- Participate, as assigned, on the AEPSC Problem Assessment Group (PAG).

Project Management and Construction Department

The AEPSC Vice President - Project Management and Construction, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Project Management and Construction Department.

Reporting to the AEPSC Vice President - Project Management and Construction are the following:

- Site Construction Manager, reporting administratively to the AEPSC Vice President - Project Management and Construction, and functionally to the Cook Nuclear Plant, Plant Manager. |* |* |

The Project Management and Construction Department is responsible for the following:

- Administer and implement construction job orders issued by the Cook Nuclear Plant organization for major modifications, replacement and maintenance work with outside contractors. |* |*
- Administer and monitor contractor's industrial safety programs and performance. |*
- Administer human resources' functions for site construction organization. |*

- Manage construction labor relations with the International Building and Construction Trades Unions. |*
- Scope, bid, recommend awards and administer construction labor and services contracts. |*
- Plan, organize and control major construction projects, as assigned by the AEPSC Senior Executive Vice President - Engineering and Construction. |*
- Maintain cognizance on matters pertaining to the Cook Nuclear Plant and corporate emergency response organization. |*
- Prepare of construction labor estimates. |*
- Provide constructability guidance when requested in support of engineering and design changes. |*
- Participate on the Nuclear Safety Design Review Committee. |

<u>Purchasing and Stores Department</u> (not charted)

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

The Purchasing and Stores Department is responsible for the following:

- Procurement of "N" items from only qualified and approved suppliers.
- Provide supervision to Cook Nuclear Plant Purchasing Section. |
- Provide ordering and stocking descriptions of "N" items and include these descriptions in the Cook Nuclear Plant inventory catalog, including necessary communications with suppliers, cognizant engineers, the Cook Nuclear Plant Stores Supervisor and other appropriate personnel. |*

- Coordinate procurement activities with AEPSC Nuclear Operations, AEPSC engineering and design divisions/departments, Cook Plant Site Purchasing Section, the AEPSC QAD and Cook Nuclear 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 Nuclear Plant.
- Establish a system of document keeping and transmittal. |*
- Establish a system of document control for controlled procedures, instructions, and purchasing documents for "N" items.
- The maintenance and control of selected standard procurement document phrases as identified by the Director - Quality Assurance, or designee.
- 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.

1.7.1.2.6 Organization (Cook Nuclear Plant)

The Plant Manager reports functionally and administratively to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) and is responsible for the Cook Nuclear Plant activities.
Reporting to the Plant Manager are the following (Figure 1.7-5):

- Assistant Plant Manager - Production
- Assistant Plant Manager - Technical Support
- Assistant Plant Manager - Projects
- Licensing Activity Coordinator
- Safety and Assessment Superintendent

- Radiation Protection Manager (reports functionally to the Plant Manager)
- Nuclear Security Manager (reports functionally to the Plant Manager)

The Cook Nuclear 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 PNSRC.
- Maintain close working relationships with the NRC, as well as local, state, and federal government regulatory officials regarding conditions which could affect, or are affected, by Cook Nuclear Plant activities. |* |* |*
- Set up plant load schedules and arrange for equipment outages.
- Develop and efficiently implement all site centralized training activities.
- Administer the centralized facility training complex, simulator, and programs ensuring that program development is consistent with the systematic approach to training, maintain INPO accreditations, regulatory and corporate requirements.
- Ensure that human resource activities include employee support programs (i.e., fitness for duty) 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 (MASP), and company policy.

- Supervise, plan, and direct the activities related to the maintenance and installation of all Cook Nuclear Plant equipment, structures, grounds, and yards.
- Prepare and maintain records and reports pertinent to equipment maintenance and regulatory agency requirements.
- Administer contracts and schedule outside contractors' work forces.
- Enforce and coordinate Cook Nuclear Plant regulations, procedures, policies, and objectives to assure safety, efficiency, and continuity in the operation of the Cook Nuclear Plant within the limits of the operating license and the Technical Specifications and formulation of related policies and procedures.
- Plan, schedule, and direct activities relating to the operation of the Cook Nuclear Plant and associated switchyards; cooperate in planning and scheduling of work and procedures for refueling and maintenance of the Cook Nuclear Plant; and direct and coordinate fuel loading operations. |*
- Review reports and records, 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 Cook Nuclear Plant safely and efficiently. |*
- Approve operator training programs administered by the Cook Nuclear 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 Nuclear 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 surveillance, and design change installations.
- Coordinate all Cook Nuclear 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 (QC) Program. This encompasses, but is not limited to, the planning and directing of quality control activities to assure that industry codes, NRC regulations, and company instructions and policies regarding quality control for Cook Nuclear Plant are implemented, qualified personnel perform the work, and that these activities are properly documented. |*
- Prepare reports of reportable events which are mandated by the NRC and the Technical Specifications.
- Direct the activities of contractor QC/nondestructive examination (NDE) personnel assigned to the Safety and Assessment Department and provide inspections of work performed.
- Prepare statistical reports utilized in NRC Appraisal Meetings and Enforcement Conference.
- Coordinate the efforts of outside agencies, such as American Nuclear Insurers (ANI), 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.
- Maintain and renew accreditation of training programs.

- Qualification and certification of I&M personnel performing inspections or tests of major modifications and non-routine maintenance to the requirements of Regulatory Guide 1.58 and ANSI N45.2.6, except as noted in Appendix B hereto, item 9.
- Qualification and certification of I&M NDE personnel to the requirements of the AEP NDE Manual.
- Qualification of I&M personnel performing inspection of normal operating activities to ANSI N18.1.
- Proper certification of contractor inspection, test and examination personnel in accordance with Regulatory Guide 1.58, ANSI N45.2.6, ASME B&PV Code and/or SNT-TC-1A, as applicable.
- Perform peer inspections of work completed by I&M personnel by independent persons qualified to ANSI N18.7.
- Conduct of the Inservice Inspection (ISI) Program.
- Procurement, receiving, quality control receipt inspection, storage, handling, issue, stock level maintenance, and overall control of stores items.
- Provide material service and support in accordance with policies and procedures required by AEP Purchasing and Stores, AEPSC QA, and the 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 Nuclear 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 Cook Nuclear 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 Cook Nuclear 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 Technical Physical Science Sections of the Cook Nuclear 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 Cook Nuclear Plant's radioactive waste programs, including volume reduction, packaging and shipping.
- Administration of the QA Records Program.
- Maintain the Cook Nuclear Plant Facility Data Base.

1.7.2 QUALITY ASSURANCE PROGRAM

1.7.2.1 SCOPE

Policies that define and establish the Cook Nuclear Plant QA 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 Cook Nuclear Plant.

Quality assurance controls apply to activities affecting the quality of safety-related structures, systems and components to an extent based on the importanceof those structures, systems, components, etc., (items) 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 items 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 Updated FSAR; or lead to a release of radioactivity in excess of 10CFR100 guidelines. |*

In general, items are classified as safety-related if they are: Seismic Class I, or Electrical Class 1E; or associated with the Engineered Safety Features Actuation System (ESFAS); or associated with the Reactor Protection System (RPS).

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

The QA Program also includes provision for Radwaste QA in accordance with the requirements of 10CFR71, part H.

QA 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 QA Program may be accomplished by AEPSC and/or Indiana Michigan Power Company or delegated in whole or in part to other AEP System companies or outside parties. However, AEPSC and/or Indiana Michigan Power Company retain full responsibility for all activities affecting safety-related items. 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 Chief Executive Officer of AEPSC has stated in a signed, formal "Statement of Policy", that it is the 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 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. The statement also 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 QA Program at AEPSC and the Cook Nuclear 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 |* shall also be periodically reviewed by independent outside parties as deemed necessary by management.

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

The QA Program applies to activities affecting the quality of safety-related structures, components, and related consumables during plant operation, maintenance, testing, and all design changes. Safety-related structures, systems and components are identified in the Facility Data Base and other documents which are developed and maintained for the plant.

As deemed necessary by the AEPSC Director - Quality Assurance, or the Plant Manager, applicable portions of the QA Program controls will be applied to nonsafety-related activities associated with the implementation of the QA Program to ensure that commitments are met (e.g., off-site records storage, training services, etc.).

1.7.2.2.3

This QAPD, organized to present the QA Program for the 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 intentionally 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 at the plant for compliance with specified criteria, and assign responsibility to the various departments, as required, for implementation. When necessary, Department Head Procedures (DHPs), and in some cases Department Head Instructions (DHIs), have been prepared to describe the detailed activities required to support safe and effective plant operation as per the PMIs.

The PMIs are reviewed by AEPSC QA for concurrence that they will satisfactorily implement regulatory requirements and commitments. They are then reviewed by the PNSRC prior to approval by the Plant Manager.

Safety-related DHPs and DHIs are reviewed by the department head of origination, 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 certain corporate QA Program requirements. AEPSC division/department and/or section procedures are also used to implement QA Program requirements.

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

Nuclear Engineering Procedures (NEPs) establish the policy in AEPSC Nuclear Engineering Department for compliance with the AEPSC GPs and assign responsibility to NED personnel, as required, for implementation. Nuclear Engineering Section Procedures (NESPs) are issued to satisfy a specialized GP requirement.

The NEPs include a procedure for the development and maintenance of procedures. This procedure contain the reviews and approvals necessary to satisfactorily implement regulatory requirements and commitments.

Other procedures used at AEPSC to implement QA Program requirements include Nuclear Design Procedures (NDPs), Nuclear Operations Department Procedures (NODs), and the Civil Engineering Organization and Procedures Manual.

When contractors perform work on-site under their own quality assurance programs, the programs are audited for compliance and consistency with the applicable requirements of the Cook Nuclear Plant's QA Program and the contract, and are approved by AEPSC QA prior to the start of work. Implementation of on-site contractor's QA programs, will be audited to assure that the contractor's programs are effective.

1.7.2.2.5

Provisions of the QA Program for the Cook Nuclear Plant apply to activities affecting the quality of safety-related items. Appendix A to this QAPD lists the Regulatory/Safety Guides and ANSI Standards 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 Cook Nuclear Plant Updated FSAR for identifying structures, systems and components to which the QA Program applies.

b) This identification process results in the Facility Data Base for the Cook Nuclear Plant. This Facility Data Base is controlled by authorized personnel. Facility Data Base items are determined by engineering analysis of the function(s) of plant items in relation to safe operation and shutdown.

c) The extent to which controls specified in the QA Program are applied to Facility Data Base items is determined for each item considering its relative importance to safety. Such determinations are based on data in such documents as the Cook Nuclear Plant Technical Specifications and the Updated FSAR.

1.7.2.2.6

Activities affecting safety-related items 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) Items are in an acceptable status.
e) Items 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 of AEPSC and Cook Nuclear Plant staff personnel are specifically designated, as follows:

a) The training and indoctrination program provides for on-going training and periodic familiarization with the QA Program for the Cook Nuclear Plant.

b) Personnel who perform inspection and examination functions are qualified in accordance with requirements of Regulatory Guide 1.8, ANSI N18.1, Regulatory Guide 1.58, ANSI N45.2.6, the ASME B&PV Code, or SNT-TC-1A, as applicable, and with exceptions as noted in Appendix B hereto. |*

c) AEPSC QAD auditors are qualified in accordance with Regulatory Guide 1.146 and ANSI N45.2.23.

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

e) The training, qualification and certification program includes, as applicable, provisions for retraining, reexamination and recertification to ensure that proficiency is maintained.

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

g) Personnel responsible for performing activities that affect safety-related items are instructed as to the purpose, scope and implementation of the applicable 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 Cook Nuclear Plant personnel is performed employing the following techniques, as applicable: 1) on the job and formal training administered by the department or section the individual works for; 2) formal training conducted by qualified instructors from the Cook Nuclear Plant Training Department or other entities (internal and external to the AEP System); and 3) formal, INPO accredited training conducted by the Cook Nuclear Plant Training Department. 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).

Cook Nuclear 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 QA Program through Plant Manager and Department Head Instructions and Procedures. These instructions also include a description of the QA Program, the use of instructions and procedures, personnel requirements for procedure compliance and the systems and components controlled by the QA Program.

1.7.3 DESIGN CONTROL

1.7.3.1 SCOPE

Design changes are accomplished in accordance with approved design. Activities to develop such designs are controlled. Depending on the type of design change, these activities include design and field engineering; the performance of physics, seismic, stress, thermal, hydraulic and radiation evaluations; update of the FSAR; review of accident analyses; the development and control of associated computer programs; studies of material compatibility; accessibility for inservice inspection and maintenance; determination of quality standards; and requirement for equipment qualification. 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

Design changes are controlled by procedures and instructions and are reviewed as required by 10CFR50.59.

Safety-related design changes are accomplished by one of two separate processes: Minor Modification (MM), or Request for Change (RFC). Those that do not alter the intended function of the item and can be determined by judgement to have a minimal overall impact on the item being modified may be implemented via the MM process. All other safety-related design changes, that are not appropriate for MM processing, are implemented via the RFC process. |*

In cases where design changes could be deemed to be within the scope of RFCs or MMs solely due to possible insignificant adverse seismic effects, the change may be implemented via the Plant Modification (PM) process.

1.7.3.2.2

RFCs (except those requiring emergency processing), MMs and PMs (having only insignificant seismic effect on safety items) are reviewed and approved prior to implementation, as a minimum, by the cognizant AEPSC engineering section, PNSRC, and Plant Manager.

RFCs and MMs are reviewed to determine their impact on nuclear safety and to determine if the proposed changes involve an unreviewed safety question as defined by 10CFR50.59. If a design change were to involve an unreviewed safety question, it would not be approved for implementation until the required NRC approval was received.

1.7.3.2.3

For RFCs, the Change Control Board established within AEPSC provides an additional review and approval level. The Change Control Board is comprised of members of the Engineering, Design, Nuclear Operations and QA organizations within AEPSC, and is supplemented by other AEPSC organizations or individuals, as required. |*

The cognizant member of the Change Control Board assigns a lead engineer for each RFC. The lead engineer is responsible for coordinating the RFC activities within AEPSC and maintaining close interface with the Cook Nuclear Plant Project Engineers.

1.7.3.2.4

Proposed RFCs which require emergency processing are originated at the plant, reviewed by the PNSRC, and approved by the Plant Manager. Cook Nuclear Plant management then contacts the AEPSC NOD, and other AEPSC management, as required, describes the change requested, and implements the change only after receiving verbal AEPSC management authorization to proceed. These reviews and approvals are documented and become a part of the RFC Packet. |*

1.7.3.2.5

When RFCs or MMs involve design interfaces between internal or external design organizations, or across technical disciplines, these interfaces are controlled. Procedures are used for the review, approval, release, distribution and revision of documents involving design interfaces to ensure that structures, systems and components are compatible geometrically and functionally with processes and the environment. Lines of communication are established for controlling the flow of needed design information across design interfaces, including changes to the |*

information as work progresses. Decisions and problem resolutions involving design interfaces are made by the AEPSC organization having responsibility for engineering direction of the design effort.

1.7.3.2.6

Checks are performed and documented to verify the dimensional accuracy and completeness of design drawings and specifications.

1.7.3.2.7

RFC design document packages are audited by AEPSC QA to assure that the documents have been prepared, verified, reviewed and approved in accordance with company procedures.

1.7.3.2.8

The extent of, and methods for, design verification are documented. The extent of design verification performed is a function of the importance of the item to safety, design complexity, degree of standardization, the state-of-the-art, and similarity with previously proven designs. Methods for design verification include evaluation of the applicability of standardized or previously proven designs, alternate calculations, qualification testing and design reviews. These methods may be used singly or in combination, depending on the needs for the design under consideration. |*

When design verification is done by evaluating standardized or previously proven designs, the applicability of such designs is confirmed. Any 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. Design changes 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.

Procedures require that minor design changes accomplished by the MM process also receive formal design verification. Applicable design verification activities shall be completed prior to declaring the design change, or portion thereof, operational. |*

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.

<u>or</u>

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

<u>and</u>

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

<u>and</u>

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

Design verification on safety-related design changes shall be completed prior to declaring a design change, or portions thereof, operational. |*

1.7.3.2.10

Cook Nuclear Plant implementation of design changes is accomplished by the Cook Nuclear Plant Project Engineering Department. 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 other cognizant engineer(s). Further, completed design changes are audited by AEPSC QA 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

Established design control procedures provide for: |

1) controlled submission of design changes,
2) engineering evaluation,
3) review for impact on nuclear safety,
4) audit 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 QA Program requirements to the extent necessary to assure adequate quality. |*

1.7.4.2 IMPLEMENTATION

1.7.4.2.1

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 (except when the Code of Federal Regulations requires upgrading), 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, or by the dedication plan process.

Procedures establish the review of procurement documents to determine that: quality requirements are correctly stated, inspectable and controllable; there are adequate acceptance criteria; and procurement documents have been prepared, reviewed and approved in accordance with established requirements. *

The manager of the originating group, with support of the cognizant AEPSC engineering group, is responsible for assuring that applicable requirements are set forth in procurement documents.

The Cook Nuclear Plant may request assistance of AEPSC cognizant engineers in any procurement activity.

1.7.4.2.2

The Facility Data Base, in conjunction with other sources, is used for equipment safety classification and procurement grade. AEPSC specifications are used to determine requirements, codes or standards that items must fulfill, and define the documentation that must accompany the item to the plant.

Procurement documents for safety related items and services are reviewed to ensure that: correct classification is made; the requirements are properly stated; and that measures have been, or will be, implemented to assure the requirements are met and adequately provided for.

Purchase requisitions for new safety related items are initiated by the cognizant engineering group which establishes initial requirements.

Replacement/spares are purchased to requirements equivalent to the original unless upgrading is required by Federal Regulations, or deemed necessary by the cognizant engineering group.

1.7.4.2.3

The contents of procurement documents vary according to the item(s) being purchased and its function(s) in the Cook Nuclear 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.
b) Technical requirements, with applicable drawings, specifications, codes and standards identified by title, document number, revision and date, with any required procedures such as special process instructions identified in such a way as to indicate source and need. Imposition of guides/standards on AEPSC/I&M suppliers and subtier suppliers will be on a case-by-case basis depending upon the item or service to be supplied and upon the degree that AEPSC/I&M relies on suppliers to invoke guides/standards. AEPSC/I&M recognizes that certain suppliers have acceptable 10CFR50, Appendix B QA programs, even though, the suppliers are not committed to Regulatory Guides or industry standards (e.g. ANSI N45.2.6). In those cases, in which suppliers are not committed to the same guides/standards as AEPSC/I&M, AEPSC/I&M will assure that (1) the supplier's QA program provides adequate QA controls, regardless of the lack of specific commitment, or (2) controls will be invoked directly by AEPSC/I&M to assure adequate quality of products/services received by suppliers.

c) Regulatory, administrative and reporting requirements.

d) Quality requirements appropriate to the complexity and scope of the wo[illegible] including necessary tests and inspections.

e) A requirement for a documented QA Program, subject to QA review and written concurrence prior to the start of work.

f) A requirement for the supplier to invoke applicable quality requirements on subtier suppliers.

g) Provisions for access to supplier and subtier suppliers' facilities and records for inspections, surveillances and audits.

h) Identification of documentation to be provided by the supplier, the schedule of submittals and documents requiring AEPSC approval.

1.7.4.2.4

The AEPSC QA Division performs audits of procurement documents to assure that QA Program requirements have been met. These audits are conducted in accordance with AEPSC QA Division procedures.

1.7.4.2.5

Changes to procurement documents are controlled in a manner commensurate with that used for the original documents.

1.7.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

1.7.5.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are accomplished using instructions, procedures and drawings appropriate to the circumstances, including acceptance criteria for determining if an activity has been satisfactorily completed.

1.7.5.2 IMPLEMENTATION

1.7.5.2.1

Instructions and procedures incorporate: 1) a description of the activity to be accomplished, and 2) appropriate quantitative (such as tolerances and operating limits) and qualitative (such as workmanship and standards) acceptance criteria sufficient to determine that the activity has been satisfactorily accomplished. Hold points for inspection are established when required.

Instructions and procedures pertaining to the specification of, and/or implementation of, the QA Program receive multiple reviews for technical adequacy and inclusion of appropriate quality requirements. Top tier instructions and procedures are reviewed and/or approved by AEPSC QA. Lower tier documents are reviewed and approved, as a minimum, by management/supervisory personnel trained to the level necessary to plan, coordinate and administer those day-to-day verification activities of the QA Program for which they are responsible. |* |* |*

Special procedures may be issued for activities which have short-term applicability.

1.7.5.2.2

AEPSC activities relative to the Cook Nuclear Plant are outlined by procedures which provide the controls for the implementation of these activities. AEPSC has two categories of QA Program implementation procedures:

1) General Procedures (GPs) which are applicable to all AEPSC divisions and departments involved with Cook Nuclear Plant.

2) Division/department/section procedures which apply to the specific division, department or section involved.

1.7.5.2.3

Activities at the Cook Nuclear Plant are controlled using plant procedures.

The PMIs have been classified into the following series:

1000 Organization and Responsibilities
2000 Administration - Document Control, Security, Training, Records, Radiation Protection and Fire Protection
3000 Procurement, Receiving, Shipping and Storage
4000 Operations, Fuel Handling, Surveillance Testing
5000 Maintenance, Repair, Modification, Special Processes, EQ and ISI
6000 Technical - Chemistry, Radiological Controls, Performance/Engineering Testing, and Instrument and Control Maintenance and Calibration
7000 Quality Assurance, Quality Control Program and Condition/Problem Reporting

Instructions and procedures identify the regulatory requirements and commitments which pertain to the subject that it will control and establish responsibilities for implementation. Instructions and procedures may either provide the guidance necessary for the development of supple mental instructions and/or procedures to implement their requirements, or provide comprehensive guidance based on the subject matter.

1.7.5.2.4

Cook Nuclear Plant drawings are produced, controlled and distributed under the control of AEPSC and the Cook Nuclear Plant. AEPSC design drawings are produced by, or under the control of, the AEPSC Nuclear Design Group under a set of procedures which direct their development and review. These procedures specify requirements for inclusion of quantitative and qualitative acceptance criteria. Specific drawings are reviewed and approved by the cognizant engineering divisions/department. |*

AEPSC has stationed an on-site design staff to provide for the revision of certain types of design drawings to reflect as-built conditions.

1.7.5.2.5

Complex Cook Nuclear Plant procedures are designated as "In Hand" procedures. Examples of "In Hand" procedures are those developed for extensive or complex jobs where reliance on memory cannot be trusted. Further, those procedures which describe a sequence which cannot be altered, or require the documentation of data during the course of the procedure, are considered. "In Hand" procedures are designated as such by double asterisks (**) which precede the procedure number on the cover sheet, all pages and attachments of a procedure and the corresponding index. |*

1.7.6 DOCUMENT CONTROL

1.7.6.1 SCOPE

Documents controlling activities within the scope defined in 1.7.2 herein are issued and changed according to established procedures. Documents such as instructions, procedures and drawings, including changes thereto, are reviewed for adequacy, approved for release by authorized personnel, and are distributed and used at the location where a prescribed activity is performed. |*

Changes to controlled documents are reviewed and approved by the same organizations that performed the original review and approval, or by other qualified, responsible organizations specifically designated in accordance with the procedures governing these documents. Obsolete or superseded documents are controlled to prevent inadvertent use.

1.7.6.2 IMPLEMENTATION

1.7.6.2.1

Controls are established for approval, issue and change of documents in the following categories:

a) Design documents (e.g., calculations, specifications, analyses)

b) Drawings and related documents
c) Procurement documents
d) Instructions and procedures
e) Updated Final Safety Analysis Report (UFSAR)
f) Plant Technical Specifications
g) Safeguards documents

1.7.6.2.2

The review, approval, issuance and change of documents are controlled by:

a) Establishment of criteria to ensure that adequate technical and quality requirements are incorporated.

b) Identification of the organization responsible for review, approval, issue and maintenance.

c) Review of changes to documents by the organization that performed the initial review and approval, or by the organization designated in accordance with the procedure governing the review and approval of specific types of documents.

Maintenance, modification and inspection procedures are audited by AEPSC QA for compliance with established inspection requirements.

1.7.6.2.3

Documents are issued and controlled so that:

a) The documents are available prior to commencing work.
b) Obsolete documents are replaced by current documents in a timely manner.

1.7.6.2.4

Master lists, or equivalent controls, are used to identify the current revision of instructions, procedures, specifications and drawings. These control documents are updated and distributed to designated personnel who are responsible for maintaining current copies of the applicable documents. The distribution of controlled documents is performed under procedures requiring receipt acknowledgement and in accordance with established distribution lists. |*

1.7.6.2.5

In the event a drawing is developed on-site to reflect an as-built configuration, the marked-up drawing is maintained in the Master Plant File and all holders of the drawing are issued appropriate notification to inform them the revision they hold is not current, cannot be used and, if required, reference must be made to the Master Plant File drawing.

1.7.6.2.6

Documents prepared for use in training or for interested parties are appropriately marked to indicate that they are for informational use |*
only and cannot be used to operate or maintain the facility, or to |*
conduct activities affecting the quality of safety-related items. At Cook Nuclear Plant, unless a document is identified as "controlled," it |*
is automatically assumed the document is for informational use only.

1.7.7 CONTROL OF PURCHASED ITEMS AND SERVICES

1.7.7.1 SCOPE

Activities that implement approved procurement requests for items and services are controlled to assure conformance with procurement document requirements. Controls include a system of supplier evaluation and selection audits, acceptance of items and documentation upon delivery, and periodic assessment of supplier performance. Objective evidence of quality that demonstrates conformance with specified procurement document requirements is available to the Cook Nuclear Plant site prior to use of equipment, material, or services.

1.7.7.2 IMPLEMENTATION

1.7.7.2.1

AEPSC qualifies suppliers and distributors by performing a documented evaluation of their capability to provide items or services specified by procurement documents. Items and services designated as safety-related are purchased from suppliers whose QA programs have been accepted in accordance with AEPSC requirements, or from commercial grade suppliers through the AEPSC dedication program. Suppliers of other items/services, such as fire protection, records storage, etc., are also evaluated using different criteria for acceptance.

Qualification of such suppliers is determined by the AEPSC QA Division. In the discharge of this responsibility, the AEPSC QA Division may use information generated by other utilities. The supplier, or distributor, must be approved before procurement can be completed. AEPSC is a member of the Nuclear Procurement Issues Council (NUPIC), participates in joint supplier audits, and shares audit information consistent with NUPIC requirements. The supplier, or distributor, must be acceptable, or acceptable subject to follow-up, before a procurement can be approved and processed. Additional audits will be conducted, as necessary, to meet requirements. Acceptance is not complete until it has been determined that the suppliers' QA program can meet the requirements for the item(s)/service(s) offered.

1.7.7.2.2

For items that are not unique to a nuclear power plant ("Commercial Grade") where requirements cannot be imposed in a practical manner at the time of procurement, programs for dedication to safety-related standards are established and accomplished by the AEPSC cognizant engineer prior to the item being accepted for safety-related use.

1.7.7.2.3

In-process audits of suppliers' activities during fabrication, inspection, testing and shipment of items are performed when deemed necessary, depending upon supplier qualification status, complexity of the item(s) being furnished, the items' importance to safety, and/or |*
previous supplier history. These audits are performed by AEPSC QA. The cognizant engineer and/or responsible Cook Nuclear Plant personnel may |*
also participate, if deemed necessary.

1.7.7.2.4

Spare and replacement parts are procured in such a manner that their performance and quality are at least equivalent to those of the parts that will be replaced.

a) Specifications and codes referenced in procurement documents for spare or replacement items are at least equivalent to those for the original items or to properly reviewed and approved revisions.

b) Parts intended as spares or replacement for "off-the-shelf" items, or other items for which quality requirements were not originally specified, are evaluated for performance at least equivalent to the original.

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, safety, interchangeability, 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 items addresses the program for inspection of incoming items, including a review of the documentation required under the procurement. Receipt inspection personnel are qualified and certified in accordance with the requirements of ANSI N45.2.6. Provisions for receipt inspection apply regardless of where the procurement originates. Additional inspections may apply if required by the procurement document. |* |*

Where items and/or services are safety-related and procurement is accomplished without assistance of AEPSC, supplier selection is limited to those companies identified as being qualified.

1.7.7.2.6

Items 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 item is placed in a designated area of the storeroom. Item traceability to procurement documents and to end use is maintained through recording of "HOLD" and "ACCEPTED" tag numbers on applicable documents. |*

Nonconforming items, or missing or questionable documentation results in items being placed on "HOLD" and maintained in a designated, controlled area of the storeroom. If the nonconformance cannot be cleared, the item 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 Cook Nuclear Plant QA Program and procedures. Prior to their working at the Cook Nuclear Plant, contractor quality assurance programs must be audited and approved by AEPSC QA. Contractor procedures must be reviewed and approved by the originating/sponsoring department supervisor, PNSRC and the Plant Manager. Further, periodic audits of site contractor activities are conducted under the direction of the AEPSC Quality Assurance Superintendent. |*

1.7.7.2.8

To the extent prescribed in specific procurement documents, suppliers furnish quality records; documentary evidence that material and equipment either conforms to requirements or identifies any requirements that have not been met; and descriptions of those nonconformances from the procurement requirements, which have been dispositioned "Use-as-is" or "repair." This evidence is retained at the Plant, or at the Service Corporation.

To the extent prescribed in specific procurement agreements, suppliers are required to maintain additional (backup) documents in their record system.

In some cases, such as with NSSS, suppliers are designated primary record retention responsibility.

1.7.7.2.9

The capability of suppliers to furnish valid certificates is evaluated during procurement document reviews, annual supplier evaluations, and during audits.

1.7.8 IDENTIFICATION AND CONTROL OF ITEMS

1.7.8.1 SCOPE

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 items (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, serial number, etc., are included in AEPSC Specifications (DCCs) and/or the procurement document.

1.7.8.2.5

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

1.7.9.2.2

Special process requirements for chemical cleaning, application of protective coatings and concrete placement are set forth in AEPSC Specifications (DCCs) 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 audited by the AEPSC QA Division.

Special process requirements for welding, heat treating and NDE are set forth in AEPSC Specifications, the AEP Welding and NDE Manuals and plant procedures. These specifications and manuals are prepared by, or are reviewed and approved by, the AEPSC Cognizant Engineer - Welding and NDE Administrator. The administrative controls portion of the NDE Manual is audited by AEPSC QA.

Special process procedures, with the exception of welding and heat treating, are prepared by Cook Nuclear Plant personnel with technical knowledge in the discipline involved. These procedures, which are also |* reviewed by other personnel with the necessary technical competence, are |* qualified by testing.

Welding is performed in accordance with procedures contained in the AEP Welding Manual, or in the approved contractor's manual. These procedures are qualified in accordance with applicable codes, and Procedure Qualification Records are prepared. Weld Procedure Qualifi cation Records are reviewed and approved by the AEPSC Cognizant Engineer - Welding. Weld qualification documentation is retained in the AEP Welding Manual, or the approved contractor's manual.

Contractor welding procedures are qualified by the contractor. These procedures and the qualification documentation are reviewed and approved by the AEPSC Cognizant Engineer - Welding. This documentation is retained by the contractor.

1.7.9.2.3

NDE personnel are qualified and certified by a Cook Nuclear Plant NDE Level III who has been qualified and certified by the designated NDE Administrator. Certification is by examination. Personnel qualification is kept current by re-examination at time intervals specified by the AEP NDE Manual, and in accordance with the ASME Code.

Cook Nuclear Plant welders are qualified by the Maintenance Department utilizing the procedures in the AEP Welding Manual. Supervision of Cook Nuclear Plant welder qualifications is performed by the Maintenance Department. Examination of specimens is performed under the supervision of the Safety and Assessment Department in accordance with the AEP Welding Manual covering welder qualification. Cook Nuclear Plant welder qualification records are maintained for each welder by the Maintenance Department. Contractor and craft welders are qualified by the contractor using procedures approved by the AEPSC Cognizant Engineer - Welding in accordance with AEPSC procedures. Contractor and craft welder qualification records are maintained by the contractor.

1.7.9.2.4

QC/NDE Technicians assigned to the Safety and Assessment Department perform nondestructive testing for work performed by Cook Nuclear Plant and contractor personnel. These individuals are qualified to either SNT-TC-1A, or ANSI N45.2.6, and records of the qualifications/ certifications are maintained at Cook Nuclear 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

Special process qualifications are reviewed during regularly scheduled QA audits. Qualification records are maintained in accordance with 1.7.17 herein.

1.7.9.2.7

The documentation resulting from welding and nondestructive testing is reviewed by appropriate 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, etc., for 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, qualification and certification programs for personnel who perform inspections are established, implemented and documented in accordance with 1.7.2 herein and as described in Appendix B hereto, item 9b, with exceptions as noted therein.

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

Inspections are conducted using the following programs:

a) <u>Work Activities Performed by I&M Personnel.</u> Work functions associated with normal operation of the plant, routine maintenance, calibrations, etc., are routinely assigned to plant personnel. I&M personnel who inspect this work are qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, and are periodically trained in their skill area using INPO "accredited" training. As a result of the qualifications and training which I&M personnel receive, a peer inspection system is used. Peer inspection personnel are independent in that they do not perform, |* or directly supervise, the work being inspected, but may be from |* the same work group. Cook Nuclear Plant Safety and Assessment Department personnel qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1 will ensure (through surveillance) that inspections have been correctly implemented and make routine reports to management.

b) Work Activities Performed by Contractors. Major modifications, non-routine maintenance, and/or other services on safety-related items are generally performed by contractors who are required to comply with the applicable requirements of Regulatory Guide 1.33 and ANSI N45.2. Inspections of these work activities are performed by inspectors qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6. A peer inspection program is not used for work activities performed by these personnel. Contractor inspection personnel are required to be qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6. I&M Cook Nuclear Plant Quality Control personnel who are also qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6 may perform inspections and/or surveillance of these activities. |* |*

1.7.10.2.5

Inspections associated with the packaging and shipment of radioactive waste and materials are conducted using the following program:

a) NRC Licensed Packagings - Inspections of NRC licensed radioactive material packagings shall be performed by individuals independent from the work being performed. The independent inspectors shall be Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum. Additionally, the inspector shall be familiar with the activities being performed. |*

b) Non-NRC Licensed Packagings and Containers - Inspections of non-NRC licensed radioactive material packagings and containers (shipping and/or burial) shall be performed by Indiana Michigan |* |* |*

Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

c) Transportation Vehicles - Inspection of transportation vehicles being shipped as "exclusive use", shall be performed by Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

d) Other inspections and Verification - Inspections and verifications of other activities associated with the packaging and shipment of radioactive materials and waste shall be performed by Indiana and |*
Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum. |*

1.7.10.2.6

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.

1.7.11 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:

Scheduled

Surveillance, preventive maintenance, post-design, qualification.

Unscheduled

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. 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 Cook Nuclear 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

Established procedures and instructions are used for calibration and control of measuring and test equipment utilized in the measurement, inspection and monitoring of structures, systems and components. These procedures and instructions describe calibration techniques and frequencies, and maintenance and control of the equipment.

AEPSC QA 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 for attaching, or affixing labels, to measuring and test equipment to display the date calibrated and the next calibration due date, or 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; or where such 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 equipment being calibrated will be within required tolerances. The basis for acceptance is documented and authorized by the responsible manager.

1.7.12.2.5

Cook Nuclear Plant procedures 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. |*

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 Cook Nuclear Plant. Controls are established in plant procedures.

1.7.13.2.3

Packaging and shipping requirements are provided to vendors with the AEPSC Specifications (DCCs) which are a part of the purchase order, or are otherwise specified on the procurement order. Controls for receipt inspection, damaged items and special handling requirements at the Cook Nuclear Plant are established by plant procedures. 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 design change activities, including item fabrication, installation and test, a program 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 program establishes a tagging system for lifted leads, etc. For those items requiring calibration, the program provides for physical indication of calibration status by calibration stickers, or a control system is used. |*

1.7.14.2.2

Application and removal of inspection and welding stamps, and of such status indicators as tags, markings, labels, etc., is controlled by plant procedures. |*

The inspection status of materials received at the Cook Nuclear Plant is identified in accordance with established instructions. 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 ANSI N18.1 qualified personnel (reference 1.7.10.2.4 herein), inspect an activity at various stages and sign off the procedural inspection steps. |* |*

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

1.7.14.2.3

Required surveillance test procedures are defined in PMIs. These instructions provide for documenting bypassed tests and rescheduling of the test. |*

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. When an immediate procedure change is required to continue in-process work or testing and the required complete review and approval process cannot be accomplished, an "On The Spot" change is processed in accordance with the PMI on plant procedures. |*

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 ITEMS

1.7.15.1 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/Problem Reports (refer to 1.7.16 herein) are used at Cook Nuclear Plant to identify nonconforming items and initiate corrective action for items which are installed or have been released to the Cook Nuclear Plant. Systems, components, or materials which require repair or inspection are controlled under the Job Order System. In addition, the various procedures identified in 1.7.14 herein 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 Division 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, corrective action is taken to correct the immediate problem, and preventive action is implemented to prevent recurrence. 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 Division.

1.7.16.2.2

Condition/Problem Reports provide the mechanism for plant and AEPSC personnel to notify management of conditions adverse to quality. Condition/Problem Reports are also used to report violations to codes, regulations and the Technical Specifications. Investigations of reported conditions adverse to quality are assigned by management. The Condition/Problem Report is used to document the investigation of a problem; and to identify the need for a design change to correct system or equipment deficiencies, or to identify the need for the initiation of Job Orders to correct minor deficiencies. Further, Condition/Problem Reports are used to identify those actions necessary to prevent recurrence of the reported condition.

Significant problems, which are so designated on Condition/Problem Reports, are reviewed by the PNSRC for evaluation of actions taken, or |*
being taken, to correct the deficiency and prevent recurrence. |*

The AEPSC NSDRC is responsible for assuring that independent reviews of violations (as specified in the Technical Specifications) are performed. These violations are considered significant problems which are documented on Condition/Problem Reports. The reviews will provide an independent evaluation of the reported problems and corrective actions.

The AEPSC QA Division 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 the quality of safety-related items 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) Problem Reports.
h) Other documentation such as drawings, specifications, dedication plans, procurement documents, calibration procedures and reports. |
i) Radiographs. |

1.7.17.2.2

Instructions and procedures establish the requirements for the identification and preparation of records for systems and equipment under the QA Program, and provide 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 the identification, collection, maintenance and storage of records generated by their department/division. Procedures 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 or equipment, as applicable. |*

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, or micrographs thereof, 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 QA 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 reviewed by |* management. Follow-up action is taken where indicated.

1.7.18.2 IMPLEMENTATION

1.7.18.2.1 <u>AEPSC QA Division Responsibilities</u>

The basic responsibility for the assessment of the QA Program is vested in the AEPSC QAD. The AEPSC QAD is primarily responsible for ensuring that proper QA programs are established and for verification of their implementation. These responsibilities are discharged in cooperation with the AEPSC and Cook Nuclear Plant management and their staffs. |*

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 one to two years.

1.7.18.2.3

The AEPSC QAD conducts audits to verify the adequacy and implementation of the QA Program at the Cook Nuclear Plant and within the AEP System. QA audit reports are distributed to appropriate Cook Nuclear Plant |* management and the NSDRC (all audits).

1.7.18.2.4

The independent off-site review and audit organization is the AEPSC NSDRC. This committee is composed of AEPSC, I&M and Cook Nuclear Plant management members. An NSDRC Manual has been developed for this committee which contains the NSDRC Charter and procedures. The NSDRC conducts periodic audits of Cook Nuclear Plant operations pursuant to established criteria (Technical Specifications, etc.).

NSDRC audit reports are submitted for review to the NSDRC membership, | the Chairman of the NSDRC, and the AEPSC Senior Executive Vice President |* - Engineering and Construction. Problem Reports provide for the recording of actions taken to correct deficiencies found during these audits.

1.7.18.2.5

The Cook Nuclear Plant on-site review group is the 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, procedures, and design changes for safety-related systems prior to approval by the Plant Manager. In addition, this committee serves to conduct investigations of violations to Technical Specifications, and reviews significant Problem Reports to determine if appropriate action has been taken. |*

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 criteria established within their quality assurance programs.

1.7.18.2.7

Regularly scheduled audits are supplemented by "special audits" when significant changes are made in the QA 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 practices, procedures, instructions, activities and items related to quality; and a 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 QA 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 QA organization, but knowledgeable in auditing and quality assurance, periodically review the effectiveness of the QA 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 Cook Nuclear Plant Fire Protection QA Program has been developed using the guidance of NRC Branch Technical Position (APCSB) 9.5-1, Appendix A, Section C, "Quality Assurance Program," and NRC clarification "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," dated June 14, 1977. As such, the Fire Protection QA Program is part of the overall QA Program for the plant. The Fire Protection QA Program encompasses design, procurement, fabrication, construction, surveillance, inspection, operation, maintenance, modification, and audits.

Implementation and assessment of the Fire Protection QA Program is the responsibility of each involved AEPSC and Indiana Michigan Power Company organization.

1.7.19.2 Organization

The Fire Protection QA Program is under the management control of AEPSC. This control consists of:

1) Verifying the effectiveness of the Fire Protection QA Program through review, surveillance, and audits.

2) Directing formulation, implementation, and assessment of the Fire Protection QA Program by procedural controls. |*

3) Assuring the QA program is acceptable to the management responsible for fire protection.

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

a) Maintenance of fire protection systems. |*
b) Testing of fire protection equipment. |*
c) Fire safety inspections. |*
d) Fire fighting procedures. |*
e) Fire drills. |*
f) Emergency remote shutdown procedures. |*
g) Emergency repair procedures (10CFR50, Appendix R). |*

The Fire Protection QA Program at the Cook Nuclear Plant also provides for inspection of fire hazards, explosion hazards, and training of fire brigade and responding fire departments.

The Assistant Shift Supervisor on duty, or designee, is designated as the Fire Brigade Leader and coordinates the fire fighting efforts of shift personnel and the Fire Brigade.

1.7.19.3 <u>Design Control and Procurement Document Control</u>

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

The Cook Nuclear Plant design was reviewed by qualified personnel to ensure inclusion of appropriate fire protection requirements. These reviews include items such as:

1) Verification as to the adequacy of electrical isolation and cable separation criteria.

2) Verification of appropriate requirements for room isolation (sealing penetrations, floors and other fire barriers).

3) Determination for increase in fire loadings.

4) Determination for the need of additional fire detection and suppression equipment.

Procurement of fire protection equipment and related items are subject to the requirements of the fire protection procurement documents. A review of these documents is performed to assure fire protection requirements and quality requirements are correctly stated, verifiable, and controllable, and that there is adequate acceptance and rejection criteria. Procurement documents must be prepared, reviewed, and approved according to QA Program requirements.

Design and procurement document changes, including field changes and design deviations, are controlled by procedure. |*

1.7.19.4 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that assist in implementing the fire protection program are prescribed by approved instructions or procedures.

Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with approved procedures. Activities associated with the fire protection systems and fire protection related systems are prescribed and accomplished in accordance with documented instructions, procedures, and drawings. Instructions and procedures for design, installation, inspection, tests, maintenance, modification and administrative controls are reviewed through audits to |* assure that the fire protection program is maintained.

Operation and maintenance information has been provided to the plant in the form of System Descriptions and equipment supplier information.

1.7.19.5 Control of Purchased Items and Services

Measures are established to assure that purchased items and services conform to 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 receipt inspection.

Source or receipt inspection is provided, as a minimum, for those items where 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 installation, maintenance and modification of fire protection systems and equipment.

b) Inspections of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed in accordance with installation specifications.

c) Inspections of cable routing to verify conformance with design requirements as specified in AEPSC Specifications and/or plant procedures.

d) Inspections to verify that appropriate requirements for fire barriers are satisfied following installation, modification, repair or replacement 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 or checklists for required inspections.

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

h) Periodic inspections of materials subject to degradation, such as fire stops, seals and fire retardant coating as required by Technical Specifications or manufacturer's recommendations. |*

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 and 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 occurs to document that fire protection equipment functions in accordance with its design. |*

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 plant Technical Specification fire protection systems are performed as described in 1.7.14 herein.

1.7.19.9 Nonconforming Items

Technical Specification fire protection equipment nonconformances are identified and dispositioned as described in 1.7.15 herein.

1.7.19.10 Corrective Action

The corrective action mechanism described in 1.7.16 herein applies to the Technical Specification fire protection equipment.

1.7.19.11 Records

Records generated to support the fire protection program are controlled as described in 1.7.17 herein.

1.7.19.12 Audits

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

Audits are periodically performed to verify compliance with the administrative controls and implementation of fire protection quality assurance criteria. The audits are performed in accordance with pre-established written procedures or checklists. 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
SUPPORT ORGANIZATION FOR THE COOK NUCLEAR PLANT

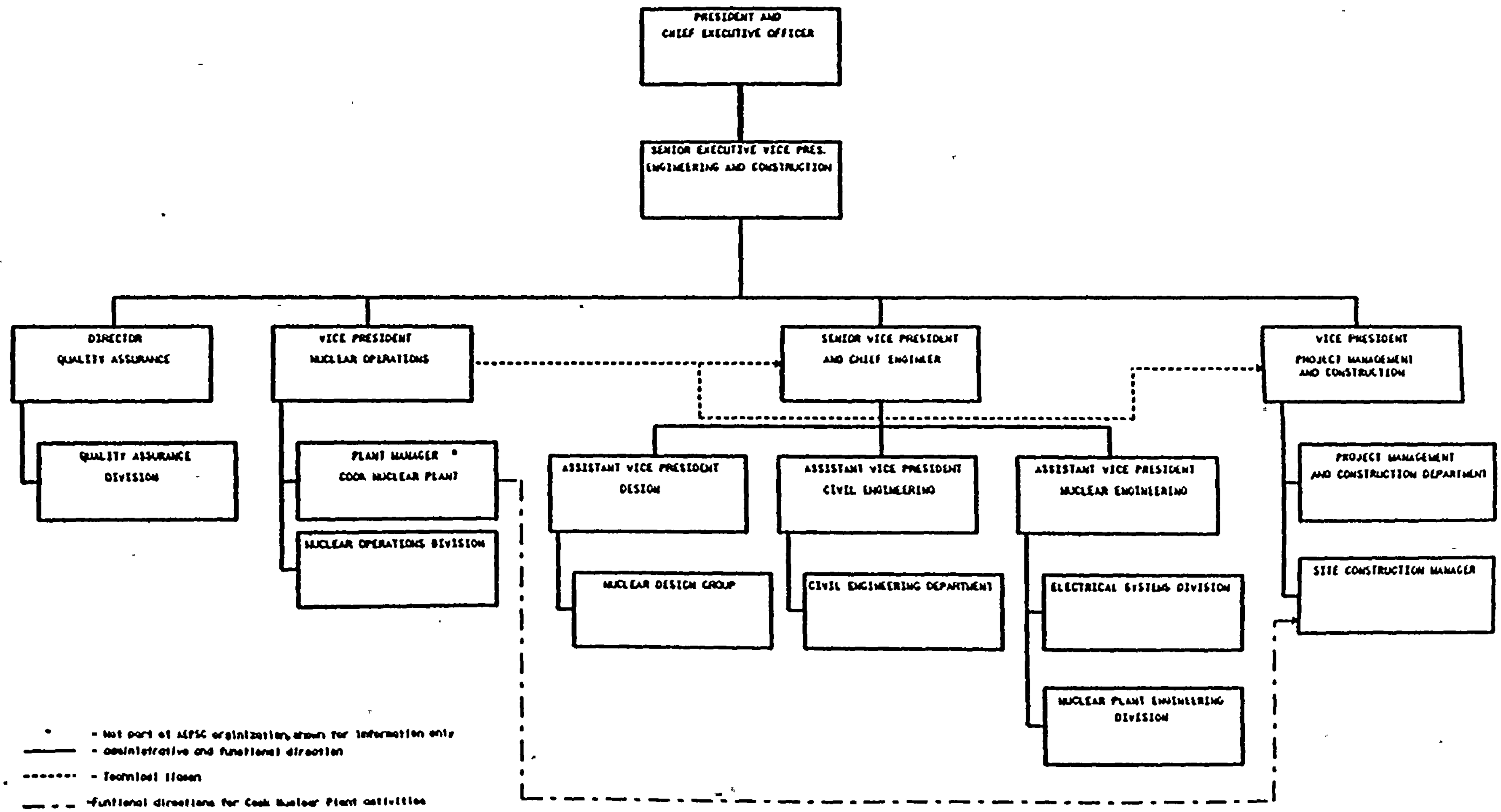
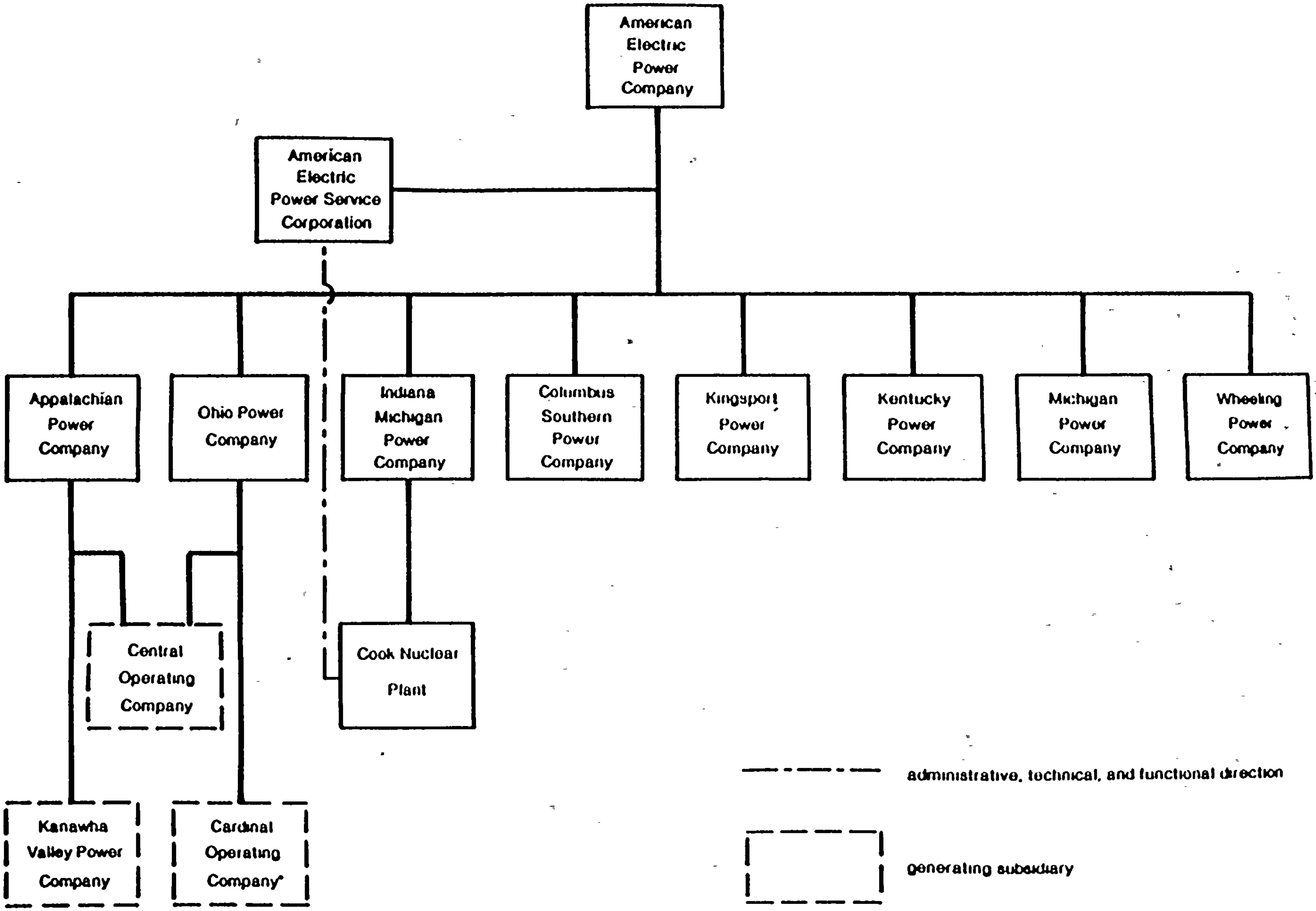


Figure No. 1.7-1

American Electric Power Company

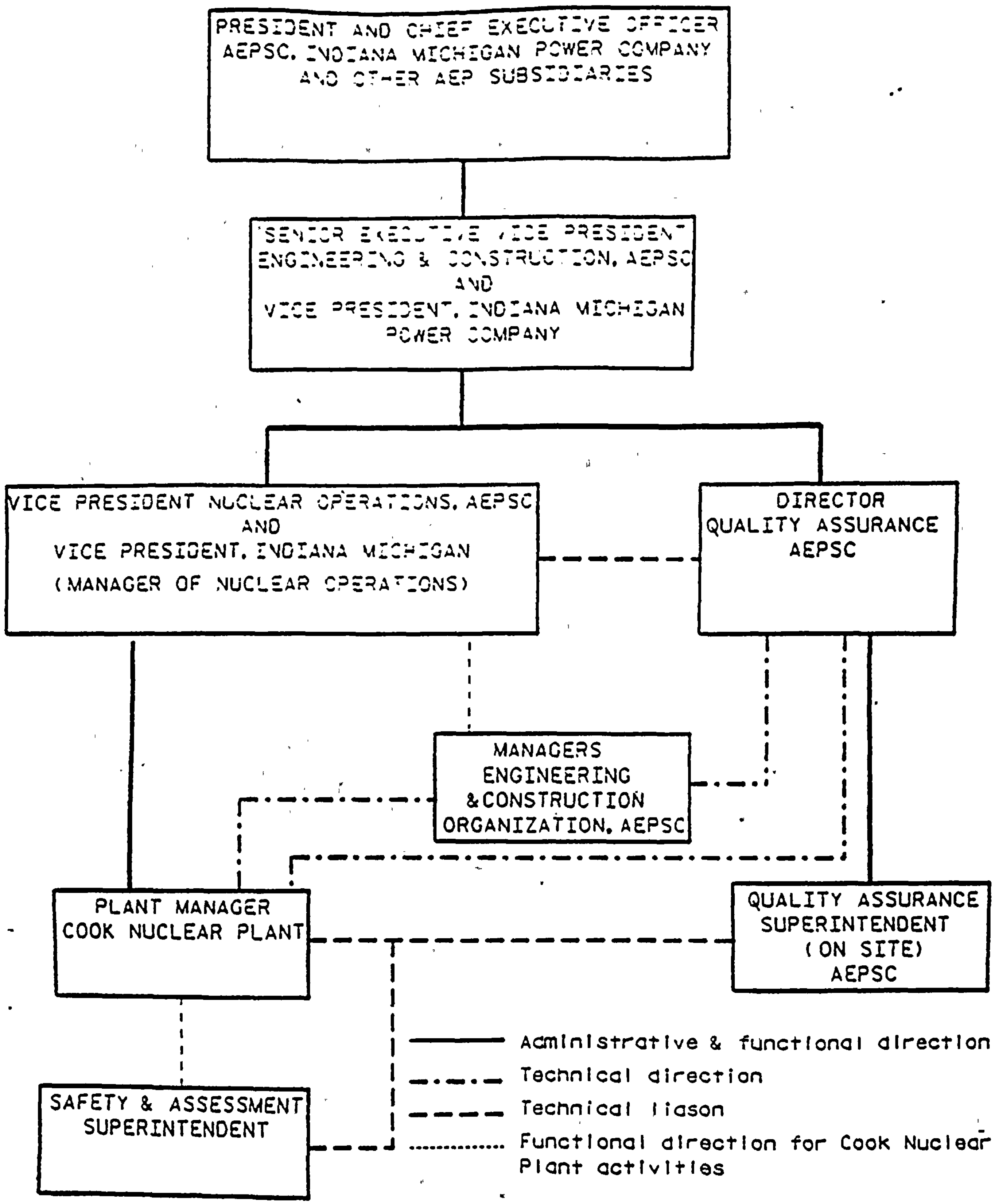


*jointly owned with Buckeye Power, Inc.

Figure No. 1.7-2

Figure No. 1.7-3

ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QUALITY ASSURANCE & QUALITY CONTROL SUPPORT OF THE COOK NUCLEAR PLANT



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AEPSC QUALITY ASSURANCE DIVISION ORGANIZATION

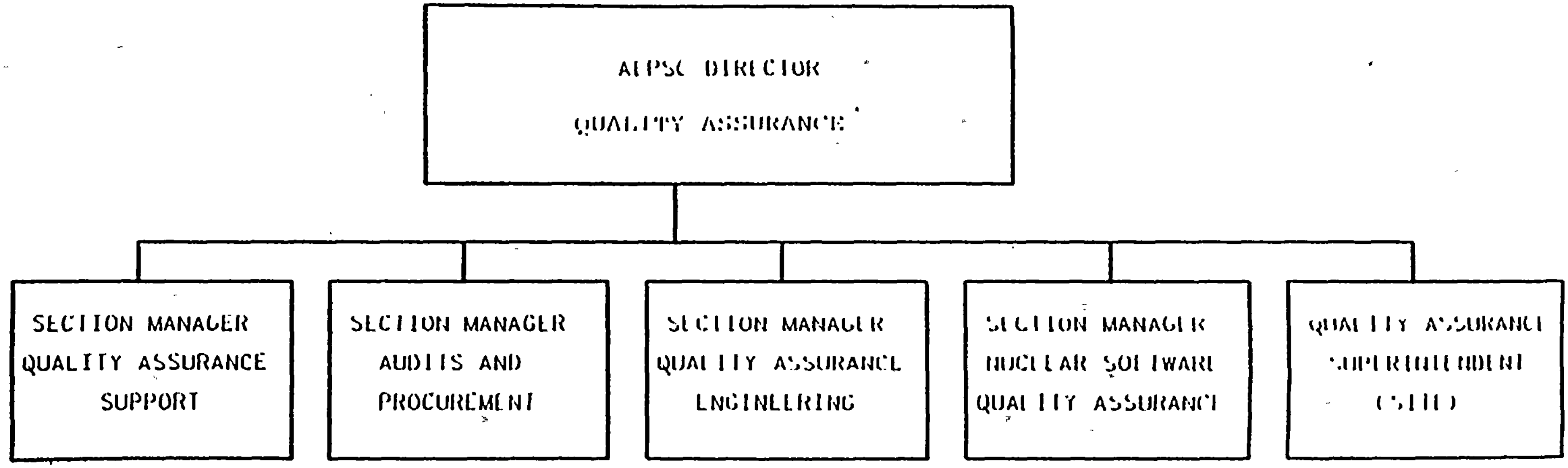
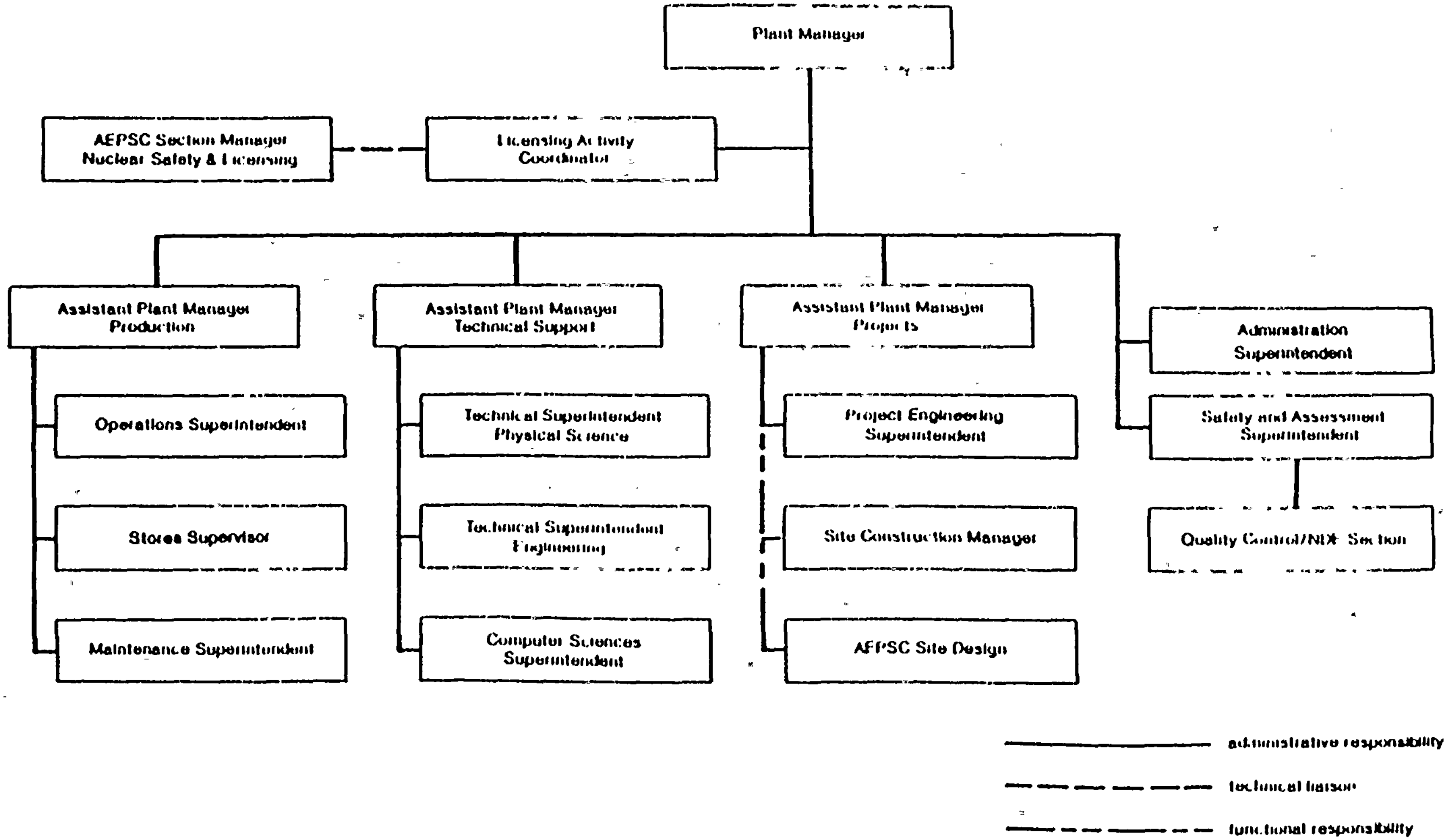


Figure No. 1.7-4

Indiana Michigan Power Company
Organization for the Cook Nuclear Plant



APPENDIX A

REGULATORY AND SAFETY GUIDES/ANSI STANDARDS

1.	Reg. Guide 1.8 (9/75)	- Personnel Selection and Training
	ANSI N18.1 (1971)	- Selection and Training of Nuclear Power Plant Personnel
2.	Reg. Guide 1.14 (8/75)	- Reactor Coolant Pump Flywheel Integrity
3.	Reg. Guide 1.16 (8/75)	- Reporting of Operating Information, Appendix A - Technical Specifications
4.	Safety Guide 30 (8/72)	- Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
	ANSI N45.2.4 (1972)	- Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations
5.	Reg. Guide 1.33 (02/78)	- Quality Assurance Program Requirements (Operation)
	ANSI N18.7 (1976) (ANS 3.2 1976)	- Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
	ANSI N45.2 (1977)	- Quality Assurance Program Requirements for Nuclear Facilities

6.	Reg. Guide 1.37 (3/73)	- Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
	ANSI N45.2.1 (1973)	- Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
7.	Reg. Guide 1.38 (10/76)	- Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
	ANSI N45.2.2 (1972)	- Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)
8.	Reg. Guide 1.39 (10/76)	- Housekeeping Requirements for Water-Cooled Nuclear Power Plants
	ANSI N45.2.3 (1973)	- Housekeeping During the Construction Phase of Nuclear Power Plants
9.	Reg. Guide 1.54 (6/73)	- Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
	ANSI N101.4 (1972)	- Quality Assurance for Protective Coatings Applied to Nuclear Facilities
10.	Reg. Guide 1.58 (9/80)	- Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel |*

	ANSI N45.2.6 (1978)	- Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
11.	Reg. Guide 1.63 (7/78)	- Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
12.	Reg. Guide 1.64 (10/73)	- Quality Assurance Requirements for the Design of Nuclear Power Plants
	ANSI N45.2.11 (1974)	- Quality Assurance Requirements for the Design of Nuclear Power Plants
13.	Reg. Guide 1.74 (2/74)	- Quality Assurance Terms and Definitions
	ANSI N45.2.10 (1973)	- Quality Assurance Terms and Definitions
14.	Reg. Guide 1.88 (10/76)	- Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
	ANSI N45.2.9 (1974)	- Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants
15.	Reg. Guide 1.94 (4/76)	- Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

	ANSI N45.2.5 (1974)	- Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
16.	Reg. Guide 1.108 (8/77)	- Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants
17.	Reg. Guide 1.123 (7/77)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
	ANSI N45.2.13 (1976)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
18.	Reg. Guide 1.144 (1/79)	- Auditing of Quality Assurance Programs for Nuclear Power Plants
	ANSI N45.2.12 (1977)	- Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
19.	Reg. Guide 1.146 (8/80)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
	ANSI N45.2.23 (1978)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

20. ANSI N45.2.8 (1975) - Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants

21. ANSI N45.4 (1972) - Leakage-Rate Testing of Containment Structures for Nuclear Reactors

APPENDIX B

AEPSC/I&M 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&M 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&M 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 alternates 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

Although the usual practice is to meet this requirement, exceptions are made to NSDRC review and approval prior to implementation in rare cases with the permission of the NSDRC Chairman and Secretary. PNSRC review and approval is always done prior to implementation of Technical Specification changes.

2f. Sec. 4.4

<u>Requirement</u>

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

<u>Exception/Interpretation</u>

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

<u>Requirement</u>

"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."

<u>Exception/Interpretation</u>

I&M 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

<u>Requirement</u>

"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&M 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 surveillance 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 - 1977."

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.

2l. Sec. 5.2.15

Requirement

"Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary or desirable."

Exception/Interpretation

Biennial reviews are not performed in that I&M has programmatic control requirements in place that make the biennial review process redundant from a regulatory perspective. These programmatic controls were effected in an effort to ensure that plant instructions and procedures are reviewed for possible revision when pertinent source material is revised, therefore maintaining the procedures current. We believe that this approach, in addition to an annual random sampling |* of procedures, better addresses the intent of the biennial review process and is more acceptable from both a technical and practical perspective than a static two-year review process.

2m. Sec. 5.2.16

Requirement

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

Exception/Interpretation

See Item 6b.

2n. 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. 3

Requirement

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

Exception/Interpretation

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

Cleanliness is maintained, consistent with the work being performed, so as to prevent the introduction of foreign material. As a minimum, cleanliness 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&M 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 and 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

Cleanliness 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&M 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. <u>N45.2.4</u>,

6a. Sec. 2.2

<u>Requirement</u>

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.

<u>Exception/Interpretation</u>

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

<u>Requirement</u>

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

<u>Exception/Interpretation</u>

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. <u>N45.2.5</u>,

7a. Sec. 2.5.2

<u>Requirement</u>

"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&M 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. Req. Guide 1.58 - General

Requirement

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

9a. C.2.a(7)

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&M 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&M 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&M does not implement the recommended "Level" designation guidelines.

It is I&M's opinion that the training guidelines of SNT-TC-1A, 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&M maintains that it meets the essence of these training guidelines. The preparation of leak test procedures and the conduct of leak tests at Cook Nuclear Plant is under the direct supervision 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&M 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 routine, 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&M 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&M 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&M 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 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&M's opinion, no useful purpose is served by classification of management, supervisory and engineering personnel into "Levels."

Therefore, I&M 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 9b, this position is not applicable to the Cook Nuclear Plant.

C-6 Replacement personnel for 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&M, as a general practice, complies with the training recommendations as set forth in this regulatory position.

C-8 All I&M 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&M 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. <u>N45.2.8</u>,

10a. Sec. 2.9e

<u>Requirement</u>

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&M assure that they are available to the site, as necessary. |*

10b. 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.

10c. 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.

11. N45.2.9

11a. Sec. 5.4, Item 2

Requirement

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.

Exception/Interpretation

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. |*

11b. Sec. 6.2

Requirement

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

Exception/Interpretation

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.

11c. 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 Cook Nuclear Plant Master File Room and other off-site record storage facilities comply with the requirements of NUREG-0800 (7/81), Section 17.1.17.4.

12. Reg. Guide 1.144/ANSI N45.2.12

12a. 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.

12b. 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&M considers that the National Institute of Standards and Technology, or other State and Federal Agencies which may provide services to AEPSC/I&M, are not required to be audited.

12c. Sec. 4.5.1

Requirement

Responses to adverse audit findings, giving results of the review and investigation, shall clearly state the corrective action taken or planned to prevent recurrence. "In the event that corrective action cannot be completed within thirty days, the audited organization's response shall include a scheduled date for the corrective action."

Exception/Interpretation

AEPSC/I&M take the position that certain circumstances warrant more than thirty (30) days to completely investigate the cause and/or total impact of an adverse finding. For these circumstances, an initial thirty (30) day response will be provided which addresses a schedule for known corrective actions, the reason why additional investigation time is needed, and a schedule for completion of the investigation. These initial responses require the approval of the Director - Quality Assurance.

13. N45.2.13,

13a. 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&M 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."

13b. Sec. 3.2.3

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.

13c. 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.

13d. 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.

13e. 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&M establishes post-installation test requirements giving due consideration to supplier recommendations.

14. Reg. Guide 1.146/ANSI N45.2.23 and ANSI N45.2.2.12

14a. 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." |*

14b. 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 Director - Quality Assurance and is administered by designated QA Division section managers/supervisor who are certified lead auditors.

Audits are, in most cases, conducted by individual auditors, not by "audit teams." These auditors are certified in accordance with |* established procedures and are assigned by the responsible QA section manager/supervisor 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 Division section managers/supervisor, thereby meeting the requirements of ANSI N45.2.23 relative to "lead auditors", and "audit team leaders."

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

3 inches of mercury and causing explosion of vulnerable structures. Because of the severity of the phenomena, very few reliable measurements of tornado intensities exist. It is therefore difficult to dissociate wind and pressure effects, but the estimates given above are considered fairly reliable maximum values. This portion of Michigan has a significant tornado probability, as is apparent in the map shown in Figure 2.2-2. The 1^{o} latitude-longitude square containing Benton Harbor has had 13 tornadoes between 1916 and 1961 while some sectors in states to the southwest have had 70 to 90. This frequency of occurrence can be translated (after Thom)(3) into a probability of a tornado affecting the site once in 1042 years.

Ice Storms

Far less destructive, but far more probable, are the ice storms that frequent the north central states. Michigan lies in the belt where such storms are common and in the years from 1898 to 1965, 33 significant ice storms have been reported in this area.

2.2.3 DISPERSION METEOROLOGY

The micrometeorology of the site seems fairly typical of the northern lake regions. The sand dunes in the immediate vicinity cause some aberration of wind flow at low levels for short distances but, in general, the wind is vigorous, turbulent and uncomplicated over the entire area. The thermal stability shows approximately the seasonal variation expected close to large lakes, exhibiting almost no stable cases during the winter months, contrasted with a slightly greater frequency in inversions in the late spring and summer when the air temperature is usually warmer than that of the lake surface. Even in the least favorable month, however, the inversion frequency is only 22%. There are almost no instances in which stable lapse rates are accompanied by winds toward the heavily populated Chicago area.

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, including 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 Nuclear Plant. 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 of 85 sampling 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 atomic 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 and 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 resulting from the plant discharge. These observations were made 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 cladophora 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 (MSL), 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. The current lake level at the Cook Nuclear Plant is 578.3 feet above MSL.

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. Lake Michigan has what appears to be very stable conditions near shore despite severe storms and winter icing. Present evidence indicates that the nearshore sandbars 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 gross current features indicated above, there appears to be a thin, elongated, counterclockwise eddy close to the shore between Michigan City, Indiana and Benton Harbor (indicated by

X on Figure 2.6-2). This eddy may be controlling 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 alongshore currents are established and controlled by interactions between local winds and the regional current pattern. It should be noted, however, that local winds are the dominant factors in establishing alongshore currents.

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 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^{o} 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^{o}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:

<u>Northward</u>	
South Haven	32 miles
Benton Harbor	11 miles
St. Joseph	9 miles
<u>Southward</u>	
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

<u>Seiches</u>

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 great quantity of information available concerning other large bodies of water, 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.

Wind Waves

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.

Coincidence of Maximum Wave and Maximum Seiche

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.

Aquatic Life

The fauna and flora of Lake Michigan are similar to those of other very large oligotrophic lakes of North America. In addition to numerous species of phytoplankton and zooplankton there are a number of species of bottom-living organisms (benthos).

The benthos in the site region is almost completely dominated by four groups; amphipods, crustaceans with laterally compressed bodies reaching a maximum size of 1 cm; oligochaetes, aquatic earthworms less than an inch in length; sphaeriids, tiny clams about one to several mm in diameter; and tendepedids, the larvae aquatic insects commonly known as "midges". All of these species are eaten by fish to some degree, with the amphipods constituting an important source of fish food.

Fishes of Lake Michigan include carp, northern pike, sculpins, smelt, lake trout, yellow perch, and brown trout. Unfortunately, the present evidence indicates that a predominate portion of the fish biomass of the lake consists of an immigrant Atlantic herring, the alewife.

Substantial die-offs of alewives have occurred during the summers of some recent years; in other summers the die-off of alewives has been negligible. The cause (or causes) of alewife die-off are not known. Other power plants operating on the Great Lakes have found no change in the pattern of alewife die-offs before and after the plant began operating. In an attempt to control the alewife population Lake Michigan has been stocked in recent years with coho salmon and the native lake trout, both of which prey on alewives.

Biofouling Bivalves

Asiatic clams (Corbicula fluminea) and zebra mussels (Dreissena polymorpha) have been introduced to the Cook Nuclear Plant area as well as other locations in Lake Michigan. An Asiatic clam shell was found at the plant in 1983 and zebra mussels were discovered in the plant intake forebay in 1990.

Asiatic clams have caused serious clogging problems in water intake systems in the southern United States over the past 30 years or so. The Nuclear Regulatory Commission issued a bulletin requiring nuclear plants to monitor for Asiatic clam infestation in 1982. Asiatic clams are heat tolerant and cold intolerant. Water temperatures at the plant will prevent this species from becoming a serious biofouling organism at Cook Nuclear Plant.

Monitoring to ensure the Asiatic clam population remained low was begun in 1982 and has been conducted annually since then. Larval Asiatic clams (veligers) are monitored in filtered intake water samples, plant raw water systems are carefully inspected during routine maintenance, and the beach is surveyed to detect the empty shells of adults washed upon the beach by waves. One live clam and about a dozen shell halves have been found in eight years of monitoring. No veligers have been collected.

Zebra mussels have been the cause of serious biofouling problems in Europe and Russia for many years[8]. Water intakes for drinking water supplies and power plants have been clogged by zebra mussels in Lake Erie since they were first discovered in the St. Clair River in 1988. Zebra mussels are cold adapted animals and are considered a potentially serious biofouling problem at the Cook Nuclear Plant.

Since the discovery of zebra mussels at Cook Nuclear Plant, monitoring for biofouling bivalves was expanded to include zebra mussels. In addition to the same procedures used for Asiatic clams, artificial substrates will be used to monitor for zebra mussels. Diver inspections of the water intake system are also part of the zebra mussel monitoring program. Zebra mussel abundance has increased from one organism per five to seven m^2 to one to fifty organisms per m^2.

Biofouling control measures initiated at the plant include eradication treatments with a proprietary molluscicide and chlorine treatment to prevent zebra mussel veligers from settling. To date, this treatment has been effective. However, the Cook Nuclear Plant experience to date is too limited to draw meaningful conclusions.

2.6.5 EFFECT OF THE PLANT ON LAKE MICHIGAN ECOLOGY

The effect of the plant on lake ecology is local in nature, and results in no large scale disruption of existing ecological patterns. The submerged discharge structure and the rising plume of discharge water prevents exposure of the local benthos to warmed water. Except directly over the outfall structure, where warmed water is rising to the surface, there is water of ambient temperature under the plume of discharged water. Under these conditions, damage to bottom-spawning fish or thermal blocking of fish migration routes is believed impossible.

Some damage from turbulence and pumps is expected to be caused to zooplankton that pass through the plant, but compared to the substantial local population of these organisms the damage will be insignificant.

Some temporary loss of photosynthetic activity, from thermal shock, is expected to be experienced by phytoplankton passing through the plant. In the huge local population of these organisms such upset will be insignificant.

Experiments in which juvenile fishes are deliberately passed through generating plants indicate that they have a high rate of survival.

As in the common experience at other plants, several species of fish are attracted to the plume of discharge water in fall, winter, and spring. There is a popular sport fishery along the edges of the Cook Plant discharge plume.

REFERENCES, SECTION 2.6

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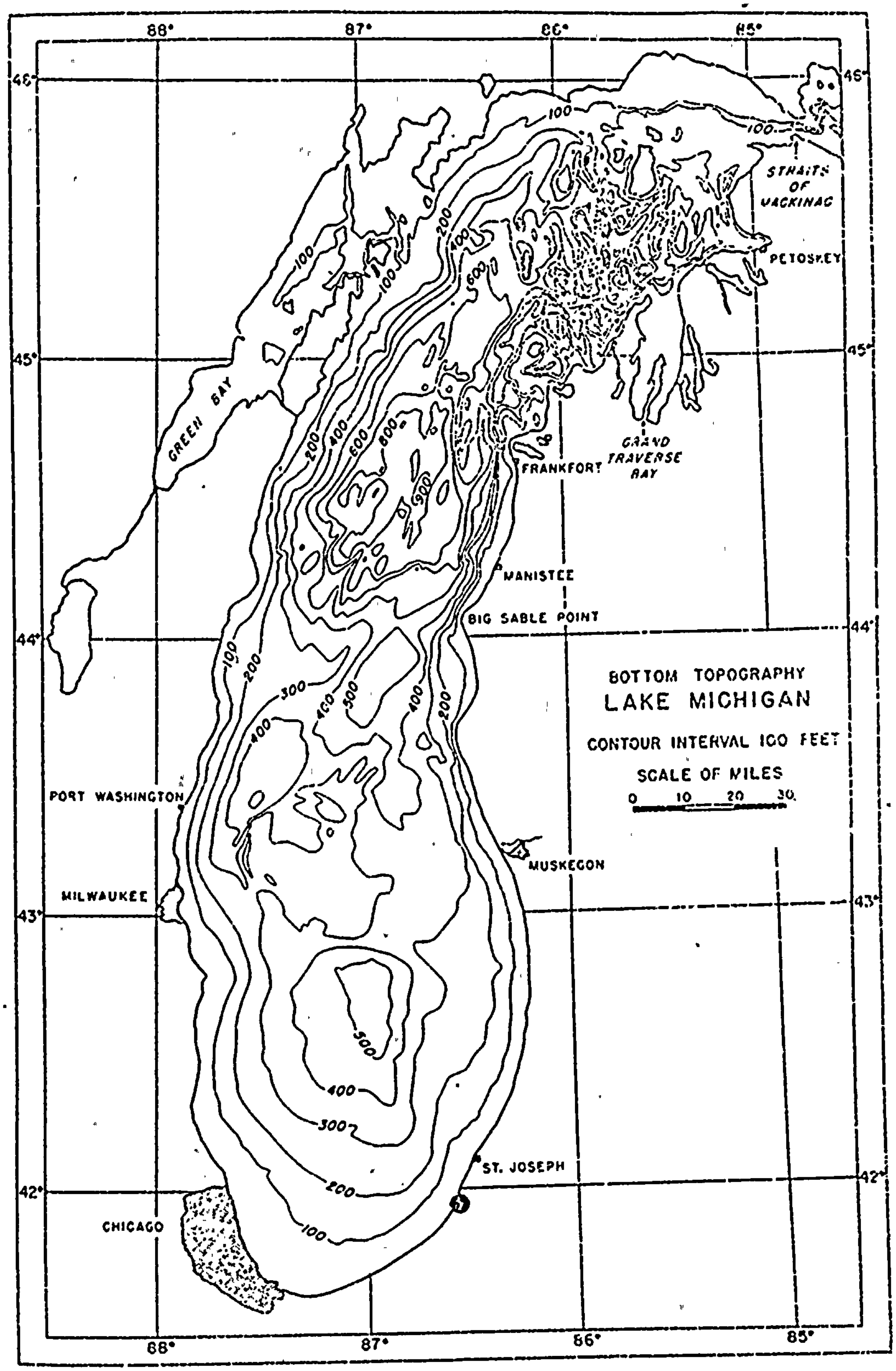
8. Mackie, G. L., W. N. Gibbons, G. W. Muncaster, and I. M. Gray, "The Zebra Mussel, *Dreissena polymorpha*: A Synthesis of European Experiences and a Preview for North America." Environment Ontario (ISBN 0-7729-5647-2), Toronto, 76 pp. and Appendices, 1989.

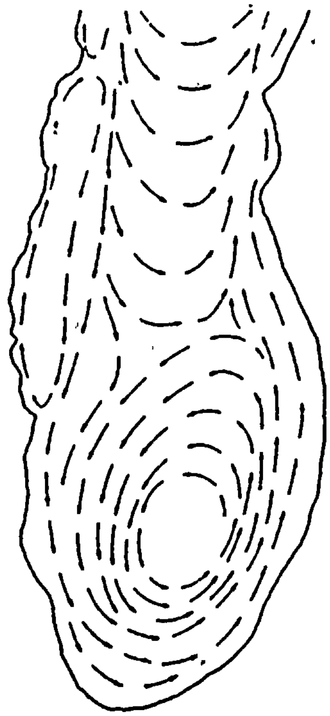
TABLE 2.6-1

EQUIPMENT

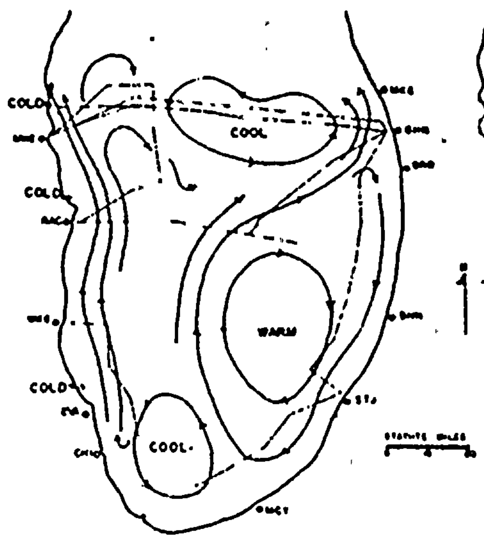
Equipment	Quantity	Location	Floor Elevation
Containment Spray Pumps	4	Auxiliary Bldg.	573 ft.
Residual Heat Removal Pumps	4	Auxiliary Bldg.	573 ft.
Spray Additive Tanks	2	Auxiliary Bldg.	587 ft.
Charging Pumps	6	Auxiliary Bldg.	587 ft.
Safety Injection Pumps	4	Auxiliary Bldg.	587 ft.
Emergency Diesel Generators	4	Auxiliary Bldg.	587 ft.
Diesel Generating Support Equipment:		Auxiliary Bldg.	587 ft.
Diesel Oil Transfer Pumps			
Jacket Water Expansion Tanks			
Jacket Water Circulating Pumps			
Lube Oil Coolers and Jacket Heat Exchangers			
Fuel Oil Day Tanks			
Starting Air Receiver Tanks			
Starting Air Compressors			
Auxiliary Feed Pumps	6	Turbine Bldg.	591 ft.
Essential Service Water Pumps	4	Screenhouse	591 ft.
Diesel Fuel Oil Storage Tanks	2	2000 from Lake Shore	Buried below 608 ft. grade



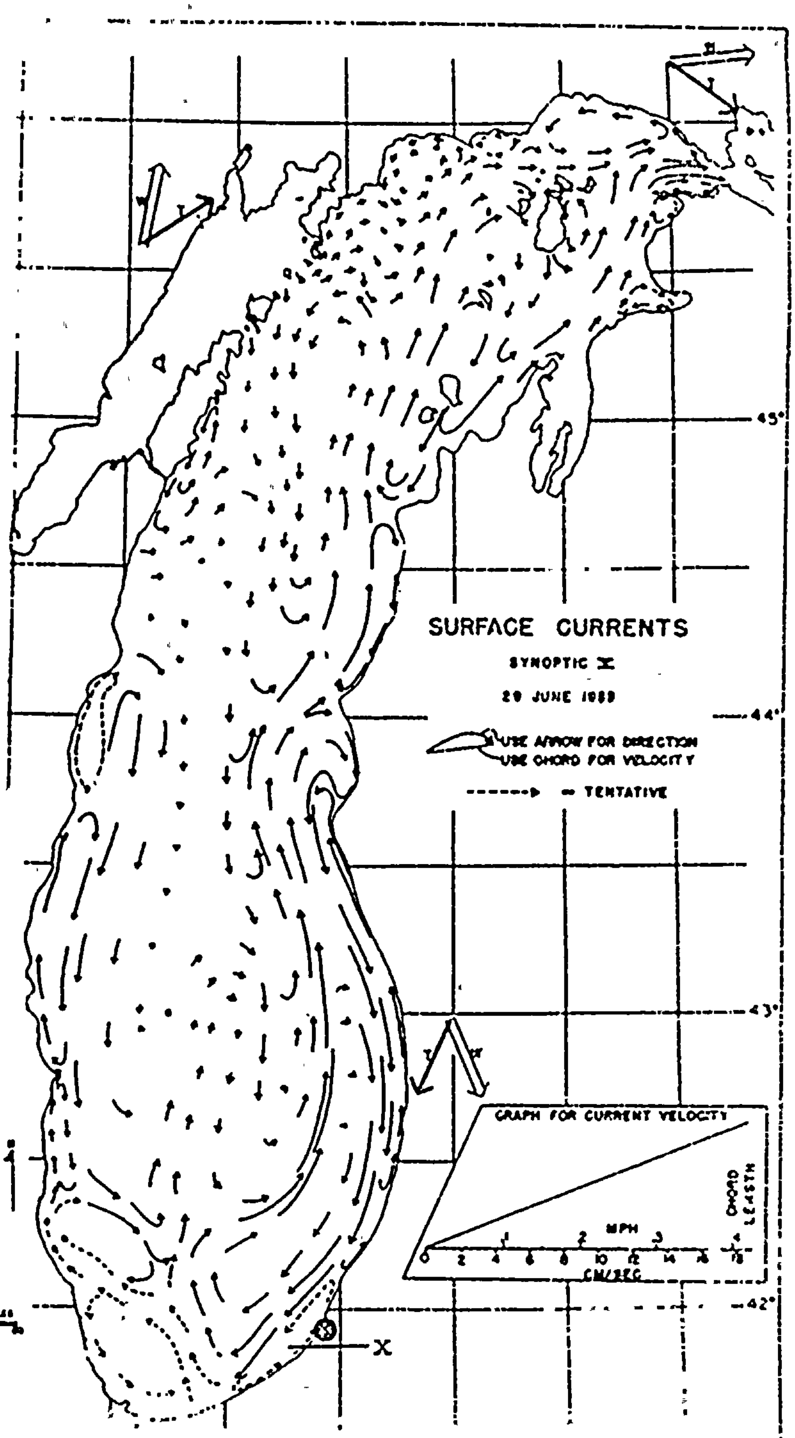
The bottom topography of Lake Michigan.
(from Hough 1958)



after Harrington (1895)
(from Hough 1958)

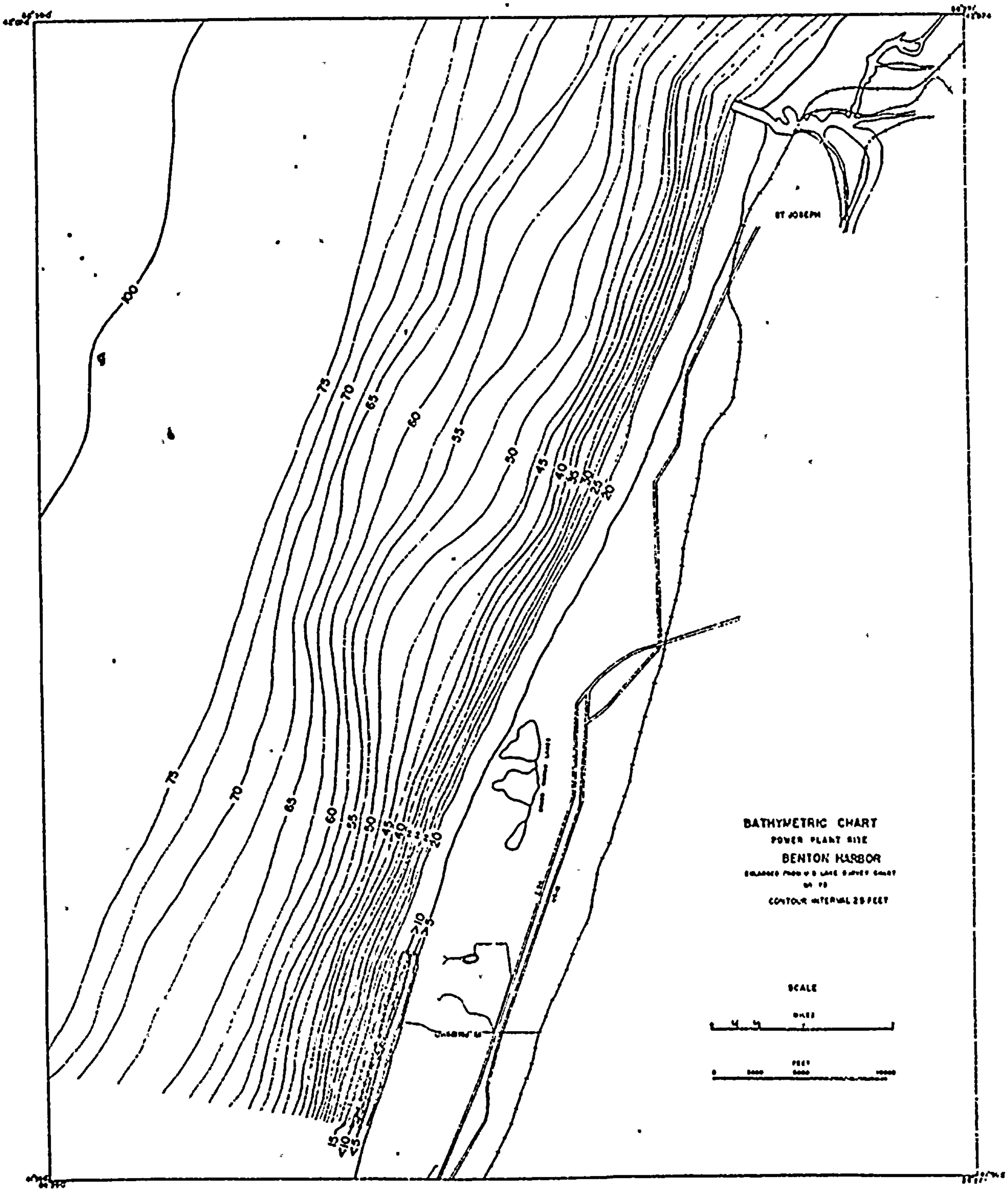


from Bellaire (1964)



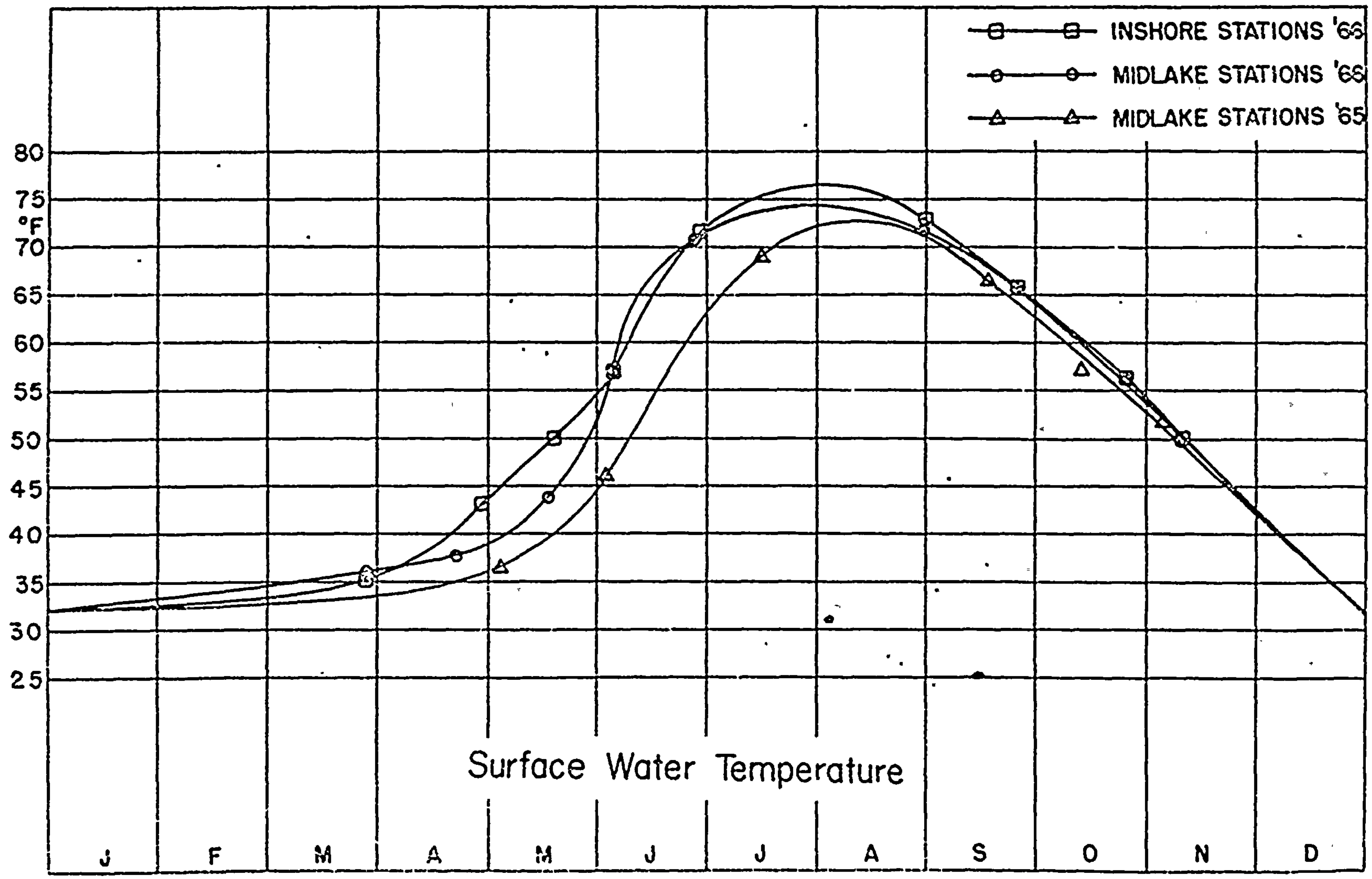
from Ayers et al (1958)

Three concepts of the surface currents of Lake Michigan.



July, 1982

FIG. 2.6-3



July, 1982

FIG. 2.6-4

2.7.2 SAMPLING STATIONS

The stations for sampling airborne particulates, volatiles, 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 2.7-3 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, 2.7-3 and 2.7-4. Twelve on-site indicator TLD stations (shown as A1 through A12 on Figure 2.7-1) are located on an approximate 2000 foot radius and eleven off-site monitoring TLD stations are within a 2 to 5 mile radius from the plant (shown as T1 through T11 on Figure 2.7-4). Four background TLD stations located about 20 miles from the plant are identified as A on Figure 2.7-3.

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 the condenser cooling water intake and at an approximate distance of 623 and 1278 feet north and 657 and 1842 feet south of the plant centerline.

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 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.

<u>Sampling of Well Water</u>

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 eleven wells (seven REMP wells and four non technical specification steam generator storage facility groundwater monitoring wells) within the owner controlled area; three are west of the plant north-south axis, and eight are east of the plant north-south axis as shown in Tables 2.7-2 and 2.7-3. 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.

<u>Sampling of Milk</u>

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

TABLE 2.7-1

LOCATIONS OF THE WATERBORNE SURFACE SAMPLING STATIONS

Indicator Stations

Condenser cooling water intake (L1).

0.35 miles southwest from plant centerline along the lake shore (L2).

0.24 miles northeast from plant centerline along the lake shore (L3).

0.12 miles southwest from plant centerline along the lake shore (L4).

0.12 miles northeast from plant centerline along the lake shore (L5).

(See Figure 2.7-1)

Background Stations (and drinking water sample stations)

Lake Township water intake, 0.40 miles south from the plant (D_A).*

St. Joseph municipal water intake, 9 miles northeast from the plant (D_B).*

(See Figure 2.7-3)

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*D_A and D_B refer to analysis performed as indicated in Table 2.7-5.

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.	Approximate Distance from Plant in Feet	Direction from North
W-1	1969	11°
W-2	2292	63°
W-3	3279	107°
W-4	418	301°
W-5	404	290°
W-6	424	273°
W-7	1895	189°

TABLE 2.7-3

NON TECHNICAL SPECIFICATION GROUNDWATER WELLS

STEAM GENERATOR STORAGE FACILITY

(Refer to Figure 2.7-2 for a map indicating the location of these sample points)

Well No.	Approximate Distance from Plant in Feet	Direction from North
SGR-1	4037	95°
SGR-2*	3879	92°
SGR-4	3699	93°
SGR-5	3649	92°

These wells are sampled and analyzed quarterly for:

- Gross alpha Activity
- Gross Beta Activity
- Gamma Isotopic Activity

*No SGR-3 well defined for this program.

TABLE 2.7-4

LOCATIONS OF THE MILK SAMPLING STATIONS

INDICATOR FARMS

G. G. Shuler & Sons Baroda, MI	Totzke Farms Baroda, MI
Paul Lozmack Galien, MI	Willie Warmbein Three Oaks, MI
Norman Zelmer Bridgman, MI	

BACKGROUND FARMS

Vic Wyant
Dowagiac, MI

Ray Livinghouse
LaPorte, IN

TABLE 2.7-5

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
1. Airborne a. Radioiodine 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 Radioactivity following Filter Change[a], composite (by location) for gamma isotopic quarterly.
2. Direct Radiation	a) A1-A12 (On-Site) b) New Buffalo, South Bend, Dowagiac, Coloma c) 11 Off-Site TLD Monitor Locations	At least once per 92 Days (Quarterly)	Gamma Dose. At Least Once Per 92 Days.
3. Waterborne a. Surface	L1, L2, L3, L4, L5	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.

[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.

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TABLE 2.7-5 (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(D_B) Lake Township(D_A)	Composite* Sample Collected over a Period of less than or equal to 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, L4, L5	Semi-Annually	Gamma Isotopic Analysis Semi-Annually.
4. Ingestion a. Milk	Indicator Farms** Background Farms	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.

**An indicator farm is defined as the nearest milk producer in each of the land sectors within 8 miles of the plant site who is willing to participate in the radiological environmental monitoring program. A background farm is defined as a milk producer in one of the less prevalent wind directions at a distance greater than 15 miles but less than 25 miles who is willing to participate in the radiological environmental monitoring program. If at least three indicator milk samples and one background milk sample cannot be obtained, vegetation sampling will be performed as a replacement for the milk sampling and no milk samples will be required.

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TABLE 2.7-5 (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 (N/S)* Off-Site (N/S)*	2/year (Semi-Annually).	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.
	3 indicator samples of broad leaf vegetation grown nearest to the offsite locations of highest calculated annual average ground level D/Q if at least three indicator milk samples and one background milk sample cannot be obtained.	Monthly when available	Gamma Isotopic and I-131 monthly when available

*N/S - North and South of Plant Site.

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TABLE 2.7-5 (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
	1 background sample of each of the similar vegetation grown 15-25 miles distant and in one of the less prevalent wind directions if at least three indicator milk samples and one background milk sample cannot be obtained.	Monthly when available	Gamma Isotopic and I-131 monthly when available

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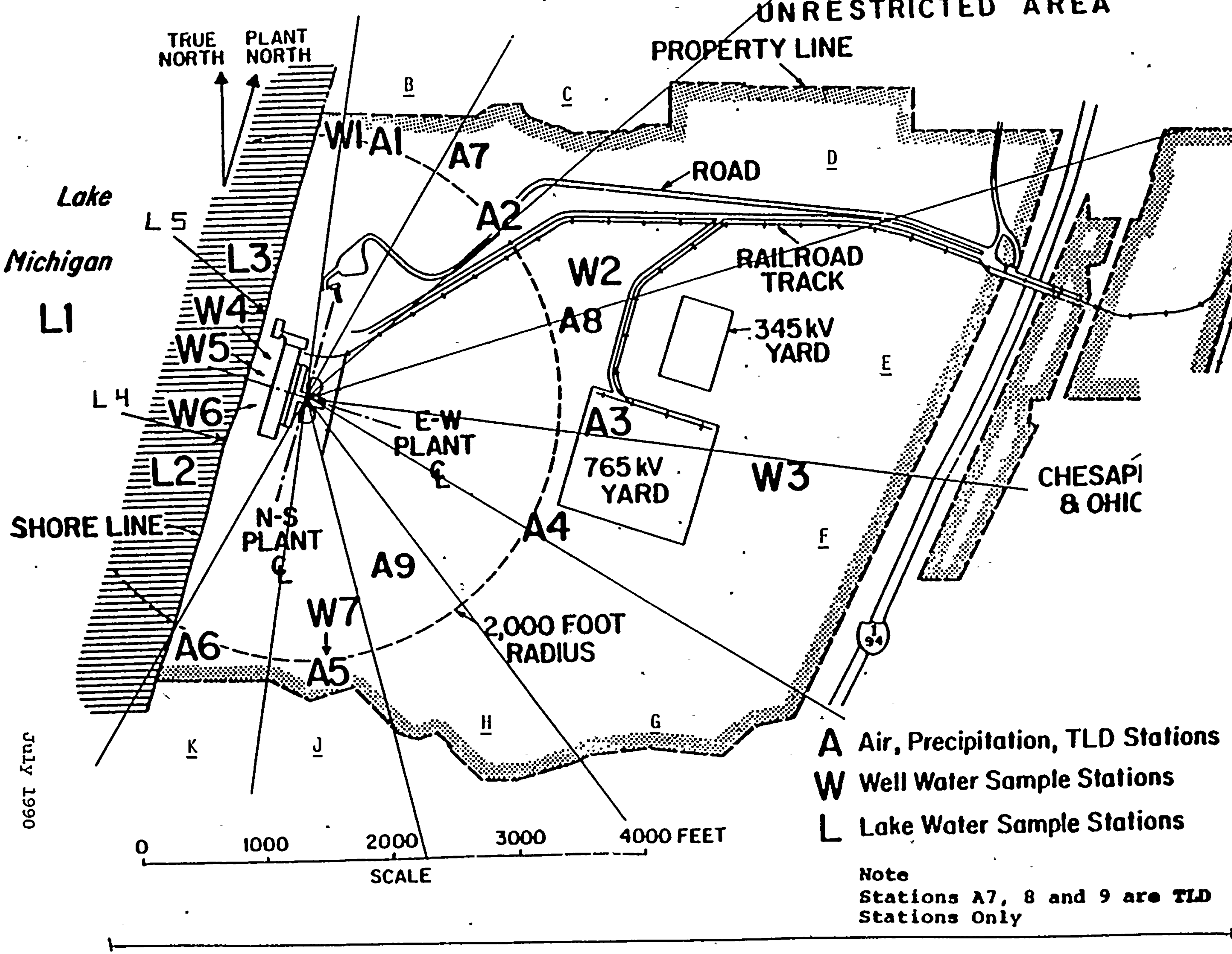
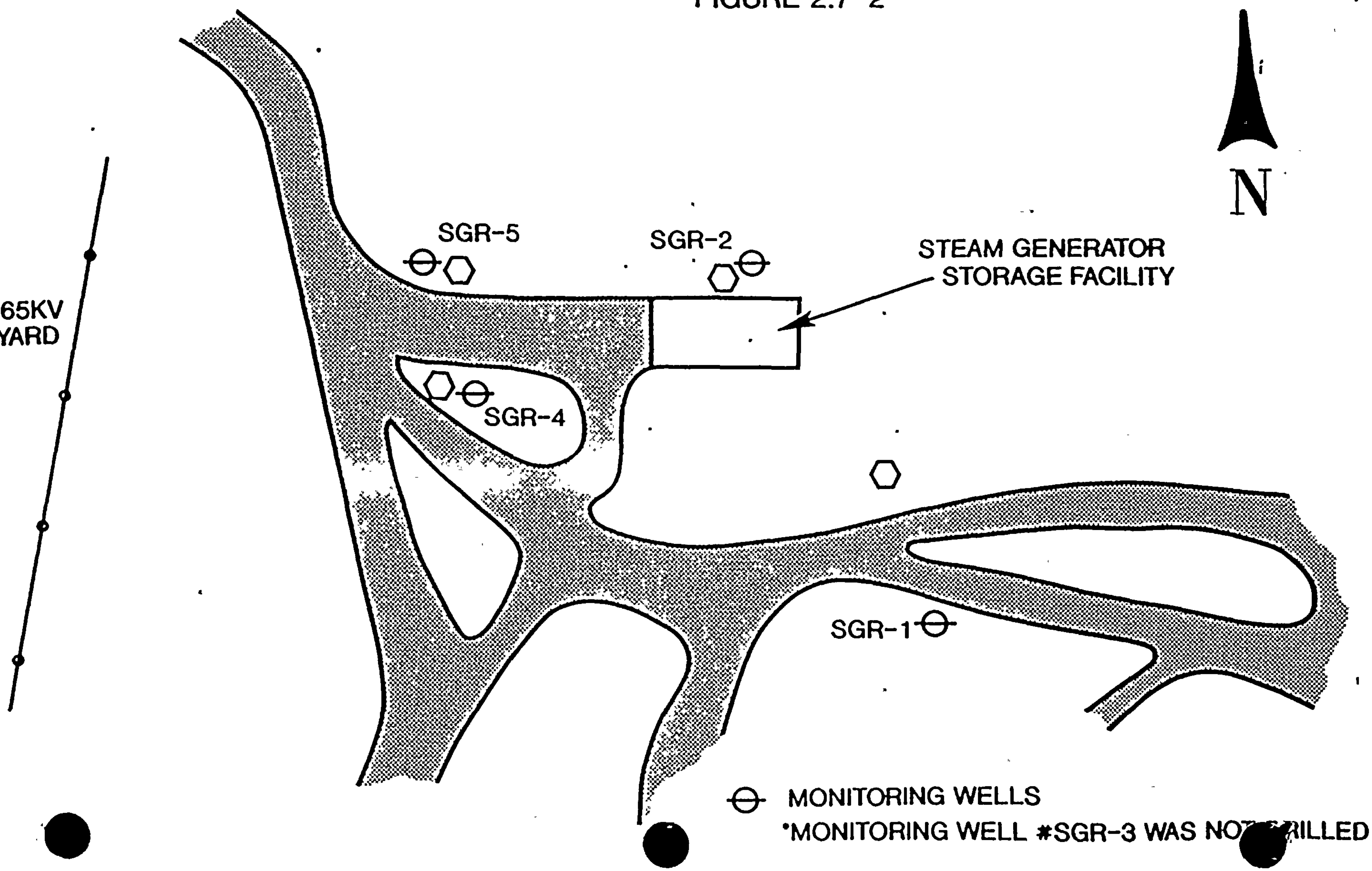
FIGURE 2.7-1


UNRESTRICTED AREA
PROPERTY LINE
TRUE NORTH
PLANT NORTH
Lake Michigan
ROAD
RAILROAD TRACK
345 kV YARD
765 kV YARD
E-W PLANT ℄
N-S PLANT ℄
SHORE LINE
2,000 FOOT RADIUS
CHESAPEAKE & OHIO
B
C
D
E
F
G
H
J
K
L1
L2
L3
L4
L5
W1
W2
W3
W4
W5
W6
W7
A1
A2
A3
A4
A5
A6
A7
A8
A9
94
0
1000
2000
3000
4000 FEET
SCALE
A Air, Precipitation, TLD Stations
W Well Water Sample Stations
L Lake Water Sample Stations
Note
Stations A7, 8 and 9 are TLD Stations Only

STEAM GENERATOR STORAGE FACILITY
NON-TECHNICAL SPECIFICATION
GROUNDWATER MONITORING WELLS

FIGURE 2.7-2



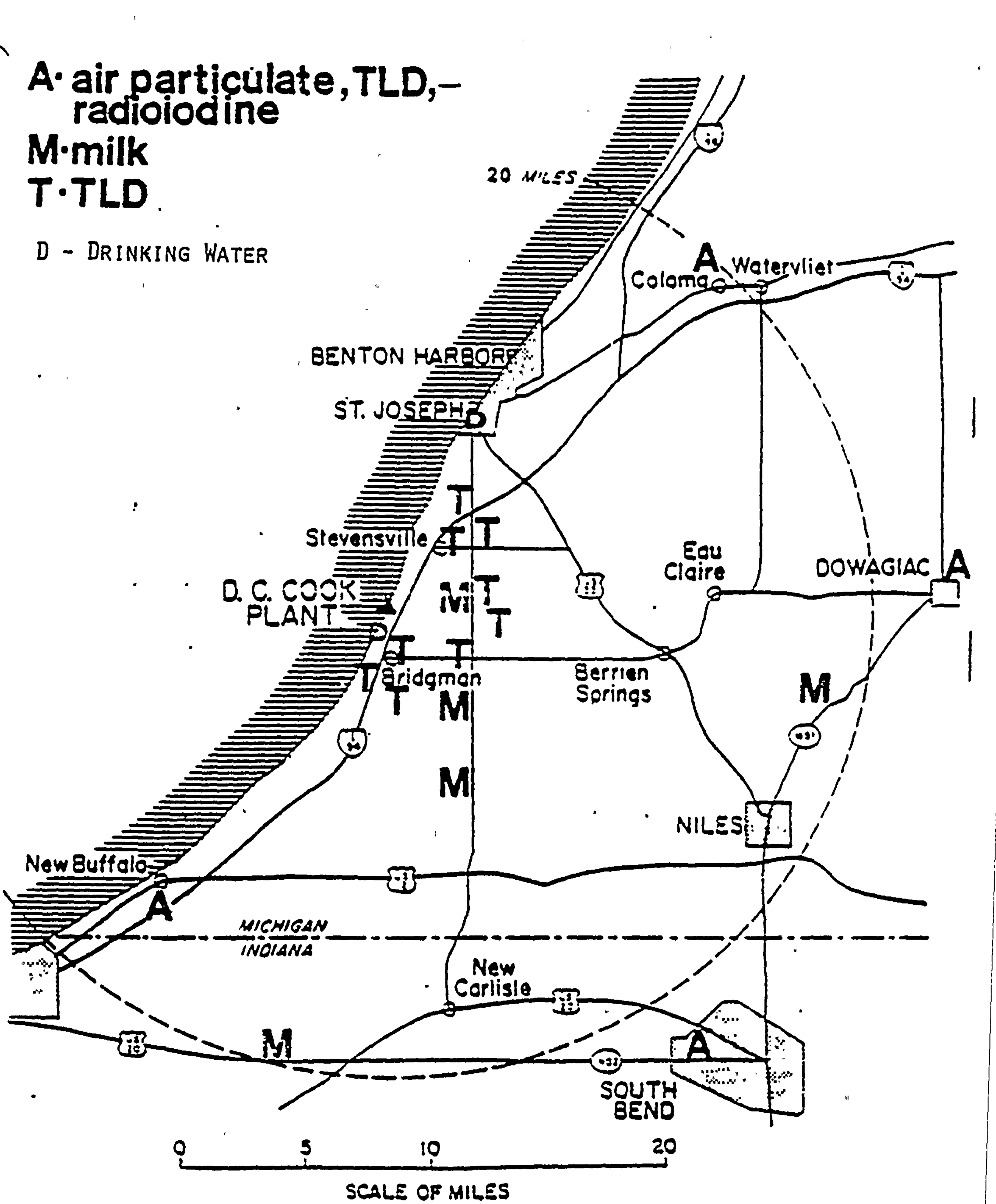


FIGURE 2.7-3

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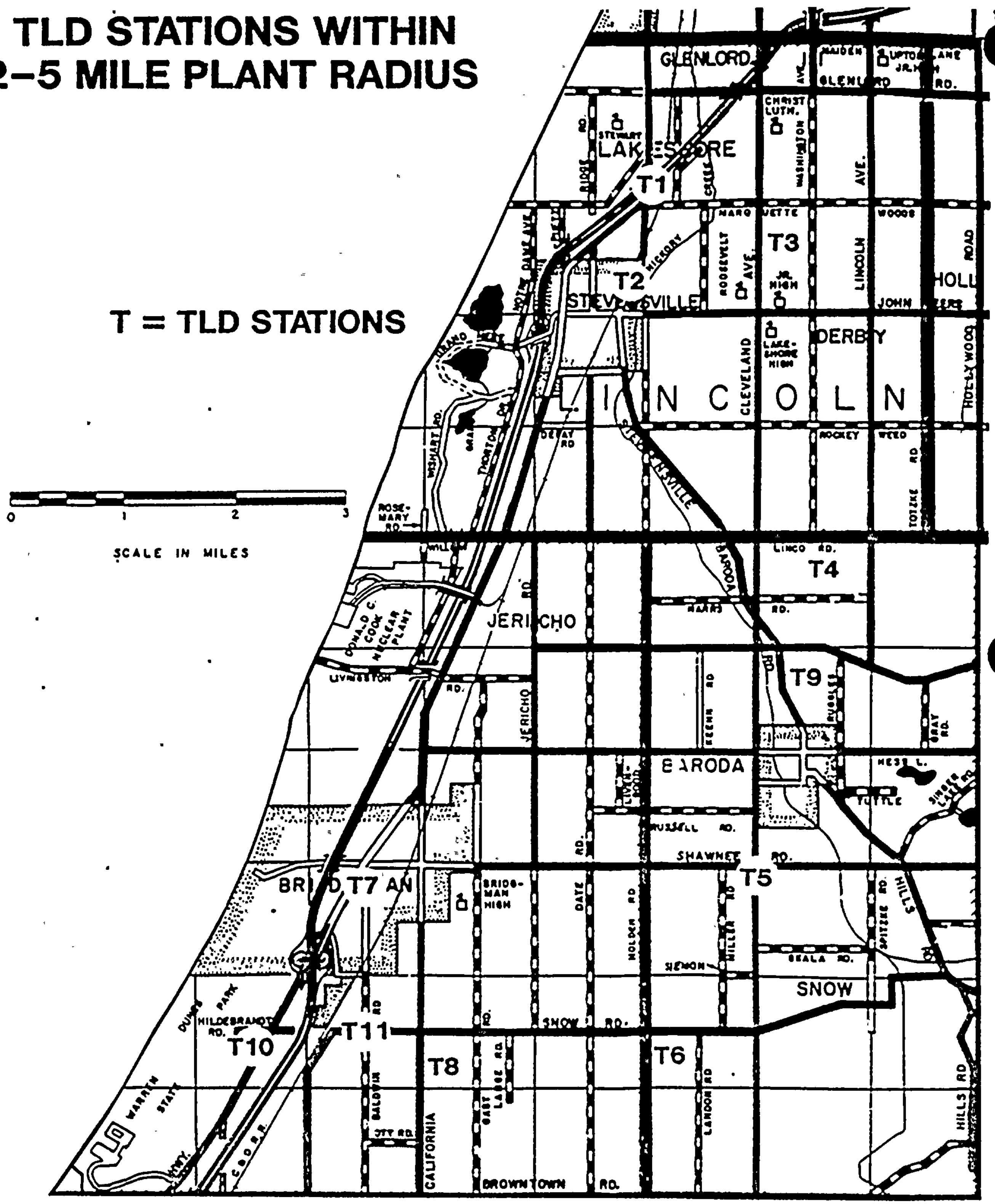
FIGURE 2.7-4

TLD STATIONS WITHIN 2-5 MILE PLANT RADIUS

T = TLD STATIONS



Item	Class
<u>Reactor Coolant System</u>	I
Piping and valves (including safety & relief valves)	
Steam generators	
Pressurizer	
Reactor coolant pumps and motors	
Reactor coolant system supports	
<u>Emergency Core Cooling System</u>	I
Accumulators	
Residual heat removal system	
Safety injection system	
Centrifugal Charging system	
Boron Injection Tank	
Refueling Water Storage Tank	
<u>Containment Spray System</u>	I
Spray additive tank	
<u>Chemical & Volume Control System</u>	
Letdown and makeup components	I
Seal water system	I
Boric acid storage tanks and transfer pumps	I
Cleanup demineralizers and filters	II
Boric acid recovery equipment	II
<u>Condensate Storage Tank</u>	II
<u>Auxiliary Feedwater System</u>	I
<u>Essential Service Water System</u>	I
<u>Component Cooling System</u>	I
<u>Emergency Power Generation and Distribution System</u>	I

Item	Class
Ventilation Systems	
Engineered safety features ventilation system	I
Control room ventilation system	I
Auxiliary feedwater pump enclosure ventilation system	I
Essential service water pump ventilation system	I
Emergency power ventilation systems	I
Containment Ventilation System	II & III
Turbine Room Ventilation System	III
Waste Disposal System	
Gas decay tanks	I
Liquid waste holdup tanks	II
Waste evaporator	II
Waste condensate tanks	III
Waste evaporator condensate pumps	III
Non-Essential Service Water System	II
Primary Water System	
Primary water storage tank	II
Primary water make-up pumps	II
Balance of System	III
Control Air System	
Air compressor and receiver	II
Fire Protection System	III

3.0 REACTOR

3.1 SUMMARY DESCRIPTION

The current Cycle 11 reactor core contains six regions of fuel in a low leakage loading pattern as described in Section 3.5.2. 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 Cycles 8 through 11, the absorber material in the fixed burnable absorber rods is in the form of annular aluminum oxide-boron carbide absorber pellets contained within two concentric Zircaloy tubes with water flowing through the center tube as well as around the outer tube.

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 Corp. (W). Reload fuel for Cycles 2 through 7 was supplied by Exxon Nuclear Co (ENC). Cycles 8 through 11 reload fuel was supplied by Westinghouse Electric Corp. The latest information regarding the current fuel cycle may be found in Sub-Chapter 3.5.

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); and a description of the current core design (Sub-Chapter 3.5).

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 Cycles 8 through 11, yield cycle burnups of between 15,000 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 3.5.3). This is accomplished for the current cycle 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 the 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 value for the fuel. For the current cycle, the design values are given in Section 3.5.3.

b) Fuel center temperature below melting point of UO_2

c) For 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 W reload fuel in the current cycle, 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 W reload fuel in the current cycle.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by analyses described in Chapter 14 to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient.

A loss of external electrical load of 100% of full power or less is normally controlled by rod cluster insertion, together with a controlled steam dump to the condenser, to prevent a large temperature and pressure increase in the reactor coolant system. In this case, the overpower-overtemperature protection would guard against any combination of pressure, temperature, and power which could result in a DNB ratio less than the applicable design value during the transient.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, and control rods can be used to supress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. (In-core instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.) The analysis, detection and control of these oscillations is discussed in Reference 2) of Sub-Chapter 3.3.

Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided.

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving chemical shimming.

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 boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of station power. Boric acid can be injected by one pump at a rate which takes the plant to 1% shutdown in the hot condition with no rods inserted in less than sixteen minutes. In sixteen additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin until approximately 15 hours after shutdown. If two boric acid pumps are available, these time periods are halved. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water storage tank. This solution can be transferred directly by the charging pumps or alternately by the safety injection pumps.

The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Chapters 14 and 9 respectively.

Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups which are used to control reactivity changes due to load changes and to control reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is 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 7.5×10^{-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 current predicted nuclear heat flux hot channel factor, F_Q does not exceed 2.15 for W 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 Nuclear Plant Unit 1 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 3.5.3.

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 the current cycle, design limit is given in Section 3.5.3.

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.

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 in the current Technical Specifications which applies to cores with W OFA reloads. These current limits are incorporated in Section 3.5. Nuclear characteristics of the current reload fuel are discussed in Section 3.5.

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 per cent 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 per cent 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 per cent 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 without 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.

Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods is also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

Control rod reactivity requirements at beginning and end of life are summarized in Table 3.3.1-2. The calculated worth of the control rods is shown in Table 3.3.1-3.

The difference is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change incurred with a change in power level due to the Doppler effect. The magnitude of this change has been established by correlating the experimental results of numerous operating cores.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range is given in

Table 3.3.1-1, line 42, while the cumulative reactivity change is shown in the first line of Table 3.3.1-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

Control Rod Bite

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five per cent per minute, or by a step load change of ten per cent. An insertion rate of 4×10^{-5} $\Delta\rho$ per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate one control bank of rods should remain partly inserted into the core.

Xenon Stability Control

Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, has shown that any induced radial or diametral xenon transients would die away naturally. A full discussion of xenon stability control can be found in Reference 2.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing one per cent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position or to prevent return to criticality following a credible steam-line break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam break accident is considered. The excess control available at the end of cycle, hot zero power condition with the highest worth rod stuck out, allowing a 10% margin for uncertainty in control rod worth, is shown in Table 3.3.1-3.

Calculated Rod Worths

The complement of 53 full length control rods arranged in the pattern shown in Figure 3.3.1-1 meets the shutdown requirements. Table 3.3.1-3 lists the calculated worths of this rod configuration for beginning and end of the first cycle. In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed are decreased in the design by 10 per cent to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating reactors show the calculation to be well within the allowed uncertainty of 10%.

Power Distributions

The Donald C. Cook Nuclear Plant is required to meet the Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors as specified in 10 CFR 50.46 and Appendix K to 10 CFR 50. It is necessary to limit the core heat flux hot channel factors, F_Q, to values which would result in peak clad temperatures below 2200°F following a loss of coolant accident and also assure other ECCS related criteria are met (see Chapter 14).

The peaking factor limits at full power for the plant can be met by operation using either the Power Distribution Control Procedure (PDC-II), or the Axial Power Distribution Monitoring System (APDMS), with both methods requiring limits on the amount of axial offset that is allowed. The material presented below provides information on the current technical basis for operation with constant axial offset control, and PDC-II that is currently reflected in the Technical Specifications.

The accuracy of power distribution calculations has been confirmed through over 1000 flux maps during over 20 plant years of operation under conditions very similar to those for the plant described herein. Details of this confirmation are given in Reference (8).

The means for maintaining power distributions within the required hot channel factor limits are described in the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWR's is included in Reference (2). Detailed background information on the following: design constraints on local power density in a Westinghouse PWR, the defined operating procedures and the measures taken to preclude exceeding design limits; is presented in the Westinghouse Topical Report on power distribution control and load following procedures. The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F_{\Delta H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition and fuel and moderator temperature feedback effects are included for the average enthalpy plane of the reactor. The steady-state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on the radial power distribution is small but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and local changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plane. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

1. Core power level.
2. Core height.
3. Coolant temperature and flow.
4. Coolant temperature as a function of reactor power.
5. Fuel cycle lifetimes.
6. Rod bank worths.
7. Rod bank overlaps.

Normal operation of the plant assumes compliance with the following conditions:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 13 steps (indicated) from the bank demand position.

2. Control banks are sequenced with overlapping banks.

3. The control full length bank insertion limits are not violated.

4. Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. Briefly they require control of the axial offset (flux difference divided by fractional power) at all power levels within a permissible operating band of a target value corresponding to the equilibrium full power value. In the first cycle, the target value changes from about -10 to 0 percent through the life of the cycle. This minimizes xenon transient effects on the axial power distribution, since the procedures essentially keep the xenon distribution in phase with the power distribution.

Calculations are performed for normal operation of the reactor including load following maneuvers. Beginning, middle and end of cycle conditions are included in the calculations. Different histories of operation are assumed prior to calculating the effect of load follow transients on the axial power distribution. These different histories assume base loaded operation and extensive load following. For a given plant and fuel cycle a finite number of maneuvers are studied to determine the general behavior of the local power density as a function of core elevation.

These cases represent many possible reactor states in the life of one fuel cycle and they have been chosen as sufficiently definitive of the cycle of comparison with much more exhaustive studies performed on different, but typical, plant and fuel cycle combinations. The cases are described in detail in Reference (6) for the Westinghouse analysis, and they are considered to be necessary and sufficient to generate a local power density limit which, when increased by 5 percent for conservatism, will not be exceeded with a 95 percent confidence level. Many of the numerous amounts of points do not approach the limiting envelope, however, they are part of the time histories which lead to the hundreds of shapes which do define the envelope. They also serve as a check that the reactor studied is typical of those studied more exhaustively.

Thus it is not possible to single out any transient or steady-state condition which defines the most limiting case. It is not even possible to separate out a small number which forms an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver which provides a limiting case for one reactor fuel cycle is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnup, coefficient, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operation conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

The calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. In these calculations the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two dimensional X-Y calculations. A 1.03 factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor F_Q^E.

For reload cores required to satisfy the Final Acceptance Criteria (10 CFR 50.46) for the Loss of Coolant Accident, the total core peaking factor (F_Q times relative power) is evaluated as a function of core height and compared to the Technical Specification limit. All of the nuclear effects which influence axial power distributions throughout the fuel cycle are included in the evaluation of the total peaking factor. Various modes of load follow and base load operation are considered. This evaluation is based on normal plant operation in compliance with the Technical Specifications.

For cores that operate within the limits of Constant Axial Offset Control (CAOC), the evaluation is initiated by determining whether the core operates within the following constraints:

1. The Technical Specification limit on the maximum height dependent F_Q is equal to or less than a value of 2.04 for ENC fuel and 2.10 for W for Cycle 8 operations, and

2. The CAOC flux difference ($\pm \Delta I$) bandwidth is less than or equal to $\pm 5\% \Delta I$.

These procedures are detailed in the Technical Specifications and are predicted only upon excore surveillance supplemented by the normal monthly full core map requirement, and by computer based alarms on deviation and time of deviation from the allowed flux difference band.

Accident analyses for this plant are presented in Chapter 14 (Unit 1) of the Cook Nuclear Plant FSAR. The results of these analyses determined a limiting value of total peaking factor, F_Q^L, under normal operation, including load following maneuvers. This value is derived from the conditions necessary to satisfy the limiting conditions specified in the LOCA analyses of Section 14.3.1, which meet Appendix K requirements. An upper bound envelope of F_Q^{ND} results from operation in accordance with Constant Axial Offset Control procedures using excore surveillance only.

The surveillance of the core hot channel factors in accordance with the above, is presented in the Cook Nuclear Plant Unit 1 Technical Specifications.

The Power Distribution Control Procedure,[7] PDC-II, enables Cook Nuclear Plant Unit 1 to manage core power distributions such that Technical Specification Limits on F_Q^T are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients.

The PDC-II procedure provides the means for predicting the maximum $F_Q^T(Z)$ distribution anticipated during operation under the PDC-II procedure taking into account the incore measured equilibrium power distribution. A comparison of this distribution with the Technical Specification limit curve determines whether the Technical Specification limit can be protected by the PDC-II procedure. If such protection can be confirmed for a given operating cycle interval, APDMS monitoring is not necessary over this interval and the excore monitored constant axial offset limits will protect the Technical Specification F_Q^T limits.

The prediction of the maximum anticipated $F_Q^T(Z)_{Max}$ distribution is made possible by controlling the distribution such that it does not increase by more than the factor V(z) times the equilibrium power distribution $F_Q^T(z)Eq$. This is accomplished by maintaining the core axial offset within a specified range of values about a target value associated with the equilibrium power distribuiton. The value of the V(z) factor is determined from analysis of plant operation data during which the axial offset is maintained within a specified band about the equilibrium (target) axial offset. The core axial offset (AO) has been previously defined in the subsection entitled, Axial Power Distributions.

A positive axial offset signifies a power shift toward the top half of the core, while a negative axial offset signifies a power shift toward the bottom half of the core.

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 cycle dependent V(z) factor, 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 11 in Figure 3.5.2-2. 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 (MWt)

P_o = reactor rated power (MWt)

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.

Moderator Temperature Coefficient*

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to

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*Chapter 14 discusses operation with a positive temperature coefficient. The value currently allowed by the Technical Specifications is 0.5×10^{-4} $\Delta k/k/^{o}F$ below 70% rated thermal power and 0×10^{-4} $\Delta k/k/^{o}F$ at or above 70% rated power. Amendment 30 to the Unit 1 Technical Specifications and the associated documentation provides the basis of this change.

the coefficient and in a chemical shim core, the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to boron being removed from the core. This is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable poison rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison and the moderator temperature coefficient will be reduced.

The Westinghouse burnable poison is in the form of borated pyrex glass rods clad in stainless steel. In Cycle 1, there were 1436 of these rods in the form of clusters distributed throughout the core in vacant rod cluster control guide tubes as illustrated in Figures 3.3.1-11 and 3.3.1-12. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in Reference 1. These rods initially controlled 9.0% of the installed excess reactivity and their addition resulted in a reduction of the initial hot full power boron concentration. The moderator temperature coefficient is negative at the operating coolant temperature with this boron concentration and with burnable poison rods installed.

The effect of burnup on the moderator temperature coefficient is calculated and the coefficient becomes more negative with increasing burnup. This is due to the buildup of fission products with burnup and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by boron, and as xenon builds up, boron is taken out. With core burnup, the coefficient will become more negative as boron is removed, and because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products.

The control rods provide a negative contribution to the moderator coefficient as can be seen from Figures 3.3.1-13, 14, and 15 which are Cycle 1 initially calculated values.

Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than a half-degree change in moderator temperature. The calculated beginning and end of life pressure coefficients are specified in Table 3.3.1-1, Line 43.

Moderator Density Coefficient

A uniform moderator density coefficient is defined as a change in the neutron multiplication* per unit change in moderator density. The range of the moderator density coefficient from BOL to EOL is specified in Table 3.3.1-1, Line 44.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication per degree change in fuel temperature. The coefficient is obtained by calculating neutron multiplication as a function of effective fuel temperature(3). The results from initial calculations are shown in Figure 3.3.1-16.

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*Neutron multiplication is defined as the ratio of the average number of neutrons produced by fission in each generation to the total number of corresponding neutrons absorbed.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power, as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse fueled cores. Figure 3.3.1-17 shows the power coefficient as a function of power obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

Nuclear Evaluation

The basis for confidence in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other Safety Analysis Reports such as the FSAR for Indian Point Unit 2, Docket No. 50-247, Section 3.2.1, and the PSAR for Cook Nuclear Plant, Docket No. 50-315-316, Section 3.2.1.

Extensive analyses on the threshold to xenon instabilities as a function of variation in core parameters (power coefficient, etc.) have been reported in Reference 4.

Finally, verification of design analysis during the startup physics tests is described in Section 3.3.2.

3.3.2 PHYSICS TESTS

Tests to Confirm Reactor Core Characteristics

A detailed series of startup physics tests were performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements were made over the entire range of operation in terms of RCCA configuration and power level by means of the incore movable detector system. In addition, rod worth, born end-point, and reactivity coefficient measurements were made.

Within relevant acceptance criteria, these test results show good agreement with design predictions[1]. To detect and eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relationship between fuel burnup and the boron concentration was normalized to accurately reflect actual core conditions. When full power was initially reached, and with the control groups in the desired positions, the boron concentration was measured and the predicted curve was adjusted to this point. As power operation continued, the measured boron concentration was compared with the predicted concentration and the slope of the predicted curve relating burnup and reactivity was corrected as necessary. This normalization was completed after about 10 percent of the total core burnup has occurred. Thereafter, actual boron concentration was compared with the predicted concentration, and the reactivity prediction of the core was continuously evaluated. No reactivity anomaly greater than one percent was observed.

In addition, periodic full-core flux maps were taken, using the incore detector system, to monitor power distribution, heat flux hot channel factors, enthalpy hot channel factors, quadrant power tilt ratios and axial flux differences. These measurements were utilized to ensure compliance with technical specifications.

3.3.3 ANTICIPATED TRANSIENTS WITHOUT TRIP

In the Code of Federal Regulations, 10 CFR 50.62(c)(1) requires that each pressurized water reactor have equipment, from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an anticipated transient without trip. Such equipment has been installed at Cook Nuclear Plant, having been designed in accordance with Reference 1. This equipment will protect against reactor coolant system overpressurization in the event that a loss of normal feedwater or a loss of load transient is not accompanied by a reactor trip after having reached the reactor trip setpoint.

3.3.4 CRITICALITY OF FUEL ASSEMBLIES

Information on criticality of the fuel assemblies outside of the reactor is presented in Section 3.3 of the Unit 2 FSAR and in Section 9.7.

REFERENCES, SECTION 3.3.1

1) Wood, P. M., Bassler, E. A., et al, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors", WCAP-7413 (October 1967).

2) Moore, J. S., "Power Distribution Control in Westinghouse Pressurized Water Reactors", WCAP-7811, December 1971.

3) Barry, R. F., "The Revised LEOPARD Code - A Spectrum Depending Non-Spatial Depletion Program", WCAP-2759, March 1965.

4) Poncelet, C. G., and Christie, A. M., "Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors", WCAP-3680-20 (1968).

5) McFarlane, A. F., "Core Power Capability in Westinghouse PWR's" WCAP-7267-L, October 1969.

6) Morita, T., et al. "Power Distribution Control and Load Follow Procedures", WCAP-8385, September 1974.

7) "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase 2" XN-NF-77-57, January 1978.

8) Langford, F. L. and Nath, R. J., "Evaluation of Nuclear Hot Channel Factor Uncertainties", WCAP-7038-L, April, 1969 (Proprietary) and WCAP-7810, December 1971 (Non-Proprietary).

9) Letter from R. F. Hering to H. R. Denton dated April 7, 1982, AEP:NRC:0665.

REFERENCES, SECTION 3.3.2

1) Nelson, J. F., et al, "Summary Report of the Startup Nuclear Test Results for Donald C. Cook Unit 1, Cycle 1", WCAP-8688, December 1975.

REFERENCES, SECTION 3.3.3

1) Adler, M. R., "AMSAC Generic Design Package," WCAP-10858, June 1985.

TABLE 3.3.1-1

NUCLEAR DESIGN DATA*

	STRUCTURAL CHARACTERISTICS	
1.	Fuel Weight (UO_2), lbs.	216,600
2.	Zircaloy Weight, lbs.	44,547
3.	Core Diameter, inches	132.7
4.	Core Height, inches	144
	Reflector Thickness and Composition	
5.	Top - Water Plus Steel	10 in.
6.	Bottom - Water Plus Steel	10 in.
7.	Side - Water Plus Steel	15 in.
8.	H_2O/U, (cold) Core	4.09
9.	Number of Fuel Assemblies	193
10.	UO_2 Rods per Assembly	204
	PERFORMANCE CHARACTERISTICS	
11.	Heat Output, MWt (initial rating)	3,250
12.	Heat Output, MWt (maximum calculated heat removal rating)	3,391
13.	Fuel Burnup, MWD/MTU First Cycle	14,040
	First Cycle Enrichments, weight %	
14.	Region 1	2.25
15.	Region 2	2.80
16.	Region 3	3.30
17.	Equilibrium Enrichment	2.90
18.	Nuclear Heat Flux Hot Channel Factor, F_Q^N	2.71**
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$	1.58***

*These data are design values for Cycle 1. Updated design values starting with Cycle 8 operation are listed in Section 3.5.

**The current technical specification limit is 2.15.

***The current technical specification limit is 1.49. These values do not include a 4% uncertainty since the ITDP was used for Cycle 8 (see Section 3.5.3).

TABLE 3.3.1-1 (cont'd.)

CONTROL CHARACTERISTICS

	Effective Multiplication (Beginning of Life) With Burnable Poison Rods in; No Boron	
20.	Cold, No Power, Clean	1.183
21.	Hot, No Power, Clean	1.154
22.	Hot, Full Power, Clean	1.132
23.	Hot, Full Power, Xe and Sm Equilibrium	1.092
24.	Absorber Material	5% Cd; 15% In; 80% Ag
25.	Full Length	53
26.	Part Length	0
27.	Number of Absorber Rods per RCC Assembly	20
28.	Total Rod Worth, BOL, %	(See Table 3.3.1-3)
	Boron Concentration for First Core Cycle Loading With Burnable Poison Rods	
29.	Fuel Loading Shutdown; Rods in (k = .87)	2000 ppm
	Rods in (k = .90)	1714 ppm
30.	Shutdown (k = .99) with Rods Inserted Clean, Cold	945 ppm
31.	Shutdown (k = .99) with Rods Inserted, Clean, Hot	602 ppm
32.	Shutdown (k = .99) with No Rods Inserted, Clean, Cold	1414 ppm
33.	Shutdown (k = .99) with No Rods Inserted, Clean, Hot	1385 ppm
	To Maintain k = 1 at Hot Full Power, No Rods Inserted:	
34.	Clean	1152
35.	Xenon	868 ppm
36.	Xenon and Samarium	838 ppm
37.	Shutdown, All But One Rod Inserted, Clean Cold (k = .99)	1031 ppm
38.	Shutdown, All But One Rod Inserted, Clean Hot (k = .99)	734 ppm

Doppler Contributions to the Power Coefficient vs. Power Level

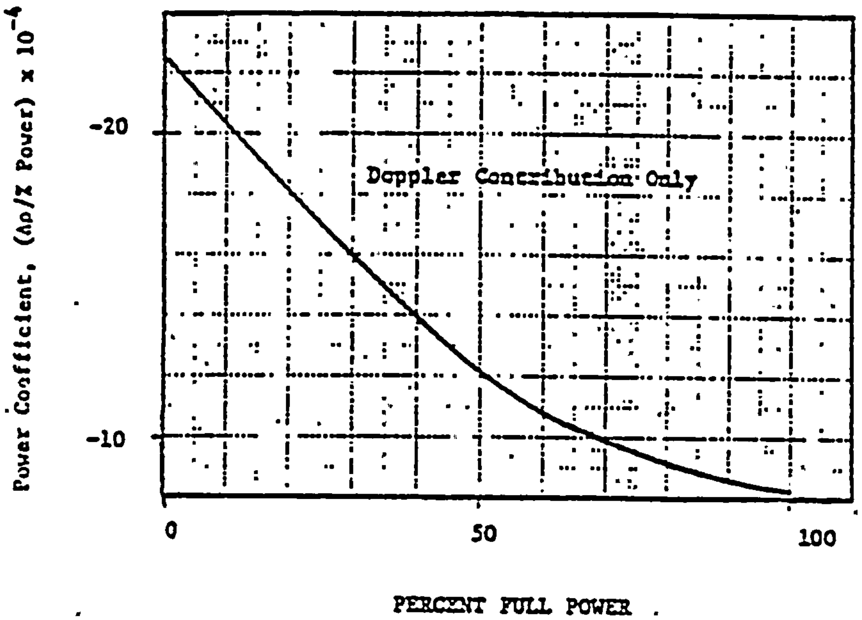


Figure 3.3.1- 17

3.5 Current Westinghouse OFA Reload Fuel

Starting with Cycle 8 operation (startup November 1983), the Cook Nuclear Plant Unit 1 has been 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. For certain criteria, OFA compatibility with the previous Exxon Nuclear Company (ENC) fuel assemblies is justified.

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 Nuclear Plant 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) Prior to the insertion of 15x15 OFAs in Cook Nuclear Plant Unit 1, operating experience had been obtained in other plants for six demonstration 17x17 OFAs which contain Zircaloy intermediate grids.(2) Two 17x17 OFAs had satisfactorily completed three cycles of irradiation to about 28,000 MWD/MTU burnup, two had completed two cycles to about 19,400 MWD/MTU, and two had completed one cycle in excess of 9,000 MWD/MTU. The demonstration OFAs had been examined and provided reason to expect good performance from the 15x15 OFA design. Thirty-four assemblies of the W OFA 15x15 design have now completed three cycles of irradiation in Cook Nuclear Plant Unit 1 and 124 other 15x15 OFAs have completed two cycles in Cook Nuclear Plant Unit 1, all with satisfactory performance.

Although Cook Nuclear Plant 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 11 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 W 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 Cook Nuclear Plant Unit 2 operation. Another feature 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 3.5.1.4.

3.5.1 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 3.5.1-1 shows an OFA fuel length schematic view, and Table 3.5.1-1 shows OFA design values. Figure 3.5.1-1 shows the difference for certain dimensions between the OFA and the previous ENC fuel.

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.

3.5.1.1 Mechanical Compatibility of Fuel Assemblies

Design Basis

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

Evaluation

Figure 3.5.1-1 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 3.5.1-1, 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 3.5.1-1, 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 W LOPAR assembly design which was the initial Cook Nuclear Plant Unit 1 fuel. W 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 3.5.1.4, to be acceptable.

A core coolable geometry must be maintained during a seismic event. 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 3.5.1-2. These reactor core reload patterns are consistent with typical reload configurations.

The maximum grid impact forces for both LOCA and seismic accidents occur at the peripheral fuel assembly locations adjacent to the baffle wall. For the four reload pattern cases in Figure 3.5.1-2 considered with the LOCA event, the results of the maximum OFA grid forces given below show that grid integrity is maintained.

	Case 1	Case 2	Case 3	Case 4
Grid Maximum Impact Force (% of Allowable Limit)	58	59	56	59

b. OFA Grid Analysis During Seismic Event

A seismic analysis of the reactor internals was performed using a synthesized time history wave which produced a response spectra that enveloped the Cook Nuclear Plant Unit 1 plant design requirement. The time history results obtained from that analysis were used as input to the model[8] to obtain the reactor core seismic response. Since the reactor core responses obtained from the LOCA analysis were essentially the same, only three of the four reload patterns were analyzed for the seismic accident.

For the seismic analysis of the three reload reference patterns considered, results of the maximum grid forces given below show that grid integrity is maintained.

	Case 1	Case 3	Case 4
Grid Maximum Impact Force (% of Allowable Limit)	80	75	54

c. OFA Component Stresses for Combined Seismic/LOCA

The stresses induced in the various fuel assembly components were assessed based on the most limiting seismic and LOCA accident conditions. The fuel assembly axial forces resulting from the LOCA accident were the primary source of the stresses in the thimble guide tube and fuel assembly nozzles. The fuel assembly component stresses which resulted from the vertical effects of the LOCA accident were directly combined with the seismic induced stresses. A summary of the combined stresses, which shows that component integrity is maintained, is given below:

	(% of Allowable Component Stress)	
Component	Uniform Stresses (Membrane/Direct)	Combined Stresses (Membrane + Bending)
Guide Thimble Tube	46.3	57.9
Fuel Rod*	13.9	14.6
Top Nozzle Plate	1.0	8.3
Bottom Nozzle Plate	1.0	31.6

*Includes primary operating stress

3.5.1.2 Fuel Assembly Structure

The fuel assembly structure consists of a bottom nozzle, top nozzle with holddown springs, twenty (20) guide thimble tubes, center instrumentation thimble tube, and seven grids (five inner Zircaloy, two end Inconel) as shown in Figure 3.5.1-1.

The design bases of the W assembly structure are the same as those given in Sections 4.2.1.4 and 4.2.1.5 of WCAP-9500-A.[6] For the Inconel and Zircaloy grids, lateral loads resulting from a seismic or LOCA event will not cause an unacceptably high plastic deformation. Each fuel assembly's geometry will be maintained such that the fuel remains in an array amenable to cooling.

3.5.1.2.1 Guide and Instrumentation Thimbles

Description

The OFA guide thimbles are structural members which also provide channels for the neutron absorber rods, burnable poison rods, neutron source, or thimble plug assemblies. Each thimble is fabricated from Zircaloy-4 tubing having two different diameters. The tube diameter at the top section provides the annular area necessary to permit rapid control rod insertion during a reactor trip. The lower portion of the guide thimble is swaged to a smaller diameter to reduce diametral clearances and produce a dashpot action near the end of the control rod travel during normal trip operation. Holes are provided on the thimble tube above the dashpot to reduce the rod drop time. The dashpot is closed at the bottom by means of an end plug which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation. The top end of the guide thimble is fastened to a tubular sleeve by three expansion swages. The sleeve fits into and is welded to the top nozzle adapter plate. The lower end of the guide thimble is fitted with an end plug which is then fastened into the bottom nozzle by a locking cup thimble screw (See Section 3.5.1.2.3).

Each grid is fastened to the guide thimble assemblies to create an integrated structure. The fastening technique depicted in Figures 3.5.1-3 and 3.5.1-4 is used for all but the top and bottom grids in a fuel assembly.

An expanding tool is inserted into the inner diameter of the Zircaloy thimble tube at the elevation of Zircaloy sleeves that have been welded into the inner five Zircaloy grid assemblies. The four lobed tool forces the thimble and sleeve outward to a predetermined diameter, thus joining the two components.

The top grid to thimble attachment is shown in Figure 3.5.1-5. The stainless steel sleeves are brazed into the Inconel grid assembly. The Zircaloy guide thimbles are fastened to the long sleeves by expanding the two members, as shown by Figure 3.5.1-5. Finally, top ends of the sleeves are welded to the top nozzle adapter plate, as shown in Figure 3.5.1-5.

The bottom grid assembly is joined to the assembly, as shown in Figure 3.5.1-6. The stainless steel insert is spot welded to the bottom grid and later captured between the guide thimble end plug and the bottom nozzle by means of a stainless steel thimble screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of Zircaloy guide thimbles in 1969.

The central instrumentation thimble of each OFA is constrained by seating in counterbores in each nozzle. This tube is a constant diameter and guides the incore neutron detectors. This thimble is expanded at the top and mid-grids in the same manner as the previously discussed expansion of the guide thimbles to the grids.

Evaluation

Due to thicker Zircaloy grid straps and a resulting reduced cell size, the OFA guide thimble tube ID (above dashpot) has a 12 mil reduction compared to the previous ENC thimble tube ID of 0.511 inches. The OFA guide tube thimble ID provides an adequate nominal diametral clearance for the control rods as well as other components. Due to the reduced OFA diametral clearance, the control rod scram time to the dashpot is increased from the current 1.8 seconds to 2.4 seconds. This increase in rod drop time was determined from conservative analytical calculations. The 2.4 second scram time is used in all the accident reanalyses. Section 3.5.1.1 shows that guide thimble mechanical integrity is maintained during a seismic and/or LOCA event(s).

The OFA instrumentation tube has a 12 mil diametral decrease compared to the ENC assembly instrumentation tube. There is sufficient diametral clearance for the instrumentation thimble to traverse the OFA instrumentation tube.

3.5.1.2.2 Top Nozzle and Holddown Springs

Description

The OFA top nozzle is a machined and welded structure approximately 8.4 inches square by 3.5 inches high. The top nozzle assembly is the uppermost structural member of the fuel assembly. The top nozzle forms a plenum, where coolant received from the fuel assembly is mixed and directed to flow holes in the upper coreplate. Four fuel assembly holddown springs are mounted to the top of the nozzle and fastened in place by bolts and clamps located at two diagonally opposite corners. Except for the screws and springs, which are Inconel, the top nozzle assembly is made from 304 stainless steel.

Evaluation

The OFA design has minor differences in the overall height of the top nozzle (0.07 inches greater), the adapter plate flow-slot configuration, and holddown leaf springs as compared to the ENC fuel assembly design. These minor differences have no adverse impact on the interaction of W 15x15 OFA and ENC assemblies during fuel handling operations or reactor operations. The W 15x15 OFA design uses a 3-leaf holddown spring design compared to the 2-leaf springs in the ENC assembly. The W OFA 3-leaf spring has been successfully used in 15x15 LOPAR assemblies, as well as on the 17x17 OFA demonstration assemblies. The 3-leaf spring provides additional holddown force margin compared to a 2-leaf spring. The ability to withstand seismic and LOCA impact loads is shown in Section 3.5.1.1.

3.5.1.2.3 Bottom Nozzle

Description

The OFA bottom nozzle is a machined and welded structure approximately 8.4 inches square by 2.7 inches in height. The bottom nozzle is made of 304 stainless steel, consisting of a top plate, containing flow holes, to which four (4) "legs" are welded, one at each corner. The bottom nozzle is the bottom structural member of the fuel assembly. The top plate portion of the bottom nozzle is designed to prevent the fuel rods from passing through, as well as to provide for coolant flow to be distributed toward the fuel assembly.

As part of its structural function, the guide thimble assemblies are attached to the bottom nozzle top plate, while the four corner legs rest on the lower coreplate and support the entire fuel assembly. Two (2) of the bottom nozzle legs, located diagonally opposite, contain holes which receive the fuel alignment pins which are mounted on the coreplate.

The OFA bottom nozzle design has a reconstitutable feature, as shown in Figure 3.5.1-7, which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard W LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

Evaluation

The OFA bottom nozzle has similar design features and dimensions compared to the ENC nozzle, thus assuring mechanical compatibility. Their cross-sectional areas are identical, and the OFA nozzle is only 0.018 inches higher than the ENC nozzle.

3.5.1.2.4 Grids

Description

Two types of grid assemblies are used in each fuel assembly. Both types consist of individual slotted straps interlocked in an "egg-crate" arrangement. The straps contain spring fingers, support dimples, and mixing vanes. One type, consisting of five (5) inner-grids per assembly, consists of Zircaloy straps arranged as described above and permanently joined by welding at their points of intersection. Their internal straps include mixing vanes which project into the coolant stream and promote mixing of the coolant. The other grid type, two (2) located at each end of a fuel assembly, does not include mixing vanes on the internal straps. The material of these grid assemblies is Inconel-718, chosen because of its corrosion resistance and high strength. Joining of the individual straps is achieved by brazing at the points of intersection. The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core. The individual grid cells at each fuel rod location provide six-point contact with the rod; four dimples and two springs.

The attachment of the five Zircaloy inner-grid and two Inconel end-grid assemblies to the guide thimble tubes is described in Section 3.5.1.2.1.

Evaluation

The fuel rods, as shown in Figure 3.5.1-1, are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods. Each fuel rod is supported within each grid by the combination of support dimples and springs. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The top and bottom Inconel grids of the OFA are the same as the Inconel grids of a W LOPAR fuel assembly. The five intermediate OFA Zircaloy-4 grids have thicker and wider straps than the OFA Inconel grids (See Table 3.5.1-1) in order to closely duplicate the Inconel grid strength. The ENC assembly grids are bimetallic, consisting of Zircaloy-4 straps with Inconel grid springs. Both the OFA Zircaloy and ENC bimetallic grids have grid heights of 2.25 inches. The OFA Inconel grid height is 1.5 inches. Elevation of the grids was established to ensure axial match-up during operation.

Impact tests that have been performed at 600°F to obtain the dynamic strength data verify that the Zircaloy grid strength at reactor operating conditions is acceptable. The 15x15 Zircaloy grids have approximately 7% less crush strength than the 15x15 Inconel grids at reactor operating temperatures.

The ability of the grids to withstand seismic and LOCA impact loads is shown in Section 3.5.1.1.

3.5.1.3 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow greater axial expansion at the center of the pellets.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel density changes during irradiation, thus avoiding overstressing of the cladding or seal welds. Shifting of the fuel within the cladding during handling or shipping prior to core loading is prevented by a stainless steel helical spring which bears on top of the fuel. At assembly, the pellets are stacked in the cladding to the required fuel height, the spring is then inserted into the top end of the fuel tube, and the end plugs are pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process in order to minimize compressive cladding stresses and prevent cladding flattening due to coolant operating pressures. Nominal fuel rod parameters are given in Table 3.5.1-1.

Design Bases

The fuel design bases and criteria for W 15x15 OFA fuel are the same as those discussed in Sections 4.2 and 4.4.1.2 of WCAP-9500[6] for the W 17x17 OFA design. The bases and criteria given in Section 3.2.1.1.1 of the UFSAR for Cook Nuclear Plant Unit 2 are also applicable, but it should be noted that the region average discharge burnups considered in the Cook Nuclear Plant Unit 1 OFA fuel design are typically in the range of 38,000 MWD/MTU. These design bases and criteria are summarized below:

a. The cladding stresses under Condition I and II events are less than the Zircaloy 0.2% offset yield stress, with due consideration of temperature and irradiation effects. While the cladding has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design basis.

b. Cladding Tensile Strain - The total tensile creep strain is less than 1% from the unirradiated condition. The elastic tensile strain during a transient is less than 1% from the pre-transient value. This limit is consistent with proven practice.

c. Strain Fatigue - The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice.

d. Wear - Potential for fretting wear of the clad surface exists due to flow induced vibrations. This condition is taken into account in the design of the fuel rod support system. The clad wear depth is limited to acceptable values by the grid support dimple and spring design.

e. 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.[10]

   Rod pressure is also limited such that extensive DNB propagation shall not occur during normal operation and accident events.[10]

f. Cladding collapse shall be precluded during the fuel rod design lifetime. The models described in Reference 11 are used for this evaluation.

g. During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature. The melting temperature of UO_2 is taken at $5080^{o}F$[7], unirradiated and decreasing $58^{o}F$ per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting, and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of $4700^{o}F$ has been selected as the overpower limit.

h. Design values for the properties of materials used for the fuel rod design and performance are given in Reference 7.

Evaluation

The detailed OFA fuel rod design establishes such parameters as pellet size and density, cladding-pellet diametral gap, gas plenum size, and helium pre-pressurization level. The design also considers effects such as fuel density changes, fission gas release, cladding creep, and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. This is achieved by designing the fuel rods to satisfy the conservative design bases in the following subsections during Condition I and Condition II events over the fuel lifetime. For each design basis, the performance of the limiting fuel rod must not exceed the limits specified. The NRC approved fuel rod design model[12,13] is used to assure that design bases are satisfied and to predict fuel operating characteristics. Additional details in the evaluation of the OFA fuel rods, which show that the design bases are satisfied, are given in Sections 4.2.3.1, 4.2.3.2 and 4.2.3.3 of WCAP-9500.[6] Also applicable are the fuel rod evaluations given in Section 3.2.1.3.1 of the Cook Nuclear Plant Unit 2 UFSAR.

The W 15x15 OFA fuel rod design is essentially the same as the LOPAR W 15x15 fuel rod design which has exhibited good in-core fuel performance.[2] The W OFA and ENC fuel rods have similar length and clad OD dimensions. Table 3.5.1-1 presents a comparison of the W OFA and ENC fuel rod designs.

As stated in the 17x17 OFA Reference Core Report,[6] for a given burnup, the magnitude of rod bow for the W OFA is conservatively assumed to be the same as that of a W LOPAR fuel assembly. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation. Since the OFA fuel rods are the same as the W LOPAR fuel rods, there will be no difference in predicted bow due to rod considerations. The OFA design will have reduced grid forces due to the Zircaloy grid springs. Therefore, this component is predicted to decrease OFA rod bow compared to LOPAR fuel. The impact of rod bow on DNBR penalties is discussed in Section 3.5.3.

The wear of fuel rod cladding is dependent on both the support provided by the grids and the flow environment to which it is subjected. Extrapolation of the results from flow tests involving OFA assemblies shows that fuel rod wear would be less than ten (10) percent of the cladding thickness for at least 48 months of reactor operation. This assures that clad wear will not impair fuel rod integrity.

The above conclusions on OFA rod wear and integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and crossflows, and for the effect of potential fuel rod to grid gaps.

3.5.1.4 Core Components

The core components consist of the rod cluster control assemblies (RCCAs), the primary and secondary source assemblies, the thimble plug assemblies, and the burnable absorber assemblies. The control rod assemblies in the Cook Nuclear Plant Unit 1 core are unchanged from previous cycles and are compatible with the OFA guide thimbles. New secondary source assemblies and OFA compatible plugging devices were supplied in Cycle 8. As discussed in Section 3.5.1.2.1, the reduced diametral clearance compared to ENC guide thimble results in an increased RCCA scram time from 1.8 to 2.4 seconds which is used in all accident reanalyses.

The guide thimble plug used with the OFA has a smaller diameter (0.485") than the current thimble plug diameter (0.498"), in order to maintain the same thimble plug to thimble tube diametral clearance. The thimble plug assembly presently used in ENC fuel cannot be used in OFAs due to insufficient diametral clearance between the current thimble plug and OFA guide thimble tube.

The optimized assemblies, their thimble plugging devices, and source assemblies are compatible with existing handling tools. A new tool is provided for handling the new Wet Annular Burnable Absorber (WABA) rods.

<u>Wet Annular Burnable Absorber (WABA)</u>

The Wet Annular Burnable Absorber (WABA) rod design is used in the Cook Nuclear Plant Unit 1 reload cores with 15x15 <u>W</u> OFA fuel. The materials, mechanical, thermal hydraulic, and nuclear design evaluations of the WABA rods are presented in a topical report,[5] which has received NRC generic approval,[5] and approval for Cook Nuclear Plant Unit 1 application[1] of WABAs.

The WABA design has annular aluminum oxide - boron carbide (Al_2O_3 - B_4C) absorber pellets contained within two concentric Zircaloy tubes with water flowing through the center tube as well as around the outer tube. The WABA design provides significantly enhanced nuclear characteristics, when compared with the <u>W</u> borosilicate absorber rod design. Fuel cycle benefits result from the reduced parasitic neutron absorption of Zircaloy compared to stainless steel tubes, increased water fraction in the burnable absorber cell, and a reduced boron penalty at the end of each cycle.

Figures 3.5.1-8 and 3.5.1-9 show the design of a WABA rod, and Table 3.5.1-2 and Figure 3.5.1-9 present a comparison between the WABA rod and a <u>W</u> borosilicate glass absorber rod.

The WABA rods inserted into each OFA are attached at their top ends to a holddown assembly and retaining plate in the same manner as burnable absorber rods previously used in Cook Nuclear Plant Unit 1 reload cores.

Based on the materials and design evaluations in Reference 5, it is concluded that the wet annular burnable absorber rod satisfies all performance and design requirements for 18,000 effective-full-power-hours irradiated life.

REFERENCES - SECTION 3.5 and 3.5.1

1. Letter from D. L. Wigginton (NRC) to John Dolan (Indiana and Michigan Electric Co.) dated September 20, 1983; Subject: Amendment 74 to Facility Operating License No. DPR-53 for Donald C. Cook Unit 1 and enclosed NRC SER, Docket No. 50-315.

2. Skaritka, J., Iorii, J. A., "Operational Experience with Westinghouse Cores (up to December 31, 1982)," WCAP-8183, Rev. 12, August 1983.

3. Letter from R. L. Tedesco (NRC) to T. M. Anderson (Westinghouse), Safety Evaluation of WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly," NRC SER letter dated May 22, 1981.

4. Bordelon, F. M., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273 (Non-Prop.), March 1978.

5. Skaritka, J., et al., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A, Revision 1 (Proprietary), October 1983. Contains NRC SER dated August 9, 1983 and W responses to NRC questions.

6. Davidson, S. L., Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," Westinghouse Report WCAP-9500-A, dated May 1982.

7. Beaumont, M. D., Iorii, J. A., (Ed.), Properties of Fuel and Core Component Materials," WCAP-9719, Revision 1 (Proprietary) and WCAP-9224, July, 1978; Appendix B, AL_2O_3-B_4C Pellets, issued October 1980; Revised pages issued September 1982.

8. R. F. Hering, (AEP), Letter to H. R. Denton (NRC), August 31, 1983.

9. Davidson, S. L., Iorii, J. A. (Ed.), "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A (Proprietary) and WCAP-9402-A, August 1981.

10. D. H. Risher, Ed., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963 (Proprietary), November 1976.

11. George, R. A., et al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

12. J. V. Miller, Ed., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary) dated October 1976.

13. W. J. Leech, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations - Application for Transient Analyses," WCAP-8720, Addendum 1 (Proprietary) dated September 1979.

TABLE 3.5.1-1

Westinghouse 15x15 OFA Design Parameters

Parameter	15x15 W Optimized Fuel Assembly Design
Fuel Assembly Length, in.	159.765
Fuel Rod Length, in.	151.85
Assembly Envelope, in.	8.426
Compatible with Core Internals	Yes
Fuel Rod Pitch, in.	0.563
Number of Fuel Rods/Ass'y	204
Number of Guide Thimbles/Ass'y	20
Number of Instrumentation Tube Ass'y	1
Compatible with Moveable In-Core Detector System	Yes
Fuel Tube Material	Zircaloy-4
Fuel Rod Clad OD, in.	0.422
Fuel Rod Clad Thickness, in.	0.0243
Fuel/Clad Gap, mil	7.5
Fuel Pellet Diameter, in.	0.3659
Guide Thimble Material	Zircaloy-4
Guide Thimble ID, in.*	0.499
Structural Material - Five Inner Grids	Zircaloy-4
Structural Material - Two End Grids	Inconel
Grid height, in., Outer Straps, Valley-to-Valley	2.25 (Inner Grids) 1.50 (End Grids)
Bottom Nozzle	Reconstitutable
Top Nozzle Holddown Springs	3-leaf

*Above dashpot

TABLE 3.5.1-2

Comparison of Burnable Absorber Rods Design Parameters

Parameter	W Wet Annular BA	W Borosilicate Glass BA 15x15 FA
Overall Length, in	150.00**	152
Absorber Length, in	134.00**	142.7
Absorber Material	Al_2O_3-B_4C	B_2O_3
Absorber Form	Annular Pellet	Glass Tube
Outer Clad O.D., in	.381	.439
Absorber Clad Material	Zircaloy	Stainless
Absorber Thickness, in	.020	.077
OFA Guide Thimble I.D., in	.499	.499

Definitions:

BA - Burnable Absorber

FA - Fuel Assembly

**Typical length which can be changed to accommodate specific plant fuel cycle application.

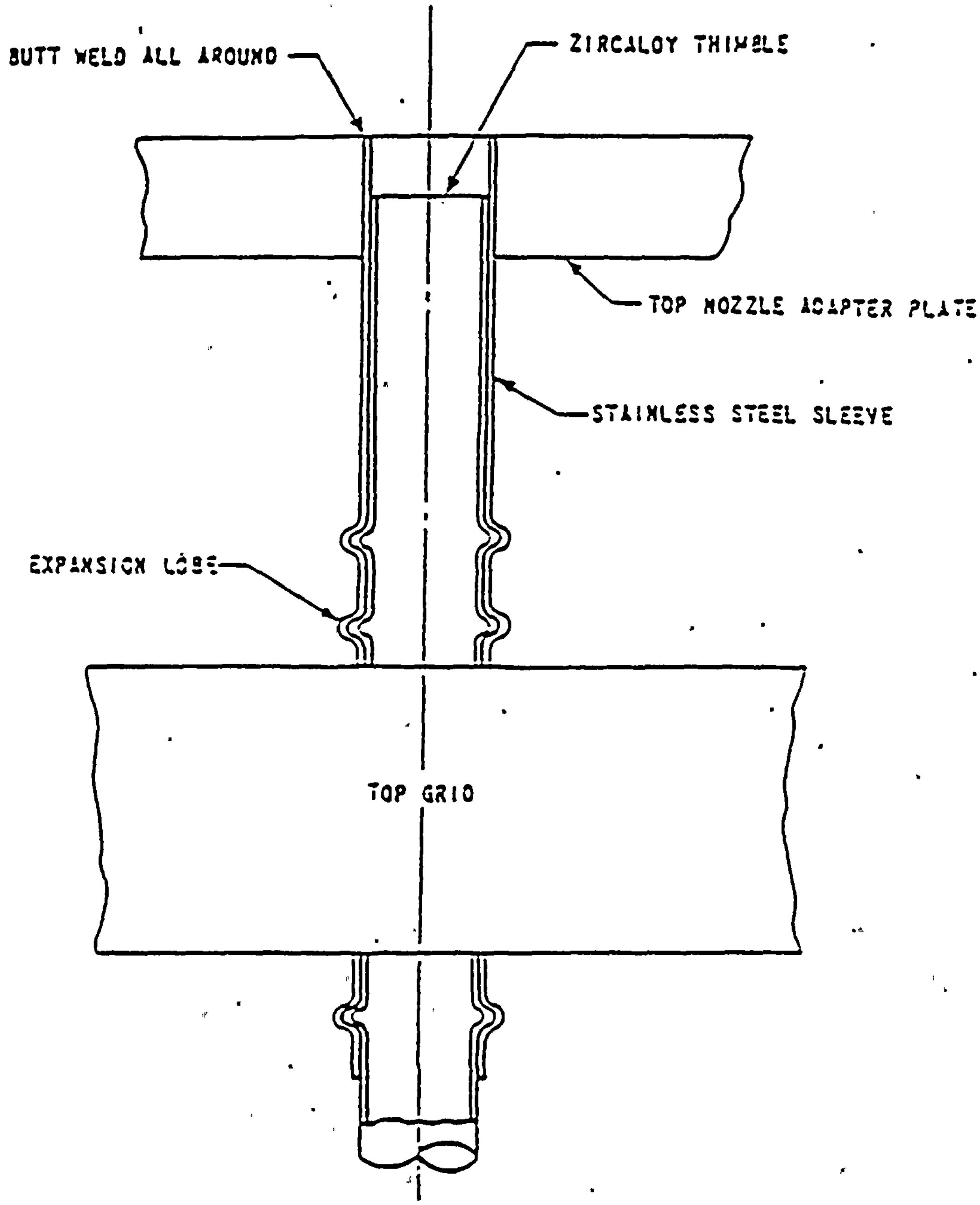


FIGURE 3.5.1-5

Top Grid to Guide Thimble and Top Nozzle Attachment

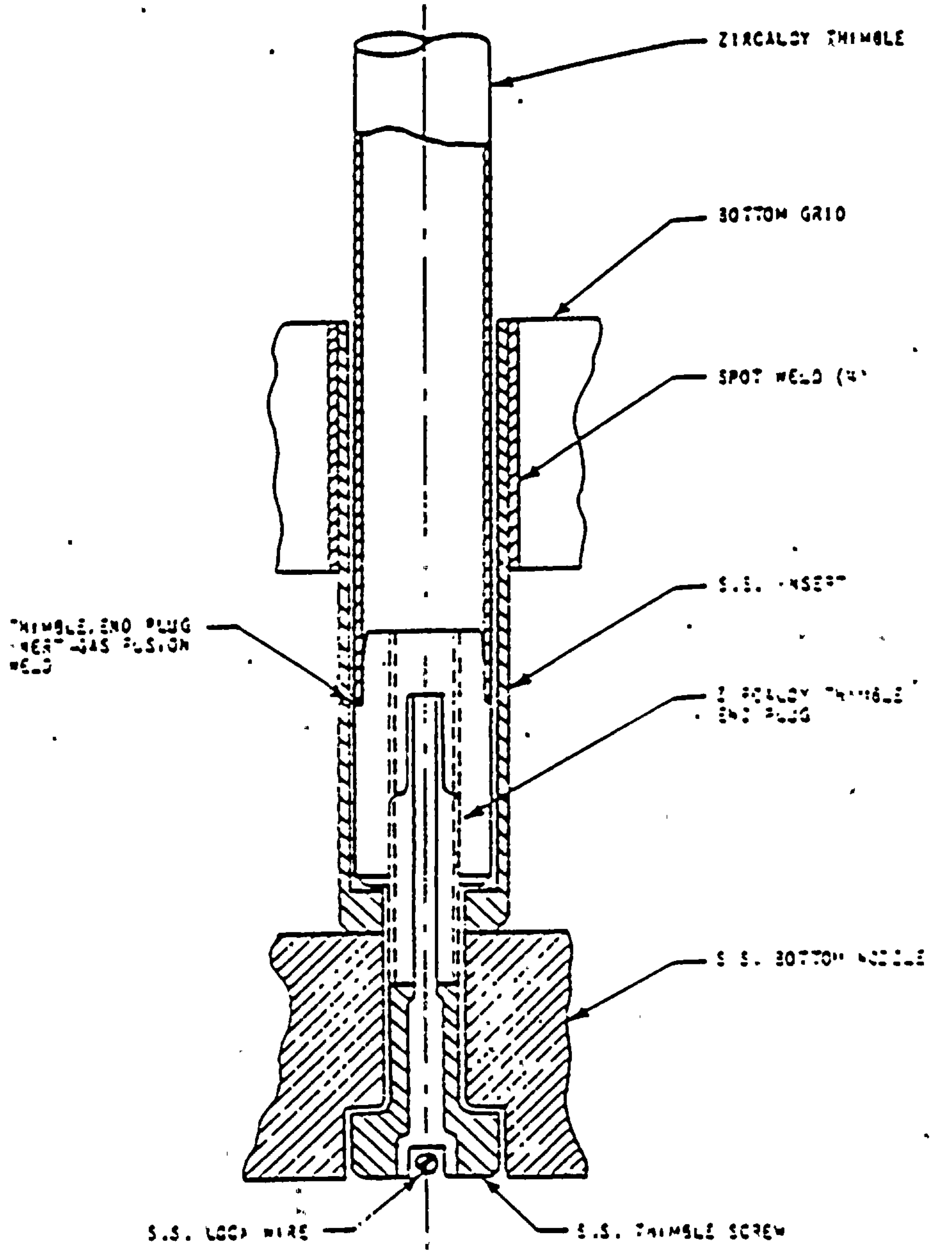
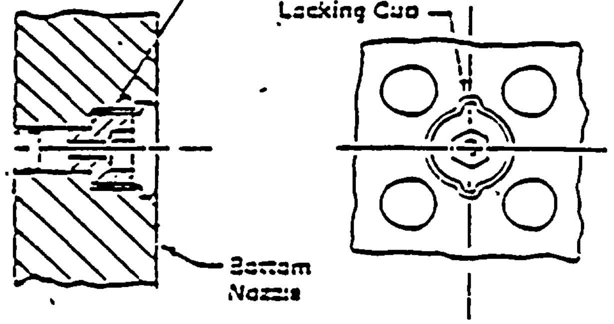
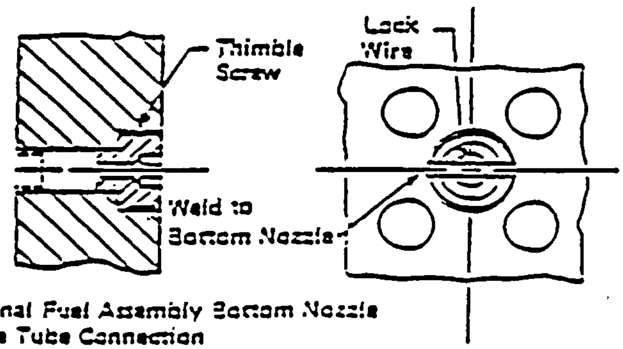


FIGURE 3.5.1-6

Guide Thimble to Bottom Grid and Nozzle Joint



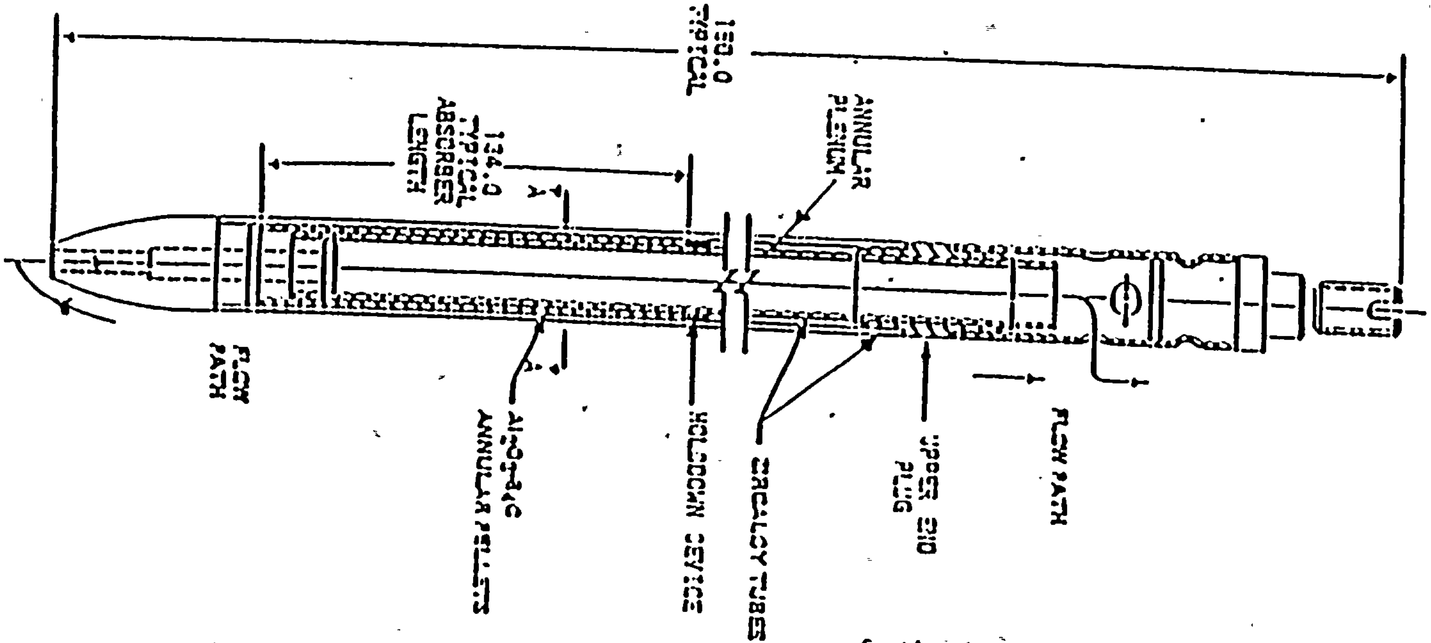
a) Reconstitutable Bottom Nozzle Design



b) Conventional Fuel Assembly Bottom Nozzle
To Thimble Tube Connection

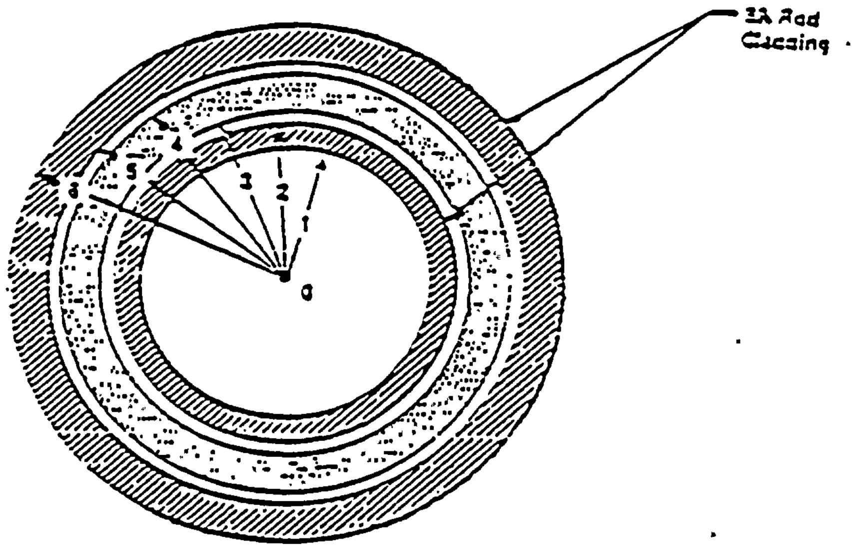
BOTTOM NOZZLE TO THIMBLE TUBE CONNECTION

FIGURE 3.5.1-7



Section A-A is shown in Figure 3.5.1-9

FIGURE 3.5.1-8 WET ANNULAR BURNABLE ABSORBER ROD



	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 3.5.1-9

COMPARISON OF BOROSILICATE GLASS

ABSORBER ROD WITH WABA ROD

3.5.2 NUCLEAR DESIGN

The nuclear design of cores with W OFA is accomplished by using the standard calculational methods as described in the W Reload Safety Evaluation Methodology.(1) In addition to Westinghouse's standard methods, starting with Cycle 10, the Westinghouse Advanced Nodal Code(9) was introduced to perform core neutronics analyses.

Each reload core design is evaluated to assure that design and safety limits for the 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 in W 15x15 OFA fuel up to and including 4.55 wt.% U-235 for the new fuel storage vault and 4.95 wt.% U-235 for the spent fuel pool. These analyses confirm that all current safety criteria applicable to fuel storage are satisfied.(2)

3.5.2.1 Computerized Methods, Codes and Cross Section Data

Three principal computer codes have been used in the nuclear design of reactor cores with W OFA; these are ARK (zero-dimensional), APOLLO (one-dimensional), and ANC (two-dimensional and three-dimensional). Descriptions and uses for these codes follow.

ARK is a point model cell-homogenization, neutron spectrum, isotopic depletion program which has evolved from the codes LEOPARD(3) and CINDER(4). Microscopic cross section data is based on the ENDF/B library(5) with minor modifications. The fast and thermal spectrum calculations are performed by the methods of the MUFT(6) and SOFOCATE(7) codes. ARK is the basis for all reactivity calculations, depletion rates, and reactivity feedback models. In this respect, aspects of core design such as the calculation of power distributions use ARK for their reactivity features.

APOLLO, an advanced version of PANDA[8], is a two-group, one-dimensional diffusion-depletion code. APOLLO utilizes the burnup dependent macroscopic cross sections generated by ARK.. The APOLLO model is used as an axial model. APOLLO is utilized to determine axial power and burnup distributions, differential rod worths, and control rod operational limits (insertion limits, etc.).

TORTISE[9] is a two-dimensional, two-group diffusion-depletion code with features similar to APOLLO. The usage of TORTISE in this design is in the generation of homogenized, macroscopic cross sections for nodal calculations.

ANC[10] is an advanced nodal theory code that is used in two-dimensional and three-dimensional calculations. ANC calculations include X-Y power and burnup distributions, critical boron concentrations, reactivity coefficients, control rod worths, and various X-Y safety analysis calculations. In addition, 3D ANC is used to validate one- and two-dimensional results and to provide information about radial (X-Y) peaking factors as a function of axial position. ANC has the capability of calculating discrete pin powers from the nodal information as well.

Additional support codes are used for special calculations such as determining fuel temperatures and control rod cross sections.

3.5.2.2 Neutronic Design of Cook Nuclear Plant Unit 1 Reactor Core

3.5.2.2.1 Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described previously with primary reliance upon the 3D ANC code.

For each cycle, the burnup history of each of the fuel assemblies retained from previous cycles for further energy production is calculated by a three-dimensional model which is utilized to simulate operation of the core for previous cycles.

As an example, Cycle 11 core calculations used assembly exposures calculated from the Cycle 10 burnup of 15,951 MWD/MTU. Axial effects in the 2D models are accounted for through the buckling term B^2_z.

3.5.2.2.2 Design Bases

For each cycle, the nuclear design bases are very similar to those for the example Cycle 11 core as follows:

1. At core full power, 3250 MWt (not including pump heat), nuclear peaking factors of 2.15 and 1.55 for F_Q^T and $F_{\Delta H}^N$ respectively, will not be exceeded. In addition, at any relative power level P ($0.0 \leq P \leq 1.0$), F_Q^T and $F_{\Delta H}^N$ shall not exceed the bases of the plant control and protection system.

2. The moderator temperature coefficient at operating conditions greater than 70% power level is a ramp function limited to +5.0 pcm/^{o}F at 70% power and 0.0 pcm/^{o}F at 100% power. Below 70% power level, the moderator temperature coefficient shall be less than +5.0 pcm/^{o}F.

3. With the most reactive control rod stuck out of the core, the remaining control rods shall be able to shut the reactor down by a sufficient reactivity to reduce the consequences of any credible accident to acceptable levels.

4. The effects of all accident situations in Cycle 11 will be acceptable and compatible with the safety bases of the Final Safety Analysis Report (FSAR), as specified in Reference 1.

5. The fuel loading specified shall be capable of generating approximately 15500 MWD/MTU at normal full power operating conditions during Cycle 11.

3.5.2.2.3 Design Description and Results

Each cycle's reactor core consists of 193 W OFA assemblies, each having a 15x15 fuel rod array. A description of the W OFAs is given in Section 3.5.1.

As an example, the Cycle 11 loading pattern is given in Figure 3.5.2-1 which shows the region number, sources, and the burnable absorber configuration. The core consists of 48 fresh W OFAs with an average enrichment of 3.253 w/o U-235, 32 fresh OFAs with an average enrichment of 3.614 w/o, 79 once burnt OFA assemblies and 34 twice burnt OFA assemblies. A low leakage loading pattern was developed which results in the scatter-loading of the fresh OFAs throughout the interior of the core. 592 new WABA rods are inserted into a number of OFAs to control power peaking and MTC. The WABA rods contain 0.0153 gm/in of B-10. Pertinent fuel assembly parameters for the Cycle 11 fuel are given in Tables 3.5.1-1 and 3.5.2-1.

Physics Characteristics

The neutronics characteristics of a reactor core with W OFA fuel are presented in Table 3.5.2-2. These reactivity coefficients are bounded by the coefficients used in the safety analysis. For an example cycle length, Cycle 11 was projected to be 15,430 MWD/MTU at a core power of 3,250 MWt with 10 ppm soluble boron remaining.

Power Distribution Considerations

Figure 3.5.2-2 shows the K(Z) function (fuel height limit for normalized $F_Q(Z)$). Each cycle's core loading satisfies the envelope shown in Figure 3.5.2-2.

Control Rod Reactivity Requirements

The Cook Nuclear Plant Unit 1 Technical Specifications require a minimum shutdown margin of 1,600 pcm in operational Modes 1, 2, 3 and 4 and 1000 pcm in operational Mode 5 at BOC and EOC. As an example, detailed calculations of shutdown margins for Cycle 11 are presented in Table 3.5.2-3. The Cycle 11 analysis indicates excess shutdown margin of 1840 pcm at BOC and 1957 pcm at EOC.

Insertion limits are specified for the control rod groups and are given in the Core Operating Limits Report, as described in Technical Specification 6.9.1.11. The control rod shutdown requirements allow for a HFP D-Bank insertion equivalent to 500 pcm at both BOC and EOC. Table 3.5.2-3 gives the shutdown requirements for the example of Cycle 11.

Moderator Temperature Coefficient

Core loadings must satisfy the Technical Specifications requirements that the moderator temperature coefficient be less than or equal to +5 pcm/^{o}F below 70% of rated thermal power and less than or equal to a linear ramp between +5 pcm/^{o}F at 70% power and 0 pcm/^{o}F at 100% power.

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TABLE 3.5.2-1

FUEL ASSEMBLY DESIGN PARAMETERS
COOK NUCLEAR PLANT UNIT 1 - CYCLE 11

Region	11A	11B	12A	12B	13A	13B
Enrichment (w/o of U 235)*	3.404	3.600	3.298	3.600	3.253	3.614
Density (percent theoretical)*	94.995	95.042	95.286	94.911	95.154	95.284
Number of Assemblies	11	23	48	31	48	32
Burnup at Beginning of Cycle 11 (MWD/MTU)**	29568	31419	18911	17139	0	0
Burnup at End of Cycle 11 (MWD/MTU)**	42057	39658	35647	31088	18991	16093
Fuel Stack Height (inches, cold)	144	144	144	144	144	144

* All values are as-built.

** Assumes a Cycle 10 actual burnup of 15,951 MWD/MTU. The end of Cycle 11 burnup is assumed to be 15,500 MWD/MTU.

TABLE 3.5.2-2

KINETICS CHARACTERISTICS

COOK NUCLEAR PLANT UNIT 1 WITH W OFA FUEL

Most Positive Moderator Temperature Coefficient (pcm/^{o}F)**	+5.0 $\leq$ 70% RTP* linear ramp to 0.0 from 70 to 100% RTP
Doppler Temperature Coefficient (pcm/^{o}F)	-0.9 to -2.9
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)	-9.55 to -6.17
Most Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)	-19.4 to -12.90
Delayed Neutron Fraction, β_{eff} (%)	0.44 to 0.70
β_{eff} (%) minimum (BOL rod ejection only)	0.5
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/sec)**	75

*RTP = Rated Thermal Power

**1 pcm = 1.0 x 10^{-5} $\Delta\rho$

TABLE 3.5.2-3

SHUTDOWN REQUIREMENTS AND MARGINS
COOK NUCLEAR PLANT UNIT 1 - CYCLE 11

	BOC	EOC
Control Rod Worth (percent $\Delta\rho$)		
All rods inserted less worst stuck rod	6.370	7.127
(A) Less 10%	5.733	6.414
Control Rod Requirements (percent $\Delta\rho$)		
Reactivity defects (Doppler, T_{avg}, Void, Redistribution)	1.793	2.357
Rod insertion allowance	0.50	0.50
(B) Total requirements	2.293	2.857
Shutdown Margin [(A)-(B)] (percent $\Delta\rho$)	3.440	3.557
Required Shutdown Margin (percent $\Delta\rho$)	1.6	1.6

Figure 3.5.2-1 Example Core Loading Pattern: Cook Nuclear Plant Unit 1 Cycle 11

180°

90°

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

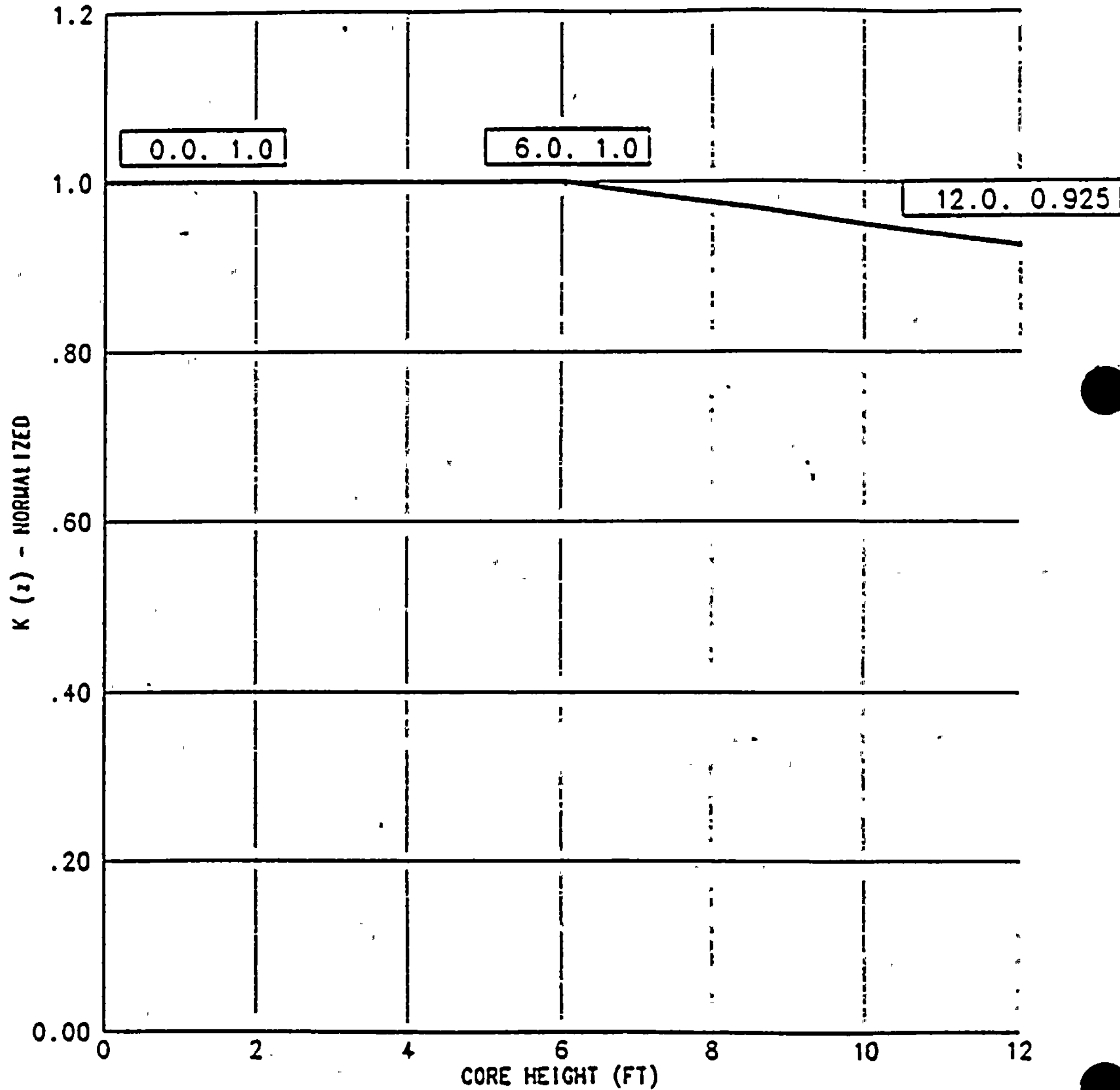
0°

X – Region Number
Y/SS – Number of Fresh Burnable Absorbers/ Location of Secondary Source Rods

FIGURE 3.5.2-2

Heat Flux Hot Channel Factor
Normalized Operating Envelope, FQ ECCS Limit = 2.15

Cook Nuclear Plant Unit 1 Cycle 11



3.5.3 Thermal and Hydraulic Design

Introduction

This section describes the thermal and hydraulic design of Cook Nuclear Plant Unit 1 core with Westinghouse Optimized Fuel Assemblies (OFA)

The thermal hydraulic design of the core is conservatively analyzed at 3411 MWt core power with a 577.1°F vessel average temperature, even though Cycle 11 will be limited to the 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.

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 table below shows the relationships which exist between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design, using the Westinghouse Improved Thermal Design Procedure (ITDP)[1].

	Typical	Thimble
Correlation Limit	1.17	1.17
Design Limit	1.33	1.32
Safety Analyses Limit	1.45	1.45

In order to bound the new Cycle 11 design conditions (reduced pressure compared to Cycle 10 conditions), some relaxation of the Cycle 10 core thermal limits were required. The Cycle 10 safety limit DNBRs of 1.69 were reduced to 1.45 for Cycle 11 to provide additional margin in the core thermal limits from those used in the Cycle 10 design. Sufficient margin is still maintained between the design/safety analysis limits to accommodate applicable rod bow penalties and provide margin for plant operation and design flexibility.

For events where conditions fall outside the range of applicability of the WRB-1 correlation, the W-3[5, 6] correlation is used.

The margin to the safety analysis DNBR limit is more than sufficient to cover the maximum rod bow penalty at full flow conditions[7].

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A maximum of 4.5 percent of this value is allotted as bypass flow. This includes RCC guide thimble cooling flow, head cooling flow, cavity flow, baffle leakage, and leakage to the vessel outlet nozzle.

The minimum measured flow used in the ITDP design of Cook Nuclear Plant Unit 1 is 366,400 gpm. Subsequent evaluations have supported a lower minimum measured flow value of 361,600 gpm[17]. These evaluations determined that sufficient margin was available in the calculation of DNBR, and allocation from that margin has been made to accommodate this lower minimum measured flow value.

Hydrodynamic Stability Design Bases

Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

Other Considerations

The above design bases together with the fuel clad and fuel assembly design bases given in Section 3.5.1 are sufficiently comprehensive so additional limits are not required.

Fuel rod diametral gap characteristics, moderator flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above mentioned design criteria are met. For instance, the fuel and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution and moderator void distribution are included in the core thermal (THINC) evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in Section 3.5.1.3, the fuel rod overly conditions change with time. A single clad temperature limit for Condition I or Condition II events is not appropriate since of necessity it would be conservative. An appropriate clad temperature limit is applied in each case to the loss of coolant accident (Section 14.3.1), control rod ejection accident, and locked rotor accident.

3.5.3.2 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in the previous section, the following discussion pertains mainly to fuel pellet temperature evaluation. A discussion of fuel clad integrity is presented in Section 3.5.1.3.

The thermal-hydraulic design assures that the maximum fuel temperature is below the melting point of UO_2. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluation. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad, gap and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, etc., have been combined into a semiempirical thermal model.[8] Conservative options of this thermal model are employed when generating fuel temperatures for use in safety analyses.[8,9]

As described in Section 3.5.1.3, fuel rod thermal evaluations (fuel centerline, average, and surface temperatures) are determined throughout the fuel rod lifetime with consideration of time dependent densification. To determine the maximum fuel temperatures, various burnup rods, including the highest burnup rod, are analyzed over the rod linear power range of interest.

The applicable range of variables is:

Pressure	:	$1440 \leq P \leq 2490$ psia
Local Mass Velocity	:	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-hr
Local Quality	:	$-0.2 \leq x_{loc} \leq 0.3$
Heated Length, Inlet to CHF Location	:	$L_h \leq 14$ feet
Grid Spacing	:	$13 \leq gsp \leq 32$ inches
Equivalent Hydraulic Diameter	:	$0.37 \leq de \leq 0.60$ inches
Equivalent Heated Hydraulic Diameter	:	$0.46 \leq dh \leq 0.58$ inches

Figure 3.5.3-1 shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

Definition of Departure from Nuclear Boiling

The DNBR as applied to this design for both typical and thimble cold wall cells is:

$$DNBR = \frac{q''_{DNB,\ N}}{q''_{loc}}$$

Where:

$$q''_{DNB,\ N} = \frac{q''_{DNB,\ EU}}{F}$$

and q''_{DNB}, EU is the equivalent uniform critical heat flux as predicted by the WRB-1 Correlation(4); q''_{loc} is the actual local heat flux.

F is the flux shape factor to account for nonuniform axial heat flux distributions.(12)

Mixing Technology

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels, the local fluid density and flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient, TDC, which is defined as:

$$TDC = \frac{w'}{\rho V a}$$

Where:

- w' = flow exchange rate per unit length, lb/ft-sec.
- ρ = fluid density, lb/ft^3.
- V = fluid velocity, ft/sec.
- a = lateral flow area between channels per unit length, ft^2/ft.

The application of the TDC in the THINC analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 11.

TDC is determined by comparing the THINC Code predictions with the measured subchannel exit temperatures. Data for 20 and 26 inch axial grid spacing have been evaluated by plotting thermal diffusion coefficient versus the Reynolds number (Figure 3.5.3-2 plots results for 26 inch grid spacing). TDC is found to be independent of Reynolds number, mass velocity, pressure and quality over the ranges tested. The two phase data (local, subcooled boiling) fell within scatter of the single phase data. The effect of two-phase flow on the value of TDC has been demonstrated by Cadek,(13) Rowe and Angle,(14,15) and Gonzalez-Santalo and Griffith.(16) In the subcooled boiling region the values of TDC were indistinguishable from the single phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in PWR reactor core geometry, the value of TDC increased with quality to a point and then decreases, but never below

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TABLE 3.5.3-1

Cook Nuclear Plant Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Parameters	Operating Parameters	Design Parameters[(1)]
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, %	----	97.4%
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		2.47
Thimble (Cold Wall) Flow Channel		2.33
Design DNBR for Design Transients		
Typical Flow Channel		≥1.69
Thimble Flow Channel		≥1.69
DNB Correlation		WRB-1

(1) Based upon Improved Thermal Design Procedure (ITDP)

TABLE 3.5.3-1 (Continued)

Cook Nuclear Plant Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Parameters	Operating Parameters	Design Parameters(1)
Coolant Conditions		
Minimum Measured Flow, 10^3 gpm	-	366.4(4)
Effective Flow Area for Heat Transfer, ft^2	51.5	51.5
Average Velocity along Fuel Rods, ft/sec	-	16.2
Average Mass Velocity, 10^6 $lbm/hr\text{-}ft^2$	-	2.61
Nominal Vessel/Core Inlet Temperature, ^oF	-	545.5
Vessel Average Temperature, ^oF	567.8	577.1
Core Average Temperature, ^oF	-	579.5
Vessel Outlet Temperature, ^oF	-	608.6
Average Temperature Rise in Vessel, ^oF	-	63.1
Average Temperature Rise in Core, ^oF	-	64.9
Average Enthalpy Rise in Core, BTU/lbm	-	86.57
Film Coefficient at Average Conditions, $BTU/hr\text{-}ft^2\text{-}^oF$	-	6131
Average Film Temperature Difference, ^oF	-	35.5

(4) Thermal hydraulic design completed for minimum measured flow of 366,400 gpm. Subsequent evaluation (Reference 17) supports a minimum measured flow value of 361,600 gpm.

TABLE 3.5.3-1 (Continued)

Cook Nuclear Plant Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Parameters	Operating Parameters	Design Parameters[1]
Heat Transfer		
Active Heat Transfer, Surface Area, ft^2	52,100	52,100
Average Heat Flux, $BTU/hr\text{-}ft^2$	207,400	217,700
Maximum Heat Flux for Normal Operation, $BTU/hr\text{-}ft^2$ [2]	481,200	505,100
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		662.4
Fuel Centerline Temperature		
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F		4700

[2] Based Upon 2.32 F_Q Peaking Factor

TABLE 3.5.3-1 (Continued)

Cook Nuclear Plant Unit 1 Thermal-Hydraulic Design Parameters

Thermal and Hydraulic Parameters	Operating Parameters	Design Parameters[1]
Calculational Factors		
Engineering Heat Flux Factor	-	1.000
Fuel Densification Factor (axial)	-	1.002
Radial Peaking Factor		
Design Nuclear Enthalpy Rise Hot Channel Factor	-	1.49[3]
Pressure Drop		
Across Core, psi (Best Estimate Flow)	-	27.2

[3] Does Not Include Measurement Uncertainty

Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated energy output of the fuel (MWD/MTU) and is a convenient means for quantifying fuel exposure criteria.

The core design lifetime or design discharge burnup is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Section 3.3.2) that meets all safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements (e.g., the controlling band at the "bite" position). In terms of chemical shim boron concentration this represents approximately 10 ppm with no control rod insertion.

A limitation on initial installed excess reactivity is not required other than as is quantified in terms of other design bases such as core negative reactivity feedback and shutdown margin discussed below.

3.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficient)

Basis

The fuel temperature coefficient is negative and the moderator temperature coefficient of reactivity is non-positive for power operation at 100% RTP, thereby providing negative reactivity feedback characteristics. The design basis meets GDC-11.

Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have an overall non-positive moderator temperature coefficient of reactivity at full power so that average coolant temperature or void content provides another, slower compensatory effect. Full power operation is permitted only in a range of overall non-positive moderator temperature coefficient. The non-positive moderator temperature coefficient can be achieved through use of fixed burnable absorber and/or control rods by limiting the reactivity held down by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a non-positive moderator temperature coefficient at power operating conditions discussed above.

3.3.1.3 Control of Power Distribution

Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

1. The fuel is not to be operated at greater than 12.4 Kw/ft for Vantage 5 fuel and 11.6 Kw/ft for ANF fuel under normal operating conditions including an allowance of 2 percent for calorimetric error and not including power spike factor due to densification.

2. Under abnormal conditions including the maximum overpower condition, the fuel peak power does not cause melting as defined in Section 3.4.1.2.

Thus it is not possible to single out any transient or steady-state condition which defines the most limiting case. It is not even possible to separate out a small number which form an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver which provides a limiting case for one reactor fuel cycle (defined as approaching the points of Figure 3.3-20) is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnup, coefficient, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operator conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

Each power shape generated is analyzed to determine if LOCA constraints are met or exceeded. The total peaking factor, F_Q^T is synthesized by combining the axial power profiles with cycle-specific radial factors appropriate for rodded and unrodded planes.

In these calculations the effects on the unrodded radial peak of xenon redistribution that occur following the withdrawal of a control blank (or banks) from a rodded region are obtained from three-dimensional X-Y-Z calculations. A height-dependent factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in References (7) and (8). The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor F_Q^E.

An envelope could be drawn over the calculated [max (F_Q^T power)] points in Figure 3.3-20. This envelope represents an upper bound on local power density versus elevation in the core. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements.

Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within an allowed deviation from a target equilibrium value of axial flux difference. These procedures are detailed in the Technical Specifications and are predicted only upon excore surveillance supplemented by the normal monthly full core map requirement and by computer based alarms on deviation and time of deviation from the allowed flux difference band.

Accident analyses for Cook Nuclear Plant Unit 2 are presented in Chapter 14 of the FSAR. The results of these analyses determined a limiting value of total peaking factor, F_Q, of 2.335 for Vantage 5 fuel and 2.10 for ANF fuel under normal operation, including load follow maneuvers. This value is derived from the conditions necessary to satisfy the limiting conditions specified in the LOCA analyses which meet Appendix K requirements. As noted previously on this section, an upper bound envelope of F_Q x power equal to 2.335 x K(z) for Vantage 5 fuel and 2.10 x K(z) for ANF fuel, as shown in Figure 3.3-20, results from operation in accordance with CAOC procedures using excore surveillance only.

The surveillance of the core hot channel factors in accordance with the above, is presented in the Cook Nuclear Plant Unit 2 Technical Specifications.

Allowing for fuel densification effects, the average kW/ft at 3411 MWt is 5.43 kW/ft. From Figure 3.3-20, the conservative upper bound value of normalized local power density, including uncertainty allowance, is 2.335 for Vantage 5 fuel and 2.10 for ANF fuel corresponding to peak local power density of 12.9 kW/ft and 11.6 kW/ft at 102 percent power, respectively, for Vantage 5 and ANF fuels.

<u>Constant Axial Offset Control Procedures</u>

The Constant Axial Offset Control Procedure[7], enables Cook Nuclear Plant Unit 2 to manage core power distributions such that Technical Specification limits on F_Q^T are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients.

3.4.1.5 Other Considerations

The above design bases together with the fuel clad and fuel assembly design bases given in Section 3.2.1.1. are sufficiently comprehensive so additional limits are not required.

Fuel rod diametral gap characteristics, moderator flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time (see Section 3.2.1.3.1) and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution (see Section 3.4.2.3) and moderator void distribution (see Section 3.4.2.5) are included in the core thermal (THINC) evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in Section 3.2.1.3.1, the fuel rod conditions change with time. A single clad temperature limit for Condition I or Condition II events is not appropriate since of necessity it would be overly conservative. An appropriate clad temperature limit is applied in each case to the loss of coolant accident (Section 14.3.1), control rod ejection accident,[6] and locked rotor accident.

3.4.2 DESCRIPTION

3.4.2.1 Summary Comparison

Table 3.4-1 provides a comparison of the current design parameters for the Cook Nuclear Plant Unit 2 core described herein with those for the initial cycle.

The Cook Nuclear Plant Unit 2 core described herein is demonstrated to meet all design bases by considering the values of plant parameters and the uncertainties in these parameters through the use of the Revised Thermal

Design Procedure[1]. The justification for the analytical techniques used in determining the values presented for the Cook Nuclear Plant Unit 2 core are presented in the relevant chapters in this document.

3.4.2.2 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in Section 3.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation. A discussion of fuel clad integrity is presented in Section 3.2.1.3.1.

The thermal-hydraulic design assures that the maximum fuel temperature is below the melting point of UO_2 (melting point of $5080^{o}F$[5] unirradiated and decreasing by $58^{o}F$ per 10,000 MWD/MTU). To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of $4700^{o}F$ has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluation as described in Section 3.4.2.10.1. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad, gap and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, etc., have been combined into a semiempirical thermal model (see Section 3.2.1.3.1) which includes a model for time dependent fuel densification, as given in Reference (7). This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to inpile fuel temperature measurements[7,8] and melt radius data[9, 10] with good results.

Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are performed for several times in the fuel rod lifetime, with consideration of time dependent densification, to determine the maximum fuel temperatures.

3.4.2.2.4 Surface Heat Transfer Coefficients

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in Section 3.4.2.8.1.

3.4.2.2.5 Fuel Clad Temperatures

The outer surface of the fuel rod at the hot spot operates at a temperature of approximately 660°F for steady-state operation at rated power throughout core life due to the onset of nucleate boiling. Initially (beginning-of-life), this temperature is that of the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of the clad temperature.

3.4.2.2.6 Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum to core average heat flux. As presented in Table 3.3-2 and discussed in Section 3.3.2.2.6, the design value of F_Q for normal operation in Cycle 8 was 2.335. This resulted in a peak local power of 12.7 Kw/ft at full power conditions.

As described in Section 3.3.2.2.6 the peak linear power resulting from overpower transients/operator errors (assuming a maximum overpower of 118 percent) is 22.5 Kw/ft. The centerline temperature Kw/ft must be below the UO_2 melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is discussed in 3.4.1.2 and results in a maximum allowable calculated centerline temperature of 4700°F. The peak linear power for prevention of centerline melt is > 22.5 kW/ft. The centerline temperature at the peak linear power resulting from overpower transients/overpower errors (assuming a maximum overpower of 118 percent) is below that required to produce melting.

3.4.2.3 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The minimum DNBRs for the rated power, design overpower and anticipated transient conditions are given in Table 3.4-1. The core average DNBR is not a safety-related item as it is not directly related to the minimum DNBR in the core, which occurs at some elevation in the limiting flow channel which is typically downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in the following Sections 3.4.2.3.1 and 3.4.2.3.2. The THINC-IV computer code (discussed in Section 3.4.3.4.1) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Section 3.4.3.2.1 (nuclear hot channel factors) and in Section 3.4.2.3.4 (engineering hot channel factors).

3.4.2.3.1 Departure from Nucleate Boiling Technology

Early experimental studies of DNB were conducted with fluid flowing inside single heated tubes or channels and with single annulus configurations with one or both walls heated. The results of the experiments were analyzed using many different physical models for describing the DNB phenomenon but all resultant correlations are highly empirical in nature. The evolution of these correlations is given by Tong [3, 31] including the W-3 Correlation which is in wide use in the pressurized water reactor industry.

The fuel assembly hold down springs are designed to keep the fuel assemblies resting on the lower core plate under all Condition I and II events with the exception of the turbine overspend transient associated with a loss of external load. Under the transient flow rate and core pressure drop resulting from increased pump outputs with turbine overspend, fuel assembly lift may occur. However, the hold down springs are designed to tolerate the over-deflection for this case and to provide the required hold down force following the transient. Maximum flow conditions are usually limiting because hydraulic loads are a maximum. The most adverse hydraulic loads occur during a LOCA. This accident is discussed in Section 14.3.

Core hydraulic loads were measured during the prototype assembly tests. Reference (2) contains a detailed discussion of the results.

3.4.2.8 Correlation and Physical Data

3.4.2.8.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the familiar Dittus-Boelter correlation, [51] with the properties evaluated at bulk fluid conditions:

$$\frac{hD_e}{k} = 0.023\left[\frac{D_e G}{\mu}\right]^{0.8} \left[\frac{C_p \mu}{k}\right]^{0.4} \qquad (3.4\text{-}6)$$

Where:

- h - heat transfer coefficient, BTU/hr-ft^2-$^\circ$F.
- D_e - equivalent diameter, ft.
- k - thermal conductivity, BTU/hr-ft-$^\circ$F.
- G - mass velocity, lb/hr-ft^2.
- 1 - dynamic viscosity, lb/ft-hr.
- C_p - heat capacity, BTU/lb-$^\circ$F.

This correlation has been shown to be conservative [55] for rod bundle geometries with pitch to diameter ratios in the range used by PWRs.

The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's[56] correlation. After this occurrence the outer clad wall temperature is determined by:

$$\Delta T_{sat} = [0.072 \exp(-P/1260)] \quad (q")^{0.5} \tag{3.4-7}$$

Where:

ΔT_{sat} = wall superheat, $T_w - T_{sat}$, ^oF.
$q"$ = wall heat flux, $BTU/hr\text{-}ft^2$.
P = pressure, psia.
T_w = outer clad wall temperature, ^oF.
T_{sat} = saturation temperature of coolant at P, ^oF.

3.4.2.8.2 Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single-phase water. These assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible (see Section 3.4.2.5 and Table 3.4-2). Two phase flow considerations in the core thermal subchannel analyses are considered and the models are discussed in Section 3.4.3.1.3. Core and vessel pressure losses are calculated by equations in the form:

$$P_L = (K+F \frac{L}{D_e}) \frac{\rho V^2}{2 g_c (144)} \tag{3.4-8}$$

Where:

P_L - unrecoverable pressure drop, lb_f/in^2

ρ - fluid density, lb_m/ft^3.

L - length, ft.

D_e - equivalent diameter, ft.

V - fluid velocity, ft/sec.

g_c - $32.174 \dfrac{lbm\text{-}ft}{lb_f\text{-}sec^2}$

K - form loss coefficient, dimensionless.

F - friction loss coefficient, dimensionless.

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in Table 3.4-1 for unrecoverable pressure loss across the reactor vessel, including the inlet and outlet nozzles, and across the core. The results of full scale tests of core components and fuel assemblies were utilized in developing the core pressure loss characteristic. The pressure drop for the vessel was obtained by combining the core loss with correlation of 1/7th scale model hydraulic test data on a number of vessels[57, 58] and form loss relationships.[59] Moody[60] curves were used to obtain the single phase friction factors.

Core pressure drops were confirmed by full-scale hydraulic flow tests performed in the fuel assembly test system (FATS) facility[2]. These hydraulic verification tests include hydraulic head losses and effects of velocity changes as well as unrecoverable pressure losses. The effects of velocity changes are small since the static pressure taps are located at elevations of approximately equal flow areas (and therefore approximately equal velocities). When wall static pressure taps are used near ambient fluid conditions, it can be shown analytically that the elevation head losses do not contribute to the measured core pressure drops. Therefore, data from the hydraulic verification tests can be directly applied to confirm the pressure drop values quoted in Table 3.4-1 which are based on unrecoverable pressure losses only.

Tests of the primary coolant loop flow rates were performed (see Section 3.4.4.1) prior to initial criticality to verify that the flow rates used in the design, which were determined in part from the pressure losses calculated by the method described here, are conservative.

3.4.2.8.3 Void Fraction Correlation

There are three separate void regions considered in flow boiling in a PWR as illustrated in Figure 3.4-7. They are the wall void region (no bubble detachment), the subcooled boiling region (bubble detachment) and the bulk boiling region.

In the wall void region, the point where local boiling begins is determined when the clad temperature reaches the amount of superheat predicted by Thom's[56] correlation (discussed in Section 3.4.2.8.1). The void fraction in this region is calculated using Maurer's[61] relationship. The bubble detachment point, where the superheated bubbles break away from the wall, is determined by using Griffith's[62] relationship.

The void fraction in the subcooled boiling region (that is, after the detachment point) is calculated from the Bowring[63] correlation. This correlation predicts the void fraction from the detachment point to the bulk boiling region.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

3.4.2.9 Thermal Effects of Operational Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power, and the axial and radial power distributions. Operation within these DNB safety limits insures that the DNB design basis is met for both steady-state operation and for anticipated operational transients that are slow with respect to fluid transport delays in the primary system. In addition, for fast transients, e.g., uncontrolled rod bank withdrawal at power incident, specific protection functions are provided as described in Chapter 7 and the use of these protection functions are described in Chapter 14. The thermal response of the fuel is discussed in Section 3.4.3.7.

3.4.2.10 Uncertainties in Estimates

3.4.2.10.1 Uncertainties in Fuel and Clad Temperatures

As discussed in 3.4.2.2, the fuel temperature is a function of crud, oxide, clad, gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to the inpile thermocouple measurements,[7,8] by out-of-pile measurements of the fuel and clad properties,[11-22] and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is also included in the calculation of the total uncertainty.

In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power, and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in Section 3.3.2.2.1.

Reactor trip setpoints are as specified in the Technical Specifications include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproducibility, temperature measurement uncertainties, noise, and heat capacity variations.

Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

3.4.2.10.2 Uncertainties in Pressure Drops

Core and vessel pressure drops based on the Best Estimate Flow, described in 3.4.2.6, are quoted in Table 3.4-1. The uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drops is to determine the primary system coolant flow rates as discussed in 3.4.2.6. In addition, as discussed in Section 3.4.4.1, tests on the primary system prior to initial criticality were made to verify that a conservative primary system coolant flow rate has been used in the design and analyses of the plant.

3.4.2.10.3 Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses is discussed in Section 3.4.3.1.2.

3.4.2.10.4 Uncertainty in DNB Correlation

The uncertainty in the DNB correlation (Section 3.4.2.3) can be written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is discussed in Section 3.4.2.3.2.

3.4.2.10.5 Uncertainties in DNBR Calculations

The uncertainties in the DNBRs calculated by THINC analysis (see Section 3.4.3.4.1) due to nuclear peaking factors are accounted for by applying conservatively high values of the nuclear peaking factors and including measurement error allowances in the statistical evaluation of the limit DNBR (see Section 3.4.1.1) using the Revised Thermal Design Procedure.[1] In addition, engineering hot channel factors are employed as discussed in Section 3.4.2.3.4.

The results of the sensitivity study[33] with THINC-IV show that the minimum DNBR in the hot channel is relatively insensitive to variations in the corewide radial power distribution (for the same value of $F^N_{\Delta H}$).

The ability of the THINC-IV computer code to accurately predict flow and enthalpy distributions in rod bundles is discussed in Section 3.4.3.4.1 and in Reference (34). Studies have been performed[33] to determine the sensitivity of the minimum DNBR in the hot channel to the void fraction correlation (see also Section 3.4.2.8.3); the inlet velocity and exit pressure distributions assumed as boundary conditions for the analysis; and the grid pressure loss coefficients. The results of these studies show that the minimum DNBR in the hot channel is relatively insensitive to variations in these parameters. The range of variations considered in these studies covered the range of possible variations in these parameters.

3.4.2.10.6 Uncertainties in Flow Rates

The uncertainties associated with loop flow rates are discussed in 3.4.2.6. For core thermal performance evaluations, a thermal design flow is used which is approximately 7 percent less than the best estimate flow.

In addition, a maximum of 5.1 percent of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses the core through the various available vessel flow paths described in Section 3.4.3.1.1.

3.4.2.10.7 Uncertainties in Hydraulic Loads

As discussed in Section 3.4.2.7.2., hydraulic loads on the fuel assembly are evaluated for a pump overspend transient which create flow rates 20 percent greater than the mechanical design flow. The mechanical design flow as stated in 3.4.2.6 is approximately 6 percent greater than the best estimate flow or most likely flow rate value for the actual plant operating condition.

3.4.2.10.8 Uncertainty in Mixing Coefficient

The value of the mixing coefficient, TDC, used in THINC analyses for this application is 0.038. The mean value of TDC obtained in the "R" grid mixing tests described in Section 3.4.2.3.1 was 0.042 (for 26 inch grid spacing). The value of 0.038 is one standard deviation below the mean value; and 90 percent of the data gives values of TDC greater than 0.038.[38]

The results of the mixing tests done on 17 x 17 geometry, as discussed in Section 3.4.2.3.3., had a mean value of TDC of 0.059 and standard deviation of 0.007. Hence the current design value of TDC is almost 3 standard deviations below the mean for 26 inch grid spacing.

3.4.2.11 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided in the appropriate Chapters 4, 6 and 9. Implementation of the emergency core cooling systems (ECCS) is discussed in Chapter 6 and evaluation of ECCS performance appears in Chapter 14. Some specific areas of interest are the following:

1. Total coolant flow rates for the reactor coolant system (RCS) and each loop are provided in 3.4.2.6. Flow rates employed in the evaluation of the core are presented in Section 3.4.3.

One standard startup test is the natural circulation test in which the core is held at a very low power (2 percent) and the pumps are turned off. The core is cooled by the natural circulation currents created by the power differences in the core. During natural circulation, a thermal siphoning effect occurs resulting in the hotter assemblies gaining flow, thereby creating significant interassembly crossflow. As described in the preceding section the most important feature of THINC-IV is the method by which crossflow is evaluated. Thus, tests with significant crossflow are of more value in the code verification. Interassembly crossflow is caused by radial variation in pressure. Radial pressure gradients are in turn caused by variations in the axial pressure drop is due mainly to friction losses. Since all assemblies have the same geometry, all assemblies have nearly the same axial pressure drops and crossflow velocities are small. However, under natural circulation conditions (low flow) the axial pressure drop is due primarily to the difference in elevation head (or coolant density) between assemblies (axial velocity is low and therefore axial friction losses are small). This phenomenon can result in relatively large radial pressure gradients and therefore higher crossflow velocities than at normal reactor operation conditions.

The incore instrumentation was used to obtain the assembly-by-assembly core power distribution during a natural circulation test. Assembly exit temperatures during the natural circulation test on a 157 assembly, three-loop plant were predicted using THINC-IV. The predicted data points were plotted as assembly temperature rise versus assembly power and a least squares fitting program used to generate an equation which best fit the data. The result is the straight line presented in Figure 3.4-9. The measured assembly exit temperatures are reasonably uniform, as indicated in this figure, and are predicted closely by the THINC-IV Code. This agreement verifies the lateral momentum equations and the crossflow resistance model used in THINC-IV. The larger crossflow resistance used in THINC-I reduces flow redistribution, so that THINC-IV gives better agreement with the experimental data.

Data has also been obtained for Westinghouse plants operating from 67 percent to 101 percent of full power. A representative cross section of the data obtained from a two-loop and three-loop reactor were analyzed to verify the THINC-IV calculational method. The THINC-IV predictions were compared with the experimental data as shown in Figures 3.4-10 and 3.4-11. The predicted assembly exit temperatures were compared with the measured exit temperature for each data run. The standard deviation of the measured and predicted assembly exit temperatures were calculated and compared for both THINC-IV and THINC-I and are given in Table 3.4-4. As the standard deviations indicate, THINC-IV generally fits the data somewhat more accurately than THINC-I. For the core inlet temperatures and power of the data examined, the coolant flow is essentially single phase. Thus, one would expect little interassembly crossflow and small differences between THINC-IV and THINC-I predictions as seen in the tables. Both codes are conservative and predict exit temperatures higher than measured values for the high powered assemblies.

An experimental verification of the THINC-IV subchannel calculation method has been obtained from exit temperature measurements in a non-uniformly heated rod bundle.(73) The inner nine heater rods were operated at approximately 20 percent more power than the outer rods to create a typical PWR intra-assembly power distribution. The rod bundle was divided into 36 subchannels and the temperature rise was calculated by THINC-IV using the measured flow and power for each experimental test.

Figure 3.4-12 shows, for a typical run, a comparison of the measured and predicted temperature rises as a function of the power density in the channel. The measurements represent an average of two to four measurements taken in various quadrants of the bundle. It is seen that the THINC-IV results predict the temperature gradient across the bundle very well. In Figure 3.4-13, the measured and predicted temperature rises are compared for a series of runs at different pressures, flows, and power levels.

3.4.3.8 Energy Release During Fuel Element Burnout

As discussed in Section 3.4.3.3 the core is protected from going through DNB over the full range of possible operating conditions. At full power nominal operation, the minimum DNBR was found to be 2.28. This means that for these conditions, the probability of a rod going through DNB is much less than 0.1 percent with 95 percent confidence. In the extremely unlikely event that DNB should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is potential for a chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following DNB, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

DNB With Physical Burnout

Westinghouse[73] has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25 rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been conducted by Westinghouse[80] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction in power of approximately 10 percent was sufficient to reestablish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

3.4.3.9 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV program. Inspection of the DNB correlation (Section 3.4.2.3 and Reference (2)) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The THINC-IV Code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. In Reference (34), it is shown that for a fuel assembly similar to the Westinghouse design, THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reference reactor operating at the nominal full power conditions specified in Table 3.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching the limiting DNBR specified in Section 3.4.1.1.

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the Westinghouse cores cause flow perturbations which are local to the blockage. For instance, A. Ohtsubo, et al.[81] shows that the mean bundle velocity is approached asymptotically about 4 inches downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked.

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TABLE 3.4-1

COOK NUCLEAR PLANT UNIT 2 REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Initial Cycle	Cycles 8 & 9	Bounding Parameters Cycle 10 & Beyond
Reactor Core Heat Output, MWt	3391	3411	3588
Reactor Core Heat Out, 10^6 BTU/hr	11,573.5	11,639	12,240
Heat Generated in Fuel, %	97.4	97.4	97.4
System Pressure, Nominal, psia(*)	2280	2280	2251
System Pressure, Minimum Steady-State, psia	2250	2250	2100
Minimum DNBR at Nominal Conditions			
Typical Flow Channel	3.03[a]	2.42	2.42
Thimble (Cold Wall) Flow Channel	2.70[a]	2.28	2.28
Design DNBR for Design Transients			
Typical Flow Channel	1.80[b]	1.69	1.69
Thimble Flow Channel	1.77[b]	1.61	1.61
DNB Correlation	WRB-1	WRB-2	WRB-1
Coolant Flow			
Total Thermal Design Flow Rate, 10^6 lb_m/hr	142.7(**)	134.3	133.2
Best Estimate Flow, 10^6 lb_m/hr	148.4	145.2	143.8
Mechanical Design Flow, 10^6 lb_m/hr	154.3	154.5	153.0
Minimum Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	136.3	127.4	126.4
Effective Flow Area for Heat Transfer, ft^2	51.1	54.1	54.1
Average Velocity Along Fuel Rods, ft/sec	16.7	14.6	14.7
Average Mass Velocity, 10^6 $lb_m/hr\text{-}ft^2$	2.72	2.35	2.34

(*) Pressure in the core. See Reference (1).

(**) Value used in DNB analyses (RTDP Transients)

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.

The 6" pipes connecting the pressurizer nozzles to their respective safety valves are shaped in the form of a loop seal. Piping is connected to the bottom of each loop seal to drain any condensate that accumulates in the loop seal. 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 390 psig for Unit 1, 425 psig for Unit 2, a safeguard circuit is energized in the control room to allow automatic opening of that Unit's two power relief valves at 400 psig for Unit 1, 435 psig for Unit 2 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.

TABLE 4.2-1 (Continued)

Component	Section	Material Unit 1	Material Unit 2
Pressurizer (cont.)	Nozzle Forgings	Integrally cast with head	SA-508 Class 2 Mn-Mo
	Internal Plate	SA-240 Type 304	SA-240 Type 304
	Inst. Tubing	SA-213 Type 316	SA-213 Type 316
	Heater Well Tubing	SA-213 Type 316 Seamless	SA-213 Type 316 Seamless
	Heater Well Adaptor	SA-182 F316	-
Pressurizer Relief Tank	Shell	ASTM A-285 Grade C	ASTM A-285 Grade C
	Heads	ASTM A-285 Grade C	ASTM A-285 Grade C
	Internal Coating	Amercoat 55	Amercoat 55
Pipe	Pipes	ASTM A-351 Grade CF8M	ASTM A-351 Grade CF8M
		ASTM A-376 Grade TP 304 or TP 316	ASTM A-376 Grade TP 304 or TP 316
	Fittings	ASTM A-351 Grade CF8M	ASTM A-351 Grade CF8M
	Nozzles	ASTM A-182 Grade F316	ASTM A-182 Grade F316
Pump	Shaft	ASTM A-182 Grade F347	ASTM A-182 Grade F347
	Impeller	ASTM A-351 Grade CF8M	ASTM A-351 Grade CF8M
	Casing	ASTM A-351 Grade CF8M	ASTM A-351 Grade CF8M
Valves	Pressure Containing Parts	ASTM A-351 Grade CF8M and ASTM A-182 Grade F316	ASTM A-351 Grade CF8M and ASTM A-182 Grade F316

TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is 1 to 40 uMhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.
Oxygen, ppm, max.	0.10
Chloride, ppm, max.	0.15
Fluoride, ppm, max.	0.15
Hydrogen, cc (STP)/kg H_2O	25-35
Total Suspended Solids, ppm, max.	1.0
pH Control Agent (Li^7OH)	0.29×10^{-4} to 5.53×10^{-4} molal (equivalent to 0.20 to 3.8 ppm Li^7)
Boric Acid as ppm B	Variable from 0 to 4000

TABLE 4.2-3

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION
FOR 100% FULL POWER

Cation Conductivity[a]	$\leq$ 0.8 umhos/cm
pH @ 25°C	$\geq$ 7.5
Chloride	$\leq$ 30 ppb
Sodium	$\leq$ 20 ppb
Sulfate	$\leq$ 30 ppb

[a]If boric acid is present in the system, the cation conductivity specification will be $\leq$)0.8 + 0.03 x (boron conc. in ppm)].

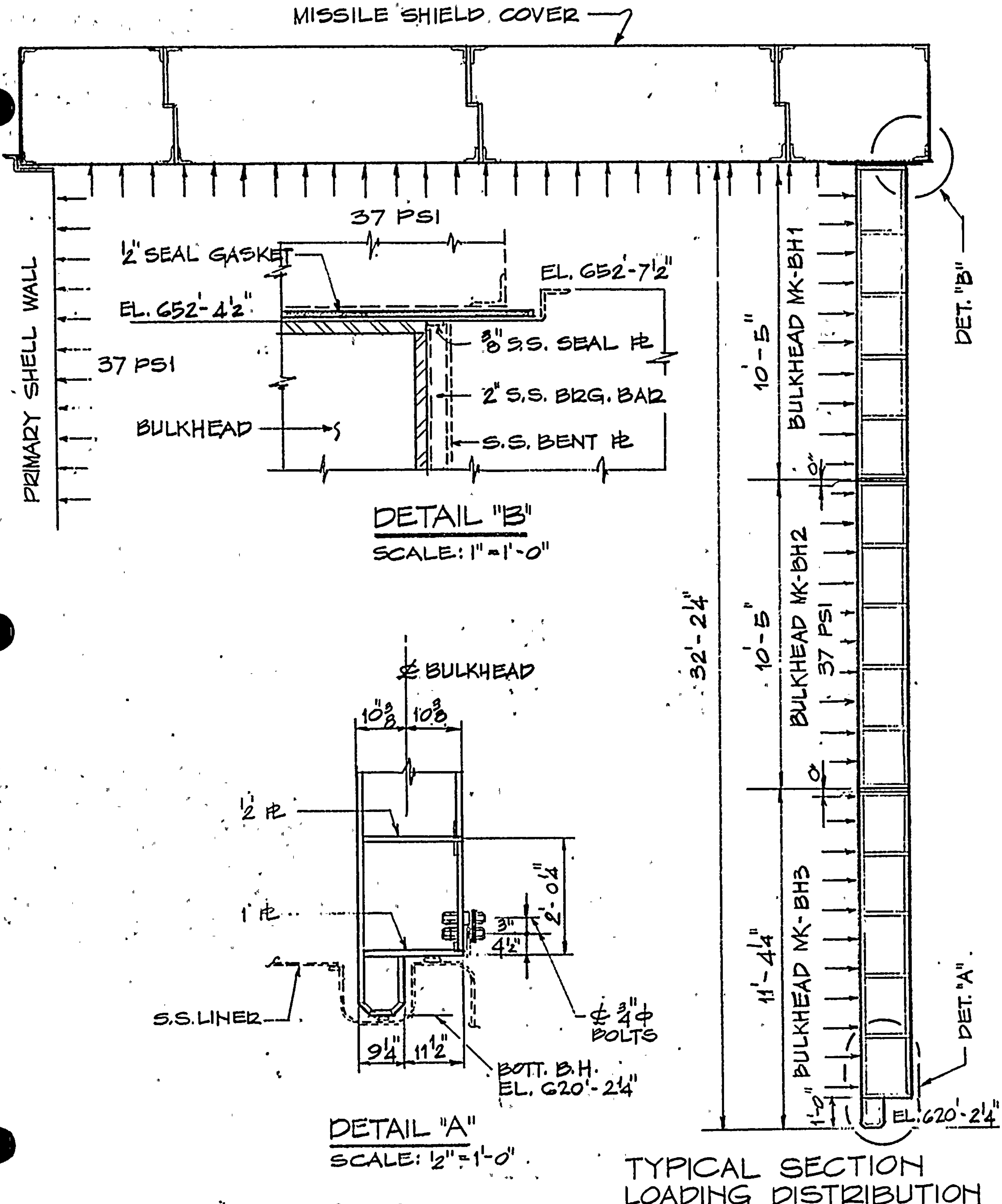

Fig. 5.2.2-54

July 1982

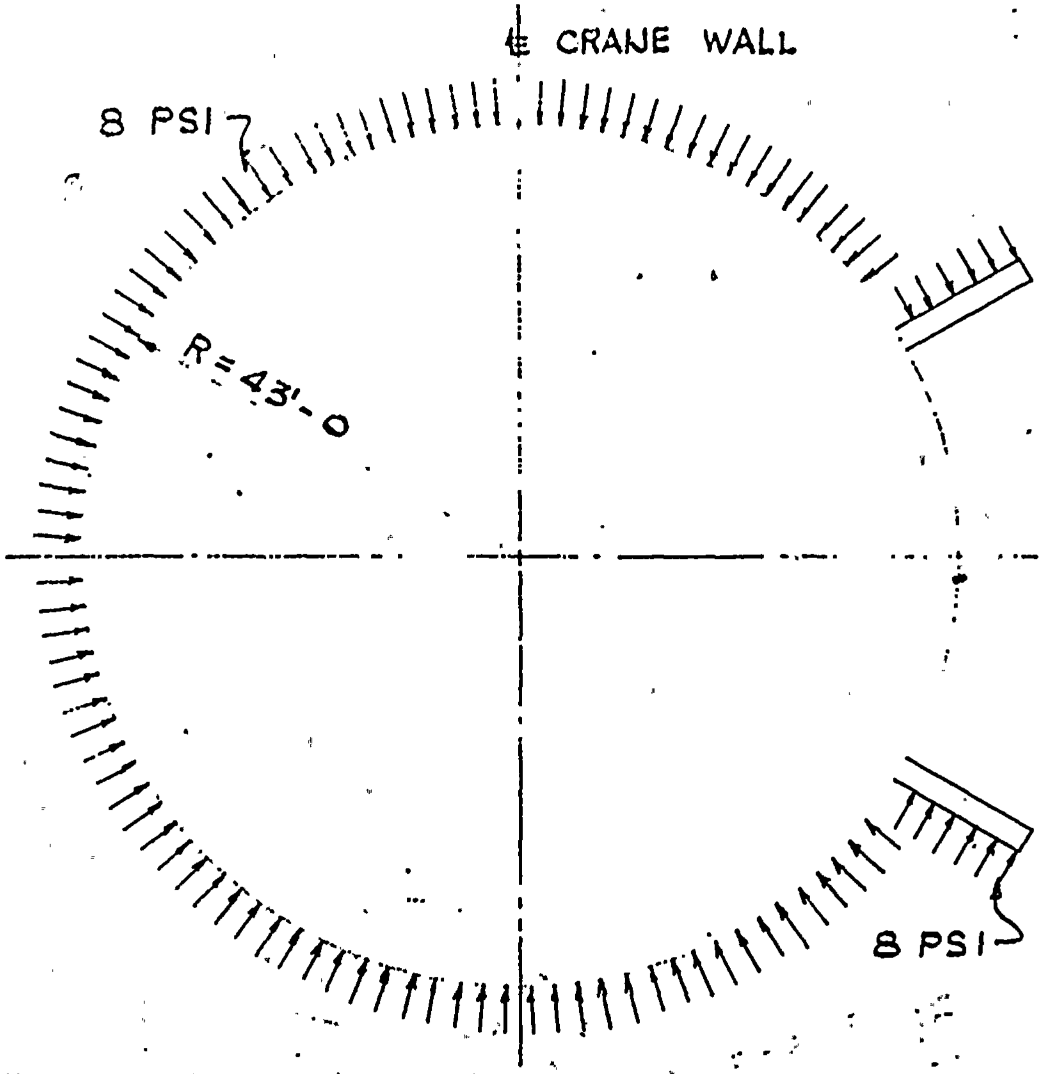

FIG 5. 26. 2-1 LOADING DIAGAM OF UPPER CRANE WALL

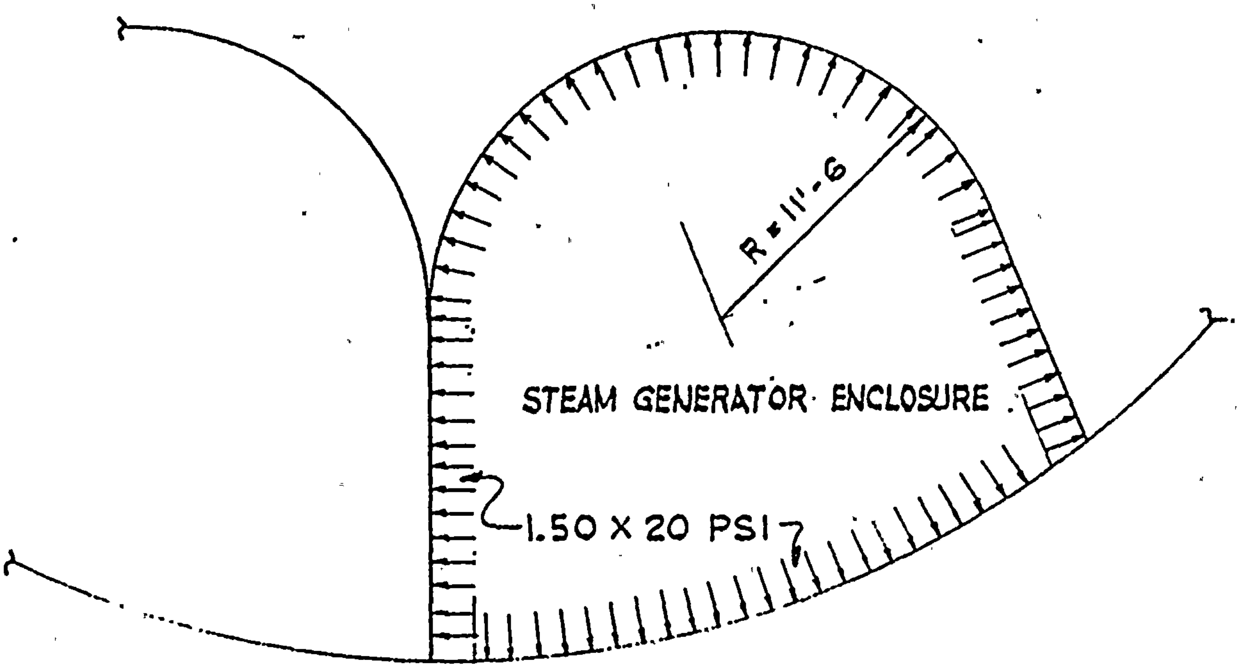

FIG. 5.2.2-54A UNSYMMETRICAL INTERNAL PRESSURE LOADING DIAGRAM OF 30 PSI OF STEAM GENERATOR ENCLOSURE
July 1982

5.4 CONTAINMENT ISOLATION SYSTEM

The Containment Isolation System provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident.

5.4.1 DESIGN BASES

The Cook Nuclear Plant was designed to the general design criteria stated in Sub-Chapter 1.4. The design of the piping, valving, penetrations, and areas in the vicinity of the penetrations was completed before July 1971 when the AEC General Design Criteria Nos. 54, 55, 56 and 57 were published in the Federal Register.

The design bases applying to all the features of the Containment Isolation System at Cook Nuclear Plant are given in the following paragraphs.

Containment Isolation System Design Basis

Subsequent to an incident, there are at least two barriers between the atmosphere outside the containment and (1) the containment atmosphere, (2) the Reactor Coolant System or (3) closed systems inside the containment which are assumed vulnerable to accident forces. The following conditions and definitions are used in the design of the Containment Isolation System to assure that the above is met:

1. The design pressure of all piping and connected equipment within the isolated boundary is greater than the design pressure of the containment.

2. Lines connected to secondary systems with a low probability of failure have at least one automatic shut-off valve.

3. All valves and equipment which are considered to be isolation barriers are protected against missiles and water jets, both inside and outside the containment.

4. Lines which, due to safety considerations, must remain in service subsequent to certain accidents have as a minimum one remote-manual valve.

5. All isolation valves and equipment are designed to operate as Class I seismic equipment.

6. The two barriers may consist of: (1) two automatic isolation valves, (2) an automatic isolation and a normally closed valve, (3) an automatic isolation valve and a closed piping system or vessel inside or outside the containment, (4) two normally closed valves, (5) a normally closed valve and blind flange and/or cap, or (6) two blind flanges and caps.

7. A check valve or a locked closed valve is considered equivalent to an automatic valve.

8. Automatic isolation is provided in all cases except for those lines which are required to be operational in post accident conditions.

Containment Isolation Testing and Reliability

The Containment Isolation System is designed to provide such functional reliability and testing facilities as are necessary to avoid undue risk to the health and safety of the public. The air operated isolation valves close on loss of control power or air. The instrumentation and control circuits are redundant in the sense that a single failure cannot prevent containment isolation. Provision is made for periodic testing of the leak tightness and functioning of the isolation valves.

Containment Isolation System Protection

Adequate protection for containment isolation, including piping, valves, and vessels, is provided against dynamic effects and missiles which might result from plant equipment failures including a loss-of-coolant accident.

Isolation valves inside the containment are located between the crane wall or some other missile shield and the outside containment wall.

Isolation valves, piping or vessels which provide one of the isolation barriers outside the containment are similarly protected.

Containment Isolation System Operation

No manual operation is required for immediate isolation of the containment. Automatic trip valves are provided in those lines which must be isolated immediately following an accident. Lines which must remain in service subsequent to certain accidents for safety reasons are provided with at least one remote manual valve.

Automatic trip valves may be operated by a manual switch. The position of each automatic trip valve is displayed in the main control room.

The instrumentation and controls for the system are described in more detail in Chapter 7.

Containment Isolation System Piping Classes

The functional classes of piping are used to further define the design bases. They are presented in Figure 5.4-1.

Class A

Class A piping is open to the outside atmosphere, and is connected to the Reactor Coolant System or is open to the containment atmosphere.

For Class A piping the following is provided, as a minimum, for isolation subsequent to an incident:

a) Incoming Lines: Two auto-trip valves or a check valve and an auto-trip valve.

b) Outgoing Lines: Two auto-trip valves or two locked closed valves.

Class B

Class B piping is connected to a closed system outside the containment, and is connected to the Reactor Coolant System or is open to the containment atmosphere.

For Class B piping the following is provided for minimum isolation subsequent to an incident:

a) Incoming Lines: One auto-trip valve or a check valve

b) Outgoing Lines: One auto-trip valve

Class C

Class C piping is connected to open systems outside the containment, and is separated from the Reactor Coolant system and the containment atmosphere by a membrane barrier.

For Class C piping, the following is provided for minimum isolation subsequent to an incident:

a) Incoming Lines: One check valve

b) Outgoing Lines: One auto-trip valve

Class D

Class D piping must remain in service after a hypothetical accident. Piping of the engineered safety features falls into this category.

For Class D piping the following is provided for minimum isolation subsequent to an incident.

a) Incoming Lines: One remote manual valve or a check valve

b) Outgoing Lines: One remote manual valve

Class E

Class E piping is connected to a normally closed system outside of the containment, and is separated from the Reactor Coolant System and the containment atmosphere by a closed valve and/or a membrane barrier.

For Class E piping the following constitutes the minimum isolation provided.

All Lines: A normally closed manual valve inside or outside the containment. (EXCEPTION: A membrane barrier outside the containment is used for sensing lines of the reactor vessel level instrumentation System.)

5.4.2 CONTAINMENT ISOLATION SYSTEM DESIGN

The general design basis covering the number and location of isolation valves required to assure reactor containment integrity is given in Section 5.4.1. A summary of the major piping penetrations is given in Table 5.4-1. This table lists the number and types of isolation valves that are provided for the lines penetrating the containment. Valve positions during normal operation, shutdown, and incident conditions are also listed.

Check valves may be employed as one of the two barriers for incoming lines.

Test connections and pressurizing means are provided to test each isolation valve or barrier for leak tightness. Either water or a gas is used as the pressurizing medium depending on the requirements of each case. Where it is necessary to make a quantative leakage test, provision is made to:

a) measure the inflow of the pressurizing medium, or
b) collect and measure the leakage, or
c) calculate the leakage from the rate of pressure drop.

The test connections are valved out and capped when not in use.

All isolation valves are missile protected. Isolation valves, actuators, and control devices required inside the containment are located between the missile barrier and the containment wall. Isolation valves, actuators and control devices outside the containment are located outside the path of potential missiles or provided with missile protection.

There are two levels of automatic containment isolation identified as Phase A and Phase B. Phase A isolation closes all lines penetrating the containment except essential lines such as Safety Injection and Containment Spray which are not isolated, and component cooling water to the reactor pumps and service water to the ventilation units which isolates on Phase B. (For Phase A and B initiating signals see Chapter 7 Instrumentation and Control.) All automatic isolation valves are able to be closed from the main control room. Position indicators are provided for each valve near its manual control switch in the main control room.

Specific administrative procedures govern the positioning of all isolation valves except check valves as well as any flanged closures during normal operation, shutdown and incident conditions. Check valves in incoming lines open only when the fluid pressure in the line coming from the outside is higher than the pressure on the containment side. Gravity or a spring holds the valve closed in the balanced pressure condition.

5.4.3 DESIGN EVALUATION

The containment isolation system provides two barriers to prevent leakage of radioactivity at each containment opening. Either barrier is sufficient to keep the leakage within limits.

5.4.4 TEST AND INSPECTION

All valve leak testing for Inservice Inspection (ISI) and Integrated Leak Rate Test (ILRT) program and surveillance requirements are performed in accordance with Appendix J to 10 CFR 50 for Type A, B and C type testing. Also certain valves will be tested for operability in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Gas Analyzer From Pressurizer Relief Tank	A	1/2" (1)	Out	Int.	Closed	Closed	Auto Trip Trip	2 Auto	A	4.2-1A	
Primary Water Supply To Pressurizer Relief Tank	A	3" (1)	In	Int.	Closed	Closed	Check	Auto Trip	A	4.2-1A	
Nitrogen Supply To Pressurizer Relief Tank	A	3/4" (1)	In	Int.	Closed	Closed	Check	Auto Trip	A	4.2-1A	
Reactor Coolant Pumps Seal Water Supplies	D	2" (4)	In	Open	Open	Open	Check	-	NA	4.2-1A	
Reactor Coolant Pumps Seal Water & Excess Letdown Heat Exchanger Discharges	B	4" (1)	Out	Open	Open	Closed	Auto Trip	Auto Trip	A	9.2-1	
Reactor Coolant Pump Motor and Thermal Barrier Cooling Water Supply	B	8" (1)	In	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	
Reactor Coolant Pump Motor Cooling Water Discharge	B	8" (1)	Out	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Letdown Line (CVCS)	B	2" (1)	Out	Open	Closed	Closed	Auto Trip	Auto Trip	A	9.2-1	
Charging Line (CVCS)	D	3" (1)	In	Open	Closed	Open	Check	-	NA	9.2-1	
Excess Letdown Heat Exchanger Component Cooling Water Inlet	C	4" (1)	In	Open	Closed	Closed	-	Auto Trip	A	9.5-1	
Excess Letdown Heat Exchanger Component Cooling Water Outlet	C	4" (1)	Out	Open	Closed	Closed	-	Auto Trip	A	9.5-1	
Reactor Coolant Drain Tank Pump Suction	A	4" (1)	Out	Int.	Int.	Closed	-	2 Auto Trip	A	11.1-1	
Containment Sump Pump Discharge to Waste Disposal	C	3" (1)	Out	Int.	Int.	Closed	-	2 Auto Trip	A	11.1-2	
Upper Containment Spray Inlet	D	8" (2)	In	Closed	Closed	Open	Check	-	NA	6.3-1	1
Lower Containment Spray Inlet	D	6" (2)	In	Closed	Closed	Open	Check	-	NA	6.3-1	1
RHR to Containment Spray	D	8" (2)	In	Closed	Closed	If Needed	Check	-	NA	6.3-1	1

1) Check valves held closed by gravity or spring in balanced pressure condition.

5.4-9 July 1990

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves			Isolation Valves		Isolation Actuation Signal	Figure Number	Notes
				N	S	I	Inside	Outside			
Residual Heat Removal Inlet to Pumps (Normal Cooldown)	B	14" (1)	Out	Closed	Open	Closed	Remote Manual	-	None	6.2-A 9.3-1	2
Residual Heat Removal To Reactor Coolant Hot Legs-Low Head S.I.	D	8" (2)	In	Open	Closed	Open	-	Remote Manual	None	6.2-A 9.3-1	
Residual Heat Removal Suction From Sump	D	18" (2)	Out	Closed	Closed	Open	-	Remote Manual	None	6.2-A 9.3-1	3
Safety Injection	D	4" (2)	In	Open	Closed	Open	-	Remote Manual	None	6.2-1	
Safety Injection Test Line and Accumulator Test Line	A	3/4" (1)	In or Out	Int.	Closed	Closed	-	2 Manual (L.C.)	NA	-	
Boron Injection Inlet	D	3" (1)	In	Closed	Closed	Open	-	Remote Manual	None	6.2-1	4
Residual Heat Removal to Reactor Coolant Cold Legs (Normal Cooldown)	B	12" (1)	In	Closed	Open	If Needed	Remote Manual	-	None	6.2-A 9.3-1	

2) Valve administratively locked closed.
3) Open during recirculation mode.
4) Open automatically on Safety Injection Signal.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Nitrogen to Accumulators	A	1" (1)	In	Int.	Int.	Closed	Check	Auto Trip	A	6.2-A 9.3-1	
Sample Line From Pressurizer Steam Space	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Line from Pressurizer Liquid Space	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Line from Hot Legs	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Line From Accumulators	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Lines from Steam Generator Steam Outlets	C	1/2" (4)	Out	Open	Closed	Closed	-	Auto Trip	A	9.6-1	
Steam Generator Main Steam Outlets	C	30" (4)	Out	Open	Closed	Closed	-	-	B	10.2-1	5
Steam Generator Blowdown Lines	C	2" (4)	Out	Int.	Closed	Closed	-	Auto Trip	A	10.2-1	

5) Steam Generator Stop Valves located outside containment also close on steamline isolation signal as described in Chapter 7.

5.4-11 July, 1989

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Steam Generator Feedwater Supply	C	14" (4)	In	Open	Closed	Closed	-	Check	NA	10.5-1	
Steam Generator Auxiliary Feed-water Supply	C	6" (4)	In	Open	Int.	If Needed	-	Check	NA	10.5-1	6
Steam Generator Chemical Feed Supply	C	1/2" (4)	In	Closed	Int.	Closed	-	Check	NA	10.5-1	6
Non Essential Service Water to Containment Ventilation Units	A	6" (4) 3" (4)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Containment Ventilation Units	A	6" (4) 3" (4)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Purge Air Inlet (Containment)	A	30" (1) 24" (1)	In	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	
Purge Air Outlet (Containment)	A	30" (1) 24" (1)	Out	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	

6) No independent containment penetrations. These lines join the Feedwater Lines between the penetrations and the isolation valves.

5.4-12 July, 1989

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Fuel Transfer Tube	A	20" (1)	In or Out	Closed	Open	Closed	Blind Flange	-	NA	-	7
Service Air	A	2" (1)	In	Closed	Open	Closed	Check	Auto Trip	A	9.8-3	
Instrument Air	A	1" (2)	In	Open	Open	Closed	-	2 Auto Trip	A	9.8-3	
Reactor Coolant Pump Thermal Barrier Cooling Water Discharge	B	4" (1)	Out	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	
Gas Analyzer From Reactor Coolant Drain Tank	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	11.1-1	
Ice Loading Line	E	5" (1)	In	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	5.3-2A	
Containment Pressure Relief Line	A	12" (1)	Out	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	
Containment Test Pressurization	E	5" (1)	In	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	9.8-3	8

7) See Sub-Chapter 5.2 for description of double gasketed seal on the Fuel Transfer Tube.
8) Same physical line as ice loading line.

5.4-13 July, 1989

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Ice Loading Return	E	5" (1)	Out	Closed	Int.	Closed	Blind Flange	Blind Flange	NA	5.3-2A	
Glycol to Ice Condenser Fan Coolers	E	3" (1)	In	Open	Open	Closed	Auto Trip	Auto Trip	A	5.3-2A	
Glycol from Ice Condenser Fan Coolers	E	3" (1)	Out	Open	Open	Closed	Auto Trip	Auto Trip	A	5.3-2A	
Bypass Glycol line to Ice Condenser Fan Coolers	E	3/8" (1)	In	Open	Open	Closed	Check	-	NA	5.3-2A	
Bypass Glycol line from Ice Condenser Fan Coolers	E	3/8" (1)	Out	Open	Open	Closed	Check	-	NA	5.3-2A	
Purge Air Inlet (Instrumentation Room)	A	14" (1)	In	Closed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	8a
Purge Air Outlet (Instrumentation Room)	A	14" (1)	Out	Closed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	8a
Reactor Coolant Drain Tank & Press. Relief Tank Vents	B	1" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	11.1-1	

8a) For status "N": "Closed" for Unit 2; Unit 1 is "If needed" (for limited purging).

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves			Isolation Valves		Isolation Actuation Signal	Figure Number	Notes
				N	S	I	Inside	Outside			
Refueling Water Supply	A	2 1/2" (1)	In	Closed	Int.	Closed	-	2 Manual (L.C.)	NA	9.4-1	
Demineralized Water Supply	A	2" (1)	In	Closed	Open	Closed	-	2 Auto Trip	A	-	
Non Essential Service Water to Reactor Coolant Pump Motor Air Coolers	A	3" (4)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Reactor Coolant Pump Motor Air Coolers	A	3" (4)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Reactor Support Cooling Inlet	C	2 1/2" (1)	In	Open	If Needed	Closed	Check	Auto Trip	A	9.5-1	
Reactor Support Cooling Outlet	C	2 1/2" (1)	Out	Open	If Needed	Closed	-	2 Auto Trip	A	9.5-1	
Refueling Cavity Drain To Purification System	A	3" (1)	Out	Closed	If Needed	Closed	-	2 Manual (L.C.)	NA	11.1-1	
Nitrogen Supply to Reactor Coolant Drain Tank	A	1" (1)	In	Open	Open	Closed	-	Auto Trip & Check	A	11.1-1	9

9) No independent containment penetration. Joins RCDT vent line between penetration and isolation valves.

5.4-15 July 1990

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Steam Generator Blowdown Samples	C	1/2" (4)	Out	Int.	Closed	Closed	-	Auto Trip	A	9.6-1	
Containment Weld Channel Pressurization Air Supply	D	1/2" (2)	In	Closed	If Needed	Open	-	Check	NA	5.6-1	10
Dead Weight Test Connection	E	1/2" (1)	-	Int.	Closed	Closed	-	Manual	NA	4.2-1A	
Relief Vent Header	B	4" (1)	In	Int.	Int.	Int.	Check	-	NA	4.2-1A	
Ice Condenser and Containment Ventilation Unit Drain to Drain Header	A	3" (1) 1" (1)	Out	Open	Open	Closed	-	2 Auto Trip (Each Line)	A	11.1-1	
Component Cooling Water to Main Steam Penetrations	D	1" (4)	In	Open	Open	Open	Check	Manual	NA	-	
Component Cooling Water from Main Steam Penetrations	D	1 1/2" (2)	Out	Open	Open	Open	-	Remote Manual	None	-	
Component Cooling Water to Pressure Equalizing Fans	D	1 1/2" (2)	In	Closed	Closed	Open	-	Remote Manual	None	-	

10) May be used for Leak Test of Channels.

5.4-16
July 1990

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Component Cooling Water from Pressure Equalizing Fans	D	1 1/2" (2)	Out	Closed	Closed	Open	-	Remote Manual	None	-	
Containment Air Particulate and Radio Gas Detector Sample Line	B	1" (2)	Out	Open	Open	Int.	-	2 Auto Trip	B	-	11
Containment Air Particulate and Radio Gas Detector Sample Return	B	1" (1)	In	Open	Open	Int.	Check	Auto Trip	B	-	11
Lower Containment Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	
Upper Containment Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	-
Instrument Room Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	-

11) May be put in service manually after incident

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TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Non Essential Service Water to Instrument Room Ventilation Units	A	2 1/2" (2)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Instrument Room Ventilation Units	A	2 1/2" (2)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Sample Lines to Hydrogen Monitoring System	D	1/2" (9)	Out	Closed	Closed	Int.	-	2 Auto Trip	A	14.3.6-12A	11
Sample Line Return From Hydrogen Monitoring System	D	1/2" (1)	In	Closed	Closed	Int.	-	Auto Trip	A	14.3.6-12A	11
Containment Pressure Transmitters	E	1/2" (6)	-	Open	Open	Open	-	Manual	NA	-	12
Containment Sump Sample to Post-Accident Sampling System	D	1/2" (1)	Out	Closed	Closed	Int.	-	Auto Trip	A	9.6-2	11
Post Accident Sampling System Return	D	1/2" (1)	In	Closed	Closed	Int.	Check	Auto Trip	A	9.6-2	11
Post Accident Sampling System Supply (Gas)	D	1/2" (1)	Out	Closed	Closed	Int.	-	2 Auto Trip	A	9.6-2	13

11) May be put in service manually after incident
12) See Fig. 7.5-1 for a functional diagram of these instruments.
13) Connected to Containment Air Particulate and Radio Gas Detector Sample Line

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Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Reactor Vessel Level Instrumentation System	E	3/16" (6)	-	Open	Open	Open	-	Membrane Barrier	NA	-	
Incore Flux Detection System	NA	8" (1)	-	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	-	13
Spare Penetrations	NA	18" (5) 6" (4)	-	Closed	Closed	Closed	Weld Cap	Weld Cap	NA	-	

13) Used for replacement of incore flux instrumentation thimbles.

N: Normal
S: shutdown
1: Incident

Int: Intermittent
L.C.: Locked Closed
NA: Not Applicable

Isolation Actuation Signals:
A: Phase A Isolation
B: Phase B Isolation
CVI: Containment Ventilation Isolation (initiated by Safety Injection Signal or High Containment Radiation)

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CONTAINMENT ISOLATION SYSTEM

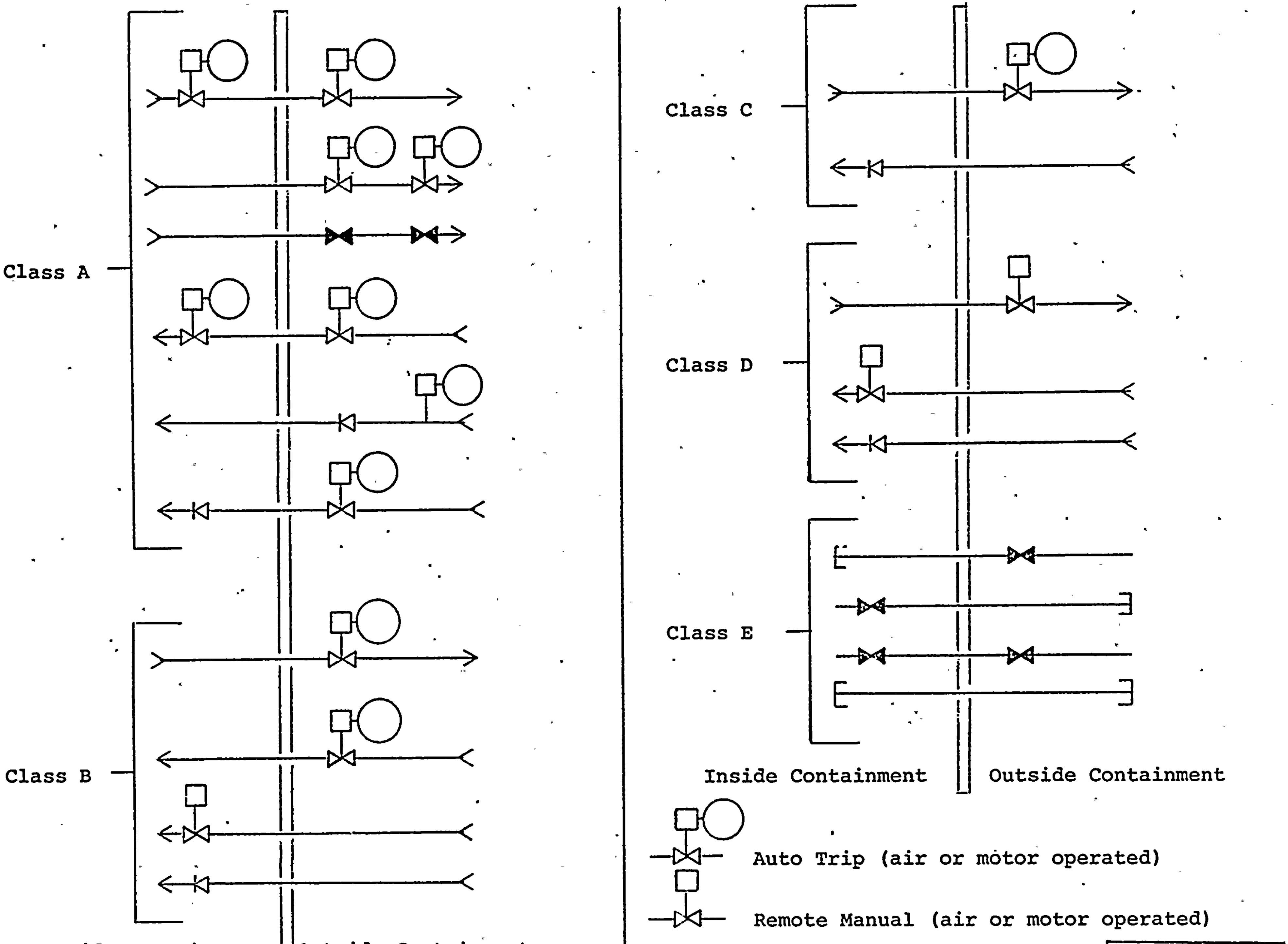

FIGURE 5.4-1
July, 1982

Upper Compartment Ventilation System

The Upper Compartment Ventilation System consists of four free standing recirculating ventilating units (3 for normal operation, 1 standby). Each unit includes a 25,000 CFM fan, water cooling coils and electric blast coil heaters.

The water for the cooling coils is supplied by the Non-Essential Service Water System. Any three of the four units have sufficient cooling capacity to maintain the temperature below 100°F during design summer conditions. Water flow to the cooling coils is regulated by modulating air-operated valves located outside the containment. These valves are controlled by proportional thermostats located on the ventilation unit intakes. Maximum water flow is 80 gpm per unit.

Normally, three ventilation units operate continuously. Cooling is performed whenever the intake air temperature exceeds 90°F. The electric blast coil heaters are energized whenever the intake air temperature drops below 75°F.

Lower Compartment Ventilation System

The Lower Compartment Ventilation System is the largest of the Containment Ventilation Systems. It consists of four recirculation ventilation units composed of fans and water cooling coils, four booster fans for Control Rod Drive Mechanism ventilation, vent fans for reactor and pressurizer enclosure ventilation and associated duct work.

The four recirculation ventilation units are located in the annular space around the periphery of the lower chamber between the crane wall and the containment liner. Each unit is composed of water cooling coils and two 36,000 CFM fans. The intake to these units is connected via a duct penetration through the crane wall to air intakes from the top of the four steam generator enclosures, the Reactor Coolant Pump Motor areas and the discharges from the control rod drive mechanism vent fans. Air is drawn from the above stated heat sources, passed through the water cooling coils and discharged into the annular space. The cooled air re-enters the lower chamber via openings in the crane wall and through the pipe tunnel below the annular space which also has openings in the crane wall into the lower chamber.

The four recirculation units are split into pairs; two units in each of the two fan rooms. Normally, both fans of one unit and one of the fans of the second unit in a given room limit the average containment air temperature to 110°F. The water to the cooling coils is fed by the non-essential service water system. Water flow to each unit is modulated by an air-operated valve outside the containment which is controlled by a proportional thermostat in the recirculation unit intake. Nominal water flow per unit is 440 gpm.

There are four 20,000 CFM fans (1 standby) which draw air through the control rod drive mechanism shroud and discharge it into the intake ducts of the four lower compartment recirculation units. The four fans are located outside the primary shield of the reactor vessel and are all connected via a common intake header to the control rod drive mechanism ventilation shroud. There are redundant temperature sensors in the intake header which actuate an alarm in the control room in the event that the air temperature leaving the shroud exceeds the setpoint.

Two 3000 cfm booster fans draw air from the pipe tunnel and discharge it into the lower reactor cavity. This operation ensures a continuous flow of cool air at the base of the reactor vessel. Two 12,000 cfm fans (1 standby) draw air from the top of the pressurizer enclosure and discharge into the suction side of the lower containment ventilation system. This operation prevents heat buildup at the top of the enclosure. (The steam generator enclosures are ventilated by ducts which are also directly connected into the suction side of the lower containment ventilation system.)

Containment Instrumentation Room Ventilation System

The In-Core Instrumentation Room is an isolated sector of the lower compartment. The temperatures in the room are controlled by two free-standing, 9,600 cfm recirculation ventilation units (1 standby). Each unit is composed of a fan, water cooling coil and electric blast coil heaters. The water for coils is supplied by the Non-Essential Service Water System. Water flow is regulated in the same manner as for the upper compartment ventilation units. Maximum water flow per unit is 50 gpm. The Instrumentation Room is kept at a constant temperature of approximately 90°F during plant operation.

Containment Auxiliary Charcoal Filter System

This system consists of two 8000 cfm fan-filter units located in the lower containment compartment. Each unit contains both absolute particulate and charcoal filters, for reduction of fission product particulate activity which may be air-borne in the lower compartment.

The containment atmosphere is monitored for radioactivity during reactor power operation, and the number of auxiliary charcoal filter units in operation (none, 1 or 2) depends on the air-borne activity levels observed.

Containment Air Recirculation/Hydrogen Skimmer System

The Containment Air Recirculation/Hydrogen Skimmer System is the only safety related ventilation system within the containment. This system functions only in the event of a hi-hi containment pressure signal. It consists of two redundant independent systems which include fans, back draft dampers, valves, piping and ductwork.

Both Containment Air Recirculation Hydrogen Skimmer System Fans are located in the upper volume. The fans discharge, via the annular space between the crane wall and the Containment liner, into the lower compartment. The fans are provided with back draft dampers on the discharge to prevent backflow during initial blowdown.

Figure 5.5-2 shows the various components of this system and Figure 5.5-3 shows the recirculation flow patterns that are created by this system. The system includes provisions for providing both 1) general recirculation of containment atmosphere between the upper and lower compartments following a loss-of-coolant accident, and 2) preventing the improbable accumulation of hydrogen in restricted areas within the containment following a loss-of-coolant accident.

The potential areas of hydrogen pocketing are the top of the containment dome, and the lower compartment enclosures which include the three rooms in the annular space between the crane wall and the liner, the steam generator enclosures, and the pressurizer enclosure. Hydrogen pocketing is prevented by continuously drawing air out of the top of each of the above areas at such a rate as to limit the potential local hydrogen concentration to less than 4% by volume.

Each of the two independent systems fan has its own intake system composed of three separate headers. These headers draw 39,000 CFM from the upper compartment in the immediate vicinity of the fan, draw 1,000 CFM from the upper compartment at the top of the dome, and draw air from the potential hydrogen pockets in the lower compartment (this is the hydrogen skimmer header). Each header has volume control dampers in the line or at the air intake to balance flow. The hydrogen skimmer header is composed of two pipe branches, one which draws 500 CFM from the top of each double steam generator enclosure and pressurizer enclosure and one which draws 100 CFM from each of three rooms in the annular space. There is a normally closed, motor-operated hydrogen skimmer valve on each main hydrogen skimmer header to prevent ice condenser bypass during initial blowdown.

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 a 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.

5.7.2 INITIAL CONTAINMENT (PRE-OPERATIONAL) LEAKAGE RATE TESTS

Integrated Leakage Rate Tests

After completion of the containment and after loading the ice condenser, an integrated leakage rate test was carried out using a test procedure which was written using the American National Standard - ANSI N45.4-1972 and 10 CFR 50, Appendix J as guidelines.

The integrated leakage rate tests were conducted with the weld channel zones open to the containment atmosphere. The containment was pressurized to 12 psig, the containment design pressure, using air dried to a dew point below the coldest temperature in the ice condenser to eliminate the possibility of condensing water vapor during the test.

The design leakage rate under accident conditions is 0.25% of the containment free volume per 24 hours.

Sensitive Leakage Rate Tests

The sensitive leakage rate tests are performed using testing procedures written for testing liner weld channels and penetrations using 10 CFR 50 Appendix J as a guide.

Since the volumes contained in the weld channels and penetrations are significantly smaller than the containment free volume, the test sensitivity is correspondingly greater than that of an integrated leakage rate test. These tests are conducted with 12 psig in the weld channels and penetrations and with the containment at atmospheric pressure.

5.7.3 CONTAINMENT PERIODIC (POST-OPERATIONAL) LEAKAGE RATE TEST

There is a small combined volume of enclosed space in the double barrier penetration, the penetration weld seam channels and the liner weld channels installed on the inside of the liner in the containment. Since it is easy to monitor these small volumes, a sensitive and accurate means of periodically monitoring their status with respect to leakage is provided.

With this provision, there is no need to perform integrated leak rate tests of the containment vessel unless major maintenance or modifications of the containment are made. To allow for this possibility, it is permissible to pressurize the containment vessel to the design pressure.

Observations of the vessel will be made from platforms or by other means with special attention given to areas of major discontinuities.

Provisions have been made in the design of the Ice Condenser structure to permit periodic inspection of the containment liner in the area behind the ice condenser. Inspection of the liner is accomplished through "Inspection Ports" located around the ice condenser, to permit access to the liner.

Periodic leak testing of the containment is performed in accordance with the Technical Specifications. The leak rate test is done to determine the leak tightness of the containment vessel and not to measure the structural response of the containment. The leak rate test is performed with the ice in place and at the design pressure of 12 psig.

The containment recirculation sump is protected at entry by coarse and fine screens supported within a substantial frame. Water flowing into the sump passes through the coarse and fine screens and downwards under the crane wall. The flow is then turned upwards and enters the twin recirculation pipes connecting the sump to the RHR and containment spray pumps. The two sets of grating act as flow straighteners and mitigate vortex formation by equalizing local velocity differences.

The sump is designed with a large flow area, allowing low water velocities, such that build-up of debris against the screens is minimized. The low velocities make it unlikely that air bubbles could be carried into the pump suction area of the sump. Each recirculation line from the sump is run outside the containment to a sump isolation valve. This valve is surrounded with a leak tight steel enclosure and the section of piping joining it to the sump is run within a guard pipe welded to the valve enclosure. Any leakage from the sump piping or valve body will be contained and cannot leak into the atmosphere or cause a loss of recirculation fluid. The pressure relief for each valve enclosure is routed to the associated residual heat removal pump room sump. The relief valve set point is 35 psig which is also the design pressure for the valve enclosure. The drain lines from the enclosures to the RHR pump room sumps are normally closed. The enclosures are ASME Section III Class B vessels which require pressure relief provision.

The sump isolation valves are interlocked with the RHR pump suction supply valves from the RWST so that the supply line(s) from the sump cannot be opened until the RHR pump suction valve(s) is (are) fully closed. These interlocks are train oriented and will prevent air from getting into the RHR pump suction. Any excessive leakage or passive failure downstream of the sump valves can be controlled and isolated by closure of the sump valve in the affected train.

Within the containment, continuity of the liner is assured by welding of the sump discharge piping to the liner plate and fitting of a weld test channel over the seal weld. The liner extends under the sump area to ensure containment integrity (see Chapter 5).

Change-Over from Injection Phase to Recirculation Phase

The general sequence, from the time of the safety injection signal, for the changeover from the injection to the recirculation is as follows:

a) First, sufficient water is delivered to the containment to provide adequate net positive suction head (NPSH) for the residual heat removal pumps.

b) Second, the low level alarm on the Refueling Water Storage Tank sounds. At this point, the operator initiates transfer to recirculation by changing the suction of one ECCS train to the recirculation sump. The other train continues to take suction from the Refueling Water Storage Tank.

c) Finally the lo-lo level alarm on the refueling water storage tank sounds. At this time, the operator completes the switch-over operation by transfering the suction of the other ECCS train to the recirculation sump.

The detailed sequence for the changeover from injection to recirculation is given in an emergency operating procedure.

The change-over from injection to recirculation is effected by the operator in the control room via a series of manual switching operations. In order to protect the residual heat removal pumps from cavitation during switchover from injection to recirculation, an automatic pump trip will occur once the refueling water storage tank (RWST) reaches lo-lo level. The power supply for each pump trip is from an independent power source, and the pump trip and associated circuitry are designed to be consistent with the remainder of the plant engineered safety features. Following a trip on lo-lo RWST level, the pump can be restarted by operator action once the RWST suction has been isolated and the recirculation sump suction opened. This automatic trip feature is a back-up to the manual switchover.

Following an accident the shortest time when the operator must take action to perform the necessary switchover is when both trains of ECCS and spray pumps are in operation at full runout conditions. This case empties the RWST at the fastest possible rate, thus requiring the most rapid operator action to perform the switchover from injection to recirculation.

The limiting times required to switchover the first train of pumps are shown on Figure 6.2-4 and the associated RWST level above the lo-lo level alarm setpoint is shown on Figure 6.2-5. Once a single train is switched over and operating, an adequate water supply is assured to both the reactor core and the containment sprays. The switchover of the second train supplies redundant capability. The switchover procedure requires stopping only the RHR pumps of the emergency core cooling system, the other pumps run continuously while the RHR pumps are shutdown. At no time in the switchover procedure are all pumps shutdown. Therefore, core uncovery is not expected to occur.

The operations used to switchover from injection mode to recirculation mode and their appromixate times in seconds are listed below: (Refer to Figures 6.2-1 and 6.2-1A.)

	Step	Valve Stroke Time (sec)	Step Time (sec)	Total Time (sec)
1.	Reset the automatic safety injection (SI) signals.	----	30	30
2.	Stop the W residual heat removal (RHR) and W containment spray (CTS) pumps.	----	12	42
3.	Close the W RHR pump and the W CTS pump suction valves from refueling water storage tank.	105	117	159
4.	Close the SI pump minimum flow valves to the refueling water storage tank.	10	8	167

5.	Open W recirculation sump isolation valve.	40	40	207
6.	Start the W RHR and W CTS pumps.	---	12	219
7.	Open SI pump suction valve from W RHR heat exchanger	15	9	228
8.	Open SI pump suction crosstie to centrifugal charging pump suction valve.	5	4	232

Steam Break Protection

Following a steam line break, the reactor control system, in response to the apparent load, would tend to increase reactor power. For larger breaks, a reactor trip would occur. Continued secondary steam blowdown cools the reactor coolant causing a positive reactivity insertion. Analyses described in Chapter 14 indicate that breaks large enough to produce a reactivity insertion sufficient to cause a return to criticality also produce sufficient depressurization and shrinkage of the primary coolant to initiate safety injection. The high pressure delivery of concentrated boric acid by the centrifugal charging pumps then re-establishes adequate shutdown margin even for the case where the most reactive control rod is stuck in the fully withdrawn position.

Components

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is isolated from the Reactor Coolant System by two check valves in series.

NOTE: Step times may in some cases (steps 4, 7 and 8) take less then the associated valve stroke time since the operator proceeds to the next step without waiting for valve movement to stop. See Figures 6.2-4 and 6.2-5.

The tank size has a minimum of 350,000 gallons of usable capacity plus sufficient reserve volume to insure that a net positive suction head is maintained for the proper operation of the safeguard pumps after the 350,000 gallons have been withdrawn from the tank. The gross capacity of the tank is 420,000 gallons.

A high level alarm is provided to alert the operator of potential overflow conditions. A minimum level alarm is provided to assure that 350,000 gallons of usable water are in the RWST.

The Unit No. 1 refueling water storage tank is heated by means of two 100% capacity heat tracing circuits with separate thermostatic controls. The tank is insulated with 2 inch thick fiberglass insulation. A temperature sensor attached to the outside of the tank will actuate a low temperature alarm in the control room in the event that the tank temperature falls below the design basis temperature requirement. The setpoint of the alarm is 85°F.

The Unit No. 2 refueling water storage tank is heated by means of a 15 gpm pump which recirculates tank water through two electric heaters. The RWST heating pump operates continuously, when required, with the heaters energizing automatically on a low RWST temperature signal. The system is seismic category I with respect to protection of the tank boundary and is designed to maintain RWST temperature at design basis conditions. The Unit 2 RWST is insulated with 2 inch thick fiberglass insulation, and has a temperature sensor and alarm similar to that of the Unit 1 tank.

Each tank is equipped with an 8 inch vent and a 10-inch overflow line. The overflow lines terminate in the pipe tunnel. Should the 8-inch vent become plugged the 10 inch overflow line would maintain sufficient venting area to prevent any adverse effect on the safety function of the tank.

Pumps

Design parameters for the emergency core cooling system pumps are included in Table 6.2-5.

The two centrifugal charging pumps are horizontal, electric motor driven multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the volume control tank or the pump suction manifold. This bypass is automatically isolated upon initiation of safety injection. The minimum flow, motor-operated valve reopens if the reactor coolant system pressure increases above 2000 psig to protect the pumps from deadheading.

The two safety injection pumps are horizontal, electric, motor-driven, multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event that the reactor coolant system pressure is above the shutoff head of the pumps.

The two residual heat removal pumps are vertical, electric, motor-driven, single-stage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or of equivalent corrosion resistant material. Pump minimum flow bypass connection is located downstream of the residual heat exchanger and the bypass flow returns to the pump suction.

The pressure containing parts of the pumps are stainless steel castings conforming to ASTM A-351 Grade CF8 or CF8M, stainless steel castings procured per ASTM A-296 Grade CA-15 or A-487 Grade CA6NM or carbon steel forgings to ASTM A-266 Class 1 and ASTM A-181 Grade 1 clad with austenitic steel or ASME SA-182 Grade F304. Parts fabricated of stainless plate are constructed to ASTM A-240 Type 304 or 316. The bolting material conforms to ASTM A-193 or ASTM A-453 Grade 660.

TABLE 6.3-3

SPRAY ADDITIVE TANK DESIGN PARAMETERS

Quantity	1 (per unit)
Volume, gal (usable)	4000
NaOH concentration, % by weight	30
Design temperature, ^{o}F	200
Design pressure, psig	10
Material	stainless steel

SPRAY ADDITIVE TANK CODE REQUIREMENTS

ASME 1968 B&PV Section VIII Div. 1

TABLE 6.3-4

CONTAINMENT SPRAY SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments and Consequences
1. Containment Spray Pump	Rupture of Pump casing	Isolate train. Redundant train continues to operate, requirement is one train.
2. Containment Spray Pump	Pump fails to start.	One of two pumps operating will supply 100 percent of required flow.
3. Containment Spray Pump	Pump suction line closed	This is prevented by pre-startup checks. During power operation, each pump is tested on a periodic basis. During these tests checks will be made to confirm that a motor-operated valve (from the Refueling Water Storage Tank) is closed. The manual valve from the Recirculation Sump is locked open. Motor-operated valve positions (open or closed) are indicated in the Control Room.
4. Containment Spray Pump	Pump discharge motor-operated valve fails to open.	Motor-operated valves are redundant and only one of the two need operate. Valve positions (open or closed) are indicated in the Control Room.

The logic test scheme uses pulse techniques to check the coincidence logic. All possible trip and non-trip combinations are checked. Pulses from the tester are applied to the inputs of the universal logic card at the same terminals that connect to the input relay contacts. This connection provides the overlap between the input relay check and the logic matrix check. Pulses are fed back from the reactor trip breaker undervoltage coil to the tester. The pulses are of such short duration that the reactor trip breaker undervoltage coil armature cannot respond mechanically.

Test indications that are provided are an annunciator in the control room indicating that reactor trips from the train have been blocked and that the train is being tested, and green and red lamps on the semi-automatic tester to indicate a good or bad logic matrix test. Protection capability provided during this portion of the test is from the train not being tested.

Reactor Trip Breaker Testing

Normally, reactor trip breakers 52/RTA and 52/RTB are in-service, and bypass breakers 52/BYA and 52/BYB are withdrawn (out of service). In testing the protection logic, pulse techniques are used to avoid tripping the reactor trip breakers. The reactor trip bypass breaker is racked in and closed to ensure a reactor trip will not occur. The following procedure describes the method used for testing the trip breakers:

a. With bypass breaker 52/BYA racked out, manually close and trip it to verify its operation.

b. Rack in and close 52/BYA. Manually trip 52/RTA through a Protection System Logic Matrix.

c. Reset 52/RTA.

d. Trip and rack out 52/BYA.

e. Repeat above steps to test trip breaker 52/RTB using bypass breaker 52/BYB.

An annunciator is provided in the control room to indicate when a breaker is bypassed.

Auxiliary contacts of the bypass breakers are connected into the alarm system of their respective trains such that if either train is placed in test while the bypass breaker of the other train is closed, both reactor trip breakers and both bypass breakers will automatically trip.

The train A and train B alarm systems operate separate annuciators in the control room. The two bypass breakers also operate an annunciator in the control room. The bypassing of a protection train with either the bypass breaker or with the test switch will result in audible and visual indications.

Response Time Testing

Tests that provide assurance that response times for various reactor trip parameters are within acceptable limits can be performed during shutdown. System design does not permit such testing during normal operation.

Primary Power Source

The primary power sources for the reactor protection system are described in Chapter 8. The source of electrical power for the measuring elements and the actuation of circuits in the engineered safety features instru- mentation is also from these buses.

Protective Actions

Reactor Trip Description

Rapid reactivity shutdown is provided by the insertion of full length rods by free fall. Duplicate series-connected circuit breakers supply all power to the full length control rod drive mechanisms. The full length rods must be energized to remain withdrawn from the core.

Automatic reactor trip occurs upon loss of power to the full length control rods. The trip breakers are opened by de-energizing the undervoltage trip coils of both breakers. A contact of an auxiliary relay, connected in parallel with the undervoltage coils, activates the shunt trip coils at the same time to provide a redundant trip actuation. The undervoltage coils and auxiliary relays, which are normally energized, become de-energized by any one of the several trip signals.

The functional diagrams for reactor protection and control may be found in Westinghouse Proprietary Class 2 Drawing No. 5654D39, (Sheets 1-16).

The analog block diagrams for the above named circuits may be found in Westinghouse Proprietary Class 2, Process Control Block Diagram Drawing No. 108D087, Sheets 1 through 35. An non-proprietary synopsis of the application of circuits are given within this text under the name of the tripping action. Table 7.2-5 is the index to these Block Diagrams (108D087).

Manual Trip

The manual actuating devices are independent of the automatic trip circuitry, and are not subject to failures which make the automatic circuity inoperable. Actuating either of two manual trip switches located in the control room initiates a reactor trip and a turbine trip.

High Neutron Flux (Power Range) Trip

This circuit trips the reactor when two out of the four power range channels read above the trip set-point. There are two independent trip ranges, a high and a low range set-point. The high range trip provides protection during normal power operation. The low range trip, which provides protection during start-up, can be manually bypassed when two out of the four power range channels read above P-10. (See Table 7.2-2 for a definition of P's and C's.) Three-out-of-the-four channels below this value automatically re-arms the trip function. The high setting is always active.

High Neutron Flux Rate (Power Range) Trip

This circuit trips the reactor when an abnormal rate of increase in nuclear power occurs in two out of four power range channels. This trip provides protection against rod ejection accidents of low worth from mid-power and is always active.

Negative Neutron Flux Rate (Power Range) Trip

This circuit trips the reactor when an abnormal rate of decrease in nuclear power occurs in two out of four power range channels. This trip provides protection against a dropped rod bank and is always active.

High Neutron Flux (Intermediate Range) Trip

This circuit trips the reactor when one out of the two intermediate range channels reads above the trip set-point. This trip, which provides protection during reactor start-up, can be manually bypassed if two out of four power range channels are above P-10. Three-out-of-four channels below this value automatically re-arms the trip function. The intermediate range channels (including detectors) are separate from the power range channels.

Initiation of automatic turbine load runback by means of an overpower ΔT signal is discussed below.

Low Pressurizer Pressure Trip

The purpose of this trip is to protect against excessive core steam voids and to limit the necessary range of protection afforded by the overtemperature ΔT trip. This trips the reactor on coincidence of two out of the four low pressurizer pressure signals. This trip is blocked when three of the four power range channels and two of two turbine first stage pressure channels read approximately 10 percent power (P-7). Each channel is lead-lag compensated.

High Pressurizer Pressure Trip

The purpose of this trip is to limit the range of required protection from the overtemperature ΔT trip and to protect against Reactor Coolant System overpressure. The reactor is tripped on coincidence of two out of the four high pressurizer pressure signals.

High Pressurizer Water Level Trip

This trip is provided as a backup to the high pressure trip. The coincidence of two out of three high water level signals trips the reactor. This trip is interlocked with permissive P-7 described in Table 7.2-2.

Low Reactor Coolant Flow Trip

This trip protects the core from DNB following a loss of flow. The means of sensing loss of flow are described below:

a. Low Primary Coolant Flow Trip

A low loop flow signal is generated by two out of three low flow signals per loop. Above the P-7 setpoint low flow in any two loops results in a reactor trip. Above the P-8 setpoint low flow in any loop results in a reactor trip.

b. Reactor Coolant Pump Breaker Position Trip

One open breaker signal is generated for each reactor coolant pump. Above the P-7 setpoint the reactor trips on two open breaker signals. Above the P-8 setpoint the reactor trips on one open breaker signal.

c. Reactor Coolant Pump Undervoltage and Underfrequency Trips

There is one underfrequency and one undervoltage sensor per bus. A 2/4 underfrequency signal directly trips all of the reactor coolant pumps, produces a direct reactor trip (interlocked by P-7), and indirectly trips the reactor through the pump breaker position trip. A time delay relay serves as an undervoltage sensor on each of the four busses. An undervoltage condition on 2/4 busses will actuate a reactor trip above P-7. A time delay is provided to prevent short duration voltage transients from causing unnecessary trips.

All of these low reactor coolant flow trips are blocked below the P-7 setpoint.

Safety Injection System (SIS) Actuation Trip

A reactor trip occurs when the safety injection system is actuated. The means of actuating the SIS trips are:

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LIST OF REACTOR TRIPS AND ACTUATION MEANS OF: ENGINEERED SAFETY FEATURES, CONTAINMENT AND STEAM LINE ISOLATION & AUXILIARY FEEDWATER

Reactor Trip	Coincidence Circuitry and Interlocks	Comments
1. Manual	1/2, no interlocks	
2. Neutron flux (Power Range):		
2A. High neutron flux	2/4, low setpoint interlocked with P-10	High and low settings; manual block and automatic reset of low setting by P-10, Table 7.2-2
2B. High neutron flux rate	2/4, no interlocks	
2C. Negative neutron flux rate	2/4, no interlocks	
3. Overtemperature T	2/4, no interlocks	
4. Overpower T	2/4, no interlocks	
5. Low pressurizer pressure	2/4, interlocked with P-7	
6. High pressurizer pressure	2/4, no interlocks	
7. High pressurizer water level	2/3, interlocked with P-7	
8. Low reactor coolant flow	2/3 signals per loop, interlocked with P-7 and P-8	Blocked below P-7. Low flow in 1 loop permitted below P-8.

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Reactor Trip	Coincidence Circuitry and Interlocks	Comments
9. Monitored electrical supply to reactor coolant pumps:		
9A. Undervoltage	2/4, interlocked with P-7	
9B. Underfrequency	2/4, interlocked with P-7	2/4 underfrequency signals will trip all reactor coolant pumps and directly actuate reactor trip: interlock with P-7.
9C. Reactor coolant pump Breaker position	Interlocked with P-7 and P-8	Blocked below P-7. Open breaker in 1 loop permitted below P-8.
10. Safety injection signal	Manual 1/2 panel switches (per train) 2/3 low pressurizer pressure 2/3 high containment pressure 2/3 differential steam line pressure signals of one line compared with the other three lines 1/2 high steam line flow signals in 2/4 steam lines coincident with 2/4 low-low Tavg or low steam line pressure (Unit 1) 2/4 low steam line pressure alone (Unit 2)	Trips main feedwater pumps. Closes feedwater control valves. Closes feedwater isolation valves. Closes pump discharge valves and initiates Phase A isolation. Actuation by pressurizer pressure may be manually blocked below P-11 and is automatically unblocked above P-11.
11. Turbine-generator trip	2/3 low auto stop oil pressure interlocked with P-7 or all stop valves closed or high steam generator water level.	

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TABLE 7.2-2 (cont'd.)

Designation	Derivation	Function
P-7 Reset	2/4 Power range neutron flux channels above setpoint (from P-10) or 1/2 turbine first stage pressure above setpoint (from P 13)	Prevents or defeats the reactor trip when any of the following conditions are sensed: - Turbine trip - Greater than one loop reactor coolant flow low - Greater than one reactor coolant pump breaker open - Reactor coolant pump bus undervoltage or underfrequency - Pressurizer low pressure - Pressurizer high level
P-8	2/4 power range channels above setpoint	Prevents or defeats the automatic block of reactor trip caused by either a low coolant flow condition in a single loop or a reactor coolant pump breaker trip on a single loop.
P-8 Reset	2/4 NIS power range channels below setpoint.	Permits the automatic block of reactor trip on low flow in a single loop.

TABLE 7.2-2 (cont'd.)

Designation	Derivation	Function
P-10	2/4 power range neutron flux channels above setpoint	Inputs to P-7 permissive. Permits the manual block of reactor trip on: - Intermediate range high neutron flux level. - Power range channel low setpoint high neutron flux level. Permits manual block of intermediate range channel rod stop. Permits automatic block of source range channel trip.
P-10 Reset	3/4 power range channels below setpoint	Prevents or defeats the manual block of reactor trip on: - Intermediate range channel high neutron flux level. - Power range channel low setpoint high neutron flux level. Prevents or defeats the manual block of intermediate range channel rod stop.
P-11	2/3 pressurizer pressure below setpoint	Permits manual block of safety injection actuation on low pressurizer pressure.
P-11 Reset	2/3 pressurizer pressure above setpoint	Prevents or defeats manual block of safety injection actuation on low pressurizer pressure.

heaters, which are used to control small pressure variations due to heat losses, including those due to a small continuous spray in the pressurizer, and backup heaters which are turned on when the pressurizer pressure controller signal is below a given value.

A spray nozzle is located in the upper portion of the pressurizer cavity. Spray is initiated when the pressure controller signal is above a given set point, and spray rate increases proportionally with increasing pressure. Steam is condensed by the spray which will return the pressurizer pressure to its Program Value. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and to help maintain uniform water chemistry and temperature in the pressurizer.

Three pressurizer power relief valves limit system pressure for large load reduction transients.

Three spring-loaded safety valves limit system pressure should a complete loss of load occur without direct reactor trip or steam dump actuation.

Pressurizer Level Control

The water inventory in the Reactor Coolant System is maintained by the Chemical and Volume Control System. During normal plant operation, the pressurizer level is controlled by the charging-flow controller which controls the charging flow control valve or the positive displacement charging-pump speed to produce the flow demanded by the pressurizer-level controller. The pressurizer water level is programmed as a function of coolant average temperature. The pressurizer water level decreases when load is reduced. This is the result of coolant contraction following programmed coolant temperature reduction from full power to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant temperature changes. To permit manual control of pressurizer level during startup and shutdown operations, the charging flow can be manually regulated from the control room.

Secondary System Control

The secondary system includes the steam from the steam generators and the condensate and feedwater systems.

Steam Dump

The steam dump system is designed to relieve steam from the steam generators to the condenser thus reducing the sensible heat in the primary system in the event of net load reduction not exceeding 100 percent.

The steam dump design capacity is 85 percent of full load steam flow at full load steam pressures. All steam dump steam flows to the main condensers via the steam lines.

When a load rejection occurs, if the difference between the required temperature set point of the Reactor Coolant System and the actual average temperature exceeds a predetermined amount, a signal will actuate the steam dump to maintain the Reactor Coolant System temperature within control range until a new equilibrium condition is reached.

The steam dump flow reduces proportionally as the control rods act to reduce the average coolant temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed equilibrium value.

Each demineralizer is sized to accommodate the maximum letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of Li^7 which builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μci/cc with 1% defective fuel. The demineralizer is used intermittently to control cesium. The flow through the demineralizer will be increased for additional cesium removal in the event that steam generator tube leaks are detected.

The demineralizer vessel is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with resin retention screens. The cation bed demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

Resin Fill Tank

The resin fill tank (shared by both units) is mobile and is used to charge fresh resin into the demineralizers. The line from the conical bottom of the tank is fitted with a valve and a flexible hose spool piece that may be connected to any one of the demineralizer fill lines. The demineralizer water and resin slurry can then be sluiced into the demineralizer by opening the valve.

Reactor Coolant Filter

The filter collects resin fines and particulates from the letdown stream. The vessel is provided with connections for draining and venting. The nominal flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements in a cage assembly are used.

Volume Control Tank

The volume control tank is an operating surge volume compensating in part for reactor coolant releases from the Reactor Coolant System as a result of level changes. The volume control tank also acts as a head tank for the charging pumps and reservoir for the leakage from the reactor coolant pump controlled leakage seal. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 35 cc per kg of water (STP).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in the tank.

TABLE 9.2-3 (cont'd.)

<u>Tube Side</u>

Design pressure, psig	600
Design temperature, ^oF	400
Fluid	Borated reactor coolant
Material of construction	Austenitic stainless steel

	Normal	Heatup (Design)	Maximum Purification
Flow, lb/hr	37,050	59,280	59,280
Inlet temperature, ^oF	290	380 (max.)	380 (max.)
Outlet temperature, ^oF	127	127	127

<u>Mixed Bed Demineralizers</u>

Number	2 (per unit)
Type	Flushable
Vessel design pressure:	
Internal, psig	200
External, psig	15
Vessel design temperature, ^oF	250
Resin volume, each, ft^3	30
Vessel volume, each, ft^3	43
Design flow rate, gpm	120
Minimum decontamination factor as measured by I-131 removal	10
Normal operating temperature, ^oF	127
Normal operating pressure, psig	150
Resin type	Cation and anion
Material of construction	Austenitic stainless steel

TABLE 9.2-3 (cont'd.)

<u>Resin Fill Tank</u>	
Number	1 (shared)
Capacity, ft^3	8
Design pressure	Atmospheric
Design temperature, ^oF	200
Normal operating temperature	Ambient
Material of construction	Austenitic stainless steel
<u>Reactor Coolant Filter</u>	
Number	1 (per unit)
Type	Disposable Cartridge Cage assembly
Design pressure, psig	200
Design temperature, ^oF	250
Flow rate,	
Nominal, gpm	120
Maximum, gpm	150
Retention of 25 micron particles	98%
Material of construction (vessel)	Austenitic stainless steel
<u>Volume Control Tank</u>	
Number	1 (per unit)
Internal volume, ft^3	400
Design pressure:	
Internal, psig	75
External, psig	15
Design temperature, ^oF	250
Operating pressure range, psig	0 - 60
Spray nozzle flow (maximum), gpm	120
Material of construction	Austenitic stainless steel

Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system between the reactor coolant and the service water system and insures that any leakage of radioactive fluid from the components being cooled is contained within the plant. The surge tank accommodates expansion and contraction, and insures a continuous component cooling water supply. Because this tank is normally vented to the auxiliary building atmosphere, a radiation monitor is provided in each component cooling heat exchanger discharge line. These monitors actuate an alarm and close the surge tank vent valve when the radiation level reaches a preset level above the normal background.

The Component Cooling System consists of two component cooling pumps, two component cooling heat exchangers, one surge tank and associated piping and valves to serve each unit. One pump and heat exchanger, with associated equipment, forms a 100% train. An additional pump is provided as an installed maintenance spare for either unit and is located in a cross tie header between the Unit 1 and 2 systems. The piping and valve arrangement is such that the maintenance spare can supply water to any one of the four trains, after the electrical controls have been transferred to it from the affected train.

One pump and one heat exchanger are required for the removal of residual and sensible heat from the reactor coolant system via the residual heat removal system during the cooldown of one unit. Full power operation of one unit, including cooling of a spent fuel pit heat exchanger, likewise requires one pump and one heat exchanger. Provision is made to add makeup to the system through lines connected to the surge tank.

The operation of the system is monitored with the following instrumentation:

a) Temperature recorder and alarm in the outlet lines for each of the component cooling heat exchangers

b) A pressure and flow indicator in the supply line to each of the component cooling heat exchangers

c) A radiation monitor in the discharge lines from the component cooling heat exchangers

d) Flow indicators and/or alarms located in the discharge lines of the major heat exchangers served by the system

e) Temperature indicators located in the discharge lines of the major heat exchangers served by the system.

In the event of a loss of coolant accident, one pump and one heat exchanger are capable of fulfilling system requirements. The remaining train therefore serves as a backup, and can be placed in service if required to increase system capability. Cooling water for the component cooling heat exchangers is supplied from the Essential Service Water System (Chapter 9) insuring a continuous source of cooling medium.

9.5.3 COMPONENTS

Component Cooling System component design data are listed in Table 9.5-3.

Component Cooling Heat Exchangers

The component cooling heat exchangers are of the shell and tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. The shell side is of carbon steel and the tubes are of arsenical copper.

Component Cooling Pumps

The component cooling pumps which circulate water through the component cooling loops are horizontal, centrifugal units and motor driven. The motors receive electric power from normal and emergency sources.

Component Cooling Surge Tank

The component cooling surge tank accommodates changes in component cooling water volume and is constructed of carbon steel. In addition to piping connections, the tank is provided with a means of adding a chemical corrosion inhibitor to the component cooling loop. The tank is internally divided (baffled) to form, in effect, two compartments. This arrangement provides redundancy for a passive failure during recirculation phase following a LOCA.

Valves

The valves used in the component cooling loop are constructed of carbon steel. Since the component cooling water is normally not radioactive, special provisions to prevent leakage to the atmosphere are not provided. Relief valves are provided for lines and components that could be pressurized beyond their design pressure by improper operation or malfunction.

The relief valves on the component cooling water lines downstream from each reactor coolant pump thermal barrier are designed to relieve excessive pressure that may be caused by over heating. The relief valve set pressure equals the design pressure of the particular segment of piping between the upstream check valve and downstream motor-operated discharge valves.

The relief valves on the cooling water lines downstream of the sample, excess letdown, seal water, spent fuel pit and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated and high temperature liquid flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The set pressure assures that the design pressure of the component cooling system is not exceeded. The discharge of this valve is directed to the waste holdup tank.

The component cooling water surge tank vent-overflow line, which is open to the auxiliary building atmosphere, is equipped with an air-operated valve that will close automatically if radiation is detected in the system. A vacuum breaker valve is also provided to prevent collapsing this tank in the event of a large loss of water in the system.

Piping

The component cooling loop piping is carbon steel with flanged joints and connections at components which might require removal for maintenance. All other joints are welded. One exception to the carbon steel is that portion of the piping between the double check valves and the motor-operated discharge isolation valves for the Reactor Coolant Pump Thermal Barrier Cooling which is stainless steel.

9.5.4 SYSTEM EVALUATION

Availability and Reliability

The component cooling pumps, heat exchangers, and associated valves, piping and instrumentation are located outside of the containment and are therefore available for maintenance and inspection during power operation. Replacement of a pump, or maintenance on a heat exchanger is practical while redundant units are in service. Sufficient cooling capability is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safeguards equipment.

Incident Control

If outleakage occurs anywhere in the Component Cooling System, including a non-seismic I component served by the Miscellaneous Service Train, detection is accomplished by falling level in the surge tank, which will actuate an alarm in the control room. Level alarms from the sumps to which this water will drain, also serve as leak indicators.

The leaking portion of the system is then shut down and isolated and the backup train is put in operation. To minimize the possibility of leakage from piping, valves, and equipment, welded construction is used wherever possible.

For leakage into the Component Cooling System, a high level alarm is provided at the surge tank.

The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the Chemical and Volume Control, Residual Heat Removal, Sampling or the Spent Fuel Pool Cooling System or from a leak in a cooling coil for the thermal barrier cooler on a reactor coolant pump.

The detection of this contamination is by a radiation monitor located in the Component Cooling Water Heat Exchanger outlet line.

Component Cooling Water flow at a reduced rate is automatically established to the Residual Heat Removal Heat Exchanger at the safety injection signal. Since the thermal demand on this heat exchanger is minimal at this time, full design Component Cooling Water flow is not required. When it has been established that both Component Cooling Water Pumps have been started, full design flow will be established to the Residual Heat Removal Heat Exchangers.

The component cooling water lines to and from the Reactor Support Coolers and the Excess Letdown Heat Exchanger have valves outside the containment wall which are automatically closed on the Phase A isolation signal.

If normal seal water supply is unavailable to the Reactor Coolant Pumps, the cooling water to the RCP thermal barriers should be available to assure that there will be no mechanical damage to the pump. Therefore, isolation valves for the component cooling water for this service are not automatically closed until a Phase B (containment spray) containment isolation signal is received. The cooling water supply line to the reactor coolant pumps contains two remote-operated valves in series outside the containment wall. The return lines from the thermal barriers and RCP motor bearings each have two remote-operated valves in series outside the containment wall. These redundant valves assure the ability to isolate this circuit if a leak

c) The reactor missile shields and the control rod drive mechanism (CRDM) seismic restraint are removed.

d) The bulkhead sections between the reactor cavity and the refueling cavity are removed.

e) CRDM cables and cooling air ducts are disconnected and removed.

f) Reactor vessel head insulation and instrument leads are removed.

g) The reactor vessel head nuts are loosened with the hydraulic tensioner.

h) The reactor vessel head studs are removed.

i) The canal drain holes are plugged and the fuel transfer tube flange is removed.

j) Checkout of the fuel transfer device and manipulator crane is started.

k) Guide studs are installed in three stud holes and the remainder of the stud holes are plugged.

l) The reactor vessel to cavity seal ring is clamped in place.

m) Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed.

n) The reactor vessel head is unseated and raised.

o) The lift of the reactor vessel head is stopped at several specified heights to check that:

- the reactor head is level
- the head is not binding on the guide studs

- the protective sleeves for the instrument port conoseals are not being lifted.

p) At the appropriate reactor vessel head lift height, a check is made that the RCCA drive shafts are clear of the CRDM housings, and are not being lifted with the head. The reactor vessel head is lifted to clear and is taken to its storage pedestal.

q) The reactor cavity and refueling canal are flooded with water to the level required for unlatching the RCCA drive shafts.

r) The control rod drive shafts are unlatched.

s) The reactor vessel internals lifting rig is lowered into position and latched to the support plate.

t) The reactor vessel upper internals are lifted out of the vessel and placed in the underwater storage rack.

u) The core is now ready for refueling.

Refueling

The refueling sequence is now started with the manipulator crane. The general sequence for fuel assemblies in non-control positions is as follows:

a) Spent fuel, which is to be discharged, is removed from the core and placed on the fuel transfer conveyor for removal to the spent fuel pool.

b) Partially spent fuel is relocated within the core.

c) New fuel assemblies are transferred from the new fuel storage area into the refueling canal and are brought through the transfer system and loaded into the core.

d) Whenever fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

The refueling sequence is modified for fuel assemblies containing rod cluster control (RCC) elements. If a transfer of the RCC elements between fuel assemblies is required, the assemblies are taken to the RCC change fixture to exchange the RCC elements from one assembly to another. Such an exchange is required whenever a spent fuel assembly containing RCC elements is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangements. The refueling sequence is modified for burnable poison rod (BPR) assemblies. The assemblies with BPR elements are moved to the spent fuel pool where the BPR element is removed using the burnable poison handling tool, and a thimble plugging device is inserted to restrict the flow through the guide thimbles. Such an operation is necessary whenever a fuel assembly containing a BPR is to be reinserted into the core.

2. Core Unload/Reload Sequence

a. All fuel (spent and partially spent) is removed from the core via the fuel transfer conveyor and place in the spent fuel pool.

b. All required insert (RCC, BPR, TP and source) changeouts occur in the spent fuel pool.

c. Fuel for the forthcoming cycle is transferred from the spent fuel pool via the fuel transfer conveyor to the core.

d. Whenever fuel is added to the reactor core a reciprocal curve of source neutron multiplication is recorded to verify subcriticality of the core.

Reactor Reassembly

The following general sequence of tasks is required following refueling:

a) The fuel transfer car is parked and the fuel transfer tube isolation valve is closed.

b) The reactor vessel internals package is replaced in the vessel. The reactor vessel internals' lifting rig is removed to storage.

c) The control rod drive shafts are relatched to RCC elements.

d) The manipulator crane is parked.

e) The old seal rings are removed from the reactor vessel head, the grooves cleaned and new rings installed.

f) The reactor vessel head is picked up and positioned over the reactor vessel.

g) The water level is lowered and the reactor vessel head is lowered.

h) The refueling cavity and refueling canal are completely drained and the flange surface is manually cleaned.

i) The reactor vessel head is seated.

j) The guide studs are removed to their storage rack. The stud hole plugs are removed.

k) The head studs are replaced and retorqued.

l) The canal drain holes are unplugged and the fuel transfer tube flange is replaced.

Spacer bars at mid-height of the storage cell maintain the cell-to-cell spacing during a seismic event. The storage module has an overall height of 15' -5-3/4".

Method of Transmitting Loads

Horizontal loads on the modules are transmitted to the storage pool structure by friction of the rack module feet on the pool floor, in combination with loads transmitted to the pool walls by wall restraint arms on the north, south and west walls and by a seismic wall brace structure on the east wall. All of the wall supports are located approximately 14 inches off the pool floor.

Vertical loads are transmitted through the rack module feet, which bear on the pool floor. Tipping is prevented by interconnecting adjacent racks with bolted connections at the top of the rack modules.

Criticality Considerations

Criticality calculations were performed using the KENO Monte Carlo code to assure that the center-to-center spacing provides an acceptable value for keff. These calculations show that the combination of the stainless steel sheets and the aluminum clad Boral poison used to form the cell around each fuel assembly results in a k_{eff} less than 0.95 under conditions of worst storage geometry, manufacturing variables, and seismic perturbation. Analyses have been performed for the following fuel types, storage configurations, and corresponding enrichments:

Fuel Type	Description	Storage Configuration	Maximum Nominal Enrichment Wt. % U235
Westinghouse	15x15 STD/OFA	A or B	4.95
Exxon/ANF	15x15	C	3.50
Westinghouse	17x17 STD/OFA/V5	A or B	4.95
Exxon/ANF	17x17	C	4.23

Storage configurations are defined as follows:

A. Region 1 three-out-of-four storage configuration, with one symmetric cell location of each 2x2 cell array vacant. No fuel assembly burnup restrictions are required for this storage configuration. The boundary between this Region 1 configuration and the rest of the spent fuel storage racks (Region 2) shall be such that the three-out-of-four storage pattern shall be carried into the rest of the storage racks by at least one row as shown in Figure 9.7-3.

B. Region 2. Use of all storage cells. A minimum accumulated burnup of 5,550 MWD/MTU is required for all fuel assemblies which have a nominal enrichment exceeding 3.95 w/o.

C. Region 2. Use of all storage cells and no burnup restrictions on fuel assemblies.

<u>Thermal Considerations</u>

A flow path for natural convection cooling of spent fuel assemblies is provided by a large hole in the module base at each fuel storage cell position. Additional cooling between cells is provided by smaller holes in the module base between cells and holes near the top of the side plate diaphragms. Additional area for flow to the bottom of the module is provided by space at the edge of the pool.

In summary, the high density (poison) spent fuel module design provides a significant increase in storage over the original non-poison spent fuel module design. The module is designed to meet all technical requirements for structural integrity, criticality, and cooling.

Pump Suction

a) The high-demand, electric fire pumps' suction source is from their respective units' circulating water discharge tunnel. The low-demand pump suction source is from the Unit 1 discharge tunnel. The diesel-engine-driven fire pump suction source is from the circulating water intake at the screen house. Self cleaning strainers are provided at the pumps' discharge line to remove foreign material from the water. Pressure differential manometers provide control room alarm if a strainer clogs.

b) Water supply for the 50 gpm pegging pump is provided by a connection to the plant non-essential service water system.

Water Distribution

a) The water from the fire pumps is distributed to an outside, buried loop header and an interior loop header in the turbine room basement.

b) The outdoor header consists of 12-inch pipe with 5 1/2 feet of earth cover for freeze protection. Isolating valves with post indicators or curb boxes are installed in this header so that the entire loop is not disabled should maintenance be required on a small section.

c) Fire hydrants are installed at regular intervals on the outdoor fire header. Each hydrant has its own buried 6-inch control valve, two 2 1/2-inch hose connections and a 4 1/2-inch pumper connection. Hose cabinets containing hose, nozzles, and fittings are associated with several of these hydrants.

d) A 10-inch interior loop header is located in the turbine and screenhouse buildings. This interior header is connected to the outdoor loop header by valved connections routed through the service building, auxiliary building and the yard. This arrangement forms a series of smaller interior-exterior loops. The interior piping network is equipped with isolating valves and supplies water to the fixed fire protection and standpipe systems.

e) Each of the fixed fire protection system valve manifolds is equipped with a manual valve to allow periodic flushing of the headers to remove silt and foreign material in the piping.

f) The standpipe connections are 2, 2 1/2 or 3 inches in size. The 2 and 2 1/2-inch connections are furnished with 1 1/2-inch hose valves. A 1 1/2-inch fire hose on a storage reel is directly connected to the 1 1/2-inch hose valve. The 3-inch connections are furnished with two hose valves, one 2 1/2-inch and one 1 1/2-inch size. The 2 1/2-inch valve is provided with a reducer to a 1 1/2-inch hose cap. The 1 1/2-inch valve is used for direct connection of the 1 1/2-inch fire hose on the storage reel.

Outside Plant Protection

a) The major piece of fire apparatus for outside plant protection is a 500 gpm capacity four wheel drive fire truck.

This truck carries: (a) 400 gallons of water in its booster tanks, (b) 500 feet of 2 1/2-inch and 500 feet of 1 1/2-inch fire hose, (c) straight stream, water spray, and foam fire hose nozzles in both the 2 1/2- and 1 1/2-inch sizes, (d) 15 gallons of 3% mechanical foam in a built-in tank as well as several 5 gallon cans, (e) a 24-foot extension ladder, (f) pike pole axes, and wrenches, (g) a 100 gpm portable gasoline engine driven pump, (h) a gasoline engine driven generator with portable spot and floodlights, (i) self-contained breathing apparatus and protective clothing, (j) miscellaneous equipment such as fittings, siamese and wye connections, strainers, suction hose, hose valves, battery operated lights, portable extinguishers and smoke ejectors.

b) The 345 kV and 765 kV switchyards are provided with 150-pound dry chemical wheeled extinguishers and/or 20-pound dry chemical or 15-pound CO_2 hand portable extinguishers.

Inside Plant Portable Equipment

a) Fire hose and various type nozzles are provided for manual fire fighting in the event of large indoor fires. This equipment is located at 75 to 100-foot spacings around the perimeter of the turbine generator building and at critical locations in the service and auxiliary buildings.

Each location consists of a hose reel, containing 75 to 100 feet of 1 1/2-inch fire hose and an adjustable water spray nozzle. In certain locations adjustable stream foam nozzles along with 5-gallon cans of 3% mechanical foam concentrate are provided. In some locations a second hose reel with up to 100 feet of 1 1/2-inch fire hose is also provided. This second hose reel is not connected to the standpipe system.

b) Wheeled dry chemical extinguishers are provided in the turbine room basement and on the turbine room main floor. The units on the main floor are equipped with special nozzles for use with quick couplers for fire fighting at the turbine bearings.

c) Hand portable extinguishers are provided in sufficient quantities to limit the distance a user need travel to obtain a unit of this type. Sizes and types of extinguishers used are 20-pound cartridge operated dry chemical, 20-pound cartridge operated all-purpose dry chemical, 15-pound carbon dioxide and 20-pound halon. Inside the lower volume of the containment, 20-pound cartridge operated all-purpose dry chemical extinguishers with brass fill caps are provided. They are secured on vehicle mounting brackets.

d) Self-contained breathing apparatus are located at critical points where fire fighting personnel must enter the various buildings and in the control rooms. The breathers used are the positive pressure type and have a one hour duration regardless of the user's level of activity. With off-shelf and cascade recharging equipment, a 5-man fire brigade team can be supported for 8 hours duration.

e) Hand portable battery operated spotlights complement the breather apparatus to allow personnel to find their way in smoke charged atmospheres.

f) Portable radios have been provided for fire brigade members.

Fixed Systems Office/Service Building

a) The service building is protected by a standard wet pipe sprinkler system on an ordinary hazard spacing. The system consists of a variable pressure alarm check valve with a retarding chamber and sprinklers of suitable temperature rating. A pressure switch on the retarding chamber, on increase of pressure, operates to give control room annunciation, activates the control room alarm and starts the appropriate fire pumps. Areas protected include storage areas and racks, machine shop, and miscellaneous rooms.

   The miscellaneous oil storage room is protected by a standard sprinkler system on an extra hazard spacing but otherwise the same as described above for the storeroom.

b) The service building extension is partially protected by a standard wet pipe sprinkler system on an ordinary hazard spacing similar to the service building. The QC record storage room on the fourth floor is protected by an automatic halon system.

C. Screen House

1. The two diesel fire pump rooms are protected by wet-pipe sprinkler systems. The systems consist of alarm check valves with retarding chambers and sprinklers of suitable temperature ratings. Pressure switches on the retarding chambers operate on pressure increase to give control room annunciation, sound the plant fire horn, and start the appropriate fire pump.

2. Ionization smoke detection systems are provided for the following:
   a. MCC Room for ESW, Basement Area - Elevation 575' (common to both Units)
   b. ESW Pump and MCC Rooms - Elevation 591' (Units 1 and 2)

<u>Auxiliary Building</u>

A 6-inch size welded steel fire protection water header supplying fire hose reels and sprinkler valves is routed through the auxiliary building. This header is isolated by remotely-operated valves outside of the auxiliary building on the east side and in the turbine generator building.

The header is not pressurized but is kept full of water. If it is desired to use one of the hose reels, the operator must actuate a local pushbutton which opens the valves to admit full header pressure. The valves can be closed by the control room operator after the emergency situation has been cleared up. Automatic sprinkler or deluge system operation also will open the remotely-operated isolation valves.

Because of the possibility of accumulations of Class A combustibles in the drumming area, this area is protected by a preaction sprinkler system on ordinary hazard spacing similar to the service building miscellaneous gas bottle shed, except that the nozzles are closed sprinklers. Similar dry pilot preaction sprinkler systems are also installed in the auxiliary building in the following areas: a) under the roof over the new fuel receiving area to protect all shipments of new fuel before transfer to the new fuel storage room, b) floor elevation 587' over normally accessible areas and in the charging and safety injection pump rooms, and to provide protection for the open stairways leading to elevations 573' and 609', c) floor elevation 609'over normally accessible areas and the component cooling water pump area (protected by extra hazard sprinkler spacing and direct closed spray nozzle application onto the pumps), and to provide protection for the open stairways leading to elevation 633', and d) floor elevation 633' over normally accessible areas (excluding the HVAC vestibule areas), and to provide protection for the open stairways leading up to elevation 650'. The sprinklers for the new fuel receiving area are baffled by the roof steel to prevent water discharge into the spent fuel pool.

The Unit 2 control room cable vault is protected by a wet-pipe sprinkler system. The system has a variable pressure alarm check valve with a retarding chamber and sprinkler of the suitable temperature rating. Pressure switches on the retarding chamber operate on pressure increase to give control room annunciation, sound the control room alarm, and start the appropriate fire pumps.

All charcoal filter equipped air handling units in the auxiliary building and for the control rooms are provided with manual water spray deluge systems to extinguish the charcoal filter fire. Continuous strip thermistors provide detection and a high temperature alarm in the associated control room. A detection alarm also sends a signal to open the isolating valves in the auxiliary building supply header and automatically opens the charcoal filter system valve. The control valve to the affected charcoal filter water spray system is then manually opened to fight the fire.

Hydrogen tubes outside the auxiliary building are equipped with a water spray dry pilot deluge system similar to that provided at the office/service building hydrogen tubes.

Ionization fire detection is provided on each floor of the auxiliary building for general alarm of fire as follows:

Elev. 573' a. Containment Spray and Residual Heat Removal Pump Cubicles (Units 1 and 2)
b. Normally accessible common areas of the Auxiliary Building

Elev. 587' a. Transformer Rooms (Units 1 and 2)
b. Sampling Room (common to both units)
c. Spray Additive Tank Room (common to both units)
d. Charging and Safety Injection Pump Cubicles (Units 1 and 2)
e. Drumming/Drum Storage (common to both units)
f. Normally accessible common areas of the Auxiliary Building

Elev. 609' a. Access Control (common to both units)
and 612' b. AB and CD (EL 625'-10") Battery Rooms (Units 1 and 2)
c. El. 617' Valve Gallery (common to both units)
d. NESW Valve Gallery (Units 1 and 2)
e. Normally accessible common areas of the Auxiliary Building

Elev. 633' a. New Fuel Storage Room (common to both units)
b. N-Train Battery Rooms (Units 1 and 2)
c. Normally accessible common areas of the Auxiliary Building

Elev. 650' a. Control Room Equipment Rooms (Units 1 and 2)
b. Normally accessible common areas of the Auxiliary Building

A combination of ionization and infrared detectors are provided in the Main Steam Valve Enclosures East and Main Steam Line Area of Units 1 and 2 at elevation 612'.

Reactor Containments

Containment cable trays, reactor coolant pumps and HVAC charcoal filters are equipped with continuous strip thermistor fire detection which will annunciate in the control rooms.

The HVAC charcoal filters have water spray deluge fire suppression systems and are actuated by the thermistor detection.

Reactor coolant pumps are equipped with preaction water spray systems, manually operated from the control rooms in the event of a lubricating oil fire. Additionally, the RCP motors are provided with an oil spillage control and retention system to preclude spreading oil from a pressure or gravity type leak.

Water supply to containment fire protection is from the non-essential service water system.

Low-Pressure Carbon Dioxide System

A 17-ton capacity low-pressure carbon dioxide system, located in the auxiliary building, is provided for automatic and/or manual protection of various areas as listed below. The amount of CO_2 in the system is sufficient to protect the largest single hazard in the plant. The CO_2 is stored in an insulated pressure vessel having an automatically operated refrigeration system. Operation of the CO_2 systems is annunciated and ~~they~~ activates the control room alarm system.

The areas protected by the low-pressure CO_2 system and the type of fire detection are as follows:

1. Turbine-Generator-Building

   a) Lubricating oil storage rooms Units No. 1 and No. 2. Continuous-strip thermistor detection.
   b) Main turbine oil tank rooms Units No. 1 and No. 2. Continuous-strip thermistor detection.

2. Auxiliary Building

   a) AB and CD emergency diesel generator rooms Units No. 1 and No. 2. Continuous-strip thermistor detection. (2 zones for each room)

b) Diesel oil pump and valve station rooms Units No. 1 and No. 2. Continuous-strip thermistor detection.

c) Electrical switchgear rooms Units No. 1 and No. 2.

1. 4.16 kV switchgear rooms. Infrared and ionization detection.
2. 4.16 kV/600 V transformers and engineered safety equipment rooms. Infrared and ionization detection.
3. 4.16 kV/600 V transformers, control rod drive and inverter rooms. Infrared and ionization detection.

d) Electrical switchgear room cable vaults Units No. 1 and No. 2. Infrared and ionization detection.

e) Auxiliary cable vaults Units No. 1 and No. 2. Ionization detection.

f) Control room cable vaults Units No. 1 and No. 2. Manual (backup to Halon 1301 systems).

g) Electrical penetration area cable tunnels Units No. 1 and No. 2.

1. Quadrant 1. Infrared and ionization detection.
2. Quadrant 2. Infrared and ionization detection.
3. Quadrant 3 north. Infrared and ionization detection.
4. Quadrant 3 middle. Infrared and ionization detection.
5. Quadrant 3 south. Infrared and ionization detection.
6. Quadrant 4. Infrared and ionization detection.

3. Carbon dioxide hose reel stations are provided for manual fire fighting in the auxiliary building, switchgear rooms, and

at the entrances to the control rooms, diesel generator rooms, and electrical penetration area cable tunnels.

Halon 1301 Systems

Halon 1301 systems are provided for automatic fire protection in various areas of the plant. Locations of these systems include the control room cable vaults, the computer rooms and underfloor, control points for the Plant Security System, and as previously mentioned, the service building extension QC record storage room, TSC computer room, TSC console room and the TSC UPS inverter room. Actuation is by two zones of ionization detection for each system.

Control Room Fire Protection

The control rooms are equipped with portable fire extinguishers. Detection systems of the ionization type are installed. The control rooms are occupied at all times by operators who have been trained in fire extinguishing procedures. All areas of the control rooms are accessible for fire fighting.

Miscellaneous Protective Features

a) Transformer decks are pitched and drained to remove oil which may be spilled from a fire-involved transformer and also to remove water discharged from the transformer water spray system.

b) The construction of most exterior and interior building walls equal or exceed fire rating requirements. Openings in walls which require fire rating are provided with appropriately rated doors, dampers and penetration fire seals. When rated components are not installed in a fire wall separating fire areas, technical evaluations are performed justifying the configurations.

9.9 AUXILIARY BUILDING VENTILATION SYSTEM

9.9.1 GENERAL DESCRIPTION

The auxiliary building ventilation systems, shown in Figures 9.9-1 and 9.9-2, consist of:

a. Engineered Safety Features Ventilation System (one per plant unit).
b. Fuel Handling Area Ventilation System (one shared system).
c. General Ventilation Systems (one per plant unit with crosstie).
d. General Supply System (one per plant unit).

The auxiliary building is basically a five-level compartmented structure containing the auxiliary nuclear equipment for both units. All equipment handling radioactive fluids is located on the lower four levels of the auxiliary building. The fourth level also houses the two control rooms and the ventilation equipment.

The auxiliary building ventilation systems are designed to maintain temperatures in the various portions of the building within design limits for operation of equipment and for personnel access for inspec-tion, maintenance and testing as required.

9.9.2 DESIGN BASES

Outside ambient conditions used for design purposes are 90°F summer dry bulb, 76°F summer wet bulb and -7°F winter dry bulb. Ventilation is based on limiting temperatures in all area to a predetermined maximum, generally 110°F, and heating is provided to maintain a 60°F minimum temperature.

All ventilation systems serving the auxiliary building are once-through systems. Supply air is introduced to the areas least likely to be contaminated, and exhausted directly from those with the greatest contamination potential. Additionally, the exhaust systems are of greater capacity than the supply systems, thus maintaining the entire auxiliary building at a slightly negative pressure.

All exhaust air from the auxiliary building is directed to the unit vents. There is a vent for each unit. Each vent has radiation detectors for continuous monitoring of the exhaust air during release to atmosphere.

Absolute filter cells are designed to remove as much as 99.97 percent of solid particulates of 0.3 micron mean diameter in size. Performance characteristics of the charcoal adsorbent provide for removal of as much as 99.9 percent of any entrained methyl iodide or iodine vapor*. Supply and exhaust unit roughing filters have a NBS duct spot efficiency (Cottrell Precipitate) of 75%.

9.9.3 SYSTEM DESCRIPTIONS

9.9.3.1 Engineered Safety Features Ventilation

The enclosures for the engineered safety features equipment for both units are located in the lower three levels of the auxiliary building. (The containment spray heat exchanger and residual heat exchanger enclosures extend up into the fourth level with access into the enclosures from the third level only.) The enclosures for each unit's safety feature equipment are ventilated by two separate ventilation systems. The areas serviced by this system are: the containment spray pump enclosures, the residual heat removal pump enclosures, the safety injection pump enclosures, the residual heat exchanger enclosures, the containment spray heat exchanger enclosures and the reciprocating and centrifugal charging pump enclosures. Figure 9.9-2 shows a flow diagram of the engineered safety features ventilation system and is typical for the system serving either unit.

*For accident analysis, the absolute filter banks of the engineered safety features ventilation system are assumed to remove 99% of all radioactive particulates with the adsorbers removing 90% of methyl iodine.

9.10 CONTROL ROOM VENTILATION SYSTEM

9.10.1 GENERAL DESCRIPTION

The Control Rooms for Unit No. 1 and Unit No. 2 are both physically located on El 633' 0" of the Auxiliary Building with normal access from the turbine building. Control Room air conditioning equipment is in an equipment room directly above the Control Room. Both Control rooms are enclosed in a missile and tornado proof structure. The Control Room Ventilation System is shown in Figure 9.10-1.

9.10.2 DESIGN BASES

The Control Room Air Conditioning System is designed to maintain room temperature within limits required for operation, maintenance and testing of plant controls and uninterrupted safe occupancy during post-accident shutdown.

The Control Room Air Conditioning System is designed to maintain a temperature of 75°F dry bulb and 50 percent relative humidity under normal operating conditions. The design is based on outside temperatures ranging from -7°F winter dry bulb to 90°F summer dry bulb- 76°F summer wet bulb. The system operates during normal or emergency conditions as required.

Conditioned air is supplied to the Control Room by either of two full-capacity 15,000 CFM air-handling Units (1 standby). Each unit includes a roughing filter, medium efficiency filter, chilled-water coil, and a fan. Downstream of each air handler in the duct system is an electric blast coil heater and an electric humidifier. Each unit is provided with chilled water from an associated 30-ton liquid-chiller. Each air-handler/liquid-chiller combination is independently capable of fulfilling design objectives. Condenser water for each liquid chiller is taken from a different header of the Essential

Service Water System. For emergency cooling the essential service water can be manually diverted directly through the air handling coil thus bypassing the liquid chillers.

Continuous pressurization of the Control Room is normally provided by the Air Conditioning System to prevent the entry of dust and dirt. Backup filtration and pressurization are provided by a separate 6,000 CFM air-handler with roughing filters, absolute particulate filters and charcoal filters. This unit can also be used in the recirculation mode as a cleanup system. The performance characteristics of the absolute particulate filters provide for removal of as much as 99.97 percent of solid particulates of 0.3 micron mean diameter. Performance characteristics of the charcoal filters provide for removal of as much as 99.9 percent of entrained methyl iodide or iodine vapor. All air conditioning equipment, pressurization fans and auxiliary equipment can be powered from emergency buses.

9.10.3 SYSTEM OPERATION

Two fresh-air intakes are provided for each Control Room. Both air conditioning units share one intake. A separate intake is provided for the pressurizer/cleanup filter unit. Both fresh-air intakes are fitted with a motor-operated isolation damper for Control Room isolation. Normally, a fixed proportion of room air and outside air is supplied to the Control Room through one of the air-handling units. Temperature is controlled by thermostats located in the Control Room. Each liquid chiller has an independent control system. Outdoor air supplied to the Control Room through the air-handling unit maintains a positive pressure within the room with respect to the surrounding environs to prevent entry of dust, etc.

A toilet facility is located in the Unit No. 2 Control Room. A small exhaust fan continuously purges this room. The exhaust vent is fitted with an isolation damper.

10. STEAM AND POWER CONVERSION SYSTEM

This chapter describes the steam (secondary) cycle for each of the two units. In general, descriptions in this chapter apply equally to either Unit No. 1 or Unit No. 2 except where specifically noted. The systems described in this chapter are included in each unit unless specifically designated as shared.

10.1 GENERAL DESCRIPTIONS

The Steam and Power Conversion System is designed to convert heat produced in the reactor to useful electric energy. Heat in the reactor coolant is transferred to the Main Steam System in the four steam generators of the Reactor Coolant System. At a reactor output of 3250 MWt (3411 MWt for Unit 2), sufficient steam is produced to drive a tandem compound reheat steam turbine with an approximate net output of 1030 MWe (1100 MWe for Unit 2) operating in a closed condensing cycle with six stages of regenerative feedwater heating. Exhaust steam is condensed in three surface type steam condensers and returned to the steam generators. The four-casing, six-flow exhaust, 1800 rpm turbine is directly coupled to a single water-and-hydrogen-cooled generator. The system is designed to receive and dispose of the total heat produced in the Reactor Coolant System following a rapid shutdown of the turbine generator from any load. Heat dissipation under this condition is accomplished by the steam dump system to the condenser and/or the steam generator power relief valves. The turbine driven main feed pumps provide water to the steam generators under normal conditions.

Radiation monitoring of secondary side discharge points is provided and described in Section 10.11.

Turbine driven and motor driven auxiliary feed pumps are provided to ensure that adequate feedwater may be supplied to the steam generators for reactor decay heat removal under all circumstances, including loss of power and loss of the normal heat sink. Auxiliary feedwater flow can be maintained until power is restored or reactor decay heat removal can be accomplished by other systems. Auxiliary feed pumps and piping are designed as Class I components. The turbine cycle is able to match the reactor rates of load change 55 MWe per minute and step load increases of 110 MWe, within the load range of 165 MWe to full load.

h) Extreme high steam generator level

i) Safety injection

j) Loss of stator cooling (low flow, low pressure, or high temperature)

k) High exhaust hood temperature (Unit 2 only)

l) Reactor trip

m) Manual operation of any of several trip levers

n) Loss of both main feedpump turbines

o) Unit or Overall differential

p) Low shaft driven oil pump pressure (Unit 1 only)

q) EHC trip system pressure low (Unit 1 only)

r) EHC master trip (Unit 1 only)

s) EHC loss of speed feedback (Unit 1 only)

t) AMSAC: less than 25% F.W. flow to 3/4 loops and above 40% power

10.3.4 LOSS OF EXTERNAL ELECTRICAL LOAD

The steam dump system, more fully described in Section 10.2.2, is designed to dump approximately 85% of full load steam flow at full load steam pressure. Dump capacity increases with any transient increase in steam pressure. The steam dump valves are capable of opening fully in three seconds. The steam is dumped to the main condensers.

The steam dump is controlled by the mismatch between reactor coolant average temperature and the value of the temperature program for the corresponding turbine load. Dump flow is reduced as the reactor coolant average temperature is reduced toward the programmed value. A turbine trip with a reactor trip will also initiate dump action. The dump valves may be manually controlled during cooldown, start-up, hot stand-by service, or physics testing.

10.3.5 TEST AND INSPECTION

The rotor has undergone the normal quality assurance and quality control tests associated with the design and manufacture of large turbine-generators. Provisions for ultrasonically testing for cracks are also included in the rotor design. These tests can be run if deemed necessary when the rotors are removed for turbine inspection.

Operational tests include full closure tests of the turbine stop and control valves (both main and reheat), and the feed-pump turbine HP stop valve (Unit 1) and partial closure tests of the feed-pump turbine LP stop valve (Unit 1) and the feed-pump turbine stop valve (Unit 2). Overspeed trip and other turbine protective devices associated with the turbines are tested as plant operating conditions permit and in accordance with accepted practice and/or manufacturers' recommendations.

11. WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 WASTE DISPOSAL SYSTEM

11.1.1 DESIGN BASES

Control of Releases of Radioactivity to the Environment

Criterion: The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR 50 Appendix I requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental requlations. The facilities are shown on Figures 11.1-1, 11.1-2, 11.1-2A, 11.1-2B, 11.1-3 and 11.1-4. The sizing of the various waste equipment was predicated on the volumes and flow rates originally expected to be handled.

Radioactive fluids entering the Waste Disposal System are collected in tanks until determination of subsequent treatment can be made. Provisions have been made for waste segregation and recycling to permit selective operation of the processing equipment to maintain radioactivity in the effluents as low as practicable. Fluids are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Liquid wastes are processed as required and then either recycled or released under controlled conditions. The system design and operation are directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 50 Appendix I.

The bulk of the radioactive liquid discharge from the Reactor Coolant System is processed and retained inside the plant by the Chemical and Volume Control System (CVCS) recycle train. This minimizes liquid input to the Waste Disposal System which processes relatively small quantities of generally low-activity level wastes. The processed water from the Waste Disposal System, from which most of the radioactive material has been removed, is either recycled to the Chemical and Volume Control System or discharged through a monitored line to the circulating water discharge.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gas is reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent.

The spent demineralizer resins, the filter cartridges, and the concentrates from the evaporators are packaged and stored on-site until shipped off-site for disposal.

11.1.2 GENERAL DESCRIPTION AND OPERATION

The Waste Disposal System Performance Data are given in Table 11.1-1. With the exception of the reactor coolant drain tanks and drain tank pumps, the Waste Disposal System is common to Units 1 and 2. The system is capable of processing all wastes generated during continuous operation of the primary system assuming that fission products

The liquids in the containment flow to the reactor coolant drain tank and are discharged by the reactor coolant drain tank pumps either directly to the CVCS holdup tanks or to the clean waste holdup tank. The pumps are operated automatically by a level controller in the tank. These pumps also return water from the refueling cavity to the refueling water storage tank. The reactor coolant drain tank pumps are located inside the auxiliary building.

Where possible, waste liquids in the auxiliary building drain to the waste holdup tanks by gravity flow. Other waste liquids drain to the sump tanks and are discharged to the waste holdup tanks by pumps operated automatically by a level controller in the sump tanks.

The activity level of waste liquid from the laundry and hot shower area will usually be low enough to permit discharge from the plant without processing. If analysis indicates that the liquid is suitable for discharge, it is pumped to waste condensate tanks where the activity is determined before discharging through a radiation monitor to the circulating water or it is pumped directly to the waste liquid discharge line upstream of the radiation monitor. Otherwise, the liquid is pumped to the station drainage waste holdup tank for processing. Similar facilities are provided for discharging low level waste from the station drainage waste holdup tank. An analysis record is maintained for all releases.

One of two CVCS boric acid evaporators is temporarily functioning as a radwaste evaporator. A 15 gpm radwaste evaporator and 2 gpm radwaste evaporator are available as backup to the 15 gpm boric acid/radwaste evaporator in case additional capacity is desired. Liquids requiring cleanup before release are processed in batches in this boric acid/radwaste evaporator. Processing liquid waste is similar to processing reactor coolant except for disposal of the processed liquids and vented gases. Liquid waste is pumped to the boric acid/radwaste evaporator via the waste evaporator feed pumps. The concentrates are discharged to the waste evaporator bottoms storage tank for drumming prior to shipment to an offsite burial facility.

Evaporator distillate (condensate) which is to be released is routed to one of two CVCS monitor tanks which are both temporarily functioning as waste condensate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms the activity level is suitable for discharge, the condensate is pumped through a flow meter and a radiation monitor to the condenser circulating water discharge. Condensate can also be released under administrative control from the other two CVCS monitor tanks which serve the other boric acid evaporator. The releases are sampled and analyzed for both tritium and non-tritium isotopes and monitored by the same radiation monitor as that previously mentioned above before release into the circulating water discharge. If analysis indicates the activity level is not suitable for discharge, the condensate is returned to the station drainage waste holdup tank for reprocessing. Although the radiochemical analysis forms the basis for recording activity released, the radiation monitor provides surveillance over the operation by closing the discharge valve if the liquid activity level exceeds a preset value.

Measures are taken to minimize the need to process fluids which contain foam causing substances. If possible, non foaming decontamination agents are used for equipment scrubdown where the decontamination agent must be processed through the evaporators. If foaming should occur a reagent tank is provided for charging the evaporator with an antifoaming reagent.

<u>Gas Processing</u>

During plant operations, gaseous wastes will originate from:

a) Degassing reactor coolant discharged to the Chemical and Volume Control System

b) Displacement of cover gases as liquids accumulate in various tanks

c) Miscellaneous equipment vents and relief valves

d) Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases

The Waste Disposal System includes nitrogen and hydrogen systems which supply these gases to primary plant components. The pressure regulator in the nitrogen system header is set at 75 psig. When the nitrogen header pressure drops below a preset pressure, an alarm alerts the operator. A backup nitrogen supply is provided for the accumulators.

Most of the gas received by the Waste Disposal System during normal operation is nitrogen cover gas displaced from the CVCS holdup tanks as they are filled with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or no aerated liquids and the vent header itself is designed to operate at a slight positive pressure (0.5 psig minimum to 2.0 psig maximum) to prevent in-leakage. Out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. From the compressors, gas flows to one of eight gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select another tank for backup. When the tank in service becomes pressurized to 100 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator so he may select a new backup tank. Pressure indicators are provided to aid the operator in selecting the backup tank. The individual tank pressures are continuously recorded on the control panel in the auxiliary building.

Gas held in the decay tanks can either be returned to the CVCS holdup tanks or, if it has decayed sufficiently for release, discharged to the atmosphere. Generally, the last tank to receive gas will be the first tank recycled to the CVCS holdup tanks. This permits the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator the option to fill, reuse, and discharge gas simultaneously. During degassing of the reactor coolant prior to a cold shutdown, for example, it may be desirable to pump the gas purged from the volume control tank into a particular gas decay tank and isolate that tank for decay rather than reuse the gas in it. This is done by opening the inlet valve to the desired tank and closing the outlet valve to the reuse header. Simultaneously, one of the other tanks can be opened to the reuse header if desired, while another is discharged to atmosphere.

Before a tank is discharged to the environment, it is sampled and analyzed to determine and record the activity to be released, and then is discharged to the plant vent at a controlled rate through a radiation monitor which enables the operator to monitor the radioactivity in the gas release. Samples of the gas to be released are taken in gas sampling vessels. During release a trip valve in the discharge line is closed automatically by a high radioactivity level indication in the plant vent.

During operation, gas samples are drawn automatically from the gas decay tanks and analyzed to determine their hydrogen and oxygen content. A second analyzer is used to monitor oxygen in the line from the discharge of the waste gas compressor in operation. There should be no significant oxygen content in the waste gas or in any of the gas decay tanks; an alarm sounds if either of the samples contains 2.5% or higher by volume of oxygen. Upon a "high-high" oxygen content of 3.0% by volume, the oxygen analyzer automatically isolates the tank being filled and places the standby gas decay tank in service. The operator then determines the source of oxygen in-leakage and purges the affected component and Waste Gas System vent header piping as required with nitrogen. The isolated waste gas decay tank and standby tank can be diluted with nitrogen if they have high oxygen concentrations.

Solids Processing

The Waste Disposal System is designed to package solid wastes for removal to disposal facilities.

Concentrates from the waste evaporator bottoms storage tank are pumped into shipping casks and mixed with the solidification agent. On-site contract personnel perform the solidification process. The casks are moved by the drumming room bridge and trolley crane to a shielded storage area until removal to a burial site.

Spent resins are either sluiced to the spent resin storage tank or pumped directly into shielded shipping casks in the drumming room. Resins in the storage tank can be sluiced by first bubbling nitrogen through the tank to the vent header to stir up the resin, then using water to transport the resin at a controlled rate into shipping casks in the drumming room. Resins are either dewatered and air dried or slurried with a solidification agent for shipment. The casks are handled and stored in a fashion identical to that for the concentrated bottoms.

Shielding is provided for each cask during filling and handling operations to reduce the dose rate in work areas. The basis for shield design and dose rate calculations is for one cycle of core operation with one percent defective fuel in each unit.

Components

Codes applying to components of the Waste Disposal System are shown in Table 11.1-2. Components summary data are shown in Table 11.1-3.

Laundry and Hot Shower Tanks

Two stainless steel tanks collect liquid wastes originating from the laundry and hot showers. When the tanks have filled, the contents are analyzed for gross activity. As dictated by the activity level, the tank contents are pumped to the station drainage waste holdup tank for processing or to the waste condensate tanks for release.

Chemical Drain Tank

A stainless steel chemical tank, collects drainage from the radiochemical laboratory. The tank contents are pumped to the station drainage waste holdup tank for processing.

Reactor Coolant Drain Tanks

The tanks serve as a drain collecting point for the Reactor Coolant Systems and other equipment located inside the reactor containments. The contents can be discharged to the waste holdup tanks, to the refueling water storage tanks, or to the CVCS holdup tanks. The tanks are of welded stainless steel construction.

11.3.3.2 Area Radiation Monitoring System

This system consists of channels which monitor radiation levels in various areas of the plant and are shown in Table 11.3-1.

Each channel consists of a fixed position gamma sensitive Geiger-Mueller detector. The radiation level is indicated at the detector, and at the Radiation Monitoring System Control Terminal where it is also recorded. High-radiation alarms are displayed on the Radiation Monitoring System Control Terminal in the control room and at the detector location. The control room annunciator sounds at the control terminal for any channel in an alarm status. Verification of which channel has alarmed is done at the Radiation Monitoring System Control Terminal. A remotely operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce a visible increase in the meter indication.

Two post-accident and high range in-containment radiation detectors are located in each containment. One detector is located in the upper containment, and the second is located in the lower containment approximately 180° apart from the other detector. These monitors function as accident detectors and consists of ion chambers. The readout modules are located in the control room.

11.3.4 Reactor Coolant Activity Monitoring

Refueling shutdown programs at operating Westinghouse PWR's indicate that, during cooldown and depressurization of the Reactor Coolant System (RCS), a release of activated corrosion products and fission products from defective fuel has been found to increase the coolant activity level above that experienced during steady state operation. However, interruptions in the refuelings attributable to high coolant activity, are avoided by implementing established shut-down procedures. These procedures include purification of the RCS through the cation and mixed bed demineralizers and system degassification.

Table 11.3-2 illustrates the calculated coolant activity increases of several isotopes for the Donald C. Cook Plant. This table lists the calculated activities during steady state operation before refueling shutdown outage and calculated peak activities during plant cooldown operations. These data are based on measurements from an operating PWR which is similar in design to the Cook Nuclear Plant and has operated with fuel defects. The measured activity levels are also included in Table 11.3-2.

The dominant non-gaseous fission product released to the coolant during system depressurization is found to be Iodine-131. The activity level in the coolant was observed to be higher than the normal oper-ating level for nearly a week following initial plant shutdown. Although lesser in magnitude, the other fission produce particulates (e.g., cesium isotopes) exhibited a similar pattern of release and removal by purification. It is reasonable to project this data to the Cook Nuclear Plant since the purification constants are similar and as it is standard operating procedure to purify the coolant through the demineralizers during plant cooldown. Fission gas data from operating plants indicate a maximum increase of approximately 1.5 over the normal coolant gas activity concentration. However, the system degassification procedures are implemented prior to and during shutdown, and have proven to be an effective means for reducing the gaseous activity concentration and controlling the activity to levels lower than the steady state value during the entire cooldown and depressurization procedure.

The corrosion product activity releases have been determined to be predominantly dissolved Cobalt-58. From Table 11.3-2, it is noted that this contribution is less than 1% of the total expected coolant activity and is hence considered to be a minor contribution.

Since continued operation of the purification system is standard operating procedure during plant cooldown and since means for system degassification are available for fission gas removal, the total activity concentration in the coolant can be maintained within Technical Specification limits throughout the plant shutdown, while considering the additional activity inventory released during system cooldown and depressurization. The coolant activity concentrations and inventories during the shutdown and prior to plant startup are established by chemical analysis of samples for the Reactor Coolant System.

11.3.5 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

The following cart-mounted, continuous air monitors are available for use in an emergency that may involve airborne radioactivity concerns:

> Two particulate and radioiodine monitors. The radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridge. The detector is connected to a single channel analyzer calibrated to 365 KeV. These monitors are dedicated to the TSC and the control room.

> Five airborne particulate, radiogas, and radioiodine monitors. The radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridges. The detector is a stabilized NaI detector connected to a two channel analyzer calibrated to 365 KeV with automatic Xe subtraction from the second channel.

> All cart-mounted iodine detectors are in three (3) inch lead shields.

In addition, there are available for use throughout the plant ten regulated air samplers, which accept either TEDA impregnated charcoal or Silver Zeolite cartridges.

In addition to the equipment normally available in the regular radiochemistry counting facility, the following analysis equipment is available for analysis of the Silver Zeolite cartridges that might be used in an emergency:

a. A 4" x 4" NaI crystal connected to a multichannel analyser is located in the low background counting facility.

b. In the basement assembly area there is a cartridge purge unit consisting of a T-size bottle of dry nitrogen, regulator, cartridge holder, and associated piping to permit purging of Silver Zeolite or charcoal cartridges with dry nitrogen.

c. Located in the basement assembly area is a single channel analyzer calibrated to 365 KeV, connected to a 2" x 2" NaI crystal in a 2 1/2" lead shield designed for counting in TEDA charcoal or Silver Zeolite cartridges.

TABLE 11.3-1

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES, AND

DETECTING MEDIUM

Monitor Name	Channel Number	Medium	Range	Detected Isotopes
Containment-Air Particulate	ERS-1301, 1401 2301, 2401	Air	1.5×10^{-4} to 7.5 μCi	Cs^{137}, Radioactive Particulates
Containment-Air Iodines	ERS-1303, 1403 2303, 2403	Air	2.3×10^{-4} to 2.3 μCi	I^{131}, Radioiodine
Containment Radio-Gas	ERS-1305, 1405 2305, 2405	Air	9.1×10^{-7} to 4.2×10^{-2} μCi/cc	Xe^{133}, Noble Gases
	ERS-1307, 1407 2307, 2407	Air	2×10^{-3} to 1.1×10^{3} μCi/cc	Xe^{133}, Noble Gases
	ERA-1309, 1409 2309, 2409	Air	1.6×10^{-1} to 9.0×10^{4} μCi/cc	Xe^{133}, Noble Gases
Steam Jet Air Ejector Gas	SRA-1905, 2905	Air	9.1×10^{-7} to 4.2×10^{-2} μCi/cc	Xe^{133}, Noble Gases
	SRA-1907, 2907	Air	2.0×10^{-3} to 1.1×10^{3} μCi/cc	Xe^{133}, Noble Gases
	SRA-1909, 2909	Air	1.6×10^{-1} to 9.0×10^{4} μCi/cc	Xe^{133}, Noble Gases
Component Cooling Loop Liquid	R-17A & B	Water	10^{-5} to 10^{-2} μCi/cc	Co^{60}, Mixed Fission Products
Waste Disposal System Liquid Effluent	R-18	Water	3×10^{-5} to 3×10^{0} μCi/cc	Co^{60}, Mixed Fission Products
Steam Generator Blowdown Liquid	R-19	Water	1.7×10^{-5} to 1.7×10 μCi/cc	Cs^{137}, Mixed Fission Products
Essential Service Water Liquid	R-20	Water	3.7×10^{-5} to 3.7×10^{-1} μCi/cc	Cs^{137}, Mixed Fission Products

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TABLE 11.3-1 (Cont'd.)

Monitor Name	Channel Number	Medium	Range	Detected Isotopes
Steam Generator Blowdown Treatment System Liquid	R-24	Water	1.4×10^{-6} to 1.3×10^{-1} µCi/cc	Co^{60}, Mixed Fission Products
Unit Vent Air Particulate	VRS-1501, 2501	Air	1.5×10^{-4} to 7.5 µCi	Cs^{137}, Radioactive Particulates
Unit Vent Radioiodine	VRS-1503, 2503	Air	2.3×10^{-4} to 2.3×10^{0} µCi/cc	I^{131}, Radioiodine
Unit Vent Radio Gas	VRS-1505, 2505	Air	9.1×10^{-7} to 4.2×10^{-2} µCi/cc	Xe^{133}, Noble Gas
	VRS-1507, 2507	Air	2.0×10^{-3} to 1.1×10^{3} µCi/cc	Xe^{133} Noble Gas
Unit Vent Hi-Level Radio Gas	VRS-1509, 2509	Air	1.6×10^{-1} to 9.0×10^{4} µCi/cc	Xe^{133}, Noble Gas
Gland Seal Condenser Exhaust Monitor	SRA-1805, 2805	Air	9.1×10^{-7} to 4.2×10^{-2} µCi/cc	Xe^{133}, Noble Gas
	SRA-1807, 2807	Air	2.0×10^{-3} to 1.1×10^{3} µCi/cc	Xe^{133}, Noble Gas
	SRA-1809, 2809	Air	1.6×10^{-1} to 9.0×10^{4} µCi/cc	Xe^{133}, Noble Gas
Essential Service Water Liquid	R-28	Water	10^{-5} to 10^{-2} µCi/cc	Co^{60}, Mixed Fission Products
Control Room Area	R-1	Air	10^{-1} to 10^{4} mr/hr	
Containment Area at Personnel Lock	VRS-1101, 2101	Air	10^{-1} to 10^{4} mr/hr	
Upper Containment Area Monitor	VRS-1201, 2201	Air	10^{-1} to 10^{4} mr/hr	
Steam Generator Relief Monitor	MRA-1600, 2600 1700, 2700	Air	3×10^{-1} to 2×10^{5} µCi/cc	Xe^{133}, Noble Gas
Radiochemistry Laboratory Area	R-3	Air	10^{-1} to 10^{4} mr/hr	
Charging Pump Room Area	R-4	Air	10^{-1} to 10^{4} mr/hr	
Spent Fuel Area	R-5	Air	10^{-1} to 10^{4} mr/hr	

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TABLE 11.3-1 (Cont'd.)

Monitor Name	Channel Number	Medium	Range	Detected Isotopes
Sampling Room Area	R-6	Air	10^{-1} to 10^{4} mr/hr	
In-Core Instrumentation Room Area	R-7	Air	10^{-1} to 10^{4} mr/hr	
Drumming Station Area	R-8	Air	10^{-1} to 10^{4} mr/hr	
High Range Containment Area Monitor	VRS-1310, -2310 -1410, -2410	Air	1×10^{0} to 1×10^{7} R/HR	
Vestibule Elevation 591'	ERS-1306, -2306	Air	10^{-3} to 10^{2} mr/hr	
Outside Containment Spray Pump Rooms Elevation 573'	ERS-1406, -2406	Air	10^{-3} to 10^{2} mr/hr	
West of Equipment Hatch, Elevation 650'	VRS-1506, -2506	Air	10^{-3} to 10^{2} mr/hr	
Turbine Building, Elevation 609'	SRA-1806, -1906, -2906	Air	10^{-3} to 10^{2} mr/hr	
Turbine Building, Elevation 591'	SRA-2806	Air	10^{-3} to 10^{2} mr/hr	
North of Boric Acid Tanks, Elevation 609'	RRS-1003	Air	5 to 500,000 cpm	
Unit 1 E CCP Room	ERA-7303	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 W CCP Room	ERA-7304	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 E RHR Pump Room	ERA-7305	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 W RHR Pump Room	ERA-7306	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 N SIS Pump Room	ERA-7307	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 S SIS Pump Room	ERA-7308	Air	10^{-3} to 1×10^{3} R/hr	

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TABLE 11.3-1 (Cont'd.)

Monitor Name	Channel Number	Medium	Range	Detected Isotopes
Unit 1 Reactor Coolant Filter Cubicle	ERA-7309	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 E CCP Room	ERA-8303	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 W CCP Room	ERA-8304	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 E RHR Pump Room	ERA-8305	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 W RHR Pump Room	ERA-8306	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 N SIS Pump Room	ERA-8307	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 S SIS Pump Room	ERA-8308	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 Reactor Coolant Filter Cubicle	ERA-8309	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 Control Room	ERS-7401	Air	10^{-5} to 10 R/hr	
Access Control Facility	ERA-7403	Air	10^{-5} to 10 R/hr	
Radio Chemistry Lab	ERA-7404	Air	10^{-5} to 10 R/hr	
Unit 1 N Seal Water Injection Filter Cubicle	ERA-7407	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 S Seal Water Injection Filter Cubicle	ERA-7408	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 Seal Water Filter Cubicle	ERA-7409	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 Control Room	ERS-8401	Air	1×10^{-4} to 10 R/hr	
609' Elevation Passageway	ERA-8403	Air	1×10^{-4} to 10 R/hr	

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TABLE 11.3-1 (Cont'd.)

Monitor Name	Channel Number	Medium	Range	Detected Isotopes
Unit 2 N Seal Water Injection Filter Cubicle	ERA-8407	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 S Seal Water Injection Filter Cubicle	ERA-8408	Air	10^{-3} to 1×10^{3} R/hr	
Unit 2 Seal Water Injection Filter Filter Cubicle	ERA-8409	Air	10^{-3} to 1×10^{3} R/hr	
587' Elevation Passageway	ERA-7504	Air	10^{-3} to 10 R/hr	
Emergency Sampling Location	ERA-7507	Air	10^{-3} to 10 R/hr	
573' Elevation Passageway	ERA-7508	Air	10^{-3} to 10 R/hr	
Refueling Water Purification Filter Cubicle	ERA-7509	Air	10^{-3} to 1×10^{3} R/hr	
Unit 1 Vent Sampling Area	ERA-7601	Air	10^{-5} to 10 R/hr	
Unit 1 Vent Sampling Flow Adjacent Area	ERA-7602	Air	10^{-4} to 1×10^{2} R/hr	
Unit 2 Vent Sampling Area	ERA-7603	Air	10^{-5} to 10 R/hr	
Unit 2 Vent Sampling Flow Adjacent Area	ERA-7604	Air	10^{-4} to 1×10^{2} R/hr	
633' Elevation Passageway	ERA-7605	Air	10^{-5} to 10 R/hr	

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TABLE 11.3-2

REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES
DURING STEADY STATE OPERATION AND PLANT SHUTDOWN OPERATION

Isotope	Operating PWR Plant: Measured Activity Before Shutdown µCi/gm	Operating PWR Plant: Measured Peak Shutdown Activity µCi/gp	Donald C. Cook Plant - 1% Fuel Defects: Calculated Activity Before Shutdown µCi/gm	Donald C. Cook Plant - 1% Fuel Defects: Expected Peak Shutdown Activity µCi/gm
I-131	0.83	14.9	2.4	43.0
Xe-133	127.0	65.0*	254.0	130.0*
Cs-134	1.29	1.7	0.19	0.25
Cs-137	1.67	2.14	1.1	1.4
Cs-144	0.00068	0.0058	0.00051	0.0044
Sr-89	0.0033	0.40	0.0042	0.51
Sr-90	0.00057	0.013	0.0001	0.0023
Co-58	---	0.95	0.025	1.0

*Activity reduced from steady state level by approximately one day of system degassification prior to plant shutdown.

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3. Steam Jet Air Ejector - A continuous release of activity exists only during periods of steam generator primary to secondary leakage. The steam jet air ejector exhaust is continuously monitored. The steam jet air ejector monitor is sensitive to total beta and gamma activity.

4. Turbine Gland Seal Exhaust - A continuous release of activity exists only during periods of steam generator primary to secondary leakage. The turbine gland seal exhaust is continuously monitored.

5. Steam Generator Blowdown Exhaust - The releases are through the main condenser during startup routine cleanup and off normal chemistry conditions, during which the release is to the atmosphere via the S/G blowdown flash tank vent. The steam generator blowdown is continuously monitored.

6. Main Steam PORV/Safety Release Valves - The main steam power operated relief valves and safety valves provide pressure relief on each steam lead if steam pressure exceeds normal operating values. They also allow plant cooldown by steam discharge to the atmosphere if the turbine by-pass system is not available. The PORV discharge lines are continuously monitored.

7. Miscellaneous Ventilation - Releases are through the unit vent from ventilation systems such as SF pool, nuclear sampling room, etc. Noble gas activity and release rates are monitored and recorded. High radioactivity will be alarmed in the control room.

Meteorological conditions during periods of release from the above systems will be obtained from the meteorological program.

The Unit Vent monitors record gross beta and gamma activity with a beta sensitivity of 5.8 x 10^{-7} μCi/cc for noble gases. This system is sensitive to both iodines and particulates.

The reactor coolant system isotopic inventory is determined by sampling and analysis to predict any change in isotopic spectrum that would lead to measurable quantities of iodine release.

The unit vent is provided with integrating type air samplers. A sample from the unit vent is drawn continuously through a particulate filter and an iodine sampling device. The sensitivity of the analysis for particulates is such that as low as 1.5 x 10^{-4} μCi collected on the filter will be measurable.

The methods and formulas for computation of doses associated with the liquid and gaseous releases are given in the Cook Nuclear Plant's Off-site Dose Calculations Manual (ODCM).

Liquid Release Pathways

Radioactive liquids are released through the Waste Disposal System Monitor Tanks, Laundry Tanks, Hot Shower Tanks, Chemical Drain Tanks, and Waste Evaporator Condensate Tanks. Activity and concentration are monitored and recorded on the Liquid Effluent monitor by batch release. Before a batch may be released, the tank is sampled and the sample analyzed. If the radioactivity level of the sample is found to be within acceptable limits, the liquid wastes will be released, monitored, and recorded. At the same time, the rate of the liquid release is measured by a flow meter. By using the rate of liquid waste releases, the rate of flow of the condenser cooling water, the activity of the liquid waste released, the rate of activity release and the concentration of activity in the condenser cooling water can be determined.

Liquid effluent and dilution volumes released are recorded. Gross beta-gamma counts are made on the liquid effluent prior to each batch release. The liquid effluent monitor sensitivity is 3.7×10^{-5} μCi/cc referenced to Cs-137.

The Radiation Monitoring System is divided into the following sub-systems:

a. The Process Radiation Monitoring System monitors various fluid streams for indication of increasing radiation levels.

b. The Area Radiation Monitoring System monitors radiation in certain areas of the plant.

c. Environmental Radiation Monitoring System Monitors radiation in the area surrounding the plant as described in Sub-Chapter 2.7.

11.3.3 GENERAL DESCRIPTION AND OPERATION

The original radiation monitoring channel equipment, including chassis with signal conditioning equipment, controls, power supplies, indicators and alarms is centralized in cabinets located in the control rooms for convenient operator access. Strip chart recorders are provided in these cabinets to sequentially record each monitoring channel.

This equipment has been supplemented and partially replaced by a system of distributed, multi-channel field data acquisition units. Each field unit services one or more detector channels. It measures and records the channel readings, performs alarm and other status checks and initiates trip functions (if applicable). Each field unit is connected via isolation devices to two data communication lines. Each line terminates at its associated system control terminal (CT). A CT is located in each control room and provides the control room operator with current channel status. A printer provides a record of the channels on a regular basis. Channel status changes are reported and recorded as they occur.

The sensitivity ranges of the various radiation monitor channels are given in Table 11.3-1 and are based on the first isotope listed in the last column of the table.

The monitor channels are checked using an internal check source, tested electronically, and calibrated by pulse injection methods and the detector calibrated using appropriate calibrated sources according to the schedule listed in the Technical Specifications. Setpoints for all release monitors are set so that the maximum release will not exceed the levels specified in the Technical Specifications.

11.3.3.1 Process Radiation Monitoring System

This system consists of (original and newer) channels which monitor radiation levels in various plant operating systems. High radiation level alarms are annunciated and identified in the control room.

The radiation monitoring channels employ instrument failure alarms at the radiation monitoring cabinets, control board annunciator, and at local indicators (where provided). Control interlocks fail in the 'high radiation' position upon instrument failure and must be manually reset. Instrument failure alarms are initiated upon failure of the radiation monitor, loss of detector signal or loss of power.

Gaseous effluents are scanned, alarmed, and recorded thereby providing a complete history of abnormal occurrences for evaluation.

These radiation monitoring channels are:

1) Containment - Air Particulate Monitor (ERS-1301, -1401, -2301, -2401)

These channels monitor air-particulate gamma radioactivity in the containment. A high radiation alarm initiates containment ventilation isolation.

The monitors take continuous air samples from the containment atmosphere. The samples are drawn from the containment through a closed, sealed system monitored by a scintillation counter-filter paper detector assembly. Filter paper collects 99% of particulate matter greater than 1 micron in size, on its surface, and it is viewed by a photomultiplier-scintillation crystal combination. The filter paper is changed on a routine schedule.

The return lines are routed back to the containment.

2) Containment Radio Gas Monitor (ERS-1305, -1307, -1309, -1405, -1407, -1409, -2305, -2307, -2309, -2405, -2407, -2409)

These channels monitor gaseous radioactivity in each containment. A high radiation alarm initiates containment ventilation isolation.

These channels take continuous air samples from the containment atmosphere after it passes through the air particulate and iodine monitors, and draws the sample through a closed, sealed system to the gas monitor assembly. The samples are returned to the containment.

3) Containment Radioiodine Monitor (ERS-1303, 1403, 2303, 2403)

These monitors are provided to measure radioiodine activity in the containment. These channels take continuous iodine air samples from the containment after it passes through a fixed air particulate filter. The gaseous vapors of iodine are absorbed in an activated charcoal cartridge where they are viewed by a sodium iodine scintillation detector.

Several components are common to the Containment Air Particulate, Radioactive Gas Monitors and Radioactive Monitors, including the flow control assembly, sample pump, and selected valves. The ERS-1300, 1400 field units can be remotely operated from the control room.

Alarms are provided to indicate high or low flow conditions.

4) Steam Jet Air Ejector Gas Monitor (SRA-1900, 2900)

These channels monitor the discharge from the condenser air ejector exhaust header for gaseous radiation which is indicative of a primary to secondary system leak. A gamma sensitive Geiger-Mueller tube is used to monitor the Mid- (channel 7) and High-range (channel 9) noble gases. The low-range (channel 5) noble gas is monitored by a sensitive beta scintillation detector.

5) Component Cooling Water Loop Liquid Monitor (R-17 A & B)

These channels continuously monitor the component cooling water system for radiation indicative of a leak of reactor coolant from the Reactor Coolant system and/or the Residual Heat Removal System into the Component Cooling Water System. Monitoring is performed using an in-line, well-mounted scintillation counter.

6) Waste Disposal System Liquid Effluent Monitor (RRS-1001)

This monitor continuously samples and measures all Waste Disposal System batch releases from the Plant. Automatic valve closure is initiated by this monitor to prevent any further release after a high-radiation level is indicated and an alarm is initiated.

7) Steam Generator Blowdown Liquid Monitor (R-19)

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak. Samples from the bottom of each of the four steam generators are mixed in a common header and the common sample is continuously monitored by a scintillation counter in a fixed volume. A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

A high-radiation signal from R-19 will close the containment isolation valves in the blowdown lines, the sample lines, and the blowdown tank condensate drain line.

8) Essential Service Water Liquid Monitors (R-20, -28)

These channels continuously monitor the essential service water system. In a post loss-of-coolant accident condition, these monitors provide a means to detect leakage in the containment spray heat exchangers.

9) Steam Generator Blowdown Treatment System Liquid Monitor (R-24)

This monitor, positioned after the second resin bed, measures the activity in the blowdown liquid after it passes the treatment demineralizer. It provides a means of monitoring the performance of the demineralizers and isolates the steam generator blowdown system in the event of saturation of the demineralizers.

10) Unit Vent Air Particulate Monitor (VRS-1501, -2501)

These monitors measure air particulate gamma radioactivity in the unit vent and to ensure that the release rate through the unit vent is maintained below specified limits.

The particulate channels take continuous air sample from the unit vent. The samples pass through a particulate filter which is monitored by a beta scintillation detector. The sample gas is then returned to the unit vent.

11) Unit Vent Radio Gas Monitor (VRS-1505, -1507, -2505, -2507)

These detectors are provided to measure gaseous gamma radioactivity in the unit vent effluents and to ensure that the radiation release rate is maintained below specified limits.

These monitors have a variable setpoint. This will serve two purposes. When the gas decay tanks are not being released, the setpoints can be kept low so as to detect small gaseous leaks in the auxiliary building. When the gas decay tank is being released, the setpoint value will be increased, to avoid automatic closure of the gas decay tank isolation valves upon high alarm.

These detectors monitor continuous air samples from the unit vent effluents after the air sample passes through particulate and iodine filters. The monitors' return lines are routed back to the unit vent.

11.3.3.2 Area Radiation Monitoring System

This system consists of channels which monitor radiation levels in various plant areas. Certain of these monitors have been upgraded. Both the original and newer monitors are shown in Table 11.3-1.

Each original monitor consists of a fixed position Geiger-Mueller detector with local indicator and check source. An associated readout drawer in the control room provides high radiation and failure alarms, and initiates trips (if required). Channel readings are logged on a multi-point recorder.

Newer monitors consist of either Geiger-Mueller or ion chamber detectors, with check sources, connected to multi-channel field-mounted data acquisition units. Each field unit reads its detectors, performs status checks, initiates trips (if required), records the readings, provides local readout and reports to the system control terminals. The control terminals poll the field units and provide channel readings and status information to the control room operators through displays, annunciators and printers. Selected channels are provided with individual indicator/alarm units near the detector.

Two high range ion chamber detectors monitor each containment. One is located in the upper containment while the second is in the lower containment about 180° apart from the first. These accident monitors are separate from the other area channels. Each has a dedicated readout module in the control room with a multi-range indicator, status lights, and test circuits. An isolated output from each module is sent to an associated field unit to provide for recording and supplemental data access via the system control terminals.

11.3.4 Reactor Coolant Activity Monitoring

Refueling shutdown programs at operating Westinghouse PWR's indicate that, during cooldown and depressurization of the Reactor Coolant System (RCS), a release of activated corrosion products and fission products from defective fuel has been found to increase the coolant activity level above that experienced during steady state operation. However, interruptions in the refuelings attributable to high coolant activity, are avoided by implementing established shut-down procedures. These procedures include purification of the RCS through the cation and mixed bed demineralizers and system degassification.

Table 11.3-2 illustrates the calculated coolant activity increases of several isotopes for the Donald C. Cook Plant. This table lists the calculated activities during steady state operation before refueling shutdown outage and calculated peak activities during plant cooldown operations. These data are based on measurements from an operating PWR which is similar in design to the Cook Nuclear Plant and has operated with fuel defects. The measured activity levels are also included in Table 11.3-2.

The dominant non-gaseous fission product released to the coolant during system depressurization is found to be Iodine-131. The activity level in the coolant was observed to be higher than the normal operating level for nearly a week following initial plant shutdown. Although lesser in magnitude, the other fission produce particulates (e.g., cesium isotopes) exhibited a similar pattern of release and removal by purification. It is reasonable to project this data to the Cook Nuclear Plant since the purification constants are similar and as it is standard operating procedure to purify the coolant through the demineralizers during plant cooldown. Fission gas data from operating plants indicate a maximum increase of approximately 1.5 over the normal coolant gas activity concentration. However, the system degassification procedures are implemented prior to and during shutdown, and have proven to be an effective means for reducing the gaseous activity concentration and controlling the activity to levels lower than the steady state value during the entire cooldown and depressurization procedure.

The corrosion product activity releases have been determined to be predominantly dissolved Cobalt-58. From Table 11.3-2, it is noted that this contribution is less than 1% of the total expected coolant activity and is hence considered to be a minor contribution.

Since continued operation of the purification system is standard operating procedure during plant cooldown and since means for system degassification are available for fission gas removal, the total activity concentration in the coolant can be maintained within Technical Specification limitsthroughout the plant shutdown, while considering the additional activity inventory released during system cooldown and depressurization. The coolant activity concentrations and inventories during the shutdown and prior to plant startup are established by chemical analysis of samples for the Reactor Coolant System.

11.3.5 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

The following cart-mounted, continuous air monitors are available for use in an emergency that may involve airborne radioactivity concerns:

Two particulate and radioiodine monitors. The radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridge. The detector is connected to a single channel analyzer calibrated to 365 KeV. These monitors are dedicated to the TSC and the control room.

Five airborne particulate, radiogas, and radioiodine monitors. The radioiodine monitor, usually used with TEDA impregnated charcoal, will also accept the Silver Zeolite cartridges. The detector is a stabilized NaI detector connected to a two channel analyzer calibrated to 365 KeV with automatic Xe subtraction from the second channel.

All cart-mounted iodine detectors are in three (3) inch lead shields.

In addition, there are available for use throughout the plant ten regulated air samplers, which accept either TEDA impregnated charcoal or Silver Zeolite cartridges.

In addition to the equipment normally available in the regular radiochemistry counting facility, the following analysis equipment is available for analysis of the Silver Zeolite cartridges that might be used in an emergency:

a. A 4" x 4" NaI crystal connected to a multichannel analyser is located in the low background counting facility.

b. In the basement assembly area there is a cartridge purge unit consisting of a T-size bottle of dry nitrogen, regulator, cartridge holder, and associated piping to permit purging of Silver Zeolite or charcoal cartridges with dry nitrogen.

c. Located in the basement assembly area is a single channel analyzer calibrated to 365 KeV, connected to a 2" x 2" NaI crystal in a 2 1/2" lead shield designed for counting in TEDA charcoal or Silver Zeolite cartridges.

12.2 Licensed Operator Requalification Program

This section has been revised in accordance with NUREG-1262, "Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operator Licenses."

A replacement training program (RO and SRO) for licensed operators, non-licensed operators, and shift technical advisors is maintained in accordance with a program based on a systems approach to training which was accredited by the National Nuclear Accrediting Board on October 15, 1986, in accordance with NRC Generic Letter 87-07. The simulator was accredited to meet the requirements of ANSI/ANS 3.5-1985 on August 24, 1990. The requalification program is also maintained in accordance with a program based on a systems approach to training for which accreditation was renewed by the National Nuclear Accrediting Board on August 23, 1990.

12.5 REVIEW AND AUDIT OF OPERATIONS

To ensure safety and efficiency of operation of the plant, administrative procedures have been established to review the following:

1. All plant operating procedures
2. Changes in plant operating procedures
3. Unusual occurrences
4. Proposed tests and experiments
5. Design modifications
6. Violations of the Technical Specifications
7. Proposed changes in the Technical Specifications

Two committees have been established for this purpose, one at the plant itself, and the other at the AEP Service Corporation. The members of these committees, their responsibilities and their authority, have been noted in the administrative control sections of the Technical Specifications.

Audits of facility operations are conducted as previously described in Section 1.7.

12.6 NUCLEAR DESIGN AND SUPPORT CAPABILITY

The American Electric Power Service Corporation (AEPSC), with offices at 1 Riverside Plaza, Columbus, Ohio, 43215, provides engineering operational support, design, legal, accounting and related services to Indiana Michigan Power Company, including Cook Nuclear Plant, and the other AEP system operating Companies. Consequently, AEPSC employs engineers, designers, and drafters who are experienced in the design and construction of electric generating stations. AEPSC acts as the architect-engineer for the AEP system and as such has designed and built nearly all of the System's present generating capacity and is performing a like function for most of that presently under construction.

AEPSC was responsible for the design of the Donald C. Cook Nuclear Plant and for construction of the entire plant. Design and fabrication of the nuclear steam supply system components and fuel was performed by the Westinghouse Electric Corporation and its subcontractors.

AEPSC began training employees in nuclear power in 1952 with the assignment of several engineers, designers and maintenance specialists to Oak Ridge National Laboratory, Bettis Atomic Power Laboratory, Knolls Atomic Power Laboratory, and various projects at the National Reactor Testing Station. Since that time, a large number of additional AEP personnel have completed assignments at various national laboratories or pursued graduate level work in nuclear engineering at leading universities, while others have attended shorter courses and seminars in various aspects of the nuclear power industry.

In 1953, AEP became one of the co-founders of the Nuclear Power Group, Inc., and in the ensuing years participated, technically and financially, in the development of the Dresden Nuclear Power Station. This group was then dissolved. It evolved into the East Central Nuclear Group (ECNG); and AEP was instrumental in the new group's formation. ECNG was comprised of 10 utility companies, including I&M. Its goal was to research and develop nuclear power. The AEP Service Corporation acted as architect-engineer administrator and research and development manager for the group.

ECNG's major undertakings were the development with the General Nuclear Engineering Corp. of the Florida West Coast Nuclear Group gas-cooled, heavy water moderated reactor from 1957-61, the joint development with Babcock & Wilcox of a Supercritical Pressure Steam Cooled Fast Breeder Reactor from 1963-65, the development of a Gas Cooled Fast Breeder Reactor in cooperation with Gulf General Atomic from 1965-67, the development with General Electric of a Steam Cooled Fast Breeder Reactor in 1967-1968, and from 1968 through 1982, a further project with General Atomic for the Gas Cooled Fast Breeder Reactor, first through an informal group of utilities and then through Helium Breeder Associates.

In addition, ECNG, with the aid of AEPSC staff and S. M. Stoller Associates, made a thorough study of the "The Outlook for Uranium", a survey of the likely demand and availability of nuclear fuel; and with the Massachusetts Institute of Technology produced a study of the "Effects of Changing Economic Conditions of Fuel Cycle Costs". This program investigated the ten-projected effects of private ownership of nuclear power economics. ECNG is now dissolved.

At the present time, the AEPSC Nuclear Engineering and Nuclear Operations Divisions consist of scientists and engineers who devote all of their energies to Cook Nuclear Plant and nuclear power industry issues. In addition, there are other individuals at AEPSC with substantial nuclear training or specific nuclear experience in key engineering, design and operating positions.

12.7 WRITTEN PROCEDURES

Detailed written procedures for normal operations, as well as for abnormal and emergency situations, have been prepared. These procedures incorporate the limits and parameters set forth in the plant Technical Specifications. The Emergency Plan includes provisions to provide the necessary facilities and personnel to deal effectively with any foreseeable emergency. However, the plant design is such that none of the credible nuclear accidents would create an undue hazard to the public. Station personnel are thoroughly familiar with the emergency plan, and practice drills are held as necessary for training.

The review and approval for all plant operating, maintenance and test procedures is described below:

- After the original or revised procedure is reviewed by supervisory personnel, it is approved by the appropriate Department Head and QA Supervisor. (Interfacing departments also give approval for those procedures affecting their departments.)

- The PNSRC reviews those procedures it is required to review by Technical Specifications (T/Ss) 6.5.1 and 6.8. The PNSRC renders a written decision on whether there is an unreviewed safety question, and recommends approval or disapproval by the Plant Manager.

- The Plant Manager (or Acting Plant Manager, per T/S 6.1 delegation) has final approval authority.

- Temporary changes are approved in accordance with T/S 6.8.3. It provides for change implementation upon approval by two members of the plant management staff, at least one of whom holds a Senior Reactor Operators License. Subsequently, final review of the documented change is made by the PNSRC and approved by the Plant Manager (or acting Plant Manager).

14.0 SAFETY ANALYSIS

This chapter presents an evaluation of the safety aspects of Unit 1 of the Donald C. Cook Nuclear Plant and demonstrates that Unit 1 can be operated safely even if highly unlikely events are postulated. It also shows that radiation exposures resulting from occurrences of these highly unlikely accidents do not exceed the guidelines of 10 CFR 100.

Cook Nuclear Plant Unit 1 is currently loaded with fuel manufactured by Westinghouse Electric Corporation, and this chapter reports on those safety analyses performed in support of operation with the current Westinghouse fuel. The current analyses additionally support operation over a range of operating reactor coolant system (RCS) temperatures and at two discrete RCS pressures, as described in Table 14.1-1.

This chapter is divided into four sections and two appendices. The first three sections each deal with a different category of fault condition, and the last is concerned with analyses performed in support of environmental qualification of structures, systems, and components.

These four sections are introduced further in the following paragraphs:

Core and Coolant Boundary Protection Analysis

The fault conditions discussed in this section may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no fault in this category shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

Standby Safeguards Analysis

The fault conditions discussed in this section are more severe but very infrequent and may lead to a breach of fission product barriers.

Primary System Pipe Rupture

The accident discussed in this section is a rupture of a reactor coolant pipe including the double ended severance of the largest pipe in the reactor coolant system, which is the worst conceivable accident and therefore is used as a basis for the design of engineered safeguards.

Environmental Qualification Analyses

This section discusses where analyses applicable to the environmental qualification of equipment important to safety may be found. It also provides one example of such analyses, that associated with high energy line breaks outside of containment.

For most accidents which are DNB limited, nominal values of initial conditions and the minimum measured flow (366,400 gpm) are assumed. The allowances on power, temperature and pressure are determined on a statistical basis and are included in the limit DNBR as described in WCAP-8567 (Reference 2). This procedure is known as the "Improved Thermal Designed Procedure" (ITDP).

Since the performance of the analyses using ITDP with a minimum measured flow of 366,400 gpm, the Technical Specifications value of minimum measured flow has been changed to 361,600 gpm. This Technical Specifications change was supported by evaluations of the impact of a lower minimum measured flow on each of the accident analyses that use ITDP.[8] For each of these analyses, the calculated system transient was evaluated not to be affected by the relatively small flow reduction. DNB margin has been allocated to offset the penalty associated with the reduction in minimum measured flow. The results and conclusions of each of the safety analyses using ITDP, as reported here, remain valid.

For accidents that are not DNB limited or in which ITDP is not employed, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following steady state errors are considered:

A.	Core Power	+ 2% calorimetric error allowance
B.	Average RCS Temperature	± 4.5°F controller deadband and measurement error allowance
C.	Pressurizer Pressure	± 35 psi - steady state fluctuations and measurement error allowance
D.	Reactor Flow	Thermal Design Flow (354,000 gpm)

Those non-ITDP analyses using as input the thermal design flow of 354,000 gpm remain unaffected by the Technical Specifications reduction of minimum measured flow to 361,600 gpm. This is because application of the measurement of uncertainty[9] associated with minimum measured flow to the value of 361,600 gpm leaves a value still greater than the thermal design flow of 354,000 gpm.

Table 14.1-3 summarizes initial conditions and computer codes used in the analysis of accidents in Sections 14.1.1 through 14.1.12 and Sections 14.2.5, 14.2.6 and 14.2.8, and shows which accidents employed a DNB analysis using the Improved Thermal Design Procedure.

Power Distribution

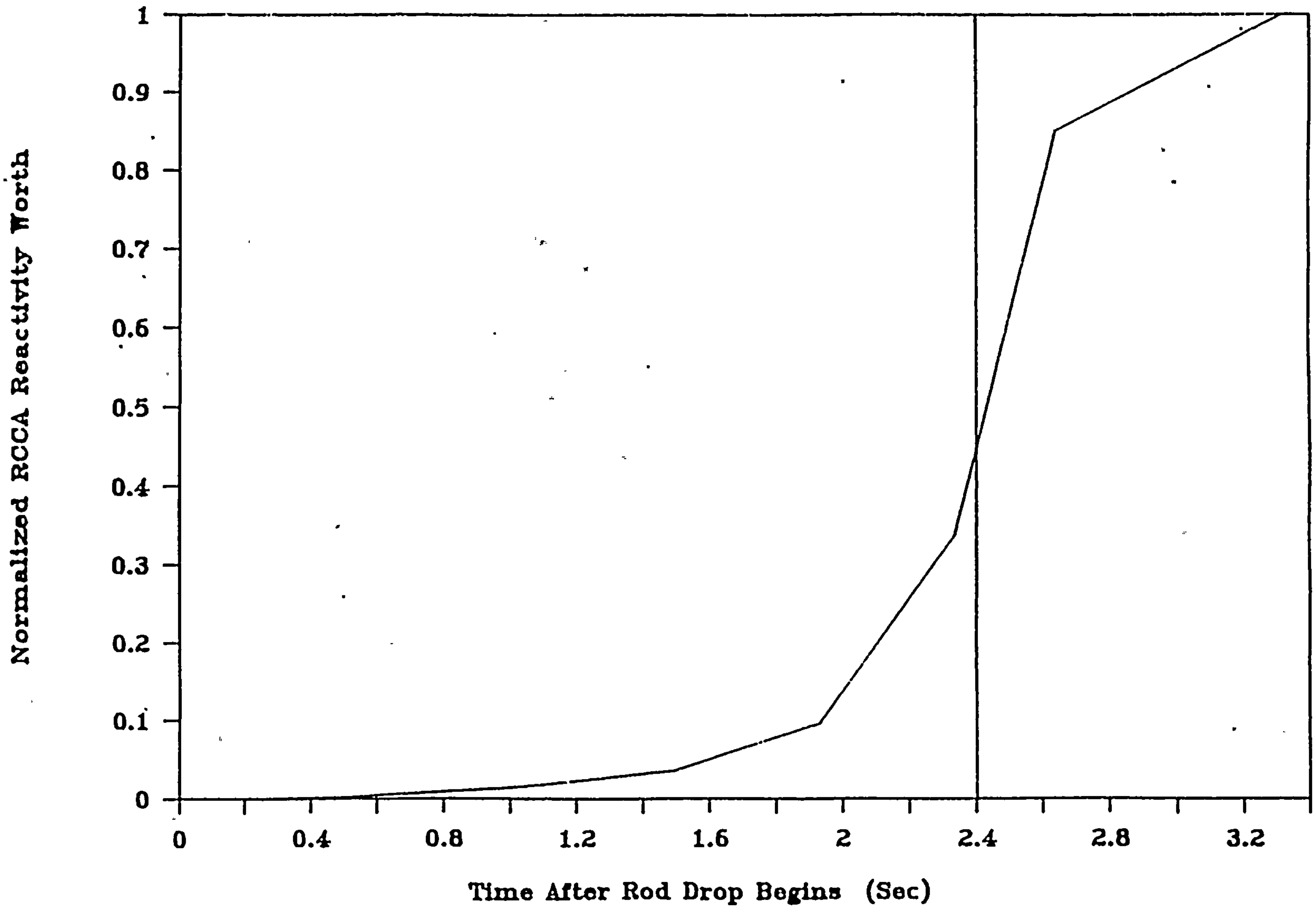
The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operation instructions. The power distribution may be characterized by the radial factor, $F_{\Delta H}$, and the total peaking factor, F_Q. The peaking factor limits are given in the Technical Specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in Technical Specifications.

The radial and axial power distributions are input to the THINC Code as described in Chapter 3.

For transients which may be overpower limited, the total peaking factor, F_Q, is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

COOK NUCLEAR PLANT UNIT 1 NORMALIZED NEGATIVE REACTIVITY INSERTION
AS A FUNCTION OF TIME USED FOR REACTOR TRIP IN TRANSIENT SAFETY ANALYSES



14.1.3 ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

Rod cluster control assembly misalignment accidents include:

A. A dropped RCCA

B. A dropped RCCA bank

C. Statically misaligned RCCA

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod bottom light. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the secondary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA or RCCA bank is detected by:

A. Sudden drop in the core power level as seen by the nuclear instrumentation system;

B. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

C. Rod bottom light;

D. Rod position deviation monitor;

E. Rod position indication.

Misaligned RCCA are detected by:

A. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

B. Rod position deviation monitor;

C. Rod position indication.

The resolution of the rod position indicator channel is ±5 percent of the 12 foot measurement span (± 12 steps). Deviation of any assembly from its group by twice this distance will not cause power distributions worse than the design limits. The rod position deviation monitor alerts the operator to rod deviation before it can exceed ten (10) percent of span (± 24 steps). If the rod position deviation monitor is not operable, the operator is required to take action as required by the Technical Specifications.

Method of Analysis

a. One or more dropped RCCAs from the same group.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code described in Section 14.1. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code described in Section 14.1. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 1.

b. Statically Misaligned RCCA

Steady state power distributions are analyzed using the methodology described in Reference 1. The peaking factors are then used as input to the THINC code to calculate the DNBR.

Results

a. One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion which may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.5 seconds following the drop of the RCCAs. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs which do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

For dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 14.1.3-1 and 14.1.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference 1. In all cases, the minimum DNBR remains above the limit value.

b. Dropped RCCA Bank

A dropped RCCA bank typically result in a reactivity insertion greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of a RCCA bank. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

c. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

14.1.3 References

1. Morits, T., et. al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10297-P-A (Proprietary) and WCAP-10298-A (Non-Proprietary), June 1983.

14.1.5 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Reactivity can be added to the core by feeding primary grade water into the reactor coolant system via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the reactor coolant system. The chemical and volume control system (CVCS) is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve supplies water to the reactor coolant system which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve. In order for makeup water to be added to the reactor coolant system, at least one charging pump must also be running in addition to the primary water pumps.

The rate of addition of unborated water makeup to the reactor coolant system is limited by the capacity of the primary water pumps. The maximum addition rate in this case is 225 gpm with both primary water pumps running. The 225 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating while the other is on standby.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board.

In order to dilute, two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode; second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the chemical and volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of the plant operation, boron dilution during refueling, startup, and power operation were examined. Included in the analysis was the effect of the difference in the density of the unborated water makeup water and the density of the reactor coolant. The analysis is to show that, from initiation of the event, sufficient time is available to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

The Technical Specifications have incorporated shutdown margin protection to ensure adequate operator response time for the Mode 4 and 5 dilution transient.(1) This is being done by applying the Westinghouse methodology described in Reference 2. The Technical Specification limits ensure that the operator has 15 minutes from initiation of event to loss of shutdown margin.

The Mode 4 and 5 boron dilution analysis is based on Cook Nuclear Plant Unit 1 plant conditions as listed below:

1. The RCS effective volume is limited to the vessel and the active portions of the hot and cold legs when on RHR, i.e., steam generator volumes are not included.

2. The plant is borated to a shutdown margin greater than or equal to 1% $\Delta k/k$.

3. Uniform mixing of clean and borated RCS water is not assumed, i.e., mixing of the clean, injected water and the affected loop is assumed but instantaneous, uniform mixing with the vessel, hot leg, and cold leg

volume upstream of the charging lines is not assumed. Thus a "dilution front" moves through the cold legs, downcomer, and lower plenum to the core volume as a single volume front. This results in subsequent decreases in shutdown margin due to dilution fronts moving through the active core region with a time constant equal to the loop transit time when on RHR. The RHR flow rate assumed is 2000 gpm, which is the Mode 5 maintenance level minimum RHR flow rate.

<u>Conclusions</u>

Because of the steps involved in the dilution process, an erroneous dilution is considered highly unlikely. Nevertheless, if it does occur numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

During certain types of operation, it is plausible that the refueling water storage tank (RWST) is at a lower boric acid concentration than the reactor coolant system water. Due to the large reactivity margins inherent in the design basis for the RWST boron concentration and slow dilution process, it has been determined that this need not be considered as a dilution source. See Reference 3.

14.1.5 <u>References</u>

1. Alexich, M. P. (I&M), letter to H. R. Denton (NRC), AEP:NRC:0916W, March 26, 1987

2. Anderson, T. M. (Westinghouse), letter to V. Stello (NRC), "Boron Dilution Concerns at Cold and Hot Shutdown," NS-TMA-2273, July 8, 1980

3. Youngblood, B. J. (NRC), letter to J. E. Dolan (I&M), "Interpretation of Technical Specifications that Apply to Borated Water Addition to the Reactor Coolant System from the Refueling Water Storage Tank," April 8, 1986

E. No credit is taken for the heat capacity of the RCS and steam generator thick metal is attenuating the resulting plant cooldown.

F. The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps and trips the turbine.

Normal reactor control system and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high-high steam generator water level conditions.

Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is 113 pcm/sec, where 1 pcm is 10^{-5} $\Delta k/k$.

An analysis has been performed to demonstrate that the applicable DNB criteria are met. A conservative reactivity insertion rate of 120 pcm/sec is assumed to bound the reactivity insertion rate calculated for the zero power feedwater malfunction analysis. The method of analysis used is the same as discussed in Section 14C.3.1 (Uncontrolled RCCA Withdrawal From A Subcritical Condition Analysis), except that the analysis assumed four (4) reactor coolant pumps to be in operation as required by the Cook Nuclear Plant Unit 1 Technical Specifications in Mode 2. Although the reactivity insertion rate for the zero power feedwater system malfunction is calculated assuming reactivity parameters representative of EOL core conditions to maximize the reactivity insertion rate, the DNB analysis is conservatively performed at BOL conditions to yield a high value of peak heat flux.

The DNB analysis performed for the hot zero power feedwater malfunction analysis with an insertion rate of 120 pcm/sec yields a minimum DNBR which remains above the safety analysis limit value.

One additional case of the feedwater flow malfunction at zero power is analyzed in which the amount of excess flow to one steam generator is assumed to be equally divided amount all 4 loops (total excess feedwater flow no greater than 200% of loop full power flow). The results show that this case is bounded by the single-loop feedwater flow malfunction analysis.

The full power case (maximum reactivity feedback coefficients, automatic rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the manual rod control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event.

For all excessive feedwater cases, continuous addition of cold feedwater is prevented by automatic closure of all feedwater isolation valves on steam generator high-high level signal. In addition, a turbine trip is initiated. A reactor trip on turbine trip was then assumed as a means of terminating the transient analysis. The reactor trip prevents reactor coolant heatup consistent with the cooldown characteristics of the feedwater malfunction event. The reactor trip on turbine trip was assumed as an anticipatory trip. If the reactor trip was not assumed, the transient would progress into a heatup event, in particular, a loss of normal feedwater due to the isolation which occurs on high-high steam generator water level signal. A reactor trip would then be provided by a low-low steam generator water level signal. The reactor trip on turbine trip was not required for core protection for this event. The results (minimum DNBR) of the feedwater malfunction analysis would be essentially unchanged if the reactor trip was not assumed to occur on turbine trip.

Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

Following isolation of the faulty steam generator, the water volume in the pressurizer will be approximately 10% of the pressurizer volume. Hence the RCS water volume at isolation will be approximately 92% of the water volume during normal operation. Further, at isolation, the faulty steam generator will be nearly 80% water full. Note that only in the event of the accident occurring at EOL will the faulty S.G. water inventory be nearly unborated.

During the time period that the plant is held in thermal equilibrium as specified by the Emergency Operating Procedures, the RCS and secondary side of the faulty S.G. are in equilibrium because of the communication via the broken tube. Therefore, fluid is not transferred between the two systems. Further, the S.I. system is delivering into the RCS with a boron concentration of 2000 ppm.

During the subsequent cooldown, heat will be transferred across the faulted S.G. tubes from the secondary to the primary, and any secondary to primary leakage through the broken tube will be small. Further, the S.I. system will be injecting water with a boron concentration of 2400 ppm during the cooldown process; therefore, any decrease in this boron concentration will be insignificant.

The shutdown margin following reactor trip will be at least 1.6 percent. It would increase substantially following actuation of the SIS due to the increasing boron concentration and at equilibrium would be greater than 10 percent. Therefore, any nuclear impact from the nearly unborated water is insignificant and the shutdown margin is only slightly different from any other shutdown resulting from actuation of the safety injection system.

The decision as to how the system would be cleaned up would be highly dependent on chemical analysis of samples taken from various parts of the system subsequent to the accident. A combination of CVCS demineralizers and evaporators, steam generator blowdown treatment system, and the waste evaporator and drumming facility would probably be required.

The recovery and cleanup would be relatively long term due to the secondary side chemicals present, and might require installation of special resins in a demineralizer to remove these.

The preferred recovery operations for which emergency procedures have been developed are given below.

The objectives of the procedure are:

a. To minimize the release of radioactive material to the outside atmosphere by reducing reactor coolant pressure below the steam generator safety valve setting.

b. To maintain the ability to remove the necessary residual heat from the reactor coolant system.

c. To maintain the reactor coolant in a subcooled state during recovery to assure proper cooling of the core.

d. To prevent flooding of the faulty steam generator and steam lines.

e. To minimize the transfer of secondary coolant to the reactor coolant system.

The operator carries out the following procedures which lead to isolation of the faulty steam generator and subsequently to unit cooldown. Note that the protection system will automatically actuate SIS, reactor and turbine trips, trip main feedwater pumps, and actuate auxiliary feedwater systems.

<u>With Offsite Power Available</u>

1. Regulate auxiliary feedwater flow to steam generators to maintain a minimum on scale water level. If water level increases in one steam generator, completely isolate auxiliary feedwater flow to that steam generator.

Additional assumptions for the steam line break are:

11. In the affected steam generator, all the water boils off and is released through the break immediately after the accident. A retention factor for iodine releases is assumed to be 0.1.

12. The primary pressure remains constant at 2235 psig for 0-2 hours and decrease linearly to atmospheric from 2235 psig during the period of 2-8 hours.

Additional assumptions for the steam generator tube rupture are:

11. Steam dump to atmosphere and the associated activity release from the non-defective steam generators is terminated at eight hours after the accident, when the residual heat removal system starts to take over cooling down the plant.

12. Thirty minutes after the accident, the pressure between the defective steam generator and the primary system is equalized. The defective steam generator is isolated. No steam and fission product activities are released from the defective steam generator thereafter.

The steam releases for the steam line break and for the steam generator tube rupture are given in Tables 14.2.7-2 and 14.2.7-3, respectively.

The thyroid and whole body doses at the site boundary and at the outer boundary of the low population zone for the steam line break accident are given as a function of primary to secondary leak rate in Figures 14.2.7-7 and 14.2.7-8.

For the steam generator tube rupture, the thyroid and whole body doses at the site boundary and at the outer boundary of the low population zone, for the original analysis power level of 3391 MWt, are presented, as a function of primary-to-secondary leak rate, in Figures 14.2.7-9 through 14.2.7-12. The doses were subsequently reevaluated at the uprated power level of 3588 MWt. This evaluation also incorporated revised reactor coolant fission product inventories which were based on current calculational methodology. Doses at the uprated power, for a 1 gpm primary-to-secondary leak rate (maximum leak rate allowed by Technical Specifications), are estimated as follows:

0 - 2 hour dose at the site boundary, rem

thyroid - 1.7
whole body - 2×10^{-1}

0 - 8 hour dose at the low population zone, rem

thyroid - 4×10^{-1}
whole body - 5×10^{-2}

The 10 CFR 100 exposure guideline is defined as 300 rem thyroid and 25 rem whole body. The doses estimated for the SGTR, considering break flow, steam releases, and source term effects, remain within a "small fraction" (10%) of the 10 CFR 100 exposure guidelines. Small fraction is the smallest of the exposure guidelines defined in NUREG-0800.

TABLE 14.2.7-3

STEAM GENERATOR TUBE RUPTURE

STEAM RELEASE

	0-2 Hours	2-8 Hours
Steam release from defective S.G., lbs (0-30 min)	55,060 (50,000)*	0
Steam release from 3 non-defective S.G.'s, lbs	413,000	978,000
Feedwater flow to 3 non-defective S.G.'s, lbs	613,000	1,074,000
Reactor coolant to the defective S.G., lbs (0-30 min)	140,265 (125,000)*	0

*Values in parenthesis were calculated for the original analysis power level of 3391 MWt.

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974)[1].

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three destinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974)[4]. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)[5]; Kelly et al. (1974)[6]; Young et al. (1987)[7]; and Bordelon et al. (1974)[4]. Code modifications are specified in References 8, 9, 10 and 11. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling throught and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill phase of the accident. The BASH code is used to determine the system response during the reflood phase of the transient. The LOTIC computer code, described by Hsieh and Raymund in WCAP-8355 (1975) and WCAP-8345 (1974)[12], calculates the containment pressure transient.

The containment pressure transient is input to BASH for the purpose of calculating the reflood transient. The LOCBART computer code calculates the thermal transient of the hottest fuel rod in the three phases. The Revised PAD Fuel Thermal Safety Model, described in References 13, generates the inital fuel rod conditions input to LOCBART.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, information on the state of the system is transferred to the WREFLOOD code which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriated to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH[7] computer code.

Figures 14.3.1.14-15 The containment pressure transient used in the analysis is provided for the minimum and maximum SI cases.

Figures 14.3.1.16-21 These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray (mainimum and maximum cases).

Figures 14.3.1.22-23 These figures show the temperature transients in both the lower and upper compartments of containment and flow from the upper to lower compartments.

The maximum clad temperature calculated for a large break is 2180.5°F, which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 11.23 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

REFERENCES, SECTION 14.3.1

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6. Kelly, R. D. et al., "Calculation Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-proprietary), 1974.

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REFERENCES, SECTION 14.3.1 (Cont'd)

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13. "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2 (Proprietary) and WCAP-8785 (Non-proprietary).

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17. Rahe, E. P. (Westinghouse). Letter to Robert L. Tedesco (USNRC), Letter No. NS-EPR-2538, December 1981.

TABLE 14.3.1-1
LARGE BREAK LOCA
RESULTS

	Case A CD=0.6 Thot=611.2	Case B CD=0.6 Pres=2100	Case C CD=0.4 Thot=611.2	Case D CD=0.6 Thot=580.7	Case E CD=0.8 Thot=611.2	Case F CD=0.6 Max SI	Case G CD=0.6 RHR Cross Tie Closed
Peak Clad Temperature (^{o}F)	2162.7	2160.1	2096.6	2095.0	1940.9	2180.5	2162.0
Peak Clad Location (ft)	6.0	6.0	6.0	6.25	6.25	6.0	6.0
Local Zr/H_2O Reaction (Max %)	10.43	10.25	9.22	9.77	5.54	11.23	11.01
Local Zr/H_2O Location (ft)	6.0	6.0	6.0	6.0	6.0	6.0	6.0
Total Zr/H_2O Reaction (%)	<0.30	<0.30	<0.30	<0.30	<0.30	<0.30	<0.30
Hot Rod Burst Time(s)	38.38	37.66	46.43	40.87	44.71	38.38	38.80
Hot Rod Burst Loc. (ft)	6.0	6.0	6.0	6.0	6.0	6.0	6.0

Calculation

Licensed Core Power (MWt) 102% of 3413

Peak Linear Power (kW/ft) 102% of 15.155

Peaking Factor (at License Rating) 2.15

Accumulator Water Volume (cu. ft.) per Accumulator 946.0

Cycle Analyzed Cycle 11

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TABLE 14.3.1-1 (Cont.)
LARGE BREAK LOCA
TIME SEQUENCE OF EVENTS

	Case A CD=0.6 Thot=611.2	Case B CD=0.6 Pres=2100	Case C CD=0.4 Thot=611.2	Case D CD=0.6 Thot=580.7	Case E CD=0.8 Thot=611.2	Case F CD=0.6 Max SI	Case G CD=0.6 RHR Cross Tie Closed
Start	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.64	0.55	0.64	0.51	0.63	0.64	0.61
Safety Injection Signal	4.60	4.14	4.90	4.07	4.46	4.60	4.55
Accumulator Injection	15.00	15.00	20.50	13.75	12.50	15.00	15.00
End of Blowdown	31.30	31.20	39.54	33.18	26.97	31.30	31.56
Bottom of Core Recovery	45.02	44.85	54.72	47.48	40.00	44.49	45.47
Accumulator Empty	60.55	60.46	67.31	61.17	57.03	61.07	60.35
Pump Injection	31.60	31.14	31.90	31.07	31.46	31.60	31.55

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- Hot spot fluid temperature,
- Cold leg break mass flow rate, (for the 2-inch case only), and
- Safety injection mass flow rate (for the 2-inch case only).

As seen in Table 14.3.2-3, the maximum clad temperatures were calculated to be less than that for the 3-inch break.

<u>Additional Analysis</u>

Calculations were also performed for Cook Nuclear Plant Unit 1 with the NOTRUMP [1,2] and LOCTA-IV[3] codes to examine the influence of initial loop fluid operating temperatures and operating pressures on small break LOCA peak clad temperature. These additional analyses confirmed that the most limiting PCT result was that from the reduced temperature and pressure limiting 3-inch diameter break described previously.

To support operation of the Cook Nuclear Plant Unit 1 RCS pressures of 2100 psia and 2250 psia for a range of loop operating temperatures, two additional analyses were performed. Calculations were performed for a 3-inch diameter break for an initial RCS pressure of 2250 psia at initial loop fluid operating temperatures corresponding to T_{avg} program setpoints of 547°F and 578°F. The results of these calculations are shown in the Results Table 14.3.2-5 and the Sequence of Events Table 14.3.2-6. Plots of the following parameters are shown in Figures 14.3.2-25 through 32 for the reduced temperature high pressure case, and Figures 14.3.2-33 through 40 for the high temperature high pressure case.

- RCS pressure,
- Core mixture level,
- Peak clad temperature,
- Core outlet steam flow,
- Hot spot rod surface heat transfer coefficient,
- Hot spot fluid temperature,
- Cold leg break mass flow rate, and
- Safety injection mass flow rate.

As seen in Table 14.3.2-5, the maximum clad temperatures were calculated to be less than that for the 3-inch break initiated at reduced temperature and pressure conditions.

NUREG-0737[5], Section II.K.3.31, required plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-65[6], generic analyses using NOTRUMP[1,2] were performed and are presented in WCAP-11145[7]. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting.

REFERENCES, Section 14.3.2

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4. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600, June 1979.

5. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

6. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31," November 2, 1983.

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TABLE 14.3.2-3

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATION

RESULTS

PARAMETER	VALUE Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Peak clad temperature (^{o}F)	1899	2122	1414
Elevation (ft)	12.00	12.00	11.25
Zr/H_2O cumulative reaction			
Maximum local (%)	7.16	7.70	0.25
Elevation (ft)	12.00	12.00	11.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

CALCULATION:

NSSS Power MWt 102% of	3588*
Peak Linear Power kW/ft 102% of	16.426
Hot Rod Power Distribution (kW/ft)	See Figure 14.3.2-2
Accumulator Water Volume, cu. ft.	946

*Does not include pump heat.

TABLE 14.3.2-4

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Small-Break Loss of Coolant Accident

Event	Time (s) Reduced Temperature, Reduced Pressure Break Size: 2-Inch	3-Inch	4-Inch
Break occurs	0	0	0
Reactor trip signal	25.37	11.24	6.85
Safety injection signal	36.54	17.10	10.74
Start of safety injection delivery	63.54	44.10	37.74
Loop seal venting	1634.4	652.1	420.4
Loop seal core uncovery	N/A	645.8	424.6
Loop seal core recovery	N/A	680.3	439.2
Boil-off core uncovery	2216.7	1045.7	696.5
Accumulator injection begins	N/A	1711.5	901,0
Peak clad temperature occurs	4143.8	1958.7	969.5
Top of core covered	N/A	N/A	1982.7
SI flow exceeds break flow	4587.5	2197.1	N/A

14.3.4 CONTAINMENT INTEGRITY ANALYSIS

This section presents the analyses to show that in the event of a high energy line break inside containment, containment design pressures will not be exceeded. A general description of the analytical methods is also presented, and this is followed by a more detailed description of the analyses.

Subsequent to the licensing of Donald C. Cook Nuclear Plant, Unit No. 1, additional analyses were performed in conjunction with the Unit No. 2 licensing process. In 1988, a new long-term containment integrity analysis was submitted to the NRC supporting operation with residual heat removal system cross-tie valves closed (reference 21). This analysis is included in Section 14.3.4 of the updated Unit 2 FSAR and is bounding for Unit 1. NRC approval of this analysis was provided through a safety evaluation dated January 30, 1989 (reference 22).

This updated FSAR subchapter is a condensation of previous material concerning containment integrity analysis. In making the condensation, certain explanatory material has been omitted. The omitted information can be found in the original FSAR material and associated questions, Appendix N in particular.

All the material preceding Section 14.3.4.1 has been added with the purpose of providing a detailed summary of the entire Section 14.3.4. This material has been obtained from the "Safety Evaluation by the Directorate of Licensing, U.S.A.E.C. in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company, Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316, September, 1973."

General Description of Analytical Methods

The containment systems for Cook Nuclear Plant include a reactor containment structure, a containment isolation system, and a combustible gas control system. The containment is equipped with redundant heat removal spray and air recirculation fan systems.

Cook Nuclear Plant was the first nuclear plant to use the ice condenser pressure suppression containment system. The basic performance and design evaluations of the ice condenser system have been the subject of both analyses and experimental programs.

The reactor containment is a steel-lined reinforced concrete structure consisting of a vertical cyclinder, a hemispherical dome, and a flat base. The design pressure for the containment is 12 psig. The containment volume of about 1,241,000 ft^3 is divided into three major subvolumes; a 368,000 ft^3 lower compartment enclosing the reactor system, a 127,000 ft^3 intermediate compartment housing an ice bed in which steam is condensed during a loss-of-coolant accident, and a 746,000 ft^3 upper compartment to accommodate air displaced from the other two compartments during the accident.

The intermediate or ice condenser compartment is an enclosed annular compartment encompassing most of the perimeter (300^{o}) of the containment structure. Borated flake ice is stored in cylindrical perforated metal baskets within the ice condenser compartment. The ice contained in baskets is provided to condense steam in the event of a loss-of-coolant accident (LOCA). During normal plant operation the ice bed is maintained at about 15^{o}F by a redundant refrigeration system. Refrigeration ducts and insulation on the ice condenser walls minimize heat losses from the ice. Thirty chiller units are provided for each containment, but only 21 are required to maintain the temperature within the ice bed. In the unlikely event of the complete loss of the refrigeration system, the insulation on the ice condenser is sufficient to prevent the ice from melting for a minimum period of 7 days during which the plant could be safely shut down.

Inlet and outlet doors are provided at the bottom and top of the ice condenser compartment. In the event of a LOCA, the rising pressure in the lower compartment pushes open the inlet doors. Air, entrained water, and steam then flow from the lower compartment into the ice condenser. The displaced air in the ice condenser forces open the doors at the top of the ice condenser and flows into the upper compartment. Steam is condensed in the ice condenser and does not reach the upper compartment. Developmental testing has confirmed this phenomenon.

The operating deck separating the upper and lower compartment ensures that the steam-air flow resulting from the LOCA is directed through the ice condenser to the upper compartment.

Condensation of the steam by the ice limits the containment pressure rise during reactor blowdown and helps reduce the containment pressure following the blowdown until melt-out (complete melting) of the ice bed occurs. Ice melt-out is predicted to occur about an hour after a design basis loss-of-coolant accident. Following ice melt-out, the containment pressure increase due to the release of decay energy from the core is limited by the containment spray system.

The lower compartment is divided into a number of subcompartments formed by equipment and internal structures. The containment pressure responses within these subcompartments were analyzed using the TMD (Transient Mass Distribution) computer code developed by Westinghouse. The code provides a means of computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. The pressure response within the subcompartments is different from the overall pressure response of the containment only during the early phase of blowdown, approximately the first second. Following this initial phase of the accident, the containment pressure responses of the upper and lower compartments were analyzed with the various versions of the Westinghouse LOTIC code.

Following a postulated reactor coolant pipe rupture, differential and local pressures build up in the subcompartments of the lower containment compartment. This occurs between 0.05 to 0.2 seconds following the break. These pressures depend on the volumes of the subcompartments, the nature of interconnecting flow paths, and the flow and thermodynamic behavior within the pressure nodes. During this phase of the transient, flow to the upper containment compartment is not significant. It is during this time period that the peak operating deck differential pressures and peak subcompartment differential pressures would occur. As the blowdown continues, however, the pressure in the upper compartment rises and eventually reaches a peak approximately equal to the lower compartment pressure. The upper compartment pressure peak is primarily the result of air being displaced from the lower compartment and forced up through the ice columns into the upper compartment.

Westinghouse uses versions of the SATAN code to determine the mass and energy addition rates to the containment during the blowdown phase of the accident.

As mentioned above, the Transient Mass Distribution code (TMD) is used to calculate the short term pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment including the containment compartment and subcompartment pressure response during the short period following a LOCA. The model includes a nodalization of 45 elements representing the containment and analyzes the pressure response of each of the subcompartments within the lower compartment (including the dead-ended compartments), the ice condenser compartment, and the upper compartment.

TMD was developed specifically to analyze the short term pressure response of the ice condenser system. The mathematical modeling in TMD is similar to that of the SATAN blowdown code in that the solution is developed by considering the conservation equations of mass, momentum, and energy, and the equation of state, and uses the control volume technique for simulating spatial variation. The governing equations for TMD are somewhat different from those in SATAN in that a two-phase (liquid water droplets and steam-air vapor), two-component (air-water) system is considered.

TMD calculates the critical flow of a two-component, two-phase fluid (air, steam and water) assuming a thermal equilibrium condition. However, a correction factor, developed by Westinghouse to account for experimental data on applicable flow regimes, is then applied to the calculated critical flow. The correction factor, as used in the code, increases the critical flow by up to 20% through the compartments as the quality of the fluid decreases and results in lower calculated differential pressures than in the uncorrected flow regime. This "augmented flow" factor is less conservative than the thermal equilibrium correlation. As a result of this fact, the short-term containment pressure response analysis without augmented flow was repeated. The safety evaluation was based on the containment subcompartment pressure without the augmented flow correlation.

The pressure response of the subcompartments of the containment lower compartment for both reactor coolant inlet and outlet pipe breaks using the non-augmented vent flow correlation is summarized in Table 14.3.4-30. In all cases the highest calculated containment subcompartment pressure is less than design.

The heat transfer model of the ice condenser as used in TMD is based on the result of some full-scale testing done during 1968 and 1969. The test program used an ice basket design which had a 64% open surface area.

Tests have also been conducted on the lower inlet doors to evaluate the dynamic characteristics of the doors during the initial transient. These tests verified the inlet door response characteristics that were calculated by the TMD code.

The containment spray system is activated after the completion of blowdown (about 30 seconds after the LOCA) and causes a slight reduction of the containment pressure. About 10 minutes after the accident, the return air fans are started and the containment pressure is reduced as air is returned from the upper volume to the lower volumes. Residual heat is still being removed almost entirely by the stored ice at this time. After ice melt-out (about 1/2-1 hour), residual heat is removed by the containment spray system. The containment experiences another pressure peak about 1.5 hours after the accident since the energy input exceeds the minimum heat removal capability of the sprays. The magnitude of this peak pressure is determined by the reactor residual heat input to the containment and heat removal by the containment spray system.

The mass and energy release rates to the containment were calculated during the reflood phase of the accident following blowdown, using a separate computer code. Proper analysis of the reflood phase of the event is important because it models the manner in which additional energy is removed from the secondary system during core refill. This is particularly true in the case of a pipe rupture in the pump suction portion of a cold leg, when because of the system design, the steam and entrained liquid carried out of the core during the accident passes through the steam generators. The water leaving the core and passing through the steam generators is assumed to be superheated to the temperature of the steam generator secondary fluid. Results of the FLECHT experiments indicate that the carryout fraction of fluid leaving the core during the reflood is about 80% of the incoming flow to the core. Therefore, the rate and amount of energy release to the containment during this phase becomes proportional to the reflood flow into the core.

The rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators, because of the low resistance flow paths between the steam generators and the broken pipe. Therefore, such a break location leads to calculation of the highest containment pressure.

The effects on containment pressure of steam from a LOCA bypassing the ice condenser were also analyzed. Drain lines in the floor of the refueling canal allow water sprayed into the upper compartment to return to the containment sump. These drains form a bypass path. Analysis of the effects of this bypass area (2.2 ft^2) indicates that a substantial amount of bypass area beyond 2.2 ft^2 could be accommodated without exceeding the design pressure of the containment. The design criterion for bypass area is at least 5 ft^2 and analysis suggests bypass well in excess of this (on the order of 35 to 50 ft^2) could be accommodated.

The containment design has been analyzed with regard to the release of hydrogen to the containment following a loss-of-coolant accident. This aspect of the analysis is discussed in Section 14.3.6.

Using the analytical methods described above, a long term peak containment pressure which is below the 12 psig design pressure of the containment structure was consistently calculated. The exact value of this peak depends on the assumptions used, and may be found in the Figures at the end of this section.

14.3.4.1 Description of the Ice Condenser Containment

14.3.4.1.1 General

The Westinghouse Ice Condenser Containment System is a design in which an insulated cold-storage compartment, filled with ice, rings the wall of the containment forming a low temperature heat sink. The ice bed is a completely enclosed annular compartment located around the perimeter of the upper compartment of the containment and penetrates the operating deck so that a portion extends into the lower compartment of the containment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment. For normal plant operation, these doors are designed to remain closed. At the top of the ice condenser is another set of doors, the top one of which is exposed to the atmosphere of the upper compartment. These also remain closed during normal plant operation.

The ice bed is held within the ice condenser in baskets arranged to promote heat transfer from steam to ice should the condenser be needed to serve its function. A refrigeration system maintains the ice in the solid state. Suitable insulation surrounding the ice compartment minimizes the heat transfer into the ice condenser enclosure.

In the event of a Design Basis Accident (DBA), the door panels located below the operating deck open due to the pressure rise in the lower compartment caused by the steam pressure due to DBA. This allows the air and steam to flow from the lower compartment into the ice condenser. The resulting buildup of pressure in the ice condenser causes the door panels at the top of the condenser to open and allow the air to flow into the upper compartment. The steam condenses quickly, as it enters the ice condenser compartment, thus limiting the peak pressure in the containment. Condensation of the steam within the ice condenser causes a pressure differential between the upper and lower compartments which results in a continual flow of steam from the lower compartment to the condensing surface of the ice. This reduces the time that the containment is at elevated pressure.

The divider deck separates the upper and lower compartments and ensures that the steam is directed through the doors into the bottom of the ice condenser. Only a small amount of steam bypasses the ice condenser through small openings in the divider deck. the melted ice drains through floor drains to the containment sump where it is available for recirculation by the emergency core cooling system.

14.3.4.1.2 Performance Criteria

The performance of the ice condenser containment is demonstrated by results and analysis of ice condenser tests performed on a full-scale section test at the Westinghouse Waltz Mill Site. These tests confirmed the ability of the ice condenser to perform satisfactorily over a wide range of conditions, exceeding the range of conditions that might be experienced in an accident inside the containment.

The ice condenser containment performance has been evaluated by testing the following important parameters. A partial list of parameters tested include blowdown rate, blowdown energy, deck leakage, compression ratio, drain performance, ice condenser hydraulic diameter, dead-ended volumes and long term performance. Analytic models have been developed to correlate and supplement these test results in the evaluation of the containment design. The results indicate that the analytical models are conservative and that the performance of the ice condenser containment is predictable relative to these variables.

The layout of the reactor containment compartments and ice condenser provides for effective and efficient use of the ice condenser to suppress pressure buildup.

The lower (Reactor Coolant System) compartment is bounded by the divider barrier such that essentially all of the energy released in this compartment is directed through doors at the bottom of the ice condenser.

Seals are provided on the boundary of compartments and hatches in the operating deck to prevent steam from bypassing the ice condenser.

Layout, size, and flow communication among compartments is arranged to minimize the containment volume compression ratio.

The energy absorption capacity of the ice condenser is at least twice that required to absorb all of the energy that can be released during the initial blowdown of the Reactor Coolant System for all reactor coolant pipe break sizes up to and including the hypothetical double-ended severance of the reactor coolant piping, or for any steam system pipe break size up to and including the hypothetical severance of the main steam line inside the containment, without exceeding the containment design pressure.

The ice bed geometry provides sufficient ice heat transfer area and flow passages so that the magnitude of the pressure transient resulting from an accident does not exceed the containment design pressure for all reactor coolant pipe breaks sizes up to and including the hypothetical double-ended severance of the reactor coolant piping.

The initial containment peak pressure and peak unsymmetrical containment pressure loads are determined by analysis. The analytical results are experimentally verified by comparison with the ice condenser tests. This analysis supplements the experimental proof of performance tests and provides pressure transients for application of the plant design.

The final peak pressure occurring at or near the end of blowdown is determined by a containment volume air compression calculation. A method of analysis of the final peak pressure was developed based on the results of full-scale section tests.

Steam bypass of the ice condenser during the postulated RCS blowdown is to be avoided. The divider deck and any other leakage paths between the lower and upper compartments are reasonably sealed to limit bypass steam flow. For the containment, the analysis considered bypass area as composed of two parts: a conservatively assumed leakage area around the various hatches in the deck, and a known leakage area through the deck drainage holes for spray located at the bottom of the refueling cavity.

Flow distribution to the ice condenser for any RCS pipe rupture that opens the ice condenser inlet doors, up to and including the double-ended RCS pipe rupture, is limited such that the maximum energy input into any section of the ice condenser does not exceed its design capability. The door port flow resistance and size provides this flow distribution for breaks that fully open the ice condenser inlet doors. For breaks that partially open the inlet doors, the lower inlet doors proportion flow into the ice bed limiting maldistribution.

Analysis of the ice condenser reactor containment performance has shown that the ice condenser alone is capable of preventing containment overpressure during the initial blowdown of the reactor coolant energy (or secondary plant energy), such that containment spray is not a requirement for overpressure protection. However, extremely small blowdown rates would not generate a differential pressure to open the ice condenser doors sufficiently. In this case the energy release, even at the extremely small rate, would eventually require containment spray operation to prevent overpressure.

For large pipe breaks, the containment final peak pressure is mainly determined by the displacement of air from the lower compartment into the upper compartment. Only a small amount of steam bypasses the ice condenser by passing through the operating deck and into the upper compartment. This steam bypass then adds a small amount to the final peak pressure.

For small pipe breaks, which generate less than the pressure drop required to fully open the spring-hinged doors, a larger than normal fraction of the break flow will pass through the deck and into the upper compartment. Also, for breaks less than approximately 10,000 gpm, only a fraction of the air may be displaced from the lower compartment by the incoming steam. For these small energy release rates, operation of the containment spray system eventually is required to limit the containment pressure rise.

Another case has been examined where it is postulated that a small break loss-of-coolant accident precedes a larger break accident which occurs before all of the coolant energy is released by the small break, (i.e., a double accident). During the small break blowdown, some quantity of steam and air will bypass the ice condenser and enter the upper compartment via leakage in the divider deck. The important design requirement for the case of a double accident is that the amount of steam leakage into the upper compartment must be limited during the first part (small break) of the accident so that only a small increase in final peak pressure results for the second part (double-ended break) of the postulated accident. The steam which reaches the upper compartment will then add to the peak pressure for the second part of the accident. Therefore, the containment spray system is used to limit the partial pressure of steam in the upper compartment due to deck bypass. The key elements which determine the double accident performance are the ice condenser lower doors, which open at a low differential pressure to admit steam to the ice condenser and limit the bypass flow of steam and thus the partial pressure of steam in the upper compartment, and the sprays which condense this bypass flow of steam and limit the partial pressure of steam in the upper compartment to a low value, less than 2 psia. The containment spray set point actuation pressure has been set at 3 psig to limit steam partial pressure to less than 2 psia in the upper compartment for the double accident use.

After a LOCA, the ice condenser has sufficient remaining heat absorption capacity such that, together with the containment spray system, subsequent assumed heat loads are absorbed without exceeding the containment design pressure. The subsequent heat loads considered include reactor core and coolant system stored heat, residual heat, substantial margin for an undefined additional energy release, and consideration of steam generators as active heat sources.

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident, thereby ensuring that containment pressure cannot exceed the containment design pressure. Protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Adequate containment heat removal capability for the Ice Condenser Containment is provided by two separate full capacity containment spray systems. The Containment Spray System is designed based on the conservative assumption that the core residual heat is continuously released to the containment as steam, eventually melting all ice in the ice condenser. The heat removal capability of each spray system is sized to keep the containment pressure below design after all the ice has melted and residual heat generated steam continues to enter the containment. The spray system is designed to keep the pressure below the design pressure with adequate margin.

14.3.4.2 General Description of Containment Pressure Analysis

The time history of conditions within an ice condenser containment during a postulated loss-of-coolant accident can be divided into two periods for calculation purposes:

a. The initial reactor coolant blowdown, which for the largest pipe break assumed can occur in approximately 15 seconds.

b. The post-blowdown period, which is of interest for two to three hours after the blowdown.

The first period is characterized by rapid pressure transients, whereas in the second period the transients occur very slowly and the pressure gradients are almost nonexistent. To calculate the pressure propagations in the first period, a detailed spatial and short-time increment analysis is necessary; for the second period, the calculations can be much simpler. The analysis for the first period will typically require a considerable amount of computer memory and also long computing times. Therefore, it is considered impractical to calculate the post-blowdown period with the model being used for the first period.

The analysis has therefore been split into two parts. The first effort has resulted in a code (TMD) to analyze a complex multicompartment containment system in great detail. This code is used only to calculate the initial few seconds of pressure and temperature transients in the containment after a postulated loss-of-coolant accident. The second effort has resulted in the LOTIC code (Long-Term Ice Condenser Code). The major feature of this code is its ability to properly describe the post-blowdown period in the ice condenser containment. Not only are the upper, lower, and ice condenser volumes described, but also the ice condenser is divided into six circumferential sections, each with two vertical divisions. In this way maldistribution and sectional burnout effects can be studied as well as the changing volume distribution during the depletion of the ice bed. Another significant feature of the code is the two sump configuration (active and stagnant sumps) such that the floodup and temperature history of the containment is accurately modeled. The code also describes the performance of the air recirculation fan in returning upper compartment air to the lower compartment. Coupling of residual and component cooling heat exchanger is provided to give an accurate indication of performance for this heat exchanger. The spray heat exchanger performance is also accurately modeled in the transients. The basic equations used are the standard transient mass and energy balances and the equations of state used in any containment transient, but appropriately coupled to the multi-volume ice condenser containment. The code also considers accumulator gas added to the containment and the displacement of free volume by the refueling water storage tank volume.

Physically, tests at the ice condenser Waltz Mill test facility have shown that the blowdown phase represents that period of time in which the lower compartment air and a portion of the ice condenser air are displaced and compressed into the upper compartment and the remainder of the ice condenser. The containment pressure at or near the end of blowdown is governed by this air compression process. In addition, the Waltz Mill tests have demonstrated the long term performance ability of the ice condenser. Specifically, these tests verified the ability of the ice condenser to reduce the containment pressure within a few minutes following the blowdown and, in addition, tests have shown excellent condenser performance for tests simulating the long term addition of residual heat.

14.3.4.3 Long Term Containment Pressure Analysis

14.3.4.3.1 Introduction

Early in the ice condenser development program it was recognized that there was a need for modeling of long term ice condenser containment performance. It was realized that the model would have to have abilities comparable to those of the dry containment (COCO) model. These abilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC Code.[1]

To understand more fully the development of the analytical model, a brief description of the arrangement of an ice condenser reactor containment is presented. The general arrangement of a containment for a four-loop reactor coolant system is shown in Figures 14.3.4-1, 14.3.4-2, and 14.3.4-3.

The containment is divided into three compartments: the reactor coolant system or lower compartment, the upper compartment, and the ice condenser compartment. Figures 14.3.4-1 through 14.3.4-3 show the boundaries of these three compartments, as well as the boundaries of dead-ended compartments within the containment whose air volumes are not displaced by steam into the upper compartment. The lower compartment completely encloses the reactor coolant system equipment. The upper compartment contains the refueling canal, refueling equipment, and the polar crane used during refueling and maintenance operations. The upper and lower compartments are separated by the operating deck, which provides a low-leakage barrier between these two compartments. The ice condenser compartment, which contains the borated ice provided to quench the loss-of-coolant accident energy, is a completely enclosed and refrigerated annular compartment located radially between the reactor coolant system compartment and outer wall of the containment, and in elevation, generally above the operating deck. The dead-ended volumes are adjacent to the lower compartment and include the auxiliary pipe tunnel, the fan accumulator compartments, and the instrument room.

The LOTIC Code uses the control volume technique to represent the physical geometry of the system. Fundamental mass and energy equations are applied to the appropriate control volumes and solved by suitable numerical procedures. The initial conditions of the containment by compartment is specified before blowdown. Ice melt is calculated for the blowdown period based on the mass and energy released to the containment. After the RCS blowdown, the basic LOTIC Code assumption is made that the total pressure in all compartments is uniform. This assumption is justified by the fact that after the initial blowdown of the RCS the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between compartments will also be relatively small. These small flow rates are unable to maintain significant pressure differentials between the containment compartments.

14.3.4.3.2 Description of the LOTIC Code (*)

14.3.4.3.2.1 Method of Solution

The model of the containment consists of five distinct control volumes: the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartments. The ice condenser control volume with unmelted and melted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases of problem time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

14.3.4.3.2.1.1 Blowdown Period

This phase coincides with the blowdown of the reactor coolant system. During this phase no attempt is made to calculate the pressure, flow, and temperature transients in the containment. Instead, this complicated analysis is accomplished with the TMD code, a code created specifically for this short term analysis. The pressure and temperatures in the containment are held constant during this phase at input values determined from TMD analyses and compression ratio calculations. Physically, tests at the Waltz Mill test facility have shown that this phase represents that period of time in which the lower compartment air and a portion of the ice condenser air are displaced and compressed into the upper compartment and the remainder of the ice condenser. (The initial pre-blowdown atmosphere in the dead-ended compartment is

(*) As of the date of this updated FSAR the approved version of the LOTIC code is LOTIC3 (See Reference 20 and Section 14.2.5, Unit 2) retained at that time.) The code represents this phenomenon through the use of an input value for the fraction of the ice bed which retains air during this phase. This fraction, determined from test data, is also used to establish the volumes of the two ice condenser control volumes which are held constant during this phase.

The temperatures in the upper and lower compartments are calculated from the input pressure. The portions of the containment which are primarily air-filled, i.e., the dead ended compartment and a portion of the ice bed, are assumed to be at upper compartment temperature during this phase. Deck leakage considerations resulted in the upper compartment atmosphere to be considered saturated at this temperature.

14.3.4.3.2.1.2 Depressurization Period

This phase of the analysis corresponds to the period of time between the end of blowdown and the establishment of a circulation flow between the control volumes. During this phase the inputted non-condensable nitrogen blowdown from the accumulator occurs, the decay heat boiloff is initiated, and the engineered safeguards come into operation. Maximum decay heat boiloff is achieved by assuming the safety injection system disabled to the point that only enough water is deliverd to the core to replace the water boiled, with the remaining safety injection spilled to the sump, although varying degrees of SIS effectiveness can be simulated. The engineered safeguards which are initiated in this phase are the recirculation fan, the safety injection system, and the spray system. The recirculation fan forces upper compartment air into the lower compartment atmosphere and has the ability to force-circulate the stagnant dead-ended atmosphere in a similar manner. During this phase the spray systems and safety injection system take water from the refueling water storage tank and pump it into the containment, with the spray flow passing through the spray heat exchanger. The models for the spray system and heat exchangers are discussed in Sections 14.3.4.3.3.2 and 14.3.4.3.3.3. At the beginning of this phase the blowdown ice melt is computed using the blowdown energy. This result is used to compute the actual volume of the melted-out portion of the ice condensers, which is used to change the ice condenser volumes from the compressed value associated with the air displacement in the blowdown phase to the actual value computed from the ice melt. The temperature of the ice condenser volume is also changed over a period of time from the original value to an input value. As soon as the recirculation fan is started the dead-ended compartment begins to undergo a conversion to lower compartment atmosphere. The conversion takes place at the input purge flow rate. This continues until all the dead-ended compartment atmosphere has been converted to lower compartment. It is also possible to input the code in such a manner that the dead-ended compartment is always treated as upper compartment volume.

As soon as recirculation fan flow is initiated the lower compartment begins to fill with an air-steam mixture, composed of the upper compartment air of the fan flow, and decay heat boiloff steam displacing the previous steam atmosphere of the lower compartment through the ice bed in a piston type manner. As this occurs the code calculates the conditions in the upper and lower compartment from the compartment conditions and the spray and flow characteristics. This phase of the analysis ends when the air-steam mixture fills both the lower compartment and the melted out portion of the ice bed.

14.3.4.3.2.1.3 <u>Long Term</u>

This phase of the analysis begins as soon as the circulation of air through the containment has been established and continues until the problem is terminated. The major occurrences of this phase are recirculation and ice meltout. Recirculation occurs when the refueling water storage tank has been drained. At this time the safety injection and spray system begin drawing from the active sump instead of the refueling water storage tank (the two sump model is discussed in Section 14.3.4.3.3.4). The spray system flow continues to be routed through the spray heat exchanger during this period, with the safety injection and residual spray flows through the residual heat exchanger.

Meltout occurs when there is no longer enough ice in the ice bed to prevent steam from flowing directly from the lower compartment to the upper compartment. As long as there is more than a foot of ice in the ice compartment the temperatures in the two ice compartment control volumes remain at constant but different values determined from Waltz Mill test data. When the ice in a sub-compartment of the ice bed volume is gone, that sub-compartment is assumed to contain lower compartment atmosphere. (Due to maldistribution which is inputted to the code the sub-compartments may melt in a sequenced manner rather than simultaneously.) During the long term phase the fan flow from the upper compartment and the flow out of the lower compartment are assumed to be at the temperature of the compartment the flow is leaving.

14.3.4.3.2.2 Assumptions*

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This asumption is justified by the fact that after the initial blowdown of the reactor coolant system the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates than are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. When the circulation fan is in operation, the fan flow and the reactor coolant system boiloff are mixed before entering the lower compartment. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

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*The assumptions associated with the current licensing basis are found in Section 14.3.4 Unit 2 of the UFSAR. Unit 1 has been analyzed for essential service water temperature of 92°F and a power level of 3250 MWt. The following text is the original licensing basis.

The condensation of steam is assumed to take place in a condensing node located, for the purpose of calculation, between the two control volumes in the ice storage compartment. The exit temperature of the air leaving this node is set equal to a specified value which is equal to the temperature of the ice filled control volume of the ice storage compartment. Lower compartment exit temperature is used if the ice bed section is melted.

14.3.4.3.3 Engineered Safeguard Systems

14.3.4.3.3.1 <u>Description</u>

The Engineered Safeguard Systems modeled in the program consist of a Safety Injection System and a Spray System. Because of decay heat produced by the reactor core after a loss-of-coolant accident, cooling water has to be supplied to the core by the Safety Injection System. Under the assumption that this cooling takes place by boiloff of the Safety Injection water, energy is added to the containment atmosphere. However, by means of the Spray System, the energy content of the containment atmosphere is reduced and absorbed by the containment sump water. Initially, the system draws in water from the Refueling Water Storage Tank for spray. After this tank is empty the system is switched to the containment sump water.

A schematic of the system as encoded in the LOTIC program is shown in Figure 14.3.4-4 for the initial period and in Figure 14.3.4-5 for the recirculation period.

The time for the system to switch to recirculation is either input or determined by the capacity of the Refueling Water Storage Tank and the combined flow rates of the spray, residual, high head and charging pumps.

The flow distribution in the system is governed by factors applied to branch points A through E of the pipe system as depicted in Figure 14.3.4-6. The factors are supplied via input data. The option exists for a change of any of the factor values during the problem time.

14.3.4.3.3.2 <u>Containment Spray System</u>

The spray system built into the D. C. Cook containment is for both the upper and lower compartments to reduce the energy contents of the compartment atmospheres.

The governing equations for the spray are:

Mass-balance:

$$m_{sp,\ out} = m_{sp,\ in} + m_c \tag{1}$$

Energy balance:

$$m_{sp,\ out}\ h_{sp,\ out} = m_{sp,\ in}\ h_{sp,\ in} + m_c\ h_s \tag{2}$$

Combining these two equations, the spray leaving the control volume is:

$$m_{sp,\ out} = m_{sp,\ in} \frac{(h_s - h_{sp,\ in})}{(h_s - h_{sp,\ out})} \tag{3}$$

By the assumption that the spray water leaves the control volume at the control volume temperature, it follows that $h_{sp,\ out} = h_w$ and the above equation can be solved directly.

14.3.4.3.3.3 <u>Heat Exchangers</u>

The component cooling heat exchanger, a counterflow heat exchanger and the residual heat exchanger, a U-tube heat exchanger, are coupled on their shell sides, while the raw water is on the tube side of the component cooling heat exchanger and safety injection and spray flow is on the tube side of the residual heat exchanger. The spray heat exchanger is of the counterflow type with raw water on the shell side and spray flow on the tube side. The spray heat exchanger can also be specified as a U-tube type. The performance equations are taken from <u>Process Heat Transfer</u> by Kern.

14.3.4.3.3.4 Two Sumps

The active sump in the ice condenser containment may have insufficient capacity to contain all the water of the reactor coolant system, the melted ice, and the refueling water storage tank. The excess water is spilled and is no longer available (inactive sump) for the safety injection or spray system. The maximum value of the active sump can be specified by input data.

The water mass and temperature in the sump is calculated as follows:

$$M_{sump,\ N} = M_{sump,\ 0} + (\Sigma m_{drn} - m_{sis} - m_{spill})\ \Delta t \qquad (4)$$

the summation is for all the flows entering the sump.

$$H_{sump,\ N} = \frac{H_{sump,\ 0} + \Sigma m_{drn} h_{dm} - (m_{sis} - m_{spill})\ h_{sump,\ 0}]\ \Delta t}{M_{sump,\ 0} + (\Sigma m_{drn} - m_{sis} - m_{spill})\ \Delta t} \qquad (5)$$

The water volume follows now from

$$V_{sump} = M_{sump,\ N}/v_w \qquad (6)$$

If the sump water volume is greater than the specified maximum volume, the spilling flow follows from

$$m_{spill} = (M_{sump,\ N} - \frac{V_{max}}{V_w})\ \Delta t \qquad (7)$$

and the water mass is reset to

$$M_{sump,\ N} = \frac{V_{max}}{V_w} \qquad (8)$$

This process is representative of the spillover into the pipe trench area after the sump inside the crane wall is filled. The water mass and temperature of the spilled water sump is calculated analogous to the main sump, the entering flow being the spilling flow having a temperature calculated by Equation (5).

The approach is conservative as the smaller active volume of this model heats more rapidly than an all-inclusive sump; this causes higher calculated spray temperatures.

14.3.4.3.4 Mathematical Models

14.3.4.3.4.1 Temperature Response in Upper and Lower Compartments

A change in the mass or energy flow rates into or out of a control volume affects the temperature of the medium in the control volume. The equation for the response of this temperature is developed using the mass and energy balances for the control volume.

The mass balance differentiated with respect to time is:

$$\frac{d}{dt} M = m_{in} - m_{out} \tag{9}$$

The energy balance also differentiated with respect to time is

$$p \frac{d}{dt} V + \frac{d}{dt} uM = (m_{in} h_{in} - m_{out} h_{out}) + q \tag{10}$$

As the control volume has stationary boundaries, the first term equals zero.

In general there exists a unique expression for the enthalpy of the medium as a function of its temperature. Considering this, the energy equation can then be written in the following way:

$$M \frac{du}{dT} \frac{d}{dt} T + u \frac{d}{dt} M = m_{in} h(T_{in}) - m_{out} h(T_{out}) + q \tag{11}$$

This form of the energy equation and some appropriate assumptions form the basis for the temperature response calculation in the control volumes.

14.3.4.3.4.2 System Pressure

As mentioned previously, the total pressure in the ice condenser containment is assumed to be uniform at a given instant. For the five control volumes this means that the sum of the steam and air pressure is equal for these control volumes.

The total air mass in the system remains constant during the whole period of the analysis, and can be calculated from the initial conditions. The time dependent air mass can also be calculated at a given instant and used in place of the initial air mass if other non-condensables are added to the system.

The fundamental equation for this calculation is the equation of state for air:

$$P_a V_a = M_a R_a \, (T_a + 460) \tag{12}$$

The total air mass in the system M_a must equal the sum of the air masses in each control volume. Rewriting Equation (12) for all 5 control volumes yields:

$$M_a = \sum_{i=1}^{5} M_{a,i} = \sum_{i=1}^{5} \frac{P_{a,i} V_i}{R_a \, (T_{a,i} + 460)} \tag{13}$$

As $P_{sys} = P_s + P_a$, one can write

$$M_a = \sum_{i=1}^{5} \frac{P_{sys} V_i}{R_a \, (T_{a,i} + 460)} - \sum_{i=1}^{5} \frac{P_{s,i} V_i}{R_a \, (T_{a,i} + 460)} \tag{14}$$

If the temperature in each compartment is known, either by calculation or assumption, and the relation between the temperature and the steam pressure is known, it follows that P_{sys} in Equation (14) is the only unknown and can be found.

14.3.4.3.4.3 <u>Filling of Lower Compartment</u>

The start of the recirculation fan during the depressurization period causes the lower compartment to be filled with upper compartment atmosphere at a specified fan flow rate. The model used is that of a growing volume displacing the original atmosphere into the ice bed. The model assumes no mixing of the fan flow and the original steam. This means nonequal temperatures for the two masses and uniform pressure in the lower compartment.

The governing equations for this process are the mass and energy balance for the volume of air-steam supplied by the fan.

Mass Balance:

$$\frac{d}{dt} M = \Sigma \dot{m} \tag{15}$$

Energy Balance:

$$\frac{d}{dt} (Mu) + p \frac{d}{dt} V = \Sigma \dot{m}h + q - W \tag{16}$$

The growing volume has an atmosphere of water, steam, and air; the only flow leaving this volume will be the steam that is condensing. As there will be no work done, the last term of Equation (16) is zero.

14.3.4.3.4.4 <u>Heat Transfer</u>

The fundamentals for the heat transfer calculations in the ice condenser or part of the containment are the test data gathered from the ice condenser Waltz Mill test facility.

These test data as applied in the calculation of the ice inventory are the following:

During the initial blowdown of the primary coolant system all the steam is condensed and the drain temperature of the ice bed for the mixture of condensed steam, melted ice, and blowdown water is equal to $175^{O}F$ as determined by early Waltz Mill tests. For the long term period of decay heat steam boiloff while the circulation fan is in operation, the drain temperature of the ice bed is $75^{O}F$. Also during this period the temperature of the air-saturated steam mixture leaving the ice condenser is $75^{O}F$, thus allowing some steam to enter the upper compartment. The temperature of that portion of the ice bed containing ice is $75^{O}F$ and that part of the ice condenser where ice has been melted is at $160^{O}F$. After the ice has melted out, the ice bed assumes lower compartment temperature and steam exits at lower compartment temperature.

The data enables one to calculate the inventory of ice at a given instant. The governing equations are:

For the mixture flowing in the ice condensers

Mass Balance:

$$M_{s, in} + m_{a, in} = m_{s, out} + m_{a, out} + m_{c} \tag{17}$$

Energy Balance:

$$m_s h_{s, in} + m_a h_{a, in} = m_s h_{s, out} + m_a h_{a, out} + m_c h_c + q \tag{18}$$

Dalton's Law for a multi-constituent mixture of gases, applied at the exit side of the ice compartment, results in:

$$m_{s, out} = \left(\frac{18}{29} m_{a, out}\right) \left(\frac{P_s}{P_a}\right) \tag{19}$$

For the Ice Bed:

$$\frac{d}{dt} (Mice) = m_m \tag{20}$$

Energy Balance:

$$q = m_m (h_m - h_{ice}) \tag{21}$$

For the resulting drain flow:

Mass Balance:

$$m_{drain} = m_m + m_c \tag{22}$$

Energy Balance:

$$m_m h_m + m_c h_c = m_{drn} h_{drn} \tag{23}$$

This set of equations can be solved to keep track of the ice inventory in the ice compartment.

14.3.4.3.4.4.1 Residual Heat

The WNES residual heat standard is incorporated in the code for the total energy release rate from isotope decay following a shutdown of the current generation of thermal reactors fueled with uranium (U-235 enriched). The three major contributors of energy are:

a. fission-product decay heat,
b. U-238 capture decay, and
c. residual fissions.

The fission-product residual heat uses a combination of the proposed ANS Sub-committee-5 data and calculations by the WNES Radiation Analysis personnel for finite fuel region cycle times of 24,000 EFPH, 16,000 EFPH and 8,000 EFPH.

14.3.4.3.4.4.2 Zirconium-Water Reaction Heat

The energy produced by a zirconium-water reaction in the core can be supplied as a time dependent table. The energy is added to the residual heat, and thus results in an additional steam release into the lower compartment.

The boiloff from the core due to residual heat and an eventual zirconium-water reaction is calculated as follows:

$$m_{boil\text{-}off} = q/\ (h_s - h_w) \tag{24}$$

where h_s is the steam enthalpy at the existing lower compartment conditions and h_w is either the enthalpy of the water in the core or the enthalpy of the safety injection water put into the core. All other safety injection water is spilled to the sump without removing heat.

If the effective safety injection option is used and the safety injection system is supplying water to the core, an energy balance is performed to calculate the enthalpy of the spilling safety injection water from the core.

$$h_{sp} = (m_{SIS} h_{SIS} + q)/m_{spill} \tag{25}$$

In case the calculated enthalpy is greater than the enthalpy corresponding to that of saturated water at the lower compartment condition, the steam boiloff is calculated by:

$$m_{boil\text{-}off} = m_{SIS}\ (h_{SIS} - h_{w,sat}) + q/h_{fg} \tag{26}$$

The remaining spilling water is at the saturated liquid condition.

14.3.4.3.4.5 Structural Wall Heat Sinks

During a postulated accident, heat is absorbed by the walls of the containment structure, as well as by interior support walls and relatively cold equipment inside the containment. There also may be considerable heat dissipation to the external air by steel vessel containment structures. In this analysis, only a slab geometry is considered, and heat transfer in any direction is neglected, with the exception of that perpendicular to the wall surface.

Figure 14.3.4-7 shows a typical multilayered wall. For illustration purposes, this wall is assumed to have two layers and to be in contact with the containment steam-air mixture on the inner surface. Walls may also be specified as having more than two layers. Each layer is divided into a small number of small elements X wide (except for the two surface elements which are X/2 wide). The thermal properties of each layer are assumed constant, and each element is assumed to be at a uniform temperature. To obtain a spatially converged temperature gradient for different materials, the element width may be varied from layer to layer.

Table 14.3.4-1 is a summary of the containment structural heat sinks used in the analysis. The material property data used is also found in Table 14.3.4-1.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami. An explanation of the manner of application is given in Reference (2).

When applying the Tagami correlations a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to 72 BTU/hr-ft^2. This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation restricts the use of the Tagami correlation to within the test range of steam-air ratios where the correlation was derived.

14.3.4.3.5 Spray System Design Basis

This section discusses the design basis and evaluation of the performance of the ice condenser and spray system as a sink for post blowdown heat sources in the containment. On completion of blowdown and the initial depressurization of the containment, a large amount (about 63 percent) of the ice remains. The ice, together with the containment spray system, provides considerable capacity to accept very large post-blowdown energy releases.

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident and thereby ensure that containment pressure is not exceeded. This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. A schematic flow diagram of the Containment Spray System is shown in Figure 14.3.4-8. Adequate containment heat removal capability for the Ice Condenser Containment is provided by two separate full capacity containment spray systems. One spray system is defined as one spray pump with heat exchanger and partial flow from the residual heat removal pump with one residual heat exchanger. For the D.C. Cook plant, the spray system pump provides 2000 gpm of spray to the upper compartment and 900 gpm to the lower compartment, plus 265 gpm to the fan rooms in the lower compartment outer annulus. Each residual heat removal pump spray capacity is 2,000 gpm directed to the upper compartment. The Containment Spray System is designed based on the conservative assumption that the core residual heat is continuously released to the containment as steam, eventually melting all ice in the ice condenser. The heat removal capability of each spray system is sized to keep the containment pressure below design after all the ice has melted and residual heat generated steam continues to enter the containment. The spray system is designed to keep the pressure below the containment design pressure (12 psig) with adequate margin.

The containment pressure transient is also evaluated assuming an energy release of 68 x 10^6 BTU between 30 and 1000 seconds, in addition to the Reactor Coolant System blowdown and release of 50 million BTU of undefined energy. This analysis is considered to be a containment capability case with the design pressure of the containment (12 psig) as the appropriate pressure limit.

The containment transient is also evaluated considering the steam generators as active heat sources during reflood and decay heat addition phases of the accident.

14.3.4.3.6 Input Assumptions In Transient Analysis(*)

The following are the major input assumptions used to calculate the containment transients for the pump suction pipe rupture cases with the steam generators considered as an active heat source for the Cook Nuclear Plant containment:

1. Minimum containment safeguards are used, in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two air recirculation fans.

2. 2.45 x 10^6 lbs. of ice initially in the ice condenser.

3. The Blowdown and Reflood mass and energy releases are described in Section 14.3.4.6 under the topic "Long Term Blowdown Analysis."

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F, respectively, and an ice bed exit temperature of 130°F were used.

(*) These assumptions correspond to the October-November, 1974, results presented in Section 14.3.4.3.7. Current assumptions are given in Section 14.3.4 of the updated Unit 2 FSAR. An evaluation was performed for Unit 1 supporting containment analysis with essential service water temperature of up to 92°F (reference 23).

5. Nitrogen from the accumulators in the amount of 5942 lbs. is included in the calculations.

6. Essential service water temperature of 76°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air recirculation fan is assumed to be effective approximately 10 minutes after the transient is initiated.

8. Water entrainment to the steam generator continues until the 10 foot level is reached during the reflood period.

9. No maldistribution of steam flow to the ice bed is assumed.

10. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

11. The initial conditions in the containment are a temperature of 110°F in the lower and dead ended volumes, a temperature of 15°F in the ice condenser, and a temperature of 75°F in the upper volume. All volumes are at a pressure of 0.3 psig and a 15% relative humidity.

12. A spray pump flow of 2000 gpm to the upper compartment and a spray pump flow of 900 gpm to the lower compartment is assumed.

13. A residual spray of 2000 gpm is assumed at the initiation of recirculation. Residual heat removal pump and spray pump take suction from the sump, after 350,000 gallons of the refueling water storage tank (100°F) have been pumped into the containment.

14. Containment structural heat sinks are assumed with conservatively low heat transfer rates.

15. The operation of one containment spray heat exchanger ($UA = 3.58 \times 10^6$) and one containment RHR heat exchanger ($UA = 2.16 \times 10^6$) has been assumed.

14.3.4.3.7 Long Term Containment Response

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design. The results of three different analyses are discussed below.

<u>Original Analysis Filed March, 1973</u>

Figure 14.3.4-9 shows two long-term pressure transients for the Cook Nuclear plant containment design basis accident. The peak pressure of 7.8 psig at the end of the Reactor Coolant System blowdown was input to the code and is based on the compression ratio of containment volumes. The air recirculation fan effectiveness at approximately 10 minutes assures the rapid reduction in the containment pressure. The pressure remains low until continued residual heat steam flow completely melts the ice bed following the blowdown. At this time the spray system must maintain the pressure below design and reduce the pressure for the remainder of the transient.

The design basis energy release includes 50×10^6 BTU undefined energy release in addition to the reactor coolant system blowdown energy release and hot metal energy. The design basis accident (solid line curve) assumes uniform ice bed blow distribution and degraded safety injection and results in a 9.8 psig pressure peak following complete ice bed melt-out. Degraded safety injection assumes all safety injection flow is spilled to the sump except that needed to make up decay heat boiloff.

No flow maldistribution to the ice bed results in the highest pressure peak following complete ice melt, at about 4000 seconds for the design basis accident. The assumption of no maldistribution is conservative, since any early melt-out of ice bed sections would allow the upper compartment spray to work on the steam leakage and thus preserve ice for a longer time in the remaining sections. The second transient shown (dashed lines) is for a peak maldistribution factor of 1.36 for the blowdown and post blowdown steam boiloff. This maldistribution is in excess of that calculated for the ice condenser but illustrates the effect of maldistribution on the peak pressure. In this transient the first ice condenser section melts out in about 1200 seconds and the final ice condenser section melts out in about 5500 seconds. This transient results in a peak pressure of 9.6 psig. Therefore maldistribution of flow to the ice bed will result in a lower pressure peak and the assumption of uniform ice bed melt-out is conservative.

Figure 14.3.4-10 shows the long term temperature transient in the containment for the design basis accident of the previous figure (9.8 psig peak pressure). This figure shows the upper and lower compartment temperature transients as well as the sump and spray temperature transients. The jump in lower and upper compartment temperature occurs at the time of ice bed melt-out which is for uniform flow distribution to the ice bed. A realistic sump flood up temperature history and active and stagnant sump model has been used in the LOTIC code. The stagnant sump is representative of the spillover into the pipe trench area after the active sump inside the crane wall is filled. The approach is conservative as the smaller active volume of this model heats more rapidly than an all-inclusive sump and causes higher spray temperatures. The residual spray temperature is determined by passing sump water through the residual heat exchanger. The spray temperature is determined by passing the refueling water storage tank water through the spray heat exchanger during the injection phase, and sump water through the spray heat exchanger during the recirculation phase.

Figure 14.3.4-13 shows the resultant pressure transient for the containment capability case. This case has the same assumptions as the design basis case but with an additional 68 x 10^6 BTU of energy released between 30 and 1000 seconds. This reaction energy is in addition to the 50 x 10^6 BTU of undefined energy release. The peak pressure for the containment capability case is 11.5 psig, below the containment design pressure of 12 psig. For this analysis and the design basis case the containment structural heat sinks were neglected. These structures provide further margin to absorb energy. Thus the ability of the ice condenser and spray system to absorb very large additional energy release is demonstrated.

Later Analysis, Filed July, 1973

The upper and lower compartment temperature transients presented in Figure 14.3.4-10 were reevaluated from time zero to 10^6 seconds. These results are shown in Figures 14.3.4-11 and 14.3.4-12. The initial twenty second portion of each transient was generated by the TMD code. The remainder of each transient was generated by the LOTIC code.

Final Analysis, Filed October-November, 1974

The containment response has been analyzed for the case of the core reflood (Section 14.3.4.6.2), considering the steam generators as active heat sources. Post-reflood froth flow split has also been considered in this analysis. Analysis of the core reflood transient has shown that the maximum energy release to the containment occurs for the double-ended pump suction break with sufficient safety injection flow to release the maximum energy to the containment. The containment pressure transient for the calculated flow split case is presented in Figure 14.3.4-14. The following plots have been provided for the bounding 100% flow split case:

Figure 14.3.4-15:	Containment Pressure
Figure 14.3.4-16:	Integrated Ice Melt
Figure 14.3.4-17:	Upper and Lower Compartment Temperature
Figures 14.3.4-18 and 14.3.4-19:	Structural Heat Sinks Heat Removal Rates
Figure 14.3.4-20:	Sump Masses
Figure 14.3.4-21:	Sump Temperatures
Figure 14.3.4-22:	Spray Temperatures
Figure 14.3.4-23:	Heat Exchanger Energy Removal Rates
Figure 14.3.4-24:	Ice Bed Energy Removal Rates

For the froth analyses presented in Figures 14.3.4-14 through 14.3.4-24, blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F were used. (These values are based on Reference (3) and discussed in Section 14.3.4.3.8.) In Figure 14.3.4-14, the peak pressure is 8.4 psig; for the 100% flow split case, the peak pressures reaches 10.4 psig over the time interval from 5842 seconds to 7028 seconds.

14.3.4.3.8 Application Of Waltz Mill Results To LOTIC

The completed Waltz Mill ice condenser blowdown tests have indicated somewhat higher ice-melt temperatures and higher ice bed exit temperatures than has been used in the long term containment transient analysis. These increased temperatures have the effect of delaying the ice bed meltout which results in lower reactor decay heat rates to be handled by the sprays after ice meltout. To quantify the effect of higher ice bed drain and exit temperatures, parametric studies were made with the LOTIC code to determine the long term containment response. Additional calculations were made to show that sufficient NPSH is available at the containment spray and RHR pump inlets with the higher sump temperatures.

LOTIC code analyses were made varying the ice bed drain temperature and ice bed exit temperature over a range exceeding conditions observed in the Waltz Mill testing. During the blowdown phase of the accident, the LOTIC code calculates the ice remaining in the ice bed based upon inputted drain temperature for the blowdown phase. It should be noted that during the blowdown phase, no attempt is made with LOTIC to find the pressure and temperature transient in the containment. Instead, this complicated analysis is accomplished with the TMD code. After the blowdown is complete, an ice bed exit temperature and ice bed drain temperature are specified. Long term containment transients performed have usually used 175° drain temperature during the blowdown phase and 75°F temperatures for both the ice bed drain and exit temperature for the post-blowdown phase of the accident.

Current test data show ice condenser drain and sump temperatures of 190°F to 200°F following blowdown for large equivalent pipe breaks. For the post-blowdown energy release period, ice bed drain and exit temperatures of 110°F to 160°F were observed.

The mass and energy release rates used in this study were similar to that presented in Section 14.3.4.3.7 for the DEPS froth analyses. For conservatism, no credit was taken for the heat removal by the containment structures following ice melt out, and is plotted on Figure 14.3.4-25 with ice bed exit temperatures as the variables. As can be seen from this figure, higher ice bed drain and exit temperatures result in lower peak containment pressures.

A base case long term accident transient using 175°F drain temperature during the blowdown phase and 75°F drain temperature and exit temperature for the post-blowdown energy release results in an ice bed melt-out time of 2050 seconds with a resultant peak containment pressure of 9.1 psig following ice bed melt. The higher drain temperatures for the ice bed preserves ice to extend the ice bed melt-out time. The higher exit temperatures also causes an extension of the melt-out time. Any steam leakage out of the bed would be condensed by sprays which are on during this period but which have little steam to condense in the upper compartment.

To illustrate the effect of ice bed drain and exit bed temperature, a comparison will be made between the base case and a case that most closely represents the results observed in the long term Waltz Mill ice condenser test. The test data is best represented by using blowdown and post-blowdown drain temperatures of 190°F and 130°F, respectively, and an ice bed exit temperature of 130°F. For this case, the ice bed melt-out time was extended to 7,400 seconds with a peak containment pressure of only 6.3 psig following the ice bed melt-out.

Figure 14.3.4-26 shows a comparison of the long term pressure transient for the base case and the ice condenser parameters representative of the current test data. As the figure shows, there is a significant improvement in peak pressure and ice bed melt-out time. Figures 14.3.4-27 and 14.3.4-28 show the comparison of lower compartment and upper compartment temperature transients. The upper compartment temperature is significantly lower because of the longer ice bed life. Figure 14.3.4-29 provides a comparison of the active sump temperature for the two cases. Although the base case sump temperature is lower earlier in the transient, at the time of peak containment pressure the sump temperatures are similar. Figure 14.3.4-30 shows the spray temperature in the upper and lower compartments to be nearly identical for both cases. Figure 14.3.4-31 also shows very similar residual heat removal spray temperatures for both cases.

Although higher ice condenser drain temperatures result in higher sump temperatures, the effect on the containment spray temperature is not significant. Thus the higher sump temperatures do not have a deleterious effect on spray heat removal capacity.

These studies show that the increased temperatures indicated by the current ice condenser tests have the effect of delaying the ice bed melt-out which results in lower peak containment pressures for the long term transient.

The higher ice condenser drain temperatures observed in the Waltz Mill ice condenser blowdown tests will result in higher pump temperatures which result in a reduction of the net positive suction head (NPSH) available to the containment spray and RHR pump inlets. To demonstrate adequate NPSH is available with the higher drain temperatures, scoping calculations were made to determine the maximum expected temperatures for the containment sump.

The Waltz Mill tests show peak sump temperatures of 190°F to 200°F following LOCA blowdown for large breaks, with lower sump temperature for small breaks. During the injection phase following the blowdown, the safety injection and containment spray pumps take suction from the refueling water storage tank at maximum temperature of 100°F. Any core boil off to the containment during the post blowdown period would melt ice. Condensate and ice melt drain temperatures from the ice bed for the post blowdown energy release rates is approximately 130°F. Thus any ice melt would lower the sump temperature and increase the NPSH available.

To provide a conservative upper bound hand calculation of sump temperature at the end of the injection phase, the following assumptions were made:

1. No additional ice melt following LOCA initial blowdown.

2. 200°F sump temperature at the end of blowdown.

3. Saturated water is assumed to spill from the reactor vessel at a rate of 1300 lb/sec, which is in excess of maximum safeguards SIS flow.

4. Only one spray pump is assumed to operate without cooling of the spray heat exchanger during the injection period, i.e., spilling of 100°F refueling water storage tank contents to the sump at the rate of one spray pump (425 lb/sec).

The results of this scoping calculation show the sump temperature is less than 190°F at the time of recirculation. At this temperature, there is sufficient NPSH available assuming 0 psig containment pressure.

For the recirculation phase, scoping hand calculations were also made to show that a sump temperature of less than 190°F will be maintained. A steady state calculation was made to demonstrate that one spray pump and one spray heat exchanger are sufficient to maintain the sump temperature below 190°F. The following conservative assumptions were made for this calculation:

1. No additional ice melt on recirculation. Any steam boiloff would melt ice and lower the sump temperature.

2. Saturated water is assumed to spill to the sump at a rate of 1300 lb/sec, which is in excess of maximum safeguards SIS flow.

3. Only one spray pump and spray heat exchanger are assumed to operate.

4. The sump temperature is assumed to be 190°F entering the spray and RHR heat exchangers. Since the SIS is assumed to spill saturated water to the sump, any heat removal by the RHR heat exchangers to provide a subcooled water spill to the sump is neglected.

The results of this calculation show the spray heat exchanger heat removal rate of 34,200 Btu/sec is sufficient to maintain the sump temperature below 190°F with the maximum saturated spill by the SIS. If the RCS heat addition is in excess of the saturated spill rate plus the RHR heat exchangers heat removal rate, steam boiloff to the containment would occur. Any steam boiloff would result in melted ice water and condensate entering the sump, further cooling the sump.

A LOTIC computer code analysis was conducted to calculate maximum sump temperature. This upper bound case was a typical DEPS froth analysis assuming maximum safeguards. The following assumptions were also made so that the sump temperature would be maximized:

A. Injection Phase

1. 200°F Ice Condenser drain temperature during the blowdown period.

2. 130°F Ice Condenser drain temperature post blowdown.

3. 130°F mixture temperature exiting the Ice Condenser bed (going to the upper compartment).

4. Saturated accumulator water spillage was considered.

5. 1300 lbs/sec of SIS flow was used. This is in excess of maximum safeguards SIS flow. This will maximize the spillage and minimize boiloff.

6. Only one spray pump (3164 gpm) is assumed with spray heat exchanger cooling at 10 minutes after LOCA.

7. During the entire transient, all the energy released is used to heat the SIS water to saturated conditions and only the remaining energy is used to calculate boiloff.

B. Recirculation Phase

1. Negligible ice melt on recirculation. Any steam boiloff would melt ice and lower the sump temperature.

2. Water is assumed to spill to the sump at a rate of 1300 lbs/sec, which is in excess of maximum safeguards SIS flow. The enthalpy of this spilled water is based on the decay heat generation.

3. Only one spray pump and one spray heat exchanger are assumed to operate.

4. Two RHR heat exchangers are assumed to operate.

The results of this analysis show that the sump temperature is maintained below 190°F. Figure 14.3.4-32 shows a plot of active sump temperatures as a function of time for this analysis.

From these scoping calculations and LOTIC code analyses, the maximum expected temperature of the pumped fluids is below 190°F. With the sump elevation head and assuming no containment pressure, sufficient NPSH is available for the RHR and spray pumps.

14.3.4.4 Short Term Containment Pressure Analysis

14.3.4.4.1 Introduction

The basic performance of the Ice Condenser Reactor Containment System has been demonstrated for a wide range of conditions by the Waltz Mill Ice Condenser Test Program. These results have clearly shown the capability and reliability of the Ice Condenser Concept to limit the containment pressure rise subsequent to a hypothetical loss-of-coolant accident.

To supplement this experimental proof of performance, a mathematical model has been developed to simulate the ice condenser pressure transients. This model, encoded as computer program TMD (Transient Mass Distribution), provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. This model is used to compute pressure differences on various structures within the containment as well as the distribution of steam flow as the air is displaced from the lower compartment. Although the TMD code can calculate the entire blowdown transient, the peak pressure differences on various structures occur within the first few seconds of the transient.

14.3.4.4.2 Description of the TMD Code

14.3.4.4.2.1 General Description

As described in WCAP-8077,[4] the control volume technique is used to spatially represent the containment. The containment is divided into 45 elements to give a detailed representation of the local pressure transient on the containment shell and internal concrete structures. This division of the containment is similar for all ice condenser plants.

The Cook plant containment has been divided into 45 elements or compartments as shown in Figures 14.3.4-33 through 14.3.4-36. The interconnection between containment elements in the TMD code is shown schematically in Figure 14.3.4-37. Flow resistance and inertia are lumped together in the flow paths connecting the elements shown. The division of the lower compartments into 6 volumes occurs at the points of greater flow resistance, i.e., the four steam generators, pressurizer and refueling cavity.

Each of these lower compartment sections delivers flow through doors into a section behind the doors and below the ice bed. Each vertical section of the ice bed is, in turn, divided into three elements. The upper plenum between the top of the ice bed and the upper doors is represented by an element. Thus, a total of thirty elements (elements 7 through 24 and 34 through 45) are used to simulate the ice condenser. The six elements at the top of the ice bed between bed and upper doors deliver to element number 25, the upper compartment. Note that cross flow in the ice bed is not accounted for in the analysis; this yields the most conservative results for the particular calculations described herein. The upper reactor cavity (element 33) is connected to the lower compartment volumes and provides cross flow for pressure equalization of the lower compartments. The less active compartments, called dead-ended compartments (elements 26, 28, 29, 30 and 34), and the fan accumulator compartments (elements 27 and 31) outside the crane wall are pressurized by ventilation openings through the crane wall into the fan compartments.

For each element in the TMD network the volume, initial pressure, and initial temperature conditions are specified. The ice condenser elements have additional inputs of mass of ice, heat transfer area, and condensate layer length. For each flow path between elements flow resistance is specified as a loss coefficient "K" or a friction loss "$f\frac{L}{D}$" or a combination of the two based on the flow area specified between elements. Friction factor, friction factor length, and hydraulic diameter are specified for the friction loss. The code input for each flow path is the flow path length used in the momentum equation. In addition the ice condenser loss coefficients have been based on 1/24 scale tests representative of the current ice condenser geometry. The test loss coefficient was increased to include basket roughness effects and to include intermediate and top deck pressure losses. The loss coefficient is based on removal of door port flow restrictors. Table 14.3.4-2 lists the flow path lengths, the flow path areas, and the loss coefficients used in the TMD analysis of the Donald C. Cook Unit 1 containment.

To better represent short term transients effects, the opening characteristics of the lower, intermediate, and top deck ice condenser doors have also been modeled in the TMD Code. The containment geometric data for the elements and flow paths used in the TMD code is confirmed to agree with the actual design by the utility and Westinghouse. An initial containment pressure of 0.3 psig was assumed in the analysis. Initial containment pressure variation about the assumed 0.3 psig value has only a slight effect on the initial pressure peak and the compression ratio pressure peak.

The reactor coolant blowdown rates used in these cases are based on the SATAN analysis of a double-ended rupture of either a hot and cold leg reactor coolant pipe using a discharge coefficient of 1.0. The blowdown analysis has been presented in Section 14.3.4.6.1.

For the Donald C. Cook Plant, the peak pressures and peak differential pressures occur within the first 3.0 seconds of the blowdown. A number of analyses have been performed using 100 percent moisture entrainment to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The maximum peak pressure and differential pressure for all cases are determined for each compartment element. Figure 14.3.4-38 is representative of the upper and lower compartment pressure transients that result from a hypothetical double-ended rupture of a reactor coolant pipe for the worst possible location in the lower compartment of the containment: a hot leg break in element 6.

14.3.4.4.2.2 Analytical Models - No Entrainment

The mathematical modeling in TMD is similar to that of the SATAN blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum, and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are given in Reference (4).

14.3.4.4.2.3 Analytical Models - Entrainment Added

The moisture entrainment modifications to the TMD Code are discussed in detail in Reference (4). These modifications comprise incorporating the additional entrainment effects into the momentum and energy equations.

14.3.4.4.2.4 Analytical Models - Additional Effects

As part of the review of the TMD Code, additional effects are considered. Changes to the analytical model required for these studies are described in Reference (4).

These studies consist of:

a. Spatial acceleration effects in ice bed.

b. Liquid entrainment in ice beds.

c. Upper limit on sonic velocity.

d. Variable ice bed loss coefficient.

e. Variable door response.

f. Wave propagation effects.

14.3.4.4.3 Experimental Verification

14.3.4.4.3.1 Early Tests

The performance of the TMD Code was verified against the 1/24 scale air tests and the 1968 Waltz Mill tests. For the 1/24 scale model the TMD Code was used to calculate flow rates to compare against experimental results. The effect of increased nodalization was also evaluated. The Waltz Mill test comparisons involved a reexamination of test data. In conducting the reanalyses, representation of the 1968 Waltz Mill test was reviewed with regard to parameters such as loss coefficients and blowdown time history. The details of this information are given in Reference (4).

14.3.4.4.3.2 1973 Waltz Mill Tests

14.3.4.4.3.2.1 Test Purpose

The Waltz Mill Ice Condenser Blowdown Test Facility was reactivated in 1973[3] to verify the ice condenser performance with the following redesigned plant hardware scaled to the test configuration:

1. Performated metal ice baskets and new design couplings.

2. Lattice frames sized to provide the correct loss coefficient relative to plant design.

3. Lower support beamed structure and turning vanes sized to provide the correct turning loss relative to the plant design.

4. No ice baskets in the lower ice condenser plenum opposite the inlet doors.

The primary objective of these tests was to determine the transient heat transfer and fluid flow performance of the ice condenser design and to confirm that conclusions derived from previous Waltz Mill tests had not been significantly changed by the redesign of plant hardware. Consequently, the design of the test hardware was configured to provide heat transfer and fluid flow characteristics which were equivalent to those in the plant design. It should be noted that test hardware was not representative of structural characteristics for the plant design since structural response to blowdown was not one of the test objectives. In addition, responses of lower, intermediate, and upper deck doors to blowdown were not included in the test objectives.

14.3.4.4.3.2.2 Test Facility

The Waltz Mill Ice Condenser Blowdown Facility consists of a boiler, receiver vessel, and instrumentation room, and also ice storage and ice machine rooms which are used in conjunction with the ice technology facility. Figure 14.3.4-39 shows the general arrangement of the facility. The boiler and receiver vessels are connected by a 12" schedule 160 pipe in which is located a rupture disc assembly.

The boiler is 3 feet in diameter and 20 feet long, mounted on a structural frame. It can be heated electrically to pressurize a maximum of 117 cubic feet of water to an allowable maximum of 1586 psig pressure at 600°F. Strip heaters mounted on the outside of the boiler shell provide the heat. The flow rate from the boiler is controlled by an orifice located in the piping between the boiler and receiver vessel.

The 12" piping between the boiler and receiver vessel is heated by strip heaters attached to the outside surface of the pipe. Figure 14.3.4-40 shows the piping is arranged into three sections as far as flow and heater capability are concerned. This permits operating the piping and sections of the piping at various subcooled temperatures relative to the boiler.

Figure 14.3.4-41 shows the internal arrangement of the receiver vessel. The ice chest section contains eight ice baskets, 12" diameter by 36 feet high, arranged in a 2 x 4 array. Lattice frames are located at six foot levels of the ice baskets. The baskets set on a lower support structure with flow blockage areas proportional to the plant. Turning vanes are located below the ice baskets and direct the flow entering the lower inlet up through the ice baskets. The vessel is divided into lower and upper compartments. The flow enters the lower compartment from the 12" pipe diffusers, is directed into the ice chest, past the ice baskets and then vents into the upper compartment. The ice chest is wood. All metal surfaces are insulated to limit the heat transfer to these surfaces.

Figure 14.3.4-42 shows the location and typical arrangement for the temperature and pressure measurements that will be made inside the receiver vessel ice chest. The outputs from the transducers are connected to a data acquisition system with scanning rates of 2000 samples per second or 200 samples per second.

14.3.4.4.3.2.3 Test Procedure

The ice baskets are filled in the penthouse at the top of the receiver vessel by a blower system before being lowered into the ice chest. Prior to installing ice baskets, the receiver vessel and building is cooled down by an air recirculation and refrigeration system. A lattice frame is installed after each six foot array of ice baskets and a hold down bar attached through the ice chest walls to prevent basket uplift. After all baskets are installed, the receiver vessel top manhole is closed and the boiler then brought to test conditions.

The boiler is evacuated and filled with demineralized water and the heatup started by energizing the strip heaters. As the water heats in the boiler and expands, it is vented through a letdown heat exchanger. The initial fill of the boiler is measured as well as the water relieved so that the total amount of water in the boiler and piping is always known. Water is circulated between the boiler and downstream piping during heatup by the recirculation system to the various sections of the 12" piping (Figure 14.3.4-40). Subcooled conditions can, thus, be obtained for the water preceeding the saturated water in the boiler itself. By using the heaters, recirculating systems, and letdown system, test energy conditions are obtained.

The flow from the boiler and subcooled piping to the receiver vessel is controlled by an orifice plate located in front of the rupture disc assembly. By varying the size of orifice, the blowdown rate can be changed in accordance with the test plans. It is calculated that the maximum orifice required is 5.5".

After the boiler has reached test pressure and temperature, the blowdown is initiated by the rupture disc assembly. This is a double disc assembly with the pressure between the discs normally at about half the boiler operating pressure. The rupture disc burst pressure rating is 60 - 75% of the boiler operating pressure. The pressure between the discs is provided by a high pressure gas cylinder of nitrogen. At blowdown, the gas pressure is quickly released from the cavity between the discs by venting it into the downstream side of the rupture disc, causing the discs to rupture and the water upstream of the discs to be released into the receiver vessel.

At the time the pressure is started to vent from between the rupture discs, the data acquisition systems is actuated so that data is recorded throughout the blowdown transient. Data recording continues for ten seconds at high speed and then is reduced to a 1/10 speed for five minutes.

A preliminary set of test conditions is presented in Table 14.3.4-3.

14.3.4.4.3.2.4 Results

Confirmation of the predicted ice condenser pressure performance was determined by comparing test results with TMD code predictions for the appropriate test conditions and configuration. Initially, the TMD code predictions were based on assumptions that provided best agreement with previous Waltz Mill test results (e.g., 30% entrainment).

The TMD Code has, as a result of the 1973 test series, been modified to match ice bed heat transfer performance.

14.3.4.4.4 Short Term Containment Response*

14.3.4.4.4.1 Early Results

A summary of the TMD input used in the subcompartment pressure analysis is given in Table 14.3.4-4. A set of ice condenser pressure transient plots is also presented in Figures 14.3.4-43 through 14.3.4-186. These plots include the double-ended hot leg (DEHL) and the double-ended cold leg (DECL) breaks for each of six lower compartment elements. These pressure transients assume loss coefficients into the ice condenser based on 1/24 scale air flow tests. For each of the twelve break analyses, the pressure response of all 45 subcompartments is illustrated. Table 14.3.4-5 is a key describing the plots found on each figure. For this early analysis, the augmented critical flow model was used.

14.3.4.4.4.2 Results Based In 1973 Waltz Mill Tests

A number of analyses have been performed to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The analyses were performed using the following assumptions and correlations:

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*The current licensing basis is located in Section 14.3.4.3, Unit 2 UFSAR. The following material represents the original licensing basis for Unit 1.

1. Flow was limited by the unaugmented critical flow correlation.

2. The TMD variable volume door model, which accounts for changes in the volumes of TMD elements as the door opens, was implemented.

3. The heat transfer calculation used was based on performance during the 1973-74 Waltz Mill test series. A higher value of the ELJAC parameter has been used and an upper bound on calculated heat transfer coefficients has been imposed (see Reference ¼3½).

Results of the analysis for the D.C. Cook plant are presented in Tables 14.3.4-6 through 14.3.4-9. The TMD computer code, using 100 percent moisture entrainment, has been used to calculate peak pressures and peak differential pressures resulting from hot and cold leg reactor coolant pipe breaks in each of the six lower compartment control volumes.

Table 14.3.4-6 presents the maximum calculated peak pressure in each of the lower compartment elements resulting from a hot leg and cold leg pipe break. Generally, the pipe break within an element results in the maximum peak pressure in the element. A break located in element 1 or 6 results in the highest pressure peak (14.4 psig) in the lower compartment because of the limited vent area from these locations in the lower compartment; a break in element 3 results in a peak lower compartment pressure of only 9.2 psig. It should be noted that these pressures exist only inside the crane wall and not on the containment shell itself.

Table 14.3.4-7 presents the maximum calculated peak pressure in each of the ice condenser compartment elements resulting from any pipe break location. The maximum value calculated anywhere in the ice condenser compartment is 10.8 psig; this value is also conservative because of blowdown rate and heat transfer assumptions. (This peak is reached in element 40.)

Table 14.3.4-8 presents the maximum calculated differential pressure across the operating deck (divider barrier) between the lower compartment elements and the upper compartment. These values are approximately the same as the maximum calculated differential pressure across the lower crane wall between the lower compartment elements and the dead ended volumes surrounding the lower compartment. The peak differential pressure of 14.1 psi was calculated to be between elements 1 and 25 or 6 and 25 for a hot leg break.

Table 14.3.4-9 presents the maximum calculated differential pressures across the upper crane wall between the ice condenser elements and the upper compartment. Because the steam generator enclosures are common with sections of the upper crane wall, each section of the crane wall is designed for different loadings. Based on the values shown in Table 14.3.4-9 the end sections of the crane wall enclosing elements 7-8-9 experience a maximum differential pressure of 8.2 psig. The sections in common with the steam generator enclosures are designed for the higher pressure conditions inside the enclosure which would occur if a steam generator steam line breaks within the enclosure; therefore, the differential pressures in Table 14.3.4-9 for elements 10-11-12 and 19-20-21 are not applicable. The remaining sections of the crane wall, enclosing elements 13-14-15 and 16-17-18, experience a maximum differential pressure of 6.0 psig.

Careful consideration is given to the design of those containment internal structures where a pipe break could cause localized compartment presure to be higher than for the design bases double ended reactor coolant pipe rupture. These subcompartments include the steam generator enclosure, fan room, pressurizer enclosure, and upper and lower reactor cavity. The TMD Code with critical flow relations has been applied to these compartments and are presented in Section 14.3.4.7.

14.3.4.4.5 Sensitivity Studies

A series of TMD runs investigated the sensitivity of peak pressures to variations in individual input parameters for the design basis blowdown rate and 100% entrainment. This analysis used a DEHL break in element 6 of the Cook Plant, and investigated effects from blowdown sensitivity to addition of the compressibility factor in the momentum equation. Table 14.3.4-10 gives these results.

The sensitivity study results demonstrate that variations in the plant geometric parameters and in ice bed loss coefficients, both of which are known with a high degree of accuracy, have little effect on the peak pressure calculated by TMD for DEHL break in element 6. However, variations in blowdown and entrainment, which are not know with great accuracy, greatly affect the pressure calculated. The highly conservative values used in the design basis analysis ensure a conservative prediction of the peak break compartment pressure.

14.3.4.4.6 Choked Flow Characteristics

The data in Figure 14.3.4-187 illustrate the behavior of mass flow rate as a function of upstream and downstream pressures, including the effects of flow choking. The upper plot shows mass flow rate as a function of upstream pressure for various assumed values of downstream pressure. For zero back pressure ($P_d = 0$), the entire curve represents choked flow conditions with the flow rate approximately proportional to upstream pressure, P_u. For higher back pressure, the flow rates are lower until the upstream pressure is high enough to provide choked flow. After the increase in upstream pressure is sufficient to provide flow choking, further increases in upstream pressure cause increases in mass flow rate along the curve for $P_d = 0$. The key point in this illustration is that flow rate continues to increase with increasing upstream pressure, even after flow choking conditions have been reached. Thus choking does not represent a threshold beyond which dramatically sharper increases in compartment presures would be expected because of limitations on flow relief to adjacent compartments.

The phenomenon of flow choking is more frequently explained by assuming a fixed upstream pressure and examining the dependence of flow rate with respect to decreasing downstream pressure. This approach is illustrated for an assumed upstream pressure of 30 psia as shown in the upper plot with the results plotted vs. downstream pressure in the lower plot. For fixed upstream conditions, flow choking represents an upper limit flow rate beyond which further decreases in back pressure will not produce any increase in mass flow rate.

The augmented choked flow relationship used in TMD is based on experimental data obtained for choked two-phase flow through long tubes, short tubes, and nozzles. The short tube data was cited by Henry and Fauske in Reference (5). Henry and Fauske conclude that an identical discharge coefficient may be applied to two-phase critical flow through sharp-edged orifices and short tubes to represent the actual critical flow rate through each geometry. On this basis, since the augmented choked flow correlation is based on short-tube data, it is applicable to sharp-edged orifices as well. Figure 14.3.4-188, from Reference (6), presents experimental data for two phase critical flow through several different geometries. The dashed line on the graph represents the augmented homogeneous equilibrium critical flow relationship used in TMD. Below a quality of 0.2 the augmentation correlation is not applicable. 0.62 is the highest quality at which critical flow is calculated by TMD to occur in a major flow path following a DEHL break in the Cook containment. It is apparent that the augmented critical flow calculated by TMD is conservative within the quality range of interest.

Carofano and McManus (7) have published data for the two-phase flow of air-water and steam-water mixtures. Actually, water vapor was present in the gas phase of the so-called air-water test, making it in effect an air-steam-water test. The data presented in Reference (7) demonstrates that the ratio of experimental air-(steam)-water critical flow values to homogeneous equilibrium model predictions is equal to or greater than the ratio of steam-water experimental critical flow values to homogeneous equilibrium model predictions. Therefore augmentation factors derived by comapring steam-water data to the homogeneous equilibrium model may be used in air-steam-water calculations.

14.3.4.4.7 Compression Ratio Analysis

As blowdown continues following the initial pressure peak from a double ended cold leg break, the pressure in the lower compartment again increases, reaching a peak at or before the end of blowdown. The pressure in the upper compartment continues to rise from beginning of blowdown and reaches a peak which is approximately equal to the lower compartment pressure. After blowdown is complete, the steam in the lower compartment continues to flow through the doors into the ice bed compartment and is condensed.

The primary factor in producing this upper containment pressure peak and, therefore, in determining design pressure, is the displacement of air from the lower compartment into the upper containment. The ice condenser quite effectively performs its function of condensing virtually all the steam that enters the ice beds. Essentially, the only source of steam entering the upper containment is from leakage through the drain holes and other leakage around crack openings in hatches in the operating deck separating the lower and upper portions of the containment building.

A method of analysis of the compression peak pressure was developed based on the results of full-scale section tests. This method consists of the calculation of the air mass compression ratio, the polytropic exponent for the compression process, and the effect of steam bypass through the operating deck on this compression.

The compression peak pressure in the upper containment for the D. C. Cook plant design is calculated to be 7.8 psig (for an initial air pressure of 0.3 psig). This compression pressure includes the effect of a pressure increase of 0.4 psi from steam bypass and also for the effects of the dead-ended volumes. The nitrogen partial pressure from the accumulators is not included since this nitrogen is not added to the containment until after the compression peak pressure has been reached, which is after blowdown is completed. This nitrogen is considered in the analysis of pressure decay following blowdown as presented in the long term performance analysis using the LOTIC code. In the following sections, a discussion of the major parameters affecting the compression peak will be discussed. Specifically they are: air compression, dead-ended volumes, steam bypass, blowdown energy, and blowdown rate.

14.3.4.4.7.1 Air Compression Process Description

The volumes of the various containment compartments determine directly the air volume compression ratio. This is basically the ratio of the total active containment air volume to the compressed air volume during blowdown. Essentially, during blowdown air is displaced from the lower compartment and compressed into the ice condenser beds and into the upper containment above the operating deck. It is this air compression process which primarily determines the peak in containment pressure following the initial blowdown release. A peak compression pressure of 7.8 psig is based on the D. C. Cook Plant design compartment volumes shown in Table 14.3.4-11. Figure 14.3.4-189 shows the sensitivity of the compression peak pressure with different air compression ratios.

14.3.4.4.7.1.1 Full Scale Section Tests

The actual Waltz Mill test compression ratios were found by performing air mass balances before the blowdown and at the time of the compression peak pressure, using the results of three special full-scale section tests. These three tests were conducted with an energy input representative of the plant design.

In the calculation of the mass balance for the ice condenser, the compartment is divided into two sub-volumes; one volume represents the flow channels and one volume represents the ice baskets. The flow channel volume is further divided into four sub-volumes, and the partial air pressure and mass in each sub-volume is found from thermocouple readings by assuming that the air is saturated with steam at the measured temperature. From these results, the average temperature of the air in the ice condenser compartment is found, and the volume occupied by the air at the total condenser pressure is found from the equation of state as follows:

$$V_{a\ 2} = \frac{M_{a\ 2} R_a T_{a\ 2}}{P_2} \tag{1}$$

Where:

$V_{a\ 2}$ = Volume of ice condenser occupied by air (ft^3)
$M_{a\ 2}$ = Mass of air in ice condenser compartment (lb)
$T_{a\ 2}$ = Average temperature of air in ice condenser (°F)
P_2 = Total ice condenser pressure (lb/ft^2)

The partial pressure and mass of air in the lower compartment are found by averaging the temperatures indicated by the thermocouples located in that compartment and assuming saturation conditions. For these three tests it was found that the partial pressure, and hence the mass of air in the lower compartment, was zero at the time of the compression peak pressure.

The actual Waltz Mill test compression ratio is then found from the following:

$$C_r = \frac{V_1 + V_2 + V_3}{V_3 + V_{a\ 2}} \tag{2}$$

Where:

V_1 = Lower compartment volume (ft^3)
V_2 = Ice condenser compartment volume (ft^3)
V_3 = Upper compartment volume (ft^3)

The polytropic exponent for these tests is then found from the measured compression pressure and the compression ratio calculated above. Also considered is the pressure increase that results from the leakage of steam through the deck into the upper compartment.

The compression peak pressure in the upper compartment for the tests or containment design is then given by:

$$P_3 = P_{3_0} (C_r)^n + \Delta P_{deck} \tag{3}$$

Where:

P_{3_0} = Initial pressure (psia)

P_3 = Compression peak pressure (psia)

C_r = Volume compression ratio

n = Polytropic exponent

ΔP_{deck} = Pressure increase caused by deck leakage (psi)

Using the method of calculation described above, the compression ratio is calculated for the three full-scale section tests. From the results of the air mass balances, it was found that air occupied 0.645 of the ice condenser compartment volume at the time of peak compression, or,

$$V_{a\ 2} = 0.645\ V_2 \tag{4}$$

The final compression volume includes the volume of the upper compartment as well as part of the volume of air in the ice condenser. The results of the full-scale section test (Figure 14.3.4-190) show a variation in steam partial pressure from 100% near the bottom of the ice condenser to essentially zero near the top. The thermocouples and pressure detectors confirm that at the time when the compression peak pressure is reached steam occupies less than half of the volume of the ice condenser. The analytical model used in defining the containment pressure peak uses the upper compartment volume plus 64.5 percent of the ice condenser air volume as the final volume. This 64.5% value was determined from appropriate test results.

The calculated volume compression ratios are shown in Figure 14.3.4-191, along with the compression peak pressures for these tests. The compression peak pressure is determined from the measured pressure, after accounting for the deck leakage contribution. From the results shown in Figure 14.3.4-191, the polytropic exponent for these tests is found to be 1.13.

14.3.4.4.7.1.2 Plant Case

For the D. C. Cook design the volume compression ratio, not accounting for the dead-ended volume effect, is calculated using Equation (2) and Table 14.3.4-11 as:

$$C_r = \frac{1,179,636}{745,896 + 0.645 \times 126,940}$$

$$C_r = 1.42$$

The peak compression pressure, based on an initial containment pressure of 15.0 psia, is then given by Equation (3) as:

$$P_3 = 15.0\ (1.42)^{1.13} + 0.4$$

$$P_3 = 22.7 \text{ psia or } 8.0 \text{ psig}$$

This peak compression pressure includes a pressure increase of 0.4 psi from steam bypass through the deck. The effect of the dead-ended compartment volumes on the compression peak pressure is considered separately; this effect reduces the above calculated value of compresion peak pressure from 8.0 psig to 7.8 psig.

14.3.4.4.7.2 Effect of Dead-Ended Volumes

There are several dead-ended compartments in the plant containment design which are connected to the lower compartment. The dead-ended volumes considered in the following analysis are the instrumentation room and the pipe trench. Additional study has shown that the fan accumulator rooms would also act as dead-ended volumes. Since the addition of dead-ended volume reduced peak compression pressure, the results presented for the following analysis are conservative.

In the preceding analysis of the containment compression ratio, it is conservatively assumed that only steam flows into the dead-ended volumes during the reactor coolant system blowdown. However, the results of certain full-scale section tests, which contained dead-ended volumes, showed that some air flowed into these volumes and remained there during the blowdown period, thus reducing the mass compression ratio for the containment. For example, one Waltz Mill test was run with the lower hemisphere of the receiver vessel vented to the lower compartment. From an air balance performed from pressure and temperature measurements at the time of peak compression pressure (9.6 psig), it was found that the ratio of the change in air mass to the initial mass in the dead-ended volume was:

$$\frac{\Delta M_a}{M_{ao}} = 0.18$$

This change in air mass is then corrected for the lower compression peak pressure of the plant design to give:

$$\frac{\Delta M_a}{M_{ao}} = 0.18 \times \frac{7.8 \text{ psig}}{9.6 \text{ psig}} = 0.15$$

The storage of air in the dead-ended volumes has the effect of reducing the mass of air stored in the downstream volumes at the time of the compression peak pressure.

The compression ratio for the Cook Nuclear plant taking into account the dead-ended volumes is found from the following:

$$C_r = \frac{V_1 + V_2 + V_3 - 0.15\ V_4}{V_3 + 0.645\ V_2} \qquad (5)$$

Where:

V_4 = Dead-ended volumes (instrument room and pipe trench)

Substituting the plant design compartment volumes as shown in Table 14.3.4-11, the compression ratio calculated from Equation (5) is:

$$C_r = \frac{1,179,636 - 0.15 \times 61,702}{745,896 + 0.645 \times 126,940}$$

$$C_r = 1.41$$

The final peak pressure is:

$$P_3 = 15.0\ (1.41)^{1.13} + 0.4$$

$$P_3 = 22.5 \text{ psia or } 7.8 \text{ psig}$$

Therefore, the effect of the dead-ended volume of 61,702 ft^3 is to decrease the final peak compression pressure by 0.2 psig. The magnitude of this effect was further substantiated by a series of tests at Waltz Mill which were run at a mass compression ratio closely representative of the Cook plant design. Tests were run with and without a dead-ended volume equivalent to 155,000 ft^3 for the containment design. In these tests, the effect of the dead-ended volume was measured to be 0.5 psig, which is equivalent to a 0.32 psi decrease in final peak pressure per 100,000 ft^3 of dead-ended volume.

14.3.4.4.7.3 Effect of Steam Bypass

The method of analysis used to obtain the maximum allowable deck leakage capacity as a function of the primary system break size is as follows.

During the blowdown transient, steam and air will flow through the ice condenser doors and also through the deck bypass area into the upper compartment. For the containment this bypass area is composed of two parts, a known leakage area of 2.2 ft^2 with a geometric loss coefficient of 1.5 through the deck drainage holes located at the bottom of the refueling canal, and an undefined deck leakage area with a conservatively small loss coefficient of 2.5. A resistance network similar to that used in TMD is used to represent 6 lower compartment volumes, each with a representative portion of the deck leakage and the lower inlet door flow resistance adjacent to the lower compartment element. The inlet door flow resistance and flow area is calculated for small breaks that would only partially open these doors.

The value of 2.8 sq. ft. was selected for the unidentified deck bypass area based upon the sensitivity of pressure response results to deck leakage and the deck design itself. At the time of early design reviews, potential leak paths from the lower to upper volume were identified and evaluated.

Based upon the engineering design of the leakage barriers, it was considered that the only meaningful path for deck leakage was through the various equipment hatches and manways from the lower to upper volume. Although the design of these hatches results in extremely tight seals, tortuous paths for leakage, and redundancy in protection, a 1/8" gap was assumed to exist around each of the hatches and manways. This, when multiplied by the periphery of the hatches and manways, yielded a bypass area of approximately 2.8 sq. ft.

Regarding other leakage paths, the direct leakage through the various drains was considered explicitly in the identified leakage component of 2.2 sq. ft. Leakage through the backdraft damper of the air return fans was determined to 0.18 sq. ft./damper. Addition of this to the 2.2 sq. ft. identified leakage yields a total of 2.5 sq. ft. Consideration of the additional 0.36 sq. ft. increases the pressure by less than 0.05 psi.

The coolant blowdown rate as a function of time is used with this flow network to calculate the differential pressure on the lower inlet doors and across the operating deck. The resultant deck leakage rate and integrated steam leakage into the upper compartment is then calculated. The lower inlet doors are initially held shut by the cold head of air behind the doors (approximately one pound per square foot). The initial blowdown from a small break opens the doors and removes the cold head on the doors. With the door differential pressure removed the door position is slightly open. An additional pressure differential of one pound per square foot is then sufficient to fully open the doors. The nominal door opening characteristic as shown in Figure 14.3.4-192 was used in the analysis.

One analysis conservatively assumed that flow through the postulated leakage paths is pure steam. During the actual blowdown transient, steam and air representative of the lower compartment mixture would leak through the holes; thus less steam would enter the upper compartment. If flow were considered to be a mixture of liquid and vapor, the total leakage mass would increase but the steam flow rate would decrease. The analysis also assumed that no condensing of the flow occurs due to structural heat sinks. The peak air compression in the upper compartment for the various break sizes is assumed with steam mass added to this value to obtain the total containment pressure. Air compression for the various break sizes is obtained from the full-scale section tests conducted at Waltz Mill.

The allowable leakage area for the following Reactor Coolant System (RCS) break sizes was determined: DE, 0.6 DE, 3 ft^2, 8 inch dia., 6 inch dia., 2.5 inch dia., and 0.5 inch dia. For break sizes 3 ft^2 and above a series of deck leakage sensitivity studies were made to establish the total steam leakage to the upper compartment over the blowdown transient. This steam was added to the air in the upper compartment to establish a peak pressure. Air and steam were assumed to be in thermal equilibrium, with the air partial pressure increased over the air compression value to account for heating effects. For these breaks sprays were neglected. Reduction in compression ratio by return of air to the lower compartment was conservatively neglected. The results of this analysis are shown in Table 14.3.4-12. This analysis is confirmed by Waltz Mill tests conducted with various deck leaks equivalent to over 50 ft^2 of deck leakage for the double ended blowdown rate and is shown in Figure 14.3.4-193.

For breaks 8 inches in diameter and smaller, the effect of containment sprays was included. The method used is as follows: For each time step of the blowdown the amount of steam leaking into the upper compartment was calculated to obtain the steam mass in the upper compartment. This steam was mixed with the air in the upper compartment assuming thermal equilibrium with air. The air partial pressure was increased to account for air heating effects. After sprays were initiated, the pressure was calculated based on the rate of accumulation of steam in the upper compartment. Reduction in pressure due to operation of the air recirculation fans has been conservatively neglected.

This analysis was conducted for the 8 inch, 6 inch and 2 1/2 inch break sizes assuming two spray pumps were operating (4000 gpm at 80°F). As shown in Table 14.3.4-12, the 8 inch break is the limiting case for this range of break sizes although the 0.6 DE is the limiting case for the entire spectrum of break sizes. With one spray pump operating (2000 gpm at 80°F) the limiting case for the entire spectrum of break sizes is the 8 inch case and results in an allowable deck leakage area of approximately 35 ft^2.

A second, more realistic method was used to analyze this limiting case. This analysis assumed a 30 percent air/70 percent steam mixture flowing through the deck leakage area. This is conservative considering the amount of air in the lower compartment during this portion of the transient. Operation of the deck fan would increase the air content of the lower compartment, thus increasing the allowable deck leakage area. Based on the LOTIC code analysis a structural heat removal rate of over 8000 BTU/sec from the upper compartment is indicated. Therefore a steam condensation rate of 8 lb/sec was used for the upper compartment. The results indicate that with one spray pump operating and a deck leakage area of 56 ft^2, the peak containment pressure will be below design for the 8 inch case.

The 1/2 inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, either the lower compartment or the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area and that which can be tolerated without exceeding containment design pressure.

14.3.4.4.7.4 Effect of Blowdown Energy

The sensitivity of the upper compartment compression pressure peak versus the amount of energy released is shown in Figure 14.3.4-194. This figure shows the magnitude of the peak compression pressure versus the amount of energy released in terms of percentage of reactor coolant system energy release. These data are based on test results wherein each of the tests were run at 110% and 200% of the initial blowdown rate equivalent to the maximum coolant pipe break flow.

These test results indicate the very large capacity of the ice condenser for additional amounts of energy with only a small effect on compression peak pressure. For example, during testing, 100% energy release gave a pressure of about 6.8 psig, while an increase up to 220% energy release gave an increase in peak pressure of only about 2 psi. It is also important to note that maldistribution of steam into different sections of the ice condenser would not cause even the small increase in peak pressure that is shown in Figure 14.3.4-194. For every section of the ice condenser which may receive more energy than that of the average section, other sections of the ice condenser would receive less energy than the average section. Thus, the compression pressure in the upper compartment would be indicated by the test performance based on 100% energy release rather than either the maximum energy release section or the minimum energy release section.

Figure 14.3.4-195 gives some insight as to the very large capacity for energy absorption of the ice condenser as obtained from test results. Figure 14.3.4-195 is a plot of the amount of ice melted versus the amount of energy released based on test results at different energies and blowdown rates. These test results indicate that a 200% energy release melts only about 74% of the ice while 100% energy release melts only 37% of the ice. Thus, even for energy release considerably in excess of 200% there would still be a substantial amount of ice remaining in the condenser.

14.3.4.4.7.5 Effect of Blowdown Rate

Figure 14.3.4-196 shows the effect of blowdown rate upon the final compression pressure in the upper compartment. Figure 14.3.4-196 is based on the results of a series of tests, all with the plant design condenser configuration, but with the important difference that all of these tests were run with 175% of the Reactor Coolant System energy release quantity. There are two important effects to note from Figure 14.3.4-196. One, the magnitude of the compression peak pressure in the upper compartment is low (about 7.8 psig) for the reactor plant design blowdown rate; and two, even an increase in this rate up to 200% blowdown rate produces only a small increase in the magnitude of this peak pressure (about 1 psi).

14.3.4.4.8 Subcritical Flow Model Studies

For high mach number subsonic flow, the TMD momentum equation incorporates a compressibility multiplier to account for compressibility effects resulting from area changes, and uses an average density along constant area flow paths. With these modifications, both inertial and density effects are modeled by the TMD computer code.

A description of the compressibility multiplier, its derivation and application, is presented in this section. A brief description of the method by which the polytropic exponent (a necessary parameter in the compressibility multiplier approach) is calculated is also provided.

These effects have been examined for the D. C. Cook plant short term transient analysis by comparing previous analyses where these methods were not used to analyses using these methods.

For the plant the worst case RCS pipe break is a DEHL rupture in the lower compartment element 6. Results are presented also, for comparison purposes, for a DECL rupture in element 6.

The results of the short term pressure analysis are summarized in Table 14.3.4-13. The values given in parentheses are those pressures calculated on the same basis but without using a compressibility multiplier. As can be seen from the table, the effects of the modifications to the TMD code are minimal.

Consideration was given to determining the effect of a varying polytropic exponent of the flow mixture across the throat section of a flow path. This was done by lowering the steam-water polytropic exponent calculated by the code by 5, 10 and 20%. The lowered polytropic exponent variance computer runs were made for a DEHL break in lower compartment element #6. The results are presented in Table 14.3.4-14 and it is apparent that the polytropic exponent variance has virtually no effect on the results.

Derivation of the Compressibility Multiplier

The system under study is shown in Figure 14.3.4-197. The flow assumptions are:

1. Steady flow
2. Zero gravity effects
3. Isentropic conditions
4. Fluid is an ideal gas
5. Channel wall is non-conducting (no heat transfer)

A detailed description of the calculation of the compressibility multiplier can be found in Question 03.5 to Appendix N of the Original FSAR. The final result is:

$$y = \left[r^{2/\gamma}\left(\frac{\gamma}{\gamma-1}\right)\left(\frac{1-r^{\frac{\gamma-1}{\gamma}}}{1-r}\right)\right]^{\frac{1}{2}} \times \left[\frac{1-B^4}{1-B^4 r^{2/\gamma}}\right]^{\frac{1}{2}}$$

The choked mass flow rate is:

$$\dot{m} = ay \left[\frac{2g\rho_1(P_1-P_2)}{1-B^4}\right]^{\frac{1}{2}}$$

Where

$$B = (a/A)^{\frac{1}{2}}$$

We next apply the compressibility multiplier to the friction term of the TMD momentum equation written as:

$$\Delta p = \frac{K + fl/D}{2\rho g} \frac{\dot{m}^2}{a^2} \qquad (1)$$

Incorporating the compressibility multiplier into the TMD momentum equation, eqn. (1) takes on the form:

$$\Delta p = \frac{(K + fl/d)\ \dot{m}^2}{2\rho g y^2 a^2} \qquad (2)$$

Coupling eqn. (2) with the inertia term presently used in TMD, the momentum equation for general flow systems (non-steady state) appears as:

$$\Delta p = \frac{L}{A}\frac{dm}{dt} + \frac{(K + flD)\ \dot{m}^2}{2\rho g y^2 a^2} \qquad (3)$$

It should be noted that the TMD computer code also employs a critical flow correlation as a check on sonic flow conditions, (see WCAP-8077). This critical flow correlation has not been modified as a result of this present work.

The compressibility multiplier as it is used in eqn. (3) (and in TMD) is calculated by the code; the only information needed as input is the B factor. The polytropic exponent is also calculated within the code, dependent upon the flow mixture conditions.

A brief explanation of the method by which the polytropic exponent, γ, is calculated, is given in Question 03.5 of Appendix N to the Original FSAR.

14.3.4.5 Containment Analysis For Steam Line Break*

14.3.4.5.1 Double-Ended Steam Line Breaks

To illustrate the substantial margin to containment design pressure following a postulated rupture of a steam line, an analysis of containment pressure was performed using extremely conservative mass and energy release rates. These mass and energy releases were established to first, maximize the energy discharged to containment, and second, minimize the time for discharge, without regard to the physical impossibility of attaining this rate or magnitude.

This provides a set of assumptions that are calculationally convenient and that are not intended to provide just a high estimate or resultant pressures, but rather an upper bound above which precisely calculated pressures would not go.

Analyses have been done and reported Section 14.3.4.6.3 for complete double-ended breaks in the steam generator doghouse and in the fan room. In that section, assumptions had been selected to maximize the rate of energy release, but not necessarily the total energy release. Compartment differential pressures are dependent on the rate of energy release, whereas the containment compression ratio pressure increases slightly with the total energy release. The non-mechanistic model described below was selected to provide a conservative, upper-bound limit on the total energy release.

The following assumptions were made for a break at the exit of the steam generator, upstream of the flow-limiting nozzle:

1. An instantaneous double-ended rupture of the 29.75 inch ID pipe was assumed, resulting in a total break area of 9.654 ft^2 (4.827 ft^2 for each end).

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*See Unit 2 FSAR Section 14.3.4 for current licensing basis. The following material represents the original licensing basis for Unit 1.

2. The break was assumed to occur at no-load conditions, at which time the steam pressure is highest (1020 psia at no-load, compared with approximately 760 psia at full load), and the fluid inventory of the steam generators is highest.

3. Initial mass velocity was assumed to be 2120 lbs/sec-ft^2, corresponding to the Moody value for dry steam and critical flow from an infinite reservoir. This assumption conservatively neglects the decompression wave which will travel away from the break and reduce the initial pipe blowdown to approximately 65% of the mass velocity assumed. The resultant mass release rate, for the mass velocity cited, is 20,466 lbs/second, or 10,233 lbs/second from each end of the pipe.

4. The initial mass release rate was assumed to continue unabated for 10 seconds, at which time steam line isolation valves were assumed to be closed, reducing the discharge rate to 10,233 lbs/second from the steam generator. This assumption neglects frictional line losses, depressurization, and the existence of the flow-limiting nozzles. The plant is equipped with flow-limiting nozzles with a 16 inch throat, or 1.4 ft^2 throat area. Although these devices are provided to minimize the steam flow in the event of a steam line rupture, their effect was conservatively neglected for this evaluation.

5. Fluid was assumed to be discharged as dry steam at an enthalpy of 1205 BTU/lb. This is the maximum enthalpy of dry steam, and can occur only in the vicinity of 450 psia. Since dry steam contains more energy per unit mass than liquid, credit for fluid entrainment was not taken.

6. Blowdown from the steam generator was assumed to continue at the initial rate of 10,233 lbs/sec. until 15 seconds. At this time, total discharge from the steam generator amounted to 153,600 lbs. This is the fluid inventory of one steam generator at no-load conditions.

7. After 15 seconds, blowdown is completed. At zero power, the main feedwater valves are closed and therefore no feed flow was considered.

With the above conservative assumptions, the total mass and energy release is 255,825 lbs. of steam (no water) and 308,269,125 BTU, respectively. As noted above, all of this energy was assumed to be discharged within 15 seconds.

The TMD computer code has been used to calculate the pressure and temperature transients following a double-ended steam line break accident. Figures 14.3.4-198 to 14.3.4-201 give the results for the break compartment and the upper containment. The TMD modeling used in this analysis assumed the break location in TMD element #2 (see Figure 14.3.4-33). The peak pressures are 9.9 psig in the break compartment occurring at 15.0 seconds. The peak temperatures are 330°F in the break compartment occurring at 10.0 seconds, and 130°F in the upper containment occurring at 15 seconds.

14.3.4.5.2 Steam Line Breaks At Other Than Test Standby Conditions

An analysis has been performed to evaluate containment pressure response to a steam break at other than test standby conditions. The initial conditions used for this evaluation were chosen to correspond to a steam break during full power operation. Feedwater temperature is highest at full power, and therefore the feedwater system would have the greatest effect on the blowdown transient. Assumptions were chosen to maximize the long term energy release to containment. The short term compartment pressurization effects are strongly rate dependent and would be less limiting for a break during power operation because of the lower initial secondary system pressure (full load steam pressure is approximately 760 psia versus approximately 1020 psia at hot shutdown).

The results presented are based on the assumed double-ended severance of a main steam line at the steam generator exit nozzle. A reactor trip and safety injection actuation was assumed to occur at 5 seconds after the rupture and the main steamline isolation valves were assumed to be closed 10 seconds after the rupture. This assumption is conservative, as the containment pressure transient analysis indicates actuation of a safety injection signal approximately 0.1 seconds after the break. The safety injection signal would trip off the feedwater pumps, close the feedwater control valves (5 second closure time) and close the feedwater pump discharge valves (2 minute closure time). For this evaluation, the feedwater control valve in the line feeding the affected steam generator was assumed to fail to close.

The following systems provide the necessary protection in the unlikely event of a main steam line rupture during power operation:

1. Reactor trip on any of the following (see Section 2.2 of the Technical Specifications):

   a. Hi neutron flux.

   b. Hi flux rate.

   c. Overtemperature ΔT.

   d. Overpower ΔT.

   e. Low Primary System Pressure.

   f. Low-low Steam Generator Water Level.

   g. Safety Injection Signal.

2. Safety Injection system actuation (and reactor trip) on any of the following:

   a. High Containment Pressure.

   b. Low Pressurizer Pressure coincident with Low Pressurizer level.*

   c. High Steamline flow coincident with Low Steamline Pressure or Low Primary coolant temperature.

*SI requires only Low Pressure, as a result of post-TMI revisions, although the analyses presented here are as stated.

3. Steamline isolation valve actuation on:

   a. High Steamline Flow coincident with either Low Steamline Pressure or Low Primary Coolant Temperature.

   b. High-High Containment Pressure.

4. Feedwater system isolation (including emergency closure of the feedwater regulating valves, trip of the feedwater pumps, and closure of the feedwater pump discharge valves) on:

   a. Any Safety Injection Signal.

   b. High-High Steam Generator Level.

All of the actuation signals above are redundant and designed as part of the protection system. For example, actuation of a feedwater regulating valve, which was assumed to fail for the purposes of this evaluation, is accomplished by redundant, independent instrument channels including redundant solenoids on the regulating valve air supply.

The failure of the feedwater regulating valve would result in water downstream of the feedwater pump discharge valve having a flow path to the steam generator throughout the transient. Since the feedwater is hot (435°F) during full power operation, it can expand into the steam generator once the steam generator pressure drops below the saturation pressure of the feedwater. Since the feedwater trains are interconnected, the volume downstream of the pump discharge valve in both trains has a path to the steam generator. This piping has a total volume of 5,000 ft^3, of which 150 ft^3 is downstream of the feedwater control valve. The mass of fluid available of expanding into the steam generator is listed in Table 14.3.4-15. In calculating the effect of vaporization, this volume was assumed to be homogeneous with an enthalpy corresponding to full power feedwater enthalpy at the steam generator inlet. Because of the large specific volume of steam at low pressures, the homogeneous mixture assumption results in the expulsion of nearly all of the mass from the feedwater train. The phase separation effect would result in a much larger amount of mass remaining in the feedwater system and correspondingly less energy release to the containment.

In addition to the feedwater expansion effect, an allowance was made for feedwater pumping following the break and auxiliary feedwater flow to the affected steam generator. Following the rupture, feedwater flow to the steam generator was assumed to increase to approximately 200% of normal flow at full load as a result of steam generator depressurization. Following the feedwater pump trip assumed at 5 seconds, the feed flow was assumed to decrease linearly to zero at 10 seconds after the rupture.

Blowdown from the steam generator would be nearly homogeneous for a large rupture. The moisture separating equipment would be completely ineffective. (For example, during normal operation, the swirl vanes throw liquid into the downcomer region: during blowdown, liquid in the downcomer region is blasting and discharging to the swirl vanes.) The only separation that would exist is due to the tendency of steam to rise faster than water.

In order to conservatively overestimate the amount of steam discharged, the Armand void correlation was applied in the following manner:

At each time during the blowdown, the void content in the entire steam generator (based on total mass of steam) was used to determine the quality of fluid discharged according to the modified Armand correlation:

$$\alpha = \frac{(0.833 + 0.167\ X)\ X\ \nu g}{(1 - X)\ \nu_f + X\nu g}$$

The Armand void fraction correlation is intended for steady flow in piping; its application to vessel blowdown underpredicts the amount of liquid discharged (conservative in this application) as shown by the following:

1. A calculation for steam generator blowdown from no-load conditions to atmospheric pressure without heat transfer has been done, and indicates a residual water inventory of 20,000 lbm, or 13% of the initial fluid inventory of 153,600 lbm. Extrapolation of available vessel blowdown data to a break area/vessel area ratio of 0.028 (break area of 4.83 sq. ft. and maximum steam generator cross-sectional area of 154 ft^2) indicates a residual water inventory less than 6% of the initial fluid mass.

2. Application of the Armand correlation implies a difference in the vertical velocities of steam and water in the steam generator of 4 to 26 ft/sec, depending on the time during blowdown. These differential velocities are a factor of 4 to 8 higher than predicted by the Davis bubble rise model.

The blowdown rate was determined from the Moody correlation with a discharge coefficient of 1.0 as a function of steam generator pressure and the quality calculated from the Armand correlation as discussed above. Average enthalpy of fluid discharged from the steam generator was 836 BTU/lb as compared to 1192 BTU/lb for dry steam. This steam generator blowdown was summed with the backflow from the other steam generator used in the evaluation of steam generator doghouse pressurization in Section 14.3.4.6.3. The calculation for backflow in 14.3.4.6.3 is based on no-load pressure (1020 psia); and is therefore conservative with respect to full load pressure (760 psia). The assumption was made for calculational convenience. The mass and energy release transients resulting from these calculations is shown in Table 14.3.4-16.

The TMD computer code has been used to conservatively calculate the pressure and temperature transients following a double-ended steam break at full power. The break was located in TMD nodal element 2. The TMD assumption of no structural heat sinks gives a conservative pressure and temperature transient late in time. The blowdown was run out until the spray system initiated at approximately 30 seconds; at this point the temperatures and pressures will begin to rapidly decrease. Since the energy released by this postulated accident will not melt out the ice bed, the pressure peak at 30 seconds will be the maximum pressure for the entire transient. Table 14.3.4-17 gives these pressure transients. The peak pressure is 7.3 psig in both the upper containment and the break compartment. The peak temperatures were 233°F in the break element and 127°F in the upper containment.

To place the above results in proper perspective, it should be noted that the D.C. Cook containment contains 2.45 x 10^6 lbs of ice. To melt this ice and bring the resultant mixture to a temperature of 175°F requires the heat addition of 722 million BTU. This would absorb the energy in 683,000 lbs of dry steam at 1200 BTU/lb, or approximately 5.8 times the amount of fluid in the steam generator at full power even if all of this fluid were hypothetically postulated to be dry steam. In addition to the ice, containment heat sinks and the containment cooling system would also absorb a substantial amount of energy.

14.3.4.6 Transient Mass And Energy Releases

14.3.4.6.1 Short Term Blowdown Analyses

14.3.4.6.1.1 Model Description

Mass and energy release rate transients generated for the TMD pressure calculation are supported by an extensive investigation of short term blowdown phenomena. The SATAN-V code was used to predict early blowdown transients. The study concerned a verification of the conservatism of the SATAN-V calculated transients. This verification was accomplished through two approaches: a review of the validity of the SATAN-V break model, and a parametric study of significant physical assumptions.

The SATAN-V code uses a control volume approach to model the behavior of the Reactor Coolant System resulting from a large break in a main coolant pipe. Release rate transients are determined by the SATAN-V break model which includes a critical flow calculation and an implicit representation of pressure wave propagation.

The SATAN-V critical flow calculation uses appropriately defined critical flow correlations applied for fluid conditions at the break element. For the early portion of blowdown, subcooled, saturated, and two-phase critical flow regimes are encountered. SATAN-V uses the Moody[8] correlation for saturated and two-phase fluid conditions and a slight modification of the Zaloudek[9] correlation for the subcooled blowdown regime.

Since most short term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition, the Zaloudek application is particularly significant. The Zaloudek correlation is modified to merge to Moody predicted mass velocities at saturation in the break element. This correlation appears in the critical flow routine of SATAN-V in the form:

$$G_{crit} = CK1 \sqrt{(5.553 \times 10^5)\ (P - C_1\ P_{sat})}$$

Where:

G_{crit} = critical flow in lb. mass/sec-ft^2

P = reservoir pressure (psia)

P_{sat} = reservoir saturation pressure (psia)

C_1 = constant where $.5 < C_1 < 1$

$CK1 = \sqrt{\frac{.1037}{1 - C_1}}$ = constant adjusted such that when $P = P_{sat}$, G_{crit} from Zaloudek matches the SATAN-V Moody critical flow calculated at zero quality. For the present analysis, C_1 equals 0.9 and CK1 equals 1.018. The modification also more conservatively accounts for the phenomena of increasing mass velocity with increasing degrees of subcooling. The slope of the subcooled G vs. P curve is steeper for the modified correlation.

The low quality portion of the SATAN-V critical flow model is presented in Figure 14.3.4-202. The Moody saturation line corresponds to the condition upstream in the break element where quality equals zero and pressure equals saturation pressure. Thus when pressure equals saturation pressure in the break element the Zaloudek and Moody critical flow values are equal. When pressure exceeds saturation pressure in the break element, the modified Zaloudek is used for the critical flow calculation. The steep slope of the Zaloudek G vs. P line indicates the over-accounting for the subcooling effect.

14.3.4.6.1.1.1 <u>Comparison To Other Critical Flow Models</u>

The Henry-Fauske critical flow correlation was considered for comparison[5, 10, 11]. This correlation models flow nonequilibrium via an approach which includes an empirical parameter. This parameter describes the deviation from equilibrium mass transfer and depends on flow geometry.

The value is selected for a particular configuration based on the range of throat equilibrium qualities. The value for constant area ducts is used in the present analysis. This choice is based on the worst possible double-ended break geometry described below.

For cold leg and hot leg breaks, the majority of the flow, about 65%, comes from the vessel side of the break. For this side, the geometry may be described as an entrance nozzle and a straight pipe of approximately 12 feet in length with a diameter of 29 inches. This length of pipe represents the distance from the reactor vessel to the periphery of the biological shield. No double-ended break can occur within the biological shield because of the restricted movement within the pipe annulus. Hence the constant area value is appropriate.

Like the SATAN-V model, the Henry-Fauske correlation yields a G_{crit} in terms of upstream conditions and like the SATAN-V model it also exhibits a steeper slope of the G vs. P line for subcooled conditions. As can be seen in Figure 14.3.4-202, the Henry-Fauske saturated liquid line is below the Moody saturated line (SATAN-V model) for pressures greater than about 1000 psia. For short term blowdown calculations, the significant pressure region is from 1000 psia to 1800 psia, with increased emphasis on subcooled conditions for the 1000 psia end. Subcooled mass velocity versus pressure is given for the two fluid temperatures corresponding to P_{sat} = 1000 and P_{sat} = 1800. It is clear from the figure that the slope of the Zaloudek G vs. P line is steeper in both cases. This increased sensitivity coupled with the higher value for Moody at saturation causes the SATAN-V model to predict higher mass velocities. Hence the SATAN-V model is a more conservative treatment of critical flow than the Henry-Fauske model.

In the original FLASH model,(12) the Moody correlation was extended to subcooled conditions. This treatment is employed in many blowdown codes and thus it is appropriate to compare the SATAN-V model to these values. This is illustrated in Figure 14.3.4-203. Again, the Zaloudek treatment yields higher mass velocities and the SATAN-V model is more conservative.

14.3.4.6.1.1.2 Comparison To Experimental Data

The margin included in the modified Zaloudek prediction of subcooled critical flow rates is demonstrated by a review of experimental subcooled critical flow data. Figures 14.3.4-204 and -205 present a plot of measured vs. predicted critical flow values for Zaloudek's own data.(9, 13) The figures indicate that when the modified correlation is applied to Zaloudek's data, the predicted critical flow values are significantly higher than measured flow rates.

The margin associated with the SATAN-V critical flow calculation may also be demonstrated by a review of the low quality data presented by Henry in ANL-7740.(11) Exit plane quality, in terms of the Moody model, is determined as a function of upstream conditions by assuming an isentropic expansion to exit plane (i.e., critical) pressure. The lowest exit plane qualities where the Moody model is applied in the SATAN-V code occur for expansion from saturated liquid conditions; a plot of these are shown in Figure 14.3.4-206. For exit plane qualities above the line, the Moody model is used in the SATAN-V code. Below the line, the Modified Zaloudek model is used.

Henry's comparison between data and model shows that for the range of exit plane quality greater than 0.02, the Moody model overpredicts the data, hence is conservative.

For the region below 0.02, it is appropriate to compare Henry's results with the Modified Zaloudek model, as used in the SATAN-V code. This is done in Figure 14.3.4-207 for all of Henry's data points. As can be seen, the Zaloudek model overpredicts the flow. A discharge coefficient of 0.6 would le more reasonable than the 1.0 value used in SATAN-V.

14.3.4.6.1.1.3 Application to Transient Conditions

The Zaloudek correlation was developed for stagnation (reservoir) pressure and quasi-steady-state critical flow conditions. It is extended to application in the SATAN-V break element and transient flow conditions. This extension is justified because of the following considerations.

The pressure in the break element differs from the value in a nearby large volume because of three effects:

1. Pressure drop due to friction
2. Pressure drop due to spatial acceleration (momentum flux)
3. Pressure drop due to the transient

The friction term in the reactor application is quantifiable; this term is less important than the other two. The sensitivity of the break flow rate to fluid friction was evaluated via a parametric study. For the purposes of this study, an analysis was made wherein the frictional resistance between the vessel and the break was reduced from the design values by a factor of one hundred. Over the period from 0.0 to 60 milliseconds (which includes the peak break flow), the integrated mass flow differed by less than 18 lbs from the design friction case; the total release over this period was about 5000 lbs.

Spatial acceleration is the major source of pressure drop upstream of the break between the reservoir and the pipe, causing steep pressure gradients in the approach region to critical flow. This term is not calculated explicitly in the SATAN-V code. Spatial acceleration is accounted for by the use of critical flow correlations (Zaloudek or Moody) which contain this effect. No credit is taken for pressure drop due to spatial acceleration for elements other than the break element. Hence the pressure calculated by SATAN-V may be interpreted as a stagnation pressure which is the appropriate pressure for the Zaloudek and Moody models.

Prior to the occurrence of the peak release rate, the break element and upstream reservoir pressures differ as a result of the transient described by pressure wave propagation. The applicability of the SATAN-V break model to this situation is verified by the code's ability to match recorded semi-scale transients. SATAN simulations of LOFT transients support the SATAN-V transient calculation. Figure 14.3.4-208 presents a comparison of LOFT pressure transients recorded near the break to the SATAN-V model of the LOFT break element transient. The graphs demonstrate the ability of the SATAN-V code to track pressure waves in the broken pipe.

Moreover, the critical flow correlation is implemented in the present analysis by combining the correlation with the appropriate momentum equation. This provides a model for predicting break flow acceleration vis-a-vis a quasi-steady simulation. This is found to have little effect on containment pressure but is a more physical representation.

Thus the SATAN-V break model is supported by subcooled critical flow data, by comparison to other correlations, and by ability to simulate short term transients.

14.3.4.6.1.2 Parametric Studies

With confirmation of the conservatism of the SATAN-V break model, a series of parametric studies were undertaken to identify the blowdown transient corresponding to the most severe TMD results. A series of basic sensitivities were first studied to set the scope of the more detailed investigations. The assumptions of break size, break type and break location were considered. The results of this analysis were evaluated using the TMD code.

14.3.4.6.1.2.1 Break Size, Type and Location

A break of an area corresponding to twice the coolant pipe area was the most severe for mass and energy release. For this size break both double-ended guillotine and double-ended split type breaks were considered. These break types differ in that the split allows full communication between approach regions at each side of the break while the guillotine models a complete severance of two ends of a broken coolant pipe.

SATAN-V transients were generated for both type double-ended breaks with the guillotine break resulting in higher mass and energy release rates. The split type break is less severe because flow is reduced from the loop side of the break. This is because communication makes the break element pressure higher than would occur for the loop end in a guillotine rupture. The higher break element pressure yields a smaller pressure gradient for driving loop side flow. The vessel end is relatively unaffected by break type because a choked condition remains at the nozzle. In particular, the split type break results in a 10,000 lb/sec reduction in peak mass flow rate.

The influence of break location on TMD peak pressure was considered by generating blowdown transients for possible worst break locations. The results indicated that a double-ended break in the pump suction leg was clearly less severe for short term blowdown release rates and that no such clear decision could be made between hot and cold leg breaks.

More detailed parametric studies were continued for the cold leg and the hot leg double-ended guillotine breaks. The two locations produce intrinsically different TMD pressure responses and therefore must be dealt with in separate parametric surveys.

14.3.4.6.1.2.2 Hot Leg Nodal Configuration

A study of the SATAN-V nodal configuration has been applied to the hot leg double-ended guillotine break. It was found that for this break the nodal configuration of the broken hot leg and the upper plenum are significant to short term transients. Spatial convergence was achieved for the upper plenum after the addition of four nodes to the standard SATAN-V two node upper plenum model. These nodes are hemispherical shells arranged concentrically from the broken hot leg nozzle and approximate the propagation of the pressure wave in the upper plenum. They are significant in that they specify the inertial response of the upper plenum. Spatial convergence was demonstrated because doubling the number of nodes yielded less than a one percent change in break flow at all times.

Sensitivity to nodal configuration in the broken hot leg pipe was also investigated. Models with from 4 to 16 nodes were used to generate transients. Increasing the number of nodes was found to give a better simulation of pressure wave propagation in the pipe.

14.3.4.6.1.2.3 Cold Leg Studies

The cold leg break transient was also reviewed in terms of significant parameters.

The Reactor Coolant System behavior is different for cold leg breaks and the peak containment pressure occurs later for cold leg breaks. The following studies were performed:

14.3.4.6.1.2.3.1 Nodal Configuration

For the cold leg break the nodal configuration of the broken cold leg and the downcomer is significant to the transient. Spatial convergence was achieved with the addition of three additional nodes to the standard SATAN-V model. These are annular rings arranged concentrically from the broken cold leg nozzle and model propagation of the pressure wave in the downcomer.

As in the hot leg sensitivity, from 4 to 16 pipe node models were tried for the cold leg transient. Again, more nodes gave a better simulation of pressure wave propagation in the broken pipe.

14.3.4.6.1.2.3.2 Pump Modeling

For the time period of interest, the variation in pump inlet density is small and the variation in pump speed is small. This model was found to have no effect.

14.3.4.6.1.3 Summary

From the hot leg and cold leg studies, the design basis mass and energy release rates have been finalized. The $\dot{m}$ and $\dot{m}h$ transients for all the design cases are given in Figures 14.3.4-209 to -218. All cases are generated from the SATAN-V break model consisting of Moody-Modified Zaloudek critical flow correlations applied at the break element. Since no mechanistic constraints have been established for full guillotine pipe rupture, an instantaneous pipe severance and disconnection is assumed for all transients. Assumptions specific to the presented transients are as follows:

For the hot leg mass and energy release rate transient to loop compartments:

FIGURES 14.3.4-209, -210

1. A double ended guillotine type break.
2. A break located just outside the biological shield.
3. A break located in the worst loop.
4. A six node upper plenum model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. A 100% power condition with T_{hot} = 606.4°F and T_{cold} = 540.4°F.

For the cold leg mass and energy release rate transient to loop compartments:

FIGURES 14.3.4-211, -212

1. A double ended guillotine type break.
2. A break located just outside the biological shield.
3. A break located in the worst loop.
4. A seven node downcomer model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. A full power condition with T_{hot} = 606.4°F and T_{cold} = 540.4°F.

For hot leg mass and energy release rate transients to subcompartments:

FIGURES 14.3.4-213, -214

1. A single ended split type break.
2. A break just outside the hot leg nozzle.
3. A break in the pressurizer loop.
4. A six node upper plenum model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. Full power condition T_{hot} = 606.4°F and T_{cold} = 540.4°F.

For the cold leg mass and energy release rate transient to subcompartments:

FIGURES 14.3.4-215, -216

1. A single ended split type break.
2. A break just outside the cold leg nozzle.
3. A break in the pressurizer loop.
4. A seven node downcomer model.
5. A 16 node broken hot leg pipe model.
6. A discharge coefficient (C_D) equal to 1.
7. A full power condition T_{hot} = 606.4°F and T_{cold} = 540.4°F.

For the mass and energy release rate transient to the pressurizer enclosure, a 6 inch safety valve pipe break was considered (Figures 14.3.4-217, -218):

1. A guillotine type break modeled as a 0.147 ft^2 split in the cold leg at the pump discharge (area of the six inch pressurizer spray feed line) and a 0.087 ft^2 split in the top of the pressurizer (area of 4 inch spray nozzle).
2. Valves in spray lines are assumed to be open.
3. No pipe resistance for the feed line considered.
4. A full power condition T_{hot} = 606.4°F and T_{cold} = 540.4°F.
5. A discharge coefficient (C_D) equal to 1.

Figures 14.3.4-213, -214, -215, and -216 present mass and energy release rate transients for hot leg and cold leg split type breaks of a single ended pipe area. For breaks of this size, the split type break is used as a design basis and this choice is justified by a generic study of the effect of break type on short term release rates. A discussion of this study and of break type influence was given as a response to question 6.71 to the Catawba PSAR (USNRC Docket No's. 50-413 and 50-414). It is sufficient for this discussion to note that for single ended breaks, a split type break results in higher release rates.

Differences in blowdown mass and energy release rates between hot leg and cold leg single ended split breaks result from the influence of the hot water in the upper plenum and hot legs. For a cold leg single ended split, the flashing fluid in the upper plenum and hot legs sustains flow to the break from both the vessel and from the loop through the broken loop pump. This flashing, then, acts to maintain a subcooled blowdown for the cold leg break.

For the hot leg single ended split no such pressurization effect occurs at the break. Flashing fluid in the hot leg and upper plenum, rather, results in an extensive two-phase blowdown condition. The broken leg pump continues to remain effective during the hot leg split transient and thus draws flow away from the break.

The hot leg and cold leg double ended release rate transients presented in the figures discussed above are the result of a guillotine type break. This basis is again justified as a result of the generic break type study referenced above. The study indicated that for breaks of twice the coolant pipe area, a guillotine type break resulted in the highest release rates.

An explanation of the differences in the release rate transients presented for hot let and cold leg double-ended breaks is complicated by the fact that these are guillotine type breaks. Since the guillotine break models a complete separation of the broken pipe, conditions at each end of the break must be considered individually. The total release rate is then the sum of contributions from each end.

Flashing of the fluid in the hot legs again accounts for the higher mass flow rates observed for the cold leg double-ended break in comparison to the hot leg double-ended transient. However, two other influences are significant for breaks of this type and area.

For the cold leg guillotine, the increased break area requires higher flows if a subcooled blowdown condition is to be maintained at the break. A subcooled blowdown occurs at the vessel end of the broken pipe but because of the broken loop pump resistance to increased flow, a two-phase blowdown occurs at the loop end of the break.

Since for both hot leg and cold leg breaks the loop side of the break experience a two-phase blowdown, the loop layout geometry determines the difference in their release rates. Higher release rates are observed for the loop side of hot leg break because it is fed from the reservoir of water in the inlet plenum of the steam generator. No such supply of water exists at the loop side of the cold leg break. In fact, flow to the cold leg loop side is restricted by the resistance of the broken loop pump.

The differences in release rates for the double-ended break are thus the result of two effects. A higher vessel side mass flow rate for the cold leg break results from a subcooled blowdown maintained by the pressurizing effect of flashing hot leg fluid.

A lower loop side mass flow is observed for the cold leg break because of the differences accountable to loop layout geometry. However, since the subcooled blowdown effect dominates the total release rate, the cold leg double-ended guillotine still results in highest total mass discharge rates.

14.3.4.6.2 Long Term Blowdown Analysis

The containment pressure response has been analyzed considering the steam generators as an active heat source during reflood. The analysis presented is for the double-ended pump suction break which has been found to be most conservative. Pump suction breaks yield the highest energy flow rate during the post blowdown period. This is because of the following: for the cold leg break, all of the fluid leaving the top of the core passes through the steam generators and may become superheated. However, the flooding rate is limited to a relatively low value by the resistance of the pump in the broken loop. For a hot leg break, the flooding rate is not so restricted but the majority of the fluid leaving the top of the core bypasses the steam generators and is not superheated. Thus the steam generators add much less energy. The pump suction break, on the other hand, has the relatively high flooding rate combined with all of the fluid passing through the primary side of the steam generators.

The calculational model may be divided into four parts: Blowdown, when the system pressure drops from 2250 psia to containment pressure; Refill, when the vessel inventory is increased to the bottom of the core; Reflood, where the water level moves into the core; and Post-Reflood, where mass/energy releases to the containment prior to the removal of steam generator sensible energy must be considered. A fifth accident phase, Post-Froth, occurs after the steam generator sensible energy is removed. This phase has been considered along with Post-Reflood for the analysis presented here.

14.3.4.6.2.1 Blowdown

The model for blowdown is similar to that used in the ECCS analysis. The SATAN code is used to simulate breaks in the various locations. All accumulators inject for breaks other than the cold leg.

The steam generator is modeled using several well known heat transfer correlations. When the heat flow in the steam generators is from primary to secondary, the heat transfer coefficient on the tube side is calculated using the Dittus-Boelter[14] correlation for subcooled forced convection. For secondary to primary heat flow, the tube side heat transfer coefficient is calculated using the Jens-Lottes[15] correlation for nucleate boiling. This calculation will be bypassed if the tubes experience Departure from Nucleate Boiling (DNB). Nucleate boiling heat transfer is continued until the DNB ratio, (DNBR) calculated using Macbeth's[16] correlation of critical heat flux, drops below the DNBR value of 0.7 input in the SATAN code. This value delays DNB in the tube until a local heat flux is achieved that is 1.43 times the critical heat flux calculated for local fluid conditions. After DNB is reached, the Dougall-Rohsenow[17] film boiling correlation is used. Should the fluid in the steam generator tubes become superheated, the superheat forced convection correlation developed by McEligot[18] is used. In the present model the heat transfer coefficient on the shell side when heat flow is from secondary to primary is calculated using McAdam's[19] recommended correlation for turbulent boundary layers on vertical surfaces. Table 14.3.4-18 lists all of the heat transfer correlations.

For the containment pressure analysis two major modifications have been made to the blowdown calculation to obtain a conservatively high energy release rate. The transition boiling correlation has been modified to give higher heat transfer coefficients and steam generator DNB time has been delayed to extend the period of nucleate boiling. For the Cook analysis a steam generator DNB time of 1.5 seconds has been used. In terms of energy released to containment, delaying steam generator tube DNB beyond 1.5 seconds has little effect due to the limiting phenomenon of heat transfer from the secondary fluid to the tube wall. If no DNB were permitted in the steam generators, an increase of only 0.2% in heat addition, as compared to the design basis case, would result!

The arbitrary extension of DNB until quality equals 100 percent (this corresponds to a time of 17 seconds for the Cook double-ended pump suction break) results in no increase in the total core heat release compared to the steam generator DNB = 1.5 second case. This is due, in large part, to the conservative modification made to the transition boiling heat transfer correlation for the purpose of containment pressure calculation. As a result of the modification an average rod film coefficient of approximately 10,000 BTU/hr-ft^2-°F is observed throughout the blowdown transient for the DNB = 1.5 second case. The 17 second DNB case also results in an average rod film coefficient of about 10,000 BTU/hr-ft^2-°F. The heat transfer coefficients predicted by the transition boiling correlation are comparable to nucleate boiling coefficients. On the other hand, the ECCS containment calculation, done with an ECCS type heat transfer treatment, results in an average rod film coefficient of approximately 100 to 300 BTU/hr-ft^2-°F.

The fluid volume contained in the primary system has been adjusted slightly. This volume reflects the correct system volume, calculated from component dimensions, plus 1.6 percent to account for thermal expansion and 1.4 percent to account for uncertainties.

The initial fluid energy is also based on coolant temperatures which are the maximum levels attained in steady state operation including allowance for instrument error and deadband (+4F). These were based on a power of 3459 MWt. The stored energy has been evaluated using a detailed temperature model of the pellet, clad and gap. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad, gap, and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, have been combined into a semiempirical thermal model. This thermal model has been incorporated into a computer code to enable the determination of these factors and their net effects on temperature profiles. The temperature predictions of the code have been compared to in-pile fuel temperature measurements and melt radius data with good results. Table 14.3.4-19 presents the results of a sensitivity study on core stored energy, in full power seconds above average coolant temperature, varying the following parameters:

1. Average power level
2. Number of nodes assumed in the pellet
3. Effect of fuel densification.

A conservative value of 7.9 (6.6 x 1.2) full power seconds, which includes fuel densification and additional margin, was used in this analysis. Moreover, core stored energy was based on a conservative value of 102% of the engineered safeguards design rating power level.

The margins cited above clearly indicate that the values used in this analysis represent a conservative upper bound of the core stored energy.

Figures 14.3.4-219 through 14.3.4-221 present the mass release rate and integrated energy release during blowdown for a spectrum of reactor coolant pump suction pipe break sizes. For the double ended pump suction guillotine with ten foot entrainment, Table 14.3.4-20 lists mass and energy release rates and integrated releases as a function of time.

The total mass and energy released into the containment for the appropriate coolant pump suction pipe breaks, along with the effective break area are presented in Table 14.3.4-21. The coolant energy release rates for these breaks are presented in Figures 14.3.4-222 through 14.3.4-224.

14.3.4.6.2.2 <u>Refill</u>

The calculations in this period use the conservative assumption that the bottom of core recovery occurs immediately after the end of blowdown, thereby increasing the head in the downcomer as soon as possible; this tends to result in immediate energy release to the containment.

14.3.4.6.2.3 <u>Reflood</u>

The SATAN calculations are performed until the completion of blowdown. In this context the end of blowdown is defined as the time at which zero break flow is first computed. At this time, the normal blowdown transient calculations are terminated and the reflooding calculations are performed.

The reflooding calculations are done in the following two steps:

1. Calculate the core inlet mass flowrate and the fraction of the inlet mass flowrate that leaves the top of the core. This hydraulic calculation yields the core flooding rate and entrainment fraction.

2. Calculate the core exit conditions due to the addition of various energy sources. Also perform calculations of the thermal conditions on the primary and secondary sides of the steam generators. This step is an energy balance calculation.

The REFLOOD Code consists of a fixed vessel model, two variable - geometry loops, and models for accumulators and pumped injection. In the vessel model, water levels in both the downcomer and core are calculated from the mass balance and momentum equations and the Westinghouse entrainment correlation for liquid carryover from the core. REFLOOD includes the effect of inertia in the core-downcomer liquid, and the pressure drop due to the elevation head of two-phase liquid above the core water front.

The model used for each of the coolant loops (broken and lumped unbroken loops) is very general. Each of the loops may have a maximum of 29 resistance elements in series. A typical schematic is shown in Figure 14.3.4-225. Provision is made for pressure drops within each element due to friction (fL/D), form factor (commonly called K-factors), and the dynamic pressure drop due to density change. The dynamic pressure drop due to area change is included at the interface between loop elements (and at the interface between the first element of each loop and the core). In the REFLOOD Code, the density of fluid flowing in each resistance element is determined from the local pressure and enthalpy. The loops are assumed to be quasi-static - there is no provision for mass buildup in any loop element.

The REFLOOD Code currently provides the following models and features:

1. The pressure at the top of the downcomer can be specified as the pressure of any element in either loop, or as containment back pressure.

2. In each loop, any element can be specified as the steam generator element. (The local enthalpy changes to that of superheated steam at the steam generator secondary side temperature at the inlet of the steam generator element.)

3. Pumped injection may be specified as a tabular head-flow curve, with delivery pressure specified as the pressure in any loop flow element, or containment back pressure.

4. Accumulator injection may be specified as a linear ramp in time. The core flooding rate is limited by the pressure in the core caused by the generation of steam when the reflood water is heated up by the hot fuel rods. Any steam generated in the core region must be vented through the intact and broken loops via the resistive paths of elements shown in Figure 14.3.4-225.

Steam which flows through the intact steam generator must encounter the injected water in the cold legs of the broken and intact loops. During the accumulator injection phase, an equilibrium calculation indicates that the amount of water available is sufficient to condense this steam, thus reducing the flow to the containment. Moreover, comparison of Donald C. Cook ECC injection data to CE tests reported in CENPD-63 REV-1 indicates that analysis with the condensation phenomena included is valid.

The pressure drops along the two paths include friction losses and dynamic pressure drops due to area and density changes. The pressure drop across the pump is calculated by assuming that the rotor is locked.

The fraction of calculated core flooding rates that is vaporized and entrained is calculated using the Westinghouse entrainment correlation, obtained from the FLECHT results. The core inlet temperature during reflood is assumed to change with time, starting at saturated conditions and decreasing with time, based on separate energy balances on the fluid in the lower plenum and the downcomer. The energy balance includes the effect of the correct distribution of hot metal heating the fluid in the lower plenum and downcomer. Figure 14.3.4-226 presents the transient core inlet temperature that is used in the entrainment correlation to calculate the carryout fraction. Entrainment is assumed to continue until the water level in the core reaches the 10' elevation. The Westinghouse entrainment correlation conservatively overpredicts the time for 10' quenching. Figure 14.3.4-227 presents a plot of quench time measured in FLECHT versus quench time calculated by the correlation. It clearly illustrates the conservatism in the present analysis.

The resulting transient values of core flooding rate and the entrainment fraction are presented in Figure 14.3.4-228. These results are used in the energy balance model to calculate mass and energy release rates to the containment for calculation of the containment pressure transient.

14.3.4.6.2.3.1 Energy Balance Model

The energy balance model consists of three reference elements which represent the core, the steam generator in the broken loop, and the steam generator in the intact loop. Figure 14.3.4-229 presents a diagram of the model where the variables shown are defined as follows:

m	= mass flow rate into the core (lbm/sec)
$(mh)_{in}$	= energy flow rate into the core (Btu/sec)
$(mh)_{exit}$	= energy flow rate out of the core (Btu/sec)
m_1	= mass flow rate to the broken loop steam generator (lbm/sec)
m_2	= mass flow rate to the intact loop steam generator (lbm/sec)
m_{hout1}	= energy flow rate from broken loop steam generator out into containment (Btu/sec)
m_{hout2}	= energy flow rate from intact loop steam generator out into containment (Btu/sec)
q_{heat}	= sum of heat sources to the core fluid (Btu/sec)
h_f	= saturated liquid enthalpy (Btu/lbm)
q_{SG1}	= heat flow rate from the broken loop steam generator (Btu/sec)
q_{SG2}	= heat flow rate from the "unbroken loop" steam generator (Btu/sec)

An energy balance is performed on the fluid entering and leaving the core to determine core exit conditions:

$$(mh)_{in} + q_{heat} = (mh)_{exit} + (m_{in} - m_{exit})h_f$$

The mass flow rate of fluid entering the core is identical to the calculated flooding rates times the product of the core area and liquid density. This fluid is taken to be at injection conditions. The heat source term is added to the fluid in the core and is the sum of the following:

1. Decay heat
2. Thick metal (reactor vessel) heat
3. Core stored energy left at end of blowdown
4. Thin metal energy remaining at end of blowdown

The decay heat contribution during the post-blowdown phase of the accident is considered in the energy balance model, and is calculated using the Westinghouse standard decay heat curve evaluated at 102% of the Engineered Safeguards Design Rating. This calculation yields a decay heat release of 16.38×10^6 BTU from end of blowdown to termination of entrainment at 8 ft. The integrated value of decay heat release from end of blowdown to 5000 seconds after the accident is 348.85×10^6 BTU's.

The core stored and thin metal energy that are remaining at end of blowdown are brought out at a constant rate over the period between the bottom of core recovery (end of blowdown) and the termination of entrainment. The thick metal energy decays exponentially with a time constant of 0.0032 seconds^{-1}.

Heat addition due to zirconium-water reaction is not a significant energy source for the double-ended pump suction break. Throughout the blowdown and post-blowdown portions of the transient, clad temperatures are limited to a range in which a negligible amount of zirconium-water reaction occurs.

The mass flow rate leaving the core is equal to the inlet flow rate times the entrainment fraction calculated from the hydraulic model. The difference between inlet and outlet flow represents the fluid which remains in the core and this is heated to saturated liquid enthalpy.

The above considerations provide sufficient information to determine the core exit enthalpy.

In this calculation feedwater flow to the steam generators is ignored throughout the transient following the double-ended pump suction break. This is conservative as pressure feedwater addition during the transient would act to reduce the temperature at the secondary side of the steam generators. This would then decrease the superheat temperature of the steam leaving the primary tubes and venting to the containment. Thus a reduction in the energy released to the containment would accompany the assumption of continued feedwater flow.

The flow split during reflood between the unbroken loop and the broken loop steam generators is determined in the hydraulic model described earlier. Separate energy balances are performed on the broken loop and intact loop steam generators. Fluid which enters the primary side of the steam generator is assumed to be heated instantaneously to the shell side temperature. This sets the outlet enthalpy; the steam generator inlet enthalpy is equal to the core exit enthalpy. Hence the energy addition from the steam generators to the fluid entering the containment is determined.

This energy flow results in a decrease in internal energy for the shell side of the steam generator. Metal heat on the secondary side is included in the internal energy calculation. The steam generator secondary side fluid mass (and hence density) is taken as constant and temperature can be found directly from the internal energy.

The fluid which leaves the steam generator primary side is assumed to flow directly into the containment. No credit is taken for the quenching effect of the accumulator water which spills to containment.

14.3.4.6.2.3.2 Results

The mass and energy release rates calculated as a function of time during reflood by the above procedure are presented in Figure 14.3.4-230.

Table 14.3.4-22 lists the mass and energy release rate data for reflood at the specified times used in the analysis. Table 14.3.4-22 also lists the integrated mass and energy releases as a function of time.

14.3.4.6.2.4 Post-Reflood

During core reflooding droplets are entrained along the rods by the steam flow. Once the core is quenched, the entire bundle is wetted and a two-phase mixture exists at upper elevations in the bundle. Since there is a void fraction difference between the bundle and the downcomer (which is filled with water) the possibility of overflow of the two-phase mixture into the collection tank exists. The collection tank is located four feet above the top of the heated length corresponding to approximately the bottom of the hot leg nozzle. (The tank is actually one foot below the proper elevation with respect to the heated length.) The overflow depends on the void fraction distribution including the amount of liquid steam separation and the resistance of the various flow paths.

For the purpose of determining the containment heat removal capability versus the addition of steam generator heat, it was assumed that a two-phase mixture, or froth, entered the steam generators after the reflood phase of a double-ended break at the suction of the reactor coolant pump. This results in secondary-to-primary heat transfer and the faster addition of steam-generator stored energy to the containment, in addition to decay heat from the core.

A major influence on release rates during the froth period is the split of decay heat boiloff between the intact loop and broken loop venting paths. The post-reflood froth model has been modified to calculate a post-froth flow split. The flow split is calculated by assuming equal entrainment of water through the intact loop and broken loop venting paths, i.e., the mixture quality is assumed equal for flow through the intact and broken loops. This assumption leads to an overprediction of the steam flow through the broken loop. Since steam velocities in the intact loop are lower than steam velocities in the broken loop, lower entrainment levels and higher quality flow would occur in the intact loops. Thus more decay heat boiloff (a lower flow split) would pass through the intact loops to match the available driving head. This additional intact loop steam flow would then be subject to condensation by ECCS injection water.

Two long term containment pressure analyses have been performed, one arbitrarily assuming 100% flow split following removal of steam generator sensible energy and the other using a calculated flow split which considers potential intact loop plugging. For both cases the current post-reflood froth model (See WCAP-8312) has been used to calculate the mass/energy releases following reflood and prior to removal of steam generator sensible energy.

14.3.4.6.2.4.1 Calculated Flow Split Case (CASE A)*

A detailed calculation has been made of the post reflood mass and energy release transient for the D. C. Cook plant. The mass and energy release during the froth period was calculated by means of the current post-reflood froth model.

The period in the long term energy release transient following the release of steam generator sensible energy (the post-froth period) was also considered in detail. Calculation of release rates during this

*In the Original FSAR, Case A was named Case B and viceversa.

period was accomplished again by means of the post-reflood froth model. The code was extended and modified to model the post-froth system behavior.

In the post-froth calculation, the possible influence of the loop seals on the flow split was specifically considered. Two concerns were addressed:

1. The reduction of intact loop steam flow from the opposing head of the two-phase mixture in the 7.5 foot vertical section of the loop seal.

2. The possibility of complete plugging of the intact loop seals by fallback of injection water through the intact loop pumps.

The reduction of intact loop flow from the opposing head in the loop seal was calculated by modification of the post-reflood froth model. The Yeh correlation, which is used to calculate mixture densities in the core, upper plenum, and steam generators, was used in a similar manner to calculate the mixture density in the vertical portion of the loop seal. The loop seal head was then subtracted from the intact loop driving head.

The possibility of complete plugging of the loop seals was investigated for two discrete periods of the post-froth transient: the pre-recirculation period when the ECCS injection rate is 625 lb/sec and the recirculation period when the injection rate falls to 125 lb/sec. Prior to recirculation, no mechanism is indicated by which there would be fallback of injection water of sufficient quantity to plug the loop seal. However, even if complete plugging is arbitrarily assumed, the available driving head in the intact loops is more than adequate to blow the loop seals clear.

In the recirculation period, data recorded for the 1/3 scale steam-water mixing tests indicates that no significant fallback of injection water occurs. The 1/3 scale tests, run specifically to simulate PWR injection zone conditions during this period, were reviewed. The tests indicated first that for the range of recirculation steam and injection flows, no pressure or steam flow oscillations were observed. The thermocouple readings taken before the pump region but upstream of the injection zone give no indication of the presence of subcooled water in this portion of the pipe.

Complete plugging and significant fallback of ECCS water during these tests would result in oscillatory behavior. Thus the lack of oscillations in these tests and the recorded upstream thermocouple data give direct evidence that no significant fallback and plugging would occur during the recirculation period.

With the modified post-reflood froth model an extended calculation of the froth and post-froth transients was accomplished. Because of the above discussion, no complete plugging of the loop seal was considered. However, as discussed earlier, the reduction in the intact loop driving head from the two-phase mixture in the loop seal was included in the calculated flow split. The resulting transient is presented as Case A. The transient has been extended to 15000 seconds. The calculated flow split is presented in Figure 14.3.4-231.

The calculated flow split would continue to approach 100% as the void fraction in the loop seal decreased. For purposes of the containment response, a flow split of 100% at 16000 seconds was arbitrarily assumed for Case A. Figure 14.3.4-14 shows the containment response.

14.3.4.6.2.4.2 <u>100% Flow Split Case (CASE B)</u>

An addition to the post-froth treatment given as Case A, a post-froth transient (case B) was generated by arbitrarily assuming a 100% flow split following the release of steam generator sensible energy. This is an exceedingly conservative treatment of the post-froth system behavior and deviates considerably from any reasonably conservative calculation of release rates for this period. Figures 14.3.4-15 through 14.3.4-24 present the containment system response for this bounding case.

14.3.4.6.2.5 Energy Balances

For the double-ended pump suction case, the energy inventories for RCS components at the initiation of blowdown, at the end of blowdown, and at the end of reflood are given in Table 14.3.4-23. The mass and energy releases to the containment for the DEPS case are presented in Table 14.3.4-24. Energy distributions through the end of froth are given in Table 14.3.4-25 for the 100% flow split case of section 14.3.4.6.2.4.2.

14.3.4.6.3 Steam Line Break Blowdown Analysis

Two cases were considered for steam line breaks inside containment:

a. Break at the exit of the steam generator, upstream of the flow-limiting nozzle, in the 32 inch pipe (inside area of 4.83 ft^2). This break discharges into the steam generator doghouse.

b. Break in the fan room in the 30 inch pipe (inside area of 4.27 ft^2), downstream of the flow-limiting nozzle.

For both cases, a complete double-ended break was postulated. A time of 0.01 seconds was then assumed for the break to open with unrestricted discharge from both ends. The failure was conservatively assumed to occur at no-load conditions, where both steam line pressure (1020 psia at no-load, versus 758 at full load), and stored energy in the steam generator are the highest.

14.3.4.6.3.1 Steam Piping Blowdown

Blowdown of the steam piping was calculated with the SATAN computer code. The SATAN code does not consider momentum flux. Neglect of this effect is conservative for high velocity steam blowdown since it overpredicts the steam pressure near the break. Since steam pressure and steam density are overpredicted, frictional losses are underpredicted.

Piping blowdown consists of steam at 1192 BTU/lb (saturation enthalpy at 1020 psia).

Steam piping blowdown consists of reverse flow (steam flow coming out the turbine end of the break), and -- for the break in the fan room -- the initial steam blowdown from the steam generator end until choking conditions are reached in the flow restrictor.

The SATAN model consists of 63 elements simulating the four steam generators and steam lines and the steam dump header. For the fan room analysis, flow restrictors with a throat area of 1.4 ft^2 were assumed in the steam line cross ties near the turbine. For the doghouse analysis, credit was taken only for the 1.4 ft^2 flow limiter inside containment.

Reverse flow was assumed to be terminated after 10 seconds as a result of steam line isolation. No credit was taken for partial isolation valve closure prior to 10 seconds.

14.3.4.6.3.2 Steam Generator Blowdown

Initial blowdown from the steam generator will be dry steam as a result of the approximately 5000 lbs of steam in the upper head. This accentuates the inertial peak compartment pressure. For the doghouse break, flow rate was based on the Moody correlation for an initial reservoir pressure of 1020 psia, and included the steam generator exit nozzle loss. Depressurization of the steam generator causes an initial decrease in steam flow.

The following assumptions were made for calculating steam generator blowdown with entrainment. Note that these assumptions are in the conservative direction for maximum water entrainment.

1. No credit was taken for the separation capability of the steam generator internals (swirl vanes and dryers).

2. Flow between regions of the steam generator was assumed as homogeneous with no slip or separation. Regions of the steam generator are the downcomer, bundle, swirl vane cylinders, and dryers.

3. Flow resistance between the steam generator regions was considered.

4. No credit was taken for flow resistance in the piping between the steam generator and the break.

5. Break flow was determined by the Moody[8] correlation with the discharge coefficient conservatively assumed as unity.

14.3.4.6.3.3 <u>Results</u>

Calculated mass and energy release rates for the doghouse and fan room breaks are tabulated on Tables 14.3.4-26 and 14.3.4-27, respectively.

14.3.4.7 <u>Containment Subcompartment Analyses</u>*

Consideration is given in the design of the containment internal structures to localized pressure pulses that could occur following a loss-of-coolant accident. If a loss-of-coolant accident were to occur due to a pipe rupture in these relatively small volumes, the pressure

*The current licensing basis is located in Section 14.3.4.9, Unit 1 UFSAR. The following material represents the original licensing basis for Unit 1.

would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures.

These subcompartments include the steam generator enclosure, fan room, pressurizer enclosure, and upper and lower reactor cavity. Each compartment is designed for the largest blowdown flow resulting from the severance of the largest connecting pipe within the enclosure or the blowdown flow into the enclosure from a break in an adjacent region.

The following sections summarize the design basis calculations.

14.3.4.7.1 Steam Generator Doghouse

The largest connecting pipe to the steam generator is the steam line. The steam generator enclosure is designed for the case of a double-ended break at the exit of the steam generator, upstream of the flow-limiting nozzle in the 32 inch pipe (inside area of 4.83 ft^2). A time of 0.01 seconds was assumed for the break to open with unrestricted discharge from both ends. The failure was conservatively assumed to occur at no-load conditions, where both steam line pressure (1020 psia at no-load, versus 758 psia at full load) and stored energy in the steam generator are the highest. The mass and energy release rates for this break are given in Table 14.3.4-26.

The free volume of the steam generator doghouse and the vent area from the enclosure are listed in Table 14.3.4-28. The TMD computer code has been used to calculate the peak pressure, and peak pressure differential for this worst case. The TMD nodal model used considered the enclosure as a single element above the minimum vent area at the support structure. The peak pressure is 20.8 psig and the peak pressure differential is 20.5 psi.

A study has been conducted to find the effect of varying the number of nodes in the steam generator enclosure. A two node model, one node from the support structure to the steam generator inlet nozzle and the other from the inlet nozzle to the roof of the steam generator enclosure, yielded a .8 psi increase in differential pressure.

In addition to this model, 3 and 4 node representation of the enclosure have been studied. These additional nodes have all been added to the volume above the steam generator inlet nozzle. For these two cases, there are no increases in peak differential pressures.

14.3.4.7.2 Pressurizer Enclosure

The largest connecting pipe within the pressurizer enclosure is the pressurizer spray line. The pressurizer enclosure is designed for the case of a double-ended break in the 6 inch line from the reactor coolant pump outlet that feeds the pressurizer spray. To have water spill from both ends of the pipe, the valves in this line were assumed to be stuck in the open position. The break area on the pressurizer side of the line was assumed equivalent to the size of the 4 inch spray nozzle; on the pump outlet side, the break area was that of the 6 inch SCH 160 pressurizer spray line. The mass and energy release rates for this break are shown in Figures 14.3.4-217 and 14.3.4-218.

The free volume and vent area for the pressurizer enclosure are given in Table 14.3.4-28. The TMD calculated values for this break show the peak pressure in the enclosure to rise to 13.9 psig, with a peak differential pressure of 13.1 psi.

14.3.4.7.3 Fan Accumulator Room

The fan room enclosure is designed for a double-ended break in the 30 inch steam line (inside area of 4.27 ft^2) downstream of the steam line flow restrictor. The break occurs in the longest line with an orifice of 1.4 ft^2 in the cross connection with the steam dump header. This orifice restricts backflow so that the entrained flow from the other three steam generators will not reach the break before the steam stop valve closes at ten seconds, reducing the pressure peak. The mass and energy release rates for this case are presented in Table 14.3.4-27.

The pressure transient calculated by the TMD code is shown in Figure 14.3.4-232. The plot shows the pressure rising quickly to an inertial peak of 12.0 psig at 0.1 second, then rapidly falling off to 7.9 psig at 1.8 seconds. At 2.0 seconds, the pressure again rises rapidly due to the entrained flow from the steam generator upstream. The peak pressure of 13.9 psig is reached at 6.2 seconds. The peak differential pressure is 13.9 psi. The fan room free volume and vent area are given in Table 14.3.4-28.

14.3.4.7.4 Reactor Cavity

The design of the concrete structure surrounding the reactor vessel is designed for the following criteria.

1. Provide support for the reactor vessel under the dead weight, seismic, and reactor coolant pipe rupture loading conditions.

2. Attenuate the neutron flux sufficiently to prevent excessive activation of plant compartments.

3. Reduce the residual radiation from the core, reactor internals, and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.

As a result of criterion 1, the reactor support concrete structure will withstand the pressure that builds up within the annulus defined by the concrete cavity and the reactor vessel, following rupture of a reactor coolant pipe, without losing its structural integrity.

Calculations show that the maximum initial differential pressure within the reactor cavity and pipe annulus surrounding the piping from the vessel nozzles are 63 psi and 419 psi, respectively. These pressures are well within the capability limits of the reactor support structure. These values are based on the following assumptions:

a. A longitudinal split in the concrete pipe sleeve of area equivalent to the cross-sectional area of a reactor coolant pipe, i.e., 4.12 ft^2. A circumferential failure of the pipe at this location would result in a much smaller flow discharge area because the vessel, pipe, and sleeve arrangement is such that no significant relative movement can take place.

b. It is assumed that the insulation in the pipe annulus blows away and that there is a 1 inch crush in the insulation along the reactor vessel to the lower reactor cavity.

c. A break flow from the break of 76,384 lb/sec. for the hot leg and 75,785 lb/sec. for the cold leg.

d. The buildup of pressure in the annulus around the failed pipe causes the reactor vessel nozzle inspection plug to blow free. This plug does not present any problem from a missile generation standpoint because it is designed to allow the sand to blow out.

e. A total flow area out of the pipe annulus of approximately 22.75 ft^2. This includes the area of the inspection plug, the flow area into the lower containment volume, and the flow area into the reactor annulus.

f. The flow entering the reactor annulus is 25% of the initial flow at the break. The reason for this is that the discharge area from the pipe annulus into the reactor annulus is 25% of the total flow area out of the pipe annulus. 63% of the flow enters the upper reactor cavity; the rest enters the lower reactor cavity.

g. All vent areas are as given in Table 14.3.4-29.

In this case the flow entering the reactor annulus enters the upper and lower cavities before being vented to the lower ice condenser volume. Also, for the case of the compartment above the reactor vessel, flow enters the cavity via the reactor vessel nozzle inspection plug above the break location. The buildup in pressure in each compartment from this flow is calculated. The TMD computer code was used to analyze the pressure buildup in the compartments.

Calculated peak pressures and differential pressures for all subcompartments are presented in Table 14.3.4-30.

14.3.4.7.5 Sensitivity Studies

The TMD computer code was used to establish peak pressures and peak pressure differentials for double-ended hot and cold leg breaks, double-ended steam line breaks in the steam generator and fan room enclosures, a 6 inch spray line break for the pressurizer enclosure, and a single-ended pipe break in the reactor cavity of the D. C. Cook Plant. These cases were analyzed with and without augmentation of the calculated homogeneous equilibrium critical mass flow rates, to study the sensitivity of compartment pressures to augmentation. The double-ended hot leg break was assumed to occur in element 6, and the double-ended cold leg break was assumed to occur in element 1 of the TMD model network given in Figure 14.3.4-35. These were the worst locations for a hot leg and a cold leg break, respectively.

The pressure response to a hot leg break is only slightly affected by augmentation; the cold leg break pressure response exhibits significant sensitivity to augmentation. Since the hot leg break parameters are limiting for the D. C. Cook Plant, omitting augmentation increases the design basis peak operating deck ΔP less than 5%.

No change in the compartment peak pressures or pressure differentials occurred when unaugmented critical flow was used in analyzing the D. C. Cook Plant fan room and steam generator enclosure. Removing augmentation increased the peak pressure and the peak differential pressure in the pressurizer enclosure by 27% and 25%, respectively, in the upper reactor cavity by 17.5% and 19.5%, respectively, and in the lower reactor cavity by 13% and 8%, respectively.

The reason that there is no change in the peak pressures in the steam generator enclosure is that both the peak pressure and peak differential pressure are due to inertia. The fan room pressures remain constant because of high resistances in the flowpaths from the fan room to the lower compartment which prevent choking.

In the reactor cavities and pressurizer enclosure, peak pressures occur in the transient coincidental with choking, and therefore a significant change in calculated pressures will occur when the critical flow model is changed (see Table 14.3.4-30).

14.3.4.8 Door, Vent, and Drain Performance

14.3.4.8.1 Inlet Door Performance*

14.3.4.8.1.1 Introduction

The ice condenser inlet doors form the barrier to air flow through the inlet ports of the ice condenser.for normal plant operation. They also provide the continuation of thermal insulation around the lower section of the crane wall to minimize heat input that would promote sublimation and mass transfer of ice in the ice condenser compartment. In the event of a loss-of-coolant incident that would cause a pressure increase in the lower compartment, the doors open, venting air and steam relatively evenly into all sections of the ice condenser.

*For an up to date description of the lower inlet doors see Section 5.3 of this FSAR and references thereto.

The inlet doors are essentially pairs of insulated composite panels vertically hinged to a rectangular shaped angle section frame that has a center post. This assembly is fastened to the crane wall support columns which frame the ports through the crane wall from the containment lower compartment to the ice condenser compartment.

The door panels are made of composite steel sheets and urethane foam construction, comprising a total thickness of 7 inches to provide proper insulating characteristics. Each door is mounted to the frame with ball bearing hinges. The door panels are normally held shut against a bulb type gasket seal by the differential pressure produced by the higher density cold air of the ice condenser, sealing against loss of the ice condenser air to the lower compartment.

The door panels are provided with tension spring mechanisms that produce a small closing torque on the door panels as they open. The magnitude of the closing torque is equivalent to providing a one pound per square foot pressure drop through the inlet ports with the door panels open to a position that develops full port flow area.

The zero load position of each spring mechanism is set so that with zero differential pressure across the door panels the gasket seal holds the door slightly open. This provides assurance that all doors will be open slightly and relatively uniformly, prior to development of sufficient lower compartment pressure to cause flow into the ice condenser, therefore eliminating significant inlet maldistribution for very small incidents. For larger incidents the doors open fully and flow distribution is controlled by the inlet ports.

14.3.4.8.1.2 Design Criteria

14.3.4.8.1.3.1 Normal Operation

a. Doors shall be instrumented to allow remote monitoring of their closed position.

b. Doors shall be capable of being inspected to determine that they are functioning properly.

c. The inlet doors shall limit the leakage of air out of the ice condenser to the minimum practical limit.

d. The inlet doors shall restrict the heat input into the ice condenser to the minimum practical limit.

e. Normal maintenance and inspection must be performed in a manner that does not hinder the ice condenser performance or availability.

14.3.4.8.1.2.2 Accident Conditions

a. All doors shall open to allow venting of energy to the ice condenser for any leak rate which results in a divider deck differential pressure in excess of the ice condenser cold head.

The force required to open the doors of the ice condenser is sufficiently low such that the energy from any leakage of steam through the divider barrier can be readily absorbed by the containment spray system without exceeding containment design pressure.

b. Doors and door ports shall limit maldistribution to 150% maximum, peak to average mass input for the accident transient which provides adequate margin in the design ice bed loadings. This is used for any reactor coolant system energy release of sufficient magnitude to cause the doors to open. The inlet doors of the ice condenser are designed to open and distribute steam to the ice condenser in accordance with design basis above, for any postulated loss-of-coolant accident.

c. The doors are designed to eliminate the possibility of doors remaining closed, even for small break conditions. In particular, two degress of freedom of rotation are incorporated in the hinges and the sealing gasket is designed to pull out for a postulated condition of sticking. The gasket material is itself selected to prevent sticking.

d. The basic performance requirement for lower inlet doors for design basis accident conditions is to open rapidly and fully, to ensure proper venting of released energy into the ice condenser. The opening rate of the inlet doors is important to ensure minimizing the pressure buildup in the lower compartment due to the rapid release of energy to that compartment.

e. Ice condenser doors shall be protected from direct steam jet following a postulated steam line break.

14.3.4.8.1.3 Performance Capability

14.3.4.8.1.3.1 Normal Operation

The normal operation mode for the ice condenser inlet doors is to serve as an insulated barrier to natural convection heat and air flow through the ice bed, providing a sufficient insulating value to limit heat input into the ice bed. In addition the design and performance of the doors must be consistent with ensuring continuous availability of the ice condenser function.

The importance of normal operation criteria is the establishment of design parameters that provide for long term ice bed life, and for constant ice condenser availability for plant protection. In this context two inlet door design parameters affect these factors. They are heat conductivity and leak tightness.

Heat input is the parameter of prime concern, as it is the major factor influencing ice bed sublimation. Leakage out of the ice bed has been reduced to an insignificant amount. The heat input is a calculated value using a two dimensional heat conductance computer program that has been verified by tests.

The main heat input within the inlet door region to the ice bed is 5.5 BTU/hr-ft^2. This produces a calculated sublimation rate of 0.43% per year, using an analytical method for this calculation that has been verified by scale model sublimation tests. This performance predicts up to 20 years continuous ice bed operation prior to any need for ice replenishment.

The inlet door leakage is predicted by tests to be significantly less than the 50 CFM total used for the ice condenser design. This predicted leakage value has negligible effect on reinforcing the convective flow developed in the ice bed, therefore not affecting sublimation rates significantly. The effect of the make-up air entering the ice condenser due to this leakage is also negligible on refrigeration load or ice condenser air handling unit coil defrost frequency.

Seismic analyses associated with response data for the Cook Nuclear Plant shows that the ice condenser inlet doors will not be opened by the maximum seismic forces. This is due to the very low frequency of the rotational or opening mode, at which the response is negligible, and the ice condenser cold head pressure holds the doors closed.

Figure 14.3.4-192 shows the door opening characteristics as a function of door differential pressure based on a linear spring constant. Notably, there is no special significance to be attached to a linear spring constant, and detail design of the door and spring system indicated that non-linear spring characteristics changed the release rate at which maximum maldistribution would occur, but did not change the maximum maldistribution valve. The performance characteristics to be expected from the bottom doors would be typical of those shown in Figure 14.3.4-192.

The effect of maximum variation of door proportioning characteristics indicates significantly less maldistribution than the 150% limit.

Figure 14.3.4-233 shows the ratio of maximum flow area (provided by the door assumed to have the spring constant 10 percent lower than the average value) divided by the flow area for the average spring constant. Thus Figure 14.3.4-233 indicates the ratio of maximum steam flow which can enter the weakest door as compared to the average steam flow going through the other doors of the ice condenser. This figure indicates a peak value of maximum to average flow area into the different sections of the ice condenser of about 1.25. Thus, for the case of small pipe breaks wherein the bottom doors of the ice condenser are partially open, these doors will limit the ratio of maximum to average flow of steam into any section of the ice condenser to a reasonably low value.

The maximum maldistribution for the limiting case of maximum variation of door opening characteristics is shown for the range of small break release rates on Figure 14.3.4-234. As shown, doors having maximum variation in opening characteristics will limit maldistribution to a reasonably low value, about 1.25. However, it is important to note that redistribution of steam will occur in the ice condenser due to lateral flow of steam from one section of the ice condenser to another. There are resistances to steam flow laterally into other sections of the ice condenser, such that maldistribution of steam flow as limited by the doors alone will be further reduced as redistribution of steam occurs within the ice condenser.

Importantly, and as discussed in other reports, the ratio of maximum to average flow of steam into the ice condenser for pipe break sizes large enough to fully open the doors is limited by the door ports themselves to a reasonably low value, about 116 percent of the average.

The equilibrium position of the inlet door panels with zero load is slightly open (about 3/8 inch), which provides a small flow area at each door for uniform inlet flow into each segment of the ice bed. The doors are designed to eliminate the possibility of doors remaining closed, even for small break conditions. In particular two degrees of freedom of rotation are incorporated in the hinges, and the sealing gasket is designed to pull out for a postulated condition of sticking. The gasket material is itself selected to prevent sticking.

Consideration is, however, given in the analysis of ice condenser performance to a hypothetical case of stuck doors at which the most severe of the above postulated malfunctions is overcome by the force on the door. Even in this hypothetical case the door panels would rupture, providing a sufficient flow path into the ice condenser to permit the ice condenser to function to limit containment pressure below design limits.

It is recognized that the springs are an important part of the lower ice condenser doors. These spring assemblies are designed such that the failure of any spring will not significantly change the operating characteristics of the ice condenser doors. This objective has been achieved in a practical manner by the use of four separate tension springs per door, which provides redundancy and assures adequate opening characteristics.

14.3.4.8.1.3.2 Accident Conditions

The basic lower inlet door performance requirement for design basis accident conditions is to open rapidly and fully, to insure proper venting of released energy into the ice condenser. The opening rate of the inlet doors is important to insure minimizing the pressure buildup in the lower compartment due to the rapid release of energy to that compartment. The rate of pressure rise and the magnitude of the peak pressure in any lower compartment region is related to the confinement of that compartment, and in particular the active volume and flow restrictions out of that compartment. The time period to reach peak lower compartment pressure due to the design basis accident is a fraction of a second. It is dependent upon flow restrictions and proximity to the break location. The opening rate of the inlet doors is wholly dependent upon the inertia of the door and the magnitude of the forcing function, which is the pressure buildup in the lower compartment due to the energy release. The ice condenser inlet door inertia is slightly less than the doors tested in the ice condenser full scale section tests. These tests, as reported in WCAP-7183, Supplement 1, demonstrate that door inertia has essentially no effect on the initial peak pressure.

The maximum inlet door structural loading is due to the design basis accident for the doors adjacent to the lower compartment in which the release occurs. Structural analysis for maximum loaded conditions show that all door members remain well below allowable stress levels. Further verification of the structural adequacy of the door is provided by the proof load testing carried out on the full prototype doors.

The necessary performance of the ice condenser is further ensured by the door design incorporating a low pressure fail open characteristic. Even if it is postulated that the doors were held rigidly along the bottom edge, they would fail open at a differential pressure sufficiently low to allow venting from the lower compartment well within the limits of pressure capability of the structures.

14.3.4.8.2 Top And Intermediate Deck Door Performance*

14.3.4.8.2.1 Introduction

The doors enclosing the top of the ice condenser and forming the roof of the upper plenum are similar to, but lighter than the lower doors. These top doors are supported by the ice condenser bridge crane support

*This description has been kept for historical purposes. For an up to date description see Section 5.3 and references thereto.

structure. The crane support structure consists of radial beams spanning the ice condenser annulus at the top of the crane wall.

The doors enclosing the ice compartment and forming the floor of the upper plenum are similar to the doors described above. These doors are supported by the lattice frame support columns.

The door panels are a 2 1/2 inch foam plastic core with bonded sheet metal facings. The doors are hinged horizontally and are normally closed. On an increase in pressure in the ice condenser compartment, these doors will open as required, allowing air to flow into the upper containment volume.

A vent area of approximately 20 sq. ft. is provided through the floor and roof of the upper plenum to equalize pressure between the ice condenser and containment volumes during normal operating pressure fluctuations.

14.3.4.8.2.2 Design Criteria

14.3.4.8.2.2.1 Normal Operations

1. The top deck will be provided with a total vent area of approximately 20 ft^2.

2. Doors will limit heat input within their immediate vicinity to the minimum practical limit.

3. Doors will be capable of being inspected during plant shutdown to determine that they are functioning properly.

14.3.4.8.2.2.2 Accident Conditions

1. All doors will open fully for a low differential pressure loading.

2. All doors will be light-weight to have a minimum effect on the initial peak pressure.

3. Doors will be of simple mechanical design to minimize the possibility of malfunction.

4. Doors will not be required to remain either open or closed following an accident.

14.3.4.8.2.3 Performance Capability

The present design of the intermediate and top deck doors is shown in Figures 14.3.4-235 and -236. On an increase in pressure in the ice condenser compartment, these doors will open as required to allow air to flow into the upper compartment. The primary design criterion for these doors is their insulating capability to limit heat flow. The flow area provided by the open doors is that area available in the compartment, considering area reduction by support structures. Both the inertia of the door with the desired insulation capability and the available flow area have been modeled in the ice condenser door tests.

The mean heat input to the plenum through the top deck is about 4.5 BTU/hr-ft^2. This heat input is removed from the plenum ambient by the ice condenser air handling units and does not affect ice bed sublimation. The effect of this heat load on refrigeration heat load and plenum ambient conditions has been investigated and provides for operation well within the ice condenser design operating parameters.

The doors for both the top and intermediate deck have the same total flow area.

The door panels weigh about 3.2 pounds per square foot, which provides slightly less inertia than the doors tested in the ice condenser test program, which showed no significant effect due to opening characteristics.

The doors are sealed to the door frame using the same bulb type gasket seal developed and tested for the lower inlet doors. The sealing force for these doors is provided by the weight of the door panels. The basic design of the top deck provides an insulated frame, maintaining the frame and seal at a warmer temperature than the plenum, eliminating any frost buildup condition.

Incorporation of the vents impose no operational problem to the ice condenser. The lower doors will effectively seal the ice condenser compartment and limit any flow of air through the compartment to a negligible value. Therefore tight seals are not necessary at the top and intermediate doors. A balanced flapper is provided to minimize migration of moisture into the ice condenser through the vents.

The open flow area through each deck for air flow due to an incident is slightly larger than the ice condenser test ratio equivalent, providing slightly less resistance to air flow. The slightly larger flow area does not produce any significant change in ice condenser performance, other than to assure flow resistances slightly less restrictive than the reference design.

The top and intermediate deck and doors have been analyzed for all loading combinations. This structural analysis shows that all members remain below allowable stress levels.

14.3.4.8.3 Vent Design and Performance

14.3.4.8.3.1 Introduction

The upper and intermediate doors are not required to remain open following the reactor coolant system blowdown and also are not required to open for small breaks. For these situations, a vent was provided through both the top and intermediate deck to allow air to flow into or from the ice condenser compartment as required. This vent air first passes into the fan cooler plenum at the top of the ice condenser compartment where it mixes with the cooling air. The temperature and humidity of the vent air that passes from this plenum into the ice condenser compartment is therefore about the same as the average temperature and humidity of the air in the condenser compartment. Accordingly, sublimation or frosting in the ice bed due to this vent flow of air will be limited to a negligible value. Specific performance requirements are given below both for large breaks and for small breaks.

14.3.4.8.3.2 Large Break Performance Requirements

Following the reactor coolant system blowdown, the vents were designed to allow air to return from the upper compartment into the ice condenser without imposing an excessive pressure drop across the upper and intermediate doors. The maximum pressure decay rate and therefore the maximum reverse flow rate of air results from the case where reactor residual heat is not released to the containment following the reactor coolant system blowdown. The pressure decay for this case was measured in a full-scale section test. In this test, the pressure decayed from 6.2 psig to 4.5 psig during the one minute period immediately following the reactor coolant system blowdown. From this pressure decay rate, the plant equivalent flow rate of air was calculated to be 98 lb/sec. This flow rate developed a pressure drop across each of the upper doors of 0.28 psi, which was well within the structural capability of the upper and intermediate doors and support structures. Further, in this calculation it was conservatively assumed that no air flowed through the deck or through the containment air recirculation fan duct.

14.3.4.8.3.3 Small Break Performance Requirements

For small breaks which generate less than the required opening pressure of the upper and intermediate doors, the vent was designed to limit the flow of steam through the deck to an acceptable level during the period of air flow through the condenser and into the upper compartment. For breaks less than approximately 5000 gpm, the full-scale section tests have shown that only a fraction of air is displaced from the lower compartment. The 20 sq. ft. vent area in both the top and intermediate deck provides a low resistance air flow path through the ice condenser to the containment upper compartment for these small break conditions.

14.3.4.8.4 Drain Design and Performance

14.3.4.8.4.1 Introduction

Drains are provided at the bottom of the ice condenser compartment to allow the melt-condensate water to flow out of the compartment during a loss-of-coolant accident. These drains are provided with check valves that are counter-weighted to seal the ice condenser during normal plant operation and to prevent steam flow through the drains into the ice condenser during a loss-of-coolant accident. These check valves will remain closed against the cold air head (1 psf) of the ice condenser and open when the water head reaches a value of 18 inches of water or less.

For a small pipe break, the water inventory in the ice condenser will be produced at a rate proportional to the rate of energy addition from the accident. The water collecting on the floor of the condenser compartment will then flow out through the drains and through the doors, which are open during the blowdown.

For a large pipe break, a short time (on the order of seconds) will be required for the water to fall from the ice condenser to the floor of the compartment. Therefore, it is possible that some water will accumulate at the bottom of the condenser compartment at the completion of the blowdown. Such water accumulation could exert a back pressure on the inlet doors, requiring an additional pressure rise in the lower compartment to open the doors and admit steam to the ice condenser. However, results of full-scale section tests indicated that, even for the design blowdown accident, a major fraction of the water drained from the ice condenser, and no increase in containment pressure was indicated even for the severe case with no drains.

14.3.4.8.4.2 Large Break Performance Requirements

A number of tests were performed with the reference flow proportional-type door installed at the inlet to the ice condenser, the reference-type hinged door installed at the top of the condenser. Tests were conducted with and without the reference water drain area, equivalent to 15 ft^2 for the plant, at the bottom of the condenser compartment.

Tests were conducted with various assumed blowdown conditions. These tests were performed with the maximum reference blowdown rate, with an initial low blowdown rate followed by the reference rate, with a low blowdown rate alone, and with the maximum reference blowdown rate followed by the simulated core residual heat rate.

The results of all of these tests showed satisfactory condenser performance with the reference type doors, vent, and drain for a wide range of blowdown rates. Also, these tests demonstrate the insensitivity of the final peak pressure to the water drain area. In particular, the results of these full-scale section tests indicated that, even for the reference blowdown rate, and with no drain area provided, the drain water did not exert a significant back pressure on the ice condenser lower doors. This showed that a major fraction of the water had drained from the ice condenser compartment by the end of the initial blowdown. The effect of this test result is that containment final peak pressure is not affected by drain performance.

Although drains are not necessary for the large break performance, 15 ft^2 of drain area are provided for small breaks.

14.3.4.8.4.3 Small Break Performance Requirements

For small breaks, water will flow through the drains at the same rate that it is produced in the ice condenser. Therefore, the water on the floor of the compartment will reach a steady height which is dependent only on the energy input rate.

To determine that the 15 ft^2 drain area met these requirements, the water height was calculated for various small break sizes up to a 30,000 gpm break. Above 30,000 gpm the ice condenser doors would be open to provide additional drainage. The maximum height of water required was calculated to be 2.2 ft above the drain check valve. Since this height resulted in a water level which was more than 1 ft below the bottom elevation of the inlet doors, it was concluded that water will not accumulate in the ice condenser for this condition and that a 15 ft^2 drain will give satisfactory performance.

14.3.4.8.4.4 Normal Operational Performance

During normal plant operation, the sole function of the valve is to remain in a closed position, minimizing air leakage across the seat. To avoid unnecessary contamination of the valve seat, a 2 inch drain line is connected to the 12 inch line immediately ahead of the valve. Any spillage or defrost water will drain off without causing the valve to be opened.

Special consideration has been given in the design to prevent freezing of the check valves and to minimize check valve leakage.

To minimize the potential for valve freezing, a low conductivity (transite) section of pipe is inserted vertically below the seal slab, while the horizontal run of pipe (steel) is embedded in a warm concrete wall before it reaches the valve. The valve itself is in the upper region of the lower compartment, where ambient temperature is generally above the freezing temperature.

The valve is held in a closed position by virtue of its design as an almost vertical flapper with a hinge at the top. The slight (10^{o}) angle from the vertical holds the flap in place by gravity.

To reduce valve leakage to an acceptable value, a sealant was applied to the seating surface after installation of the valves. Tests show that this will reduce leakage to practically zero. Maximum allowable leakage rate would be approached as a limit only if all the sealant were to disappear completely from all the valves, which is unlikely. Sealant is replaced as necessary.

14.3.4.9 Short-Term Containment Analysis for Reduced Temperature and Pressure Operation

14.4.3.9.1 Introduction and Background

The containment building subcompartments are the fully or partially enclosed volumes within the containment which contain high energy lines. These subcompartments are designed to limit the adverse effects of a postulated high energy pipe rupture within them.

The short term mass and energy subcompartment analysis represents the initial seconds of the blowdown phase of the postulated rupture. The short-term analyses results are used in the design of the subcompartment walls in the ice condenser containment. The methodology that is currently used for the LOCA short-term M&E analysis, the SATAN-V computer code, is documented in Reference 26. The TMD computer program, assuming 100% entrainment and unaugmented critical flow, is used for the short-term containment response. The SATAN-V and TMD computer programs form the basis for the current analysis.

Section 14.3.4.3 of the Unit 2 FSAR describes the methodology and details of the TMD Short-Term Analysis. Information from this section, applicable to the analyses and evaluations discussed herein, includes: ice condenser performance, TMD analytical modeling and experimental verification.

Also included in the Unit 2 FSAR is a discussion of the specific application for the Cook Nuclear Plant design. Specific loop compartment results and sensitivities are presented in Section 14.3.4.3, and specific results for the pressurizer enclosure and steam generator doghouse are presented in Section 14.3.4.7.

The specific subcompartments analyzed as part of this effort include: the pressurizer enclosure, the fan accumulator room, the loop or lower compartments, the steam generator doghouse and the reactor cavity cubicle. These subcompartments represent a complete set of the major structures forming subcompartment boundaries. The analysis addressed the Cook Nuclear Plant Units 1 and 2, at rerated conditions assuming an NSSS power level of 3600 MWt, for a range of conditions which bound those shown in Table 14.3.4-31.

In some of the subcompartments analyzed, the calculated pressures resulting from the rerating conditions exceed the original structural design basis. The structural adequacy of these compartments was evaluated using acceptance criteria found in Section 5.2.2.3 of the FSAR and was confirmed.

14.3.4.9.2 Pressurizer Enclosure

The largest break possible in the pressurizer enclosure, a double-ended break of the spray line from the reactor coolant system, is postulated to occur at the top of the enclosure. A conservative determination of the effects of revised RCS parameters, such as enthalpy, on the mass and energy releases was made. Additionally, the effect of initial conditions, including:

- Temperature range of 60-160°F
- Pressure range of 13.2-15.0 psia
- Humidity range of 15-100 percent

on the containment response was determined. Generic parametric studies and specific computer runs made as part of the overall subcompartment evaluation were utilized to determine these effects. An evaluation in lieu of detailed computer analyses, because of significant design margin, was conducted for this subcompartment. The analysis and evaluation described in Section 14.3.4.7 of the Unit 2 FSAR were utilized as a starting basis for the evaluation. Figures 14.3.4-480 and 14.3.4-481 illustrate the noding, flowpaths and TMD network for the pressurizer enclosure.

The concrete structure is designed for a differential pressure of 80.00 psi. The maximum calculated differential pressure is 8.10 psi. Therefore structural integrity has been maintained for this subcompartment. Additionally, the pressurizer supports have been shown to be adequate for a differential pressure across the vessel as high as 1.30 psi. The maximum calculated differential pressure across the vessel is 0.38 psi, therefore the vessel supports are adequate.

14.3.4.9.3 Fan Accumulator Room

The largest break possible in the fan/accumulator room is a double-ended break of the steam line. The limiting mass and energy releases from this break are from the hot shutdown condition (no load), which do not change for the rerating. Therefore the analysis and evaluation described in Section 14.3.4.7.3 of the Unit 1 FSAR is still applicable.

14.3.4.9.4 <u>Loop Compartments</u>

Analyses and evaluations were conducted for this subcompartment. The containment, as described in Section 14.3.4.3 of the Unit 2 FSAR, was divided into 45 elements or compartments as shown in Figures 14.3.4-6 through 14.3.4-9. The interconnection between containment elements in the TMD code is shown schematically in Figure 14.3.4-10.

Mass and energy releases were developed for a double-ended hot leg break (DEHL) and double-ended cold leg break (DECL), the limiting breaks for the loop compartments. SATAN-V models, consistent with the methodology of reference (1), were developed utilizing the appropriate RCS data, such as enthalpies, pressures and flows. Peak rates increased by approximately 10% and 20% for the revised DECL and DEHL cases respectively when compared to those utilized in the Unit 2 loop compartment analysis described in FSAR Section 14.3.4.3.

Subcompartment pressurization effects were determined by making TMD runs including compressibility effects for the DEHL break in compartment 1, DEHL in compartment 2 and the DECL break in compartment 1. These breaks represent the limiting cases for the operating deck differential pressure, the upper and lower crane wall differential pressure and the peak shell pressure. The basis for selecting the break compartment was the previous analysis with results tabulated in FSAR Tables 14.3.4-6 through 14.3.4-9 for Unit 1 and FSAR Tables 14.3.4-4 through 14.3.4-7 for Unit 2.

The new base TMD DEHL in compartment 1 and DECL in compartment 1 model were identical to the TMD model discussed in the Unit 2 FSAR Section 14.3 but included revised mass and energy, 2,110,000 lbs of ice, and revised volumes, with the biggest effect being the mass and energy. The minor volume changes were for elements 25 through 33. The volume data in cubic feet respectively, for these elements is 734830., 10380., 27414., 10380., 17111., 10380., 27414., 10380. and 16147. As in the original FSAR, the DEHL case was found to be limiting for the operating deck differential pressure. The DECL case was limiting for the shell. Figures 14.3.4-237 through 14.3.4-244 present the pressure time histories for the lower compartment elements (1-6), the upper compartment (25) and element 40 on the shell for the DEHL break in compartment 1 case. Figures 14.3.4-245 and 14.3.4-246 present the pressure time histories for element 2 and element 25 for the DEHL in compartment 2 case. Figure 14.3.4-247 presents the pressure time history for element 40 on the shell for the DECL case.

Additionally included in the evaluation were 15% flow blockage, the effect of the following initial conditions:

- Temperature range of 60-120°F
- Pressure range of 13.2-15.0 psia
- Humidity range of 15-100 percent

and an uncertainty allowance. The new base TMD DEHL in compartment 2 case, in addition to including new mass and energy, new ice mass and the revised volumes, included the limiting initial conditions directly in the run.

Following are the results illustrating the original design values, the results for the revised base limiting case and the new calculated pressure loadings including all previously mentioned effects:

Item	Peak Differ. Pressure DP[1-25] DP[6-25]	Peak Differ. Pressure DP[2-25] DP[5-25]	Peak Differ. Pressure DP(7,8,9 to 25)	Peak Pressure SHELL P40, P45
Structural Design	16.6 psi	12.0 psi	12.0 psi	12.0 psi
Original Base	14.1	10.6	8.2	10.8
New Base	16.8 psi	12.2 psi	10.7 psi	13.1 psi
New Total	18.7 psi	13.0 psi	11.2 psi	14.0 psi

Additionally, the peak calculated pressure for the internal shell elements 41-44 and the peak calculated differential pressure across the operating deck for elements 3 and 4 were below the 12.0 psi structural design value.

The previously mentioned calculated pressures and differential pressures exceed the original structural design basis. The structural adequacy of this compartment was evaluated using acceptance criteria found in Section 5.2.2.3 of the FSAR and was confirmed.

FSAR Table 14.3.4-10 and FSAR Table 14.3.4-8, for Unit 1 and Unit 2 respectively, demonstrated the effects of changes in certain variables on the operating deck differential pressure and the shell pressure. The purpose of that study was to illustrate the sensitivity of the TMD code results to different input and assumption conditions and to illustrate the inherent analysis conservatisms. The purpose of the tables was not to supply an extrapolation tool for all subcompartments since the work was done for a specific subcompartment and trends may be different for other compartments. For example, the effect of initial compartment pressure on the peak differential pressure can be either a benefit or a penalty depending upon the flow regime before and during the peak. Additionally, if the peak occurs later in time the trend will be geometry dependent. That is, the pertinent downstream element would pressurize differently based upon specific key variables, such as flow areas and resistance into and out of the element. A combination of both sonic and subsonic flow regime periods could occur over the total transient. Since the new analysis is sufficiently different when compared to the original sensitivity basis, FSAR tables 14.3.4-10 and 8 for Units 1 and 2 respectively should only be used for guidance.

14.3.4.9.5 Steam Generator Enclosure

The largest break possible in the steam generator enclosure is a double-ended break of the steam line. This can be postulated to occur at the nozzle (or top of the enclosure) and at the side of the vessel. The limiting mass and energy releases from these breaks are from the hot shutdown condition (no load), which do not change for rerating. Therefore, the analysis and evaluation as described in Section 14.3.4.7 of the Unit 2 FSAR is applicable. Figure 14.3.4-496 of the Unit 2 FSAR shows the 9 node TMD model which was used in the analytical model.

14.3.4.9.6 Short Term Containment Analysis Conclusions

The results of the short-term containment analyses and evaluations for the Cook Nuclear Power Plants demonstrate that, for the pressurizer enclosure, the fan accumulator room and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. For the loop compartments, the peak calculated pressures at the rerated conditions are higher than the FSAR design allowables; for these areas, structural evaluations were performed for these compartments for the revised peak pressures. The structural adequacy was confirmed through evaluations using Section 5.2.2.3 of the FSAR as acceptance criteria.

14.3.4.10 Reactor Cavity Pressure Analysis

14.3.4.10.1 Introduction

The reactor cavity pressure analysis was performed for Cook Nuclear Plant Units 1 and 2, for the rerated conditions (including the assumption of an NSSS power level of 3600 MWt). The purpose of this analysis is to calculate the initial pressure response in the reactor cavity to a loss of coolant accident. The reactor cavity pressure analysis was performed for the upper and lower reactor cavities, the reactor vessel annulus and the reactor pipe annulus.

As in Table 14.3.4-28 of Unit 1 FSAR vent areas from the upper and lower reactor cavities were 175 and 70 square feet, respectively. The LOCA break flow split is also maintained. The use of this flow split is justified because the flow split does not depend on the mass and energy release rates, but only on the flow path characteristics of the pipe annulus, reactor annulus, and sand plug holes. This split is such that 75% of the break flow discharges to the upper reactor cavity and loop compartments, with the remaining 25% entering the reactor annulus. Of this 25%, 63% enters the upper reactor cavity.

In this evaluation, the effect of the following initial conditions was also assessed:

- Temperature range of 60-160°F in the loop compartments.
- Temperature range of 60-120°F in the upper and lower reactor cavities.
- Pressure range of 13.2-15.0 psia.
- Humidity range of 15-100 percent.

14.3.4.10.2 <u>Upper Reactor Cavity</u>

The limiting break for the upper reactor cavity is a single-ended break of the primary cold leg. Mass and energy releases were developed for this break using SATAN-V models. Upper reactor cavity pressurization effects were calculated with the TMD code, using the new mass and energy release rates and assuming unaugmented critical flow. The following results are summarized:

	Peak Upper Cavity Pressure	Peak Missile Shield Differential Pressure	Peak Cavity Wall Differential Pressure
Structural Design	N/A	48 psi	48 psi
Calculated	56.4 psi	54.3 psi	48.4 psi
Previous Calculation	47.0 psi	44.1 psi	44.1 psi

The previously mentioned calculated pressures and differential pressures for the rerated conditions exceed the original structural design basis. The structural adequacy of the reactor cavity was evaluated using acceptance criteria found in Section 5.2.2.3 of the FSAR and was confirmed.

The pressures above are higher than those in the previous FSAR analysis because the new mass and energy release rates are 10-20% higher than those on which the previous FSAR peak pressures were based. These higher peak pressures above reflect the fact that, with higher input mass and energy influx, the driving pressure must increase to force the same efflux through the 175 square feet of vent area available to the upper reactor cavity.

14.3.4.10.3 Lower Reactor Cavity

As in the upper reactor cavity analysis, the limiting break for the lower reactor cavity is a single-ended break of the primary cold leg. The analytic assumptions used for this analysis are the same as those of the upper reactor cavity.

The results are summarized below:

	Peak Lower Cavity Pressure	Peak Differential Pressure Between Lower Cavity and Loop Compartments
Structural Design	15 psi	15 psi
Calculated	18.5 psi	8.0 psi
Previous Calculation	13.8 psi	12.3 psi

Again, the lower cavity pressure above is higher than in the previous FSAR analysis because the new mass and energy release rates are 10-20% higher than those on which the previous FSAR lower cavity peak pressure is based.

14.3.4.10.4 Reactor Vessel Annulus and Reactor Pipe Annulus

The reactor vessel annulus and pipe annuli peak pressures were evaluated using a homogeneous, unaugmented critical flow model. The peak break flow rates for the single-ended cold leg (SECL) and single-ended hot leg (SEHL) breaks were considered. The limiting break was found to be the SEHL break because, even though the peak break flow rate was higher for the SECL, the enthalpy was higher for the SEHL. The results are summarized below:

	Peak Pipe Annulus Pressure (SEHL)	Peak Reactor Vessel Annulus Pressure (SEHL)
Structural Design	2000 psi	1000 psi
Calculated	850 psi	115 psi
Previous Calculation	735 psi	95 psi

The calculated values are well below the design values. Therefore, structural integrity is ensured for the pipe annuli and reactor vessel annulus.

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TABLE 14.3.4-1

COOK NUCLEAR PLANT PASSIVE HEAT SINKS

A. MATERIAL PROPERTIES

Material	Thermal Conductivity Btu/hr-^{o}F-ft	Heat Capacity Btu/ft^{3}-^{o}F
Paint	0.0833	28.4
Concrete	0.8	28.8
Carbon Steel	26.0	56.4
Steel and Insulation	0.2	3.663

B. SURFACES

		Area Ft^{2}	Thickness ft
Upper Compartment Material			
1.	Paint	32500.	0.001083
	Carbon Steel	32500.	0.0469
	Concrete	32500.	2.0
2.	Paint	10086.	0.001083
	Concrete	10086.	2.0
3.	Paint	5880.	0.001250
	Concrete	5880.	1.5
4.	Paint	11970.	0.00125
	Concrete	11970.	1.0
Lower Compartment Material			
5.	Paint	5069.	0.00125
	Concrete	5069.	2.0
6.	Paint	13660.	0.00125
	Concrete	13660.	1.5
7.	Paint	16730	0.00125
	Concrete	16730	1.0
8.	Paint	8665.	0.00125
	Concrete	8665.	2.0

TABLE 14.3.4-1 (Cont'd)

COOK NUCLEAR PLANT PASSIVE HEAT SINKS

		Area Ft^2	Thickness ft
Ice Condenser			
9.	Steel	180600.	0.00663
10.	Steel	76650.	0.0217
11.	Steel	28670.	0.0267
12.	Paint	3336.	0.000833
	Concrete	3336.	0.333
13.	Steel and Insulation	19100.	1.0
	Steel	19100.	0.0625
14.	Steel and Insulation	13055.	1.0
	Concrete	13055.	1.0

TABLE 14.3.4-2

TMD FLOW PATH INPUT DATA

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft^2)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
1 to 2	16.7	635	0.3		.529
2 to 3	21.2	585	0.34		.488
3 to 4	26.2	740	0.22		.617
4 to 5	17.3	585	0.34		.488
5 to 6	16.7	635	0.3		.529
6 to 1	30.0	72	1.45		.060
26 to 32	34.81	20	1.6		.134
27 to 26	18.4	43	2.7		.125
28 to 3	29	40	4.2		.0349
29 to 30	21.81	11.09	1.5		.0322
30 to 28	47	55	1.6		.368
31 to 30	18.41	43	2.7		.125
32 to 30	70	100	0.5		.669
33 to 2	5.5	24	1.5		.038
40 to 1	10.36	121.9	0.89		.225
41 to 2	10.36	144	0.89		.225
42 to 3	10.36	288	0.89		.225

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TABLE 14.3.4-2 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft^2)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
43 to 4	10.36	199.4	0.89		.225
44 to 5	10.36	155.1	0.89		.225
45 to 6	10.36	155.1	0.89		.225
1 to 33	5.5	20	1.5		.038
2 to 27	15	154	4.2		.0349
3 to 33	8	56	1.5		.038
4 to 33	6.5	32.6	1.5		.038
5 to 31	15	154	4.2		.0349
6 to 33	5.2	18	1.5		.038
7 to 8	12.28	112.8		0.516	.727
8 to 9	12.28	112.8		0.516	.727
9 to 34	8.86	112.8	0.812	0.258	.727
10 to 11	12.28	131.3		0.516	.727
11 to 12	12.28	131.3		0.516	.727
12 to 35	8.86	131.3	0.812	0.258	.727
13 to 14	12.28	266.6		0.516	.727
14 to 15	12.28	266.6		0.516	.727
15 to 36	8.86	266.6	0.812	0.258	.727
16 to 17	12.28	184.6		0.516	.727

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TABLE 14.3.4-2 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft^2)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
32 to 31	18.4	43	2.7		.125
33 to 5	5.5	24	1.5		.038
34 to 25	2.8	233.8	1.45		.659
35 to 25	2.8	267.6	1.43		.659
36 to 25	2.8	539.5	1.43		.625
37 to 25	2.8	376.5	1.41		.636
38 to 25	2.8	289.4	1.44		.646
39 to 25	2.8	296.3	1.43		.249
40 to 7	8.222	106.7	0.227	0.142	.33
41 to 10	8.222	126.1	0.227	0.142	.33
42 to 13	8.222	252.2	0.227	0.142	.33
43 to 16	8.222	174.6	0.227	0.142	.33
44 to 19	8.222	135.8	0.227	0.142	.33
45 to 22	8.222	135.8	0.227	0.142	.33
40 to 41	13.8	24.7	7.5		.075
41 to 42	22.4	24.7	12.5		.046
42 to 43	25.3	24.7	12.5		.041
43 to 44	18.4	24.7	10.0		.056
44 to 45	16.1	24.7	10.0		.064

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TABLE 14.3.4-2 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft^2)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
17 to 18	12.28	184.6		0.516	.727
18 to 37	8.86	184.6	0.812	0.258	.727
19 to 20	12.28	143.6		0.516	.727
20 to 21	12.28	143.6		0.516	.727
21 to 38	8.86	143.6	0.812	0.258	.727
22 to 23	12.28	143.6		0.516	.727
23 to 24	12.28	143.6		0.516	.727
24 to 39	8.86	143.6	0.812	0.258	.727
26 to 28	70.3	100	0.5		.669
27 to 29	14.3	10	3.0		.0291
28 to 27	18.4	43	2.7		.125
29 to 28	21.81	11.09	1.5		.0322
30 to 4	29	40	4.2		.0349
31 to 29	14.3	10	3.0		.0291

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TABLE 14.3.4-3

1973 WALTZ MILL PRELIMINARY TEST CONDITIONS

Test Series	$\frac{G_{test}}{G_{plant}}$	$\frac{(\text{Total Energy/Ice Wt.})_{test}}{(\text{Total Energy/Ice Wt.})_{plant}}$	Nominal Conditions: Subcooling In Piping	Nominal Conditions: Varied in Test Setup
Blowdown Rate Series	37%	100%	40°F	Variable Orifice Sizes
	75%			
	100%			
	150%			
	10%			
	1.5%			
Blowdown Energy Series	75%	150%	~40°F	Variable Boiler Water Levels
	100%	150%		
	100%	200%		
Blowdown Transient Shape Series	75%	100%	~10°F	Variable Conditions In Subcooled Leg
	75%		~25°F	
	100%		~10°F	
	100%		~25°F	

TABLE 14.3.4-4

TMD VOLUME INPUT FOR AEP

Element	Volume (Ft^3)
1	27250
2	38000
3	55000
4	35000
5	38000
6	22500
7	3925
8	3925
9	3925
10	3895
11	3895
12	3895
13	7789
14	7789
15	7789
16	5393
17	5393
18	5393

TABLE 14.3.4-4 (Cont'd)

TMD VOLUME INPUT FOR AEP

Element	Volume (Ft^3)
19	4194
20	4194
21	4194
22	4194
23	4194
24	4194
25	734830
26	10380
27	27414
28	10380
29	17111
30	10380
31	27414
32	10380
33	16147
34	5385
35	6365
36	12729

TABLE 14.3.4-4 (Cont'd)

TMD VOLUME INPUT FOR AEP

Element	Volume (Ft^3)
37	8813
38	6857
39	6854
40	2778
41	3283
42	6565
43	4545
44	3535
45	3535

TABLE 14.3.4-5

FRAME #	1	2	3	4	5	6	7	8	9	10	11	12
						ELEMENT NUMBER						
X	1	5	9	13	17	21	25	29	33	37	41	45
O	2	6	10	14	18	22	26	30	34	38	42	
I	3	7	11	15	19	23	27	31	35	39	43	
+	4	8	12	16	20	24	28	32	36	40	44	

TABLE 14.3.4-6

CALCULATED MAXIMUM PEAK PRESSURES IN LOWER COMPARTMENT ELEMENTS ASSUMING UNAUGMENTED FLOW

Element	1	2	3	4	5	6	
Peak Pressure (psig)	13.6	11.6	10.5	10.6	11.5	13.0	DECL - 100% Ent.
Peak Pressure (psig)	14.4	11.0	9.2	9.1	10.8	14.4	DEHL - 100% Ent.

TABLE 14.3.4-7

CALCULATED MAXIMUM PEAK PRESSURES IN THE ICE CONDENSER COMPARTMENT ASSUMING UNAUGMENTED FLOW

Element	40	41	42	43	44	45	
Peak Pressure (psig)	10.4	8.8	7.9	7.9	8.7	9.9	DECL - 100% Ent.
Peak Pressure (psig)	10.8	8.3	7.2	7.5	8.3	10.6	DEHL - 100% Ent.

TABLE 14.3.4-8

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES ACROSS THE OPERATING DECK OR LOWER CRANE WALL ASSUMING UNAUGMENTED FLOW

Element	1	2	3	4	5	6	
Peak Δp (psi)	12.1	8.9	7.4	7.2	8.7	11.7	DECL - 100% Ent.
Peak Δp (psi)	14.1	10.6	8.1	8.3	10.5	14.1	DEHL - 100% Ent.

TABLE 14.3.4-9

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES ACROSS THE UPPER CRANE WALL ASSUMING UNAUGMENTED FLOW

Element	7-8-9.	10-11-12	13-14-15	16-17-18	19-20-21	22-23-24	
Peak Δp (psi)	6.9	9.5	5.1	5.1	5.6	6.8	DECL - 100% Ent.
Peak Δp (psi	8.2	6.8	6.0	6.0	6.7	8.2	DEHL - 100% Ent.

TABLE 14.3.4-10

SENSITIVITY STUDIES FOR COOK NUCLEAR PLANT

PARAMETER	CHANGE MADE FROM BASE VALUE	CHANGE IN OPERATING DECK ΔP	CHANGE IN PEAK PRESSURE AGAINST THE SHELL
Blowdown	+ 10%	+ 11%	+ 12%
Blowdown	- 10%	- 10%	- 12%
Blowdown	- 20%	- 20%	- 23%
Blowdown	- 50%	- 50%	- 53%
Break Compartment Inertial Length	+ 10%	+ 4%	+ 1%
Break Compartment Inertial Length	- 10%	- 4%	- 1%
Break Compartment Volume	+ 10%	- 2%	- 1%
Break Compartment Volume	- 10%	+ 2%	+ 1%
Break Compartment Vent Areas	+ 10%	- 6%	- 5%
Break Compartment Vent Areas	- 10%	+ 8%	+ 5%
Door Port Failure in Break Compartment	one door port fails to open	+ 1%	- 1%
Ice Mass	+ 10%	0	0
Ice Mass	- 10%	0	0
Door Inertia	+ 10%	+ 1%	0
Door Inertia	- 10%	- 1%	0
All Inertial Lengths	+ 10%	+ 5%	+ 4%
All Inertial Lengths	- 10%	- 5%	- 3%
Ice Bed Loss Coefficients	+ 10%	0	0
Ice Bed Loss Coefficients	- 10%	0	0
Entrainment Level	0% Ent.	- 27%	- 11%
Entrainment Level	30% Ent.	- 19%	- 15%

TABLE 14.3.4-10 (Cont'd)

SENSITIVITY STUDIES FOR COOK NUCLEAR PLANT

PARAMETER	CHANGE MADE FROM BASE VALUE	CHANGE IN OPERATING DECK ΔP	CHANGE IN PEAK PRESSURE AGAINST THE SHELL
Entrainment Level	50% Ent.	- 13%	- 12%
Entrainment Level	75% Ent.	- 6%	- 6%
Lower Compartment Loss Coefficients	+ 10%	0	0
Lower Compartment Loss Coefficients	- 10%	0	0
Cross Flow in Lower Plenum	Low estimate of resistance	0	- 7%
Cross Flow in Lower Plenum	High estimate of resistance	0	- 3%
Ice Condenser Flow Area	+ 10%	0	- 3%
Ice Condenser Flow Area	- 10%	0	+ 4%
Ice Condenser	+ 20%	0	- 6%
Ice Condenser Flow Area	- 50%	0	+ 8%
Initial Pressure in Containment	+0.3 psi	+ 2%	+ 2%
Initial Pressure in Containment	- 0.3 psi	- 2%	- 2%
Reactor Coolant Break Enthalpy	- 13.0%	+ 6%	+ 3%
Compressibility Factor	Addition of the compressibility factor	+ 4%	0

All values shown are to the nearest percent.

TABLE 14.3.4-11

COOK NUCLEAR PLANT ICE CONDENSER DESIGN PARAMETERS

Reactor Containment Volume (net free volume)	
Upper Compartment, ft^3	745,896
Ice Condenser, ft^3	126,940
Lower Compartment (active), ft^3	306,800
Total Active Volume, ft^3	1,179,636
Lower Compartment (dead-ended), ft^3	61,702
Total Containment Volume, ft^3	1,241,338
Reactor Containment Air Compression Ratio	1.41
Reactor Power, MWt	3391
Design Energy Release to Containment	
Initial blowdown mass release, lb	549,000
Initial blowdown energy release, Btu	346.7×10^6
Allowance for undefined energy release in addition to core residual heat, Btu	50×10^6
Ice Condenser Parameters	
Weight of ice in condenser, lb	2.45×10^6

TABLE 14.3.4-12

Break Size	5 ft^2 Deck Leak Air Compression Peak (psig)	Deck Leakage Area (ft^2)	Spray Flow Rate (gpm)	Resultant Peak Containment Pressure (psig)
Double-ended	7.8	54.	0	12.0
0.6 double-ended	6.6	46.	0	12.0
3 ft^2	6.25	50.	0	12.0
8 inch diameter	5.5	56.	4000	12.2
8 inch diameter	5.5	35.	2000	12.0
8 inch diameter*	5.5	56.	2000	11.3
6 inch diameter	5.0	56.	4000	10.4
2-1/2 inch diameter	4.0	56.	4000	8.5
1/2 inch diameter	3.0	>50.	4000	3.0

* This case assumes upper compartment structural heat sink steam condensation of 8 lb/sec and 30 percent of deck leakage is air.

TABLE 14.3.4-13

PEAK PRESSURES/DIFFERENTIALS

	DEHL Break In Element #6	DECL Break In Element #6
Pressure In Element #6 (Psig)	14.8 (14.4)	13.4 (13.0)
Peak Pressure In Ice Condenser Compartments (Psig)	10.6 (10.6)	9.8 (9.9)
Peak Differential Pressure Across Operating Deck or Lower Crane Wall (Psi)	14.5 (14.1)	12.3 (11.7)

TABLE 14.3.4-14

EFFECTS OF VARYING POLYTROPIC EXPONENT

	Base Case	5% Decrease	10% Decrease	20% Decrease
Pressure In Element #6 (Psig)	14.8	14.8	14.8	14.9
Peak Pressure In Ice Condenser Compartment (Psig)	10.6	10.6	10.6	10.6
Peak Differential Pressure Across Operating Deck or Lower Crane Wall (Psi)	14.5	14.5	14.5	14.6

TABLE 14.3.4-15

SOURCES OF SECONDARY SYSTEM FLUID FOR BLOWDOWN CALCULATION

	Source	Mass (lb)	Energy (10^6 BTU)
1.	Steam Generator Initial Inventory	117,500	63.0
2.	Feedwater System	310,100	117.2
3.	Reverse flow from steam piping	28,000	33.4
Total		455,600	213.6
Heat Addition from Reactor			180.0
Total			393.6

TABLE 14.3.4-16

INTEGRATED MASS AND ENERGY RELEASE

Time (seconds)	Mass (lbs)	Energy (10^6 BTUs)
0	0	0
5	100,000	71
10	143,000	110
20	175,000	137
30	202,000	161
40	227,000	184
60	273,000	224
80	312,000	260
100	345,000	290
140	394,000	336
200	422,000	365
600	446,000	393

TABLE 14.3.4-17

PRESSURE TRANSIENT FOLLOWING A STEAMLINE BREAK ACCIDENT

Time (sec)	TMD Nodal Element 2 psia	Upper Containment psia
0	15.0	15.0
.1	19.0	15.0
.2	18.3	15.0
.3	18.6	15.6
.5	18.1	16.6
.6	18.0	17.0
.7	17.8	17.2
.8	18.2	17.5
.9	18.5	17.7
1.0	18.8	17.9
1.2	19.3	18.3
1.4	19.6	18.6
2.0	20.0	19.3
3.0	20.5	20.0
4.0	20.7	20.3
5.0	20.8	20.6
6.0	21.2	20.7
7.0	20.9	20.8
8.0	21.0	20.9
9.0	21.2	21.1
10.0	21.3	21.2
12.0	21.3	21.3
14.0	21.3	21.3
18.0	21.4	21.4
20.0	21.5	21.5
24.0	21.7	21.7
30.0	22.0	22.0

TABLE 14.3.4-18

SUMMARY OF HEAT TRANSFER CORRELATIONS USED TO CALCULATE STEAM GENERATOR HEAT FLOW IN THE SATAN CODE

Primary To Secondary Heat Flow

Primary (Tube Side)	Secondary (Shell Side)
Dittus-Boelter	Jens-Lottes

Secondary To Primary Heat Flow

Primary	Secondary
Jens-Lottes	McAdams
Dougall-Rohsenow	
McEligot	

TABLE 14.3.4-19

SENSITIVITY OF CORE
STORED ENERGY TO POWER LEVEL,
NODES IN THE PELLET AND FUEL DENSIFICATION

Power (kw/ft)	Number of Radial Nodes in Pellet	Stored Energy (# full power seconds)
I. Instantaneous Isotropic Densification to 96.5% TD		
7.66*	10	6.59
7.66	1 (average pellet)	6.53
15.04	10	6.80
15.04	1	6.69
II. No Densification (94% TD)		
7.66	1	5.78

* This value of kw/ft can be considered a typical core average power and was used as a reference value for core stored energy calculations.

TABLE 14.3.4-20

MASS AND ENERGY RELEASE TO CONTAINMENT (BLOWDOWN PHASE)

TIME (sec.)	MASS OUT (lbs.)	ENERGY OUT (BTU)	TBAR (sec.)	MASS RATE (lbs./sec.)	ENERGY RATE (BTU/sec.)
1.00000E-08	0.	0.	1.00000E-08	6.21319E+04	3.36476E+07
2.00691E-02	1.24692E+03	6.75276E+05	1.00346E-02	6.21315E+04	3.36476E+07
1.80044E-01	1.30774E+04	7.08602E+06	1.00057E-01	7.39522E+04	4.00734E+07
3.40095E-01	2.60401E+04	1.41558E+07	2.60070E-01	8.09905E+04	4.41719E+07
5.00256E-01	3.82043E+04	2.08592E+07	4.20176E-01	7.59503E+04	4.18542E+07
6.60341E-01	4.95630E+04	2.72065E+07	5.80298E-01	7.09541E+04	3.96496E+07
8.40195E-01	6.16336E+04	3.40406E+07	7.50268E-01	6.71136E+04	3.79986E+07
1.20061E+00	8.47838E+04	4.73383E+07	1.02040E+00	6.42316E+04	3.68953E+07
2.10019E+00	1.64821E+05	7.66874E+07	1.65040E+00	5.56228E+04	3.26254E+07
3.10047E+00	1.79557E+05	1.03698E+08	2.60033E+00	4.47238E+04	2.70032E+07
4.10000E+00	2.16468E+05	1.26634E+08	3.60024E+00	3.69291E+04	2.29463E+07
5.20014E+00	2.51871E+05	1.48969E+08	4.65007E+00	3.21794E+04	2.03024E+08
6.30019E+00	2.84768E+05	1.69717E+08	5.75017E+00	2.99056E+04	1.88603E+07
7.30098E+00	3.12893E+05	1.87406E+08	6.80058E+00	2.81026E+04	1.76756E+07
8.30016E+00	3.39498E+05	2.04043E+08	7.80057E+00	2.66270E+04	1.66500E+07
9.30030E+00	3.64057E+05	2.20037E+08	8.80023E+00	2.45548E+04	1.59927E+07
1.04003E+01	3.88769E+05	2.36161E+08	9.85032E+00	2.24652E+04	1.46568E+07
1.17051E+01	4.16156E+05	2.53978E+08	1.10509E+01	2.10474E+04	1.36932E+07
1.32009E+01	4.43910E+05	2.72384E+08	1.24512E+01	1.85100E+04	1.22757E+07
1.47013E+01	4.67723E+05	2.88628E+08	1.39511E+01	1.58709E+04	1.08264E+07
1.58008E+01	4.62611E+01	2.99137E+08	1.52511E+01	1.35406E+04	9.55753E+06
1.68009E+01	4.35052E+05	3.07611E+08	1.63009E+01	1.24410E+04	8.47366E+06
1.81006E+01	5.08626E+05	3.16365E+08	1.74508E+01	1.04435E+04	6.73542E+06
1.93010E+01	5.17828E+05	3.22166E+08	1.87008E+01	7.66560E+03	4.83251E+06
2.08001E+01	5.25866E+05	3.27118E+08	2.00506E+01	5.36224E+03	3.30328E+06
2.28005E+01	5.33791E+05	3.30971E+08	2.18003E+01	3.96129E+03	1.92598E+06
2.50001E+01	5.40138E+05	3.33451E+08	2.39003E+01	2.88561E+03	1.12775E+06
2.67988E+01	5.43330E+05	3.34617E+08	2.58995E+01	1.77458E+03	6.48009E+05
2.67988E+01	5.43330E+05	3.34617E+08	2.67988E+01	0.	0.
1.00000E+05	5.43330E+05	3.34617E+08	1.00000E+05	0.	0.

TABLE 14.3.4-21

TYPE OF BREAK	EFFECTIVE AREA (FT^2)	TOTAL MASS 16M ($X10^{-5}$)	TOTAL ENERGY BTU ($X10^{-8}$)
Double-Ended Pump Suction (Guillotine) 20 Foot Entrainment	10.48	5.43	3.35
.6 Double-Ended Pump Suction (Guillotine) 10 Foot Entrainment	6.29	5.41	3.32
3 Foot Square Pump Suction Split 10 Foot Entrainment	3.00	5.33	3.30

TABLE 14.3.4-22

MASS AND ENERGY RELEASE TO CONTAINMENT (REFLOOD PHASE)

TIME (SEC)	MASS RATE (LBM/SEC)	ENERGY RATE (BTU/SEC)	MASS (LBM)	BTU (ENERGY)
2.6790000E+01	0.	0.	0.	0.
2.6800000E+01	0.	0.	0.	0.
2.8800000E+01	2.1865209E+02	2.8217140E+05	3.3252981E+02	6.8724691E+05
3.2800000E+01	6.7732797E+02	8.7385399E+05	2.0950811E+03	2.7037112E+06
3.3800000E+01	7.6077275E+02	9.8812806E+05	2.8026868E+03	3.6165120E+06
3.6800000E+01	7.4261228E+02	9.5663032E+05	5.0618291E+03	6.5284772E+06
4.1800000E+01	7.1724655E+02	9.2218456E+05	8.7892951E+03	1.1222491E+07
4.6800000E+01	6.9426154E+02	8.9088350E+05	1.2236755E+04	1.5753346E+07
5.0000000E+01	6.8596255E+02	8.7922416E+05	1.4444550E+04	1.8584807E+07
5.6800000E+01	6.6823534E+02	8.5443003E+05	1.9046691E+04	2.4476194E+07
7.6800000E+01	5.9910021E+02	7.6080333E+05	3.1723254E+04	4.0629040E+07
8.6800000E+01	5.6770399E+02	7.1877472E+05	3.7561602E+04	4.8032373E+07
1.0000000E+02	5.3082617E+02	6.6966681E+05	4.4819249E+04	5.7205232E+07
1.0680000E+02	5.1307699E+02	6.4624952E+05	4.8369626E+04	6.1680734E+07
1.2680000E+02	4.6114931E+02	5.7824717E+05	5.8114797E+04	7.3928609E+07
1.4680000E+02	4.1981504E+02	5.2458757E+05	6.6922507E+04	8.4952331E+07
1.8040900E+02	3.7752371E+02	4.6945746E+05	8.0298768E+04	1.0161359E+08
1.8041100E+02	1.4529151E+02	1.8064268E+05	8.0290289E+04	1.0161424E+08
2.0000000E+02	1.3992356E+02	1.7392386E+05	8.3086491E+04	1.0509055E+08
5.0000000E+02	1.0040971E+02	1.2459523E+05	1.1836478E+05	1.4890624E+08
1.0000000E+03	7.4969299E+01	9.2833114E+04	1.6115969E+05	2.0195335E+08
2.0000000E+03	5.7040149E+01	7.0421873E+04	2.2563203E+05	2.8166767E+08
5.0000000E+03	4.2429118E+01	5.2067853E+04	3.7027285E+05	4.5969062E+08
1.0000000E+04	3.3709915E+01	4.1106320E+04	5.5762436E+05	6.8885108E+08

Entrainment ends at 180.41 seconds.

TABLE 14.3.4-23

SOURCE ENERGY INVENTORIES

Initial Energy Distribution

Source	Referenced at T=32°F Energy x 10^6 BTU	Referenced at T_{sat} for P containment
Reactor Coolant	314.01	199.0
Accumulators	18.51	-26.08
Core Stored	37.98	33.95
RCS Thin Metal	24.33	14.89
RCS thick Metal	29.64	18.01
Steam Generator Secondary Side Fluid	237.87	147.06
Steam Generator Metal (Based on the mass of all the metal in the steam generators)	158.77	90.77

Energy Distribution at End of Blowdown

Source	Referenced to 32°F Energy x 10^6 BTU	Referenced T_{sat} at P containment
Reactor Coolant	16.37	.89
Accumulators	11.96	-16.88
Core Stored	15.99	11.96
RCS Thin Metal	18.97	9.53
RCS Thick Metal	29.64	18.01
Steam Generator Secondary Side Fluid	239.45	147.71
Steam Generator Metal	158.48	90.44

TABLE 14.3.4-23 (Cont'd)

SOURCE ENERGY INVENTORIES

Energy Distribution at End of Reflood

Source	Referenced at T=32^{o}F Energy x 10^6 BTU	Referenced at T_{sat} for containment
Reactor Coolant	16.37	.89
Accumulators	0.	0.
Core Stored	4.03	0.
RCS Thin Metal	9.44	0.
RCS Thick Metal	22.62	11.0
Steam Generator Secondary Side Fluid	208.56	116.82
Steam Generator Metal	141.35	73.35

Energy Distribution at 5000 Seconds

Source	Referenced at T= 32^{o}F Energy x 10^6 BTU	Referenced at T_{sat} Energy x 10^6 BTU
Reactor Coolant	16.37	.89
Accumulators	0.	0.
Core Stored	4.03	0.
RCS Thin Metal	9.44	0.
RCS Thick Metal	11.62	0.
Steam Generator Secondary Side Fluid	195.19	103.45
Steam Generator Metal	133.55	65.55

TABLE 14.3.4-24

MASS AND ENERGY RELEASED TO THE CONTAINMENT

Period	Time Seconds	Boiloff Mass (lbs)	Boiloff Energy (BTU's)
End of Blowdown	26.8	0.543 (10^6)	334.6 (10^6)
End of Reflood (10' entrainment)	180.4	0.624 (10^6)	436.2 (10^6)
Broken Loop Steam Generators In Equilibrium With Containment Atmosphere	345.4	0.674 (10^6)	495.4 (10^6)
Unbroken Loop Steam Generators In Equilibrium With Containment Atmosphere	1660.4	0.822 (10^6)	667.2 (10^6)
Recirculation Flow is Initiated	2755	0.892 (10^6)	747.3 (10^6)
Ice Condenser Ice is Depleted	5314	1.026 (10^6)	902.3 (10^6)
------	10^4	1.23 (10^6)	1.14 (10^9)
------	10^5	3.55 (10^6)	3.82 (10^9)
------	10^6	14.2 (10^6)	16.1 (10^9)

TABLE 14.3.4-25

FROTH ENERGY BALANCES

Sink	Blowdown (BTU)	End Of Reflood (BTU)	Approximate End Of Froth (BTU) (1651 SECS)
Ice Heat Removal	209 (10^6)	299 (10^6)	490 (10^6)
Structural Heat Sinks	10.9 (10^6)	39.6 (10^6)	54.3 (10^6)
RHR Heat Exchanger Heat Removal	0.0	0.0	0.0
Spray Heat Exchanger Heat Removal	0.0	0.0	0.0
Lower Compartment Spray Heat Removal	0.0	1.89 (10^6)	20.3 (10^6)
Energy Content Of Sump	190 (10^6)	240 (10^6)	541 (10^6)
Ice Melted (Pounds)	0.674 (10^6)	1.04 (10^6)	1.80 (10^6)

TABLE 14.3.4-26
STEAM LINE RUPTURE IN STEAM GENERATOR DOGHOUSE

Time Seconds	Forward Flow (from steam generator) Flow 10^3 lbs/sec	Forward Flow (from steam generator) Energy 10^6 BTU/sec	Back Flow (from piping) Flow 10^3 lbs/sec	Back Flow (from piping) Energy 10^6 BTU/sec
0	0	0	0	0
.01	9.58	11.41	7.19	8.57
.02	9.39	11.19	6.36	7.58
.03	9.25	11.04	6.08	7.25
.04	9.15	10.92	5.85	6.97
.05	9.05	10.80	4.80	5.72
.06	8.94	10.68	3.51	4.18
.10	8.48	10.14	2.76	3.29
.15	8.06	9.65	2.60	3.10
.20	7.97	9.54	2.52	3.00
.30	7.84	9.39	2.50	2.98
.50	7.66	9.17	2.48	2.96
.70	28.54	15.82	2.65	3.16
1.0	29.35	16.24	2.74	3.27
1.5	28.52	15.75	2.75	3.28
2.0	27.56	15.21	2.80	3.34
3.0	24.95	13.82	2.80	3.34

TABLE 14.3.4-27
FAN ROOM - BACKFLOW CONTRIBUTION

Time (sec)	Mass Flow Rate Lbs/sec 10^3	Energy Flow Rate BTU/sec 10^6
0	7.54	8.99
.1	4.68	5.58
.2	4.48	5.34
.3	4.41	5.26
.4	4.32	5.15
.5	4.27	5.09
.6	4.15	4.95
.7	3.93	4.68
.8	3.59	4.28
.9	3.62	4.32
1.0	3.53	4.21
1.5	3.17	3.78
2.0	3.00	3.58
2.5	2.93	3.49
3.0	2.87	3.42
3.5	2.87	3.42
4.0	2.83	3.37
4.5	2.79	3.33
5.0	2.81	3.35
5.5	2.77	3.30
6.0	2.72	3.24
6.5	2.72	3.24
7.0	2.69	3.21
8.0	2.65	3.16
8.5	2.65	3.16
9.0	2.65	3.16
9.5	2.65	3.16
10.0	2.65	3.16

With 1.4 ft^2 orifice in cross-connect to steam dump header; break in longest line.

TABLE 14.3.4-27 (Cont'd)
FAN ROOM - FORWARD FLOW CONTRIBUTION

t	10^3 #/sec m	10^6 BTU/sec mh
0	5.55	6.62
.1	4.15	4.94
.2	3.05	3.64
.3		
.4	2.95	3.52
.5		
.6	2.90	3.46
.7		
.8	2.78	3.31
.9		
1.0	2.75	3.28
1.5	2.67	3.19
2.0	3.45	3.38
2.5	9.50	5.26
3.0	9.42	5.21
3.5	9.38	5.19
4.0	9.33	5.16
4.5	9.28	5.10
5.0	9.23	5.10
5.5	9.16	5.07
6.0	9.10	5.04
6.5	9.03	5.01
7.0	8.95	4.97
7.5	8.86	4.93
8.0	8.80	4.91
8.5	8.70	4.86
9.0	8.58	4.81
9.5	8.46	4.76
10.0	8.33	4.70

TABLE 14.3.4-28

Compartment	Free Volume (ft^3)	Vent Area (ft^2)
Upper Reactor Cavity	19,731	175
Lower Reactor Cavity	14,335	70
Steam Generator	7,956	264
Pressurizer	3,537	42
Fan Room	26,423	226

TABLE 14.3.4-29

VENT AREAS USED IN SUBCOMPARTMENT ANALYSES

A. The following are the vent areas from the reactor vessel annulus:

1. Vent area from broken pipe to annulus - 5.64 $ft.^2$
2. Vent area along vessel to upper reactor cavity - 9.38 $ft.^2$
3. Vent area along vessel to lower reactor cavity - 17.32 $ft.^2$
4. Vent area to other pipe sleeves - 40.50 $ft.^2$

B. The following are the vent areas from the pipe sleeve:

1. Vent area to reactor vessel annulus - 5.64 $ft.^2$
2. Vent area to lower containment - 5.64 $ft.^2$
3. Vent area to upper reactor cavity through sand plug hole - 11.47 $ft.^2$

C. The volumes of the reactor vessel annulus and pipe sleeve are:

1. Reactor vessel annulus - 1,474 $ft.^3$
2. Pipe sleeve - 36.7 $ft.^3$

D. The modes of communication of the other doghouses with the remainder of the containment follow:

1. Fan Room

   a. Vent area to lower containment - 140 $ft.^2$
   b. Vent area to other fan room - 86 $ft.^2$

2. Steam Generator Doghouse

   a. Vent area to lower containment - 162 $ft.^2$
   b. Vent area to other steam generator doghouse - 102 $ft.^2$

3. Pressurizer Doghouse

   a. Vent area to lower containment - 42 $ft.^2$

E. The inside diameter of the pressurizer spray nozzle is 3.25"

F. The moment of inertia of the ice condenser lower inlet door is 1225 $lb.-ft.^2$

TABLE 14.3.4-30

CALCULATED MAXIMUM PEAK PRESSURES COMPARED WITH DESIGN PRESSURE

Type of Break	Location[a]	Peak Pressure		Peak Differential Pressure[c]		Design
		Augmented	Unaugmented	Augmented	Unaugmented	
DECL	Element 1	13.7	14.1	10.8	12.7	16.6
DECL	Element 2	10.8	12.2[b]	8.6[b]	10.5[b]	12.0
DECL	Element 3	9.8	11.2[b]	7.5[b]	9.4[b]	12.0
DECL	Element 4	9.7	11.1[b]	7.6[b]	9.5[b]	12.0
DECL	Element 5	10.5	11.9[b]	8.6[b]	10.5[b]	12.0
DECL	Element 6	11.6	13.0[b]	10.4[b]	12.3[b]	16.6
DEHL	Element 1	13.3	13.7[b]	13.0[b]	13.5[b]	16.6
DEHL	Element 2	10.6	11.0[b]	10.3[b]	10.8[b]	12.0
DEHL	Element 3	8.9	9.3[b]	8.3[b]	8.8[b]	12.0
DEHL	Element 4	9.0	9.4[b]	8.0[b]	8.5[b]	12.0
DEHL	Element 5	10.5	10.9[b]	10.2[b]	10.7[b]	12.0
DEHL	Element 6	13.6	14.0	13.2	13.7	16.6
DECL	Element 40	9.8	10.6	9.8	10.6	12.0
DECL	Element 41	8.7	9.5[b]	8.7	9.5	12.0
DECL	Element 42	7.8	8.6[b]	7.8	8.6	12.0
DECL	Element 43	7.8	8.6[b]	7.8	8.6	12.0
DECL	Element 44	8.5	9.3[b]	8.5	9.3	12.0
DECL	Element 45	9.5	10.3[b]	9.5	10.3	12.0
DEHL	Element 40	10.7	10.8[b]	10.7	10.8	12.0
DEHL	Element 41	8.3	8.4[b]	8.3	8.4	12.0
DEHL	Element 42	7.0	8.1[b]	7.0	8.1	12.0
DEHL	Element 43	7.1	7.2[b]	7.1	7.2	12.0
DEHL	Element 44	8.4	8.5[b]	8.4	8.5	12.0
DEHL	Element 45	10.7	10.8	10.7	10.8	12.0
DECL	Elements 7-8-9	6.1	6.1	6.6	6.6[b]	12.0
DECL	10-11-12	5.9	6.1	5.9	6.1[b]	12.0
DECL	13-14-15	5.6	6.0	5.2	5.6[b]	12.0
DECL	16-17-18	6.0	6.2	5.4	5.6[b]	12.0
DECL	19-20-21	6.7	6.7	6.0	6.0[b]	12.0
DECL	22-23-24	6.0	6.1	6.6	6.7[b]	12.0

TABLE 14.3.4-30 (Cont'd)

CALCULATED MAXIMUM PEAK PRESSURES COMPARED WITH DESIGN PRESSURE

Type of Break	Location[a]	Peak Pressure Augmented PSIG	Peak Pressure Unaugmented PSIG	Peak Differential Pressure[c] Augmented PSIG	Peak Differential Pressure[c] Unaugmented PSIG	Design PSIG
DEHL	Elements 7-8-9	7.1	7.2	7.8	7.9[b]	12.0
DEHL	10-11-12	7.6	7.6	6.8	6.8[b]	12.0
DEHL	13-14-15	6.4	6.8	6.0	6.4[b]	12.0
DEHL	16-17-18	6.0	6.5	6.1	6.6[b]	12.0
DEHL	19-20-21	6.8	6.8	6.9	6.9[b]	12.0
DEHL	22-23-24	7.1	7.5	7.6	8.0[b]	12.0
STEAMLINE	S.G. Doghouse	20.8	20.8	20.5	20.5	26.4
STEAMLINE	Fan Room	13.9	13.9	13.9	13.9	16.0
SECL	Lower Rx Cavity	12.2	13.8	11.4	12.3	15.0
SECL	Upper Rx Cavity	40.4	47.0	36.9	44.1	48.0
6" Spray Line	Pressurizer Enclosure	14.0	17.8	13.1	16.4	80.0
LOCA	Reactor Vessel Annulus	63.0	95.0	63.0	95.0	1000.0
LOCA	Reactor Pipe Annulus	419.0	735.0	419.0	735.0	2000.0

---

[a]Element 1-6 are break locations

[b]The unaugmented peak pressure and peak differential pressure other than Elements 1/40 (DECL) and 6/45 (DEHL) are conservatively estimated by taking the ΔP (unaug-aug) and adding it to the augmented pressure. Elements 2 through 6 and 41 through 45 for DECL and 1 through 5 and 40 through 44 for DEHL reflect this change.

In Elements 7 through 24 the ΔP (unaug-aug) for peak pressure was used to estimate the unaugmented peak differential pressure.

[c]For Elements 1 through 6 the peak differential pressure is across the operating deck or the lower crane wall. For Elements 7 through 24 the peak differential pressure is across the upper crane wall. For Elements 40 through 45 the peak differential pressure is across the containment shell.

TABLE 14.3.4-31

COOK NUCLEAR PLANT UNIT 1 DESIGN POWER CAPABILITY PARAMETERS

Parameter	(Original) Case 1	(Revised) Case 2	(Revised) Case 3
NSSS Power, MWt	3250	3262	3262
Core Power, MWt	3250	3250	3250
RCS Flow, (gpm/loop)*	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)**	366,400	366,400	366,400
RCS Temperature, °F			
Core Outlet	602.0	582.0	610.1
Vessel Outlet	599.3	579.1	607.5
Core Average	570.5	549.5	579.2
Vessel Average	567.8	547.0	576.3
Vessel/Core Inlet	536.3	514.9	545.2
Steam Generator Outlet	536.3	514.6	545.0
Zero Load	547.0	547.0	547.0
RCS Pressure, psia	2250	2250 or 2100	2250 or 2100
Steam Pressure, psia	758	618	820
Steam Flow, (10^6 lb/hr. tot.)	14.12	14.12	14.2
Feedwater Temperature, °F	434.8	434.8	434.8
% SG Tube Plugging	0	10	10

Flow Definitions:

*RCS Flow (Thermal Design Flow) -- The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow -- The flow specified in the Technical Specifications which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

14.4 ENVIRONMENTAL QUALIFICATION ANALYSES

14.4.1 BASIS OF DISCUSSION

The Unit 2 updated FSAR provides a brief discussion of environmental qualification applicable to Cook Nuclear Plant Unit 2. Although there are minor differences in equipment used between units, that discussion is similar in intent to what is applicable to Unit 1, and includes, by reference, the various accident analyses that are being used for equipment qualification on Unit 1.

14.4.2 POSTULATED PIPE FAILURE ANALYSIS OUTSIDE CONTAINMENT

Presented below is a discussion of analyses associated with high energy line breaks (HELBs) outside of containment. Much of this analysis was performed early in the Unit 1 licensing process, and the results were used for equipment environmental qualification studies.* The results for steamline breaks have been updated to reflect the latest available information regarding the effects of superheated steam.

14.4.2.1 Description Of High Energy Systems Definition

High energy piping systems are defined as those having a normal service temperature above 200^{o}F, a normal operating pressure above 275 psig, and a nominal diameter greater than one inch.

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*This analysis was prepared specifically for Unit 1 and discusses Unit 1 equipment, pipe routings, etc. Because of the similarity between Unit 1 and Unit 2 designs, many of the analyses showing temperatures and pressures in various compartments are applicable to both units. Where specific features of the plants were different, separate analyses had to be done, or the analyses presented in this subchapter were evaluated for applicability as part of the normal Unit 2 design process.

Identification of Systems

The systems of the Donald C. Cook Nuclear Plant that fall under the above definition are:

1. Main Steam System
2. Feedwater System
3. Steam Generator Blowdown System
4. Chemical and Volume Control System
5. Steam Supply to Auxiliary Feedwater Pump Turbine

Specific high energy lines in each of the above systems are presented in Section 14.4.2.6. Where a routing description or analysis is presented for Unit 1 only, a similar description or analysis applies for Unit 2.

Equipment Necessary To Assure Safe Shutdown Of The Plant

Table 14.4.2-1 presents the list of required equipment necessary to assure safe shutdown of the plant following a high energy line pipe failure outside of the containment. The equipment required under each piping failure condition is similar. Figures 14.4.2-1 through 14.4.2-9 graphically depict the location of most of the items identified in Table 14.4.2-1.

Other High Energy Line Systems

High energy lines other than those listed above were walked during plant construction to assure that inadvertent rupture of these lines would not impair the ability to safely shut down the plant. Problem areas were resolved by the use of impingement barriers or other appropriate means before plant operation. A list of the systems that have been considered is presented in Table 14.4.2-2.

REFERENCES, SECTION 14.4.6

1. MSLB Environmental Analysis, Donald C. Cook Units 1 and 2, Impell Report No. 01-0120-1524, Revision 0, September 1986.

2. NRC IE Information Notice No. 84-90, "MSLB Effect on Environmental Qualification of Equipment," December 7, 1984.

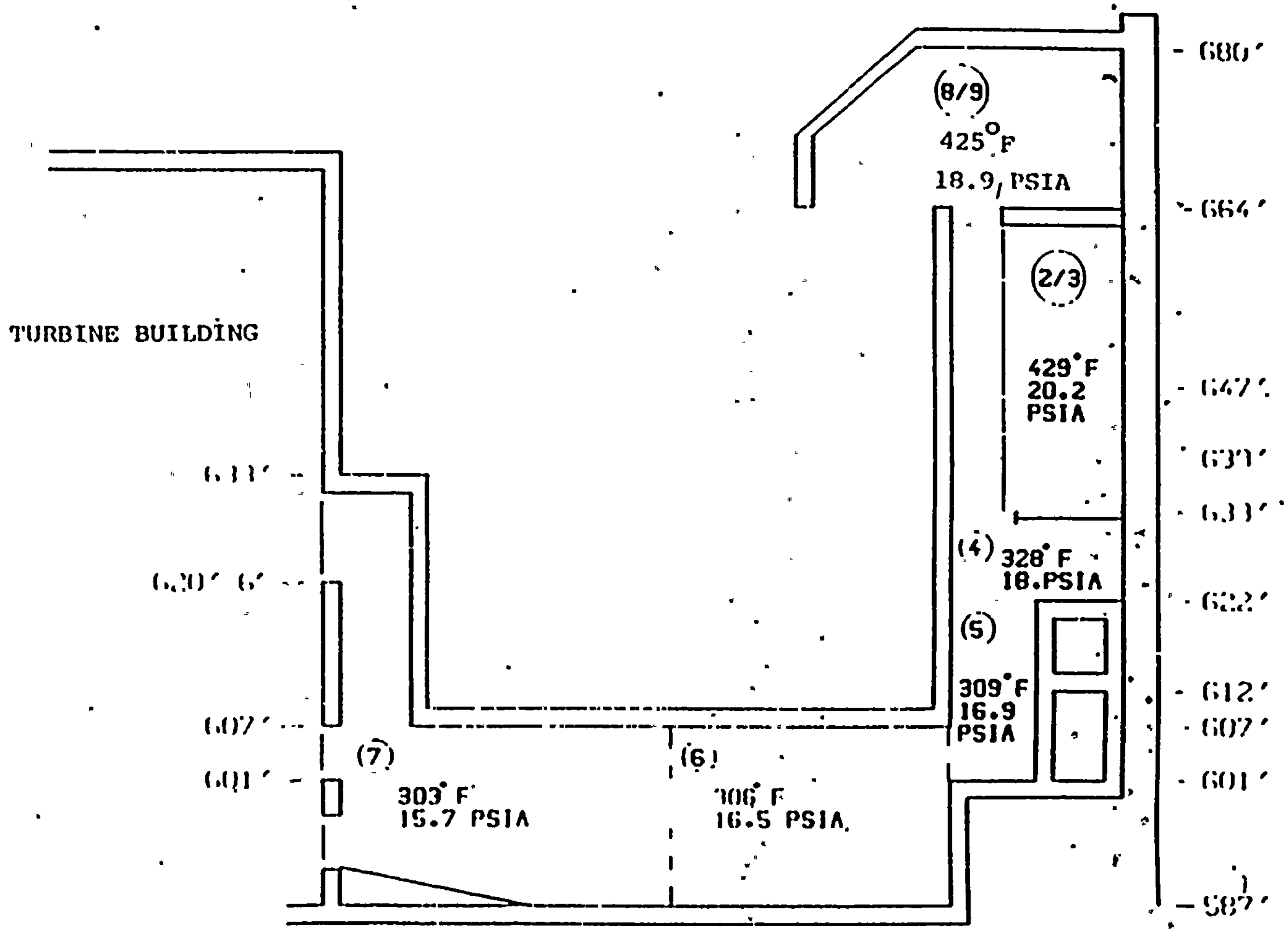

Figure 14.4.6-5 – Peak Environmental Parameters (West Main Steam Enclosure and Accessway)

UNIT 1

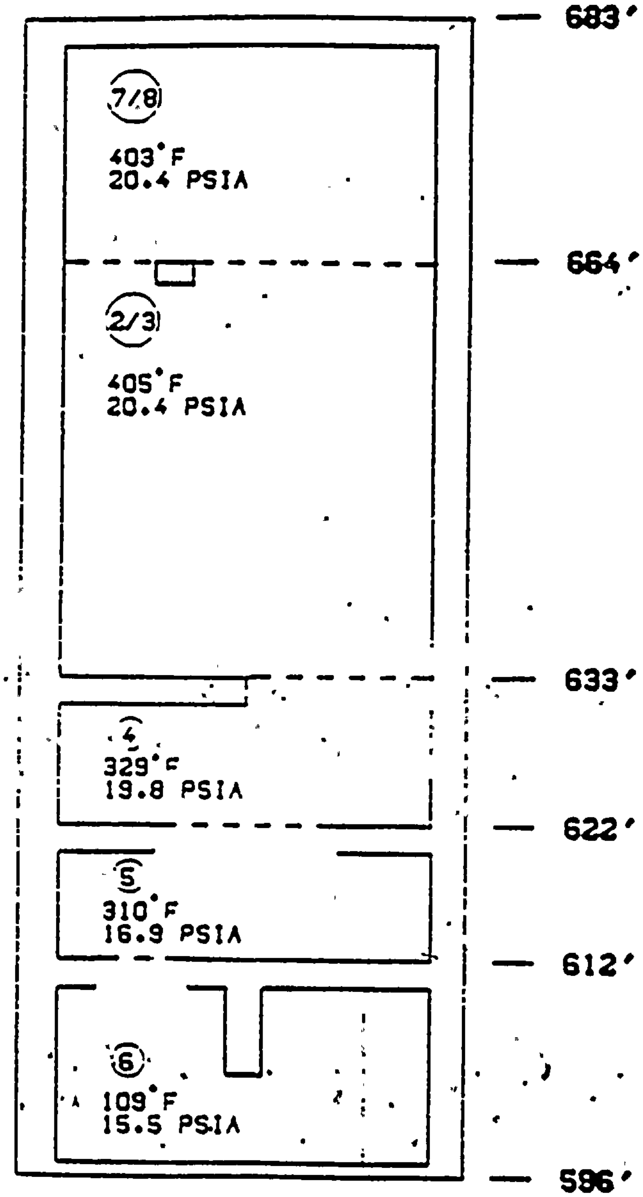

FIGURE 14.4.6-6 - Peak Environmental Parameters (East Main Steam Enclosure)

UNIT 1

AEPSC/DC COOK UNITS 1&2

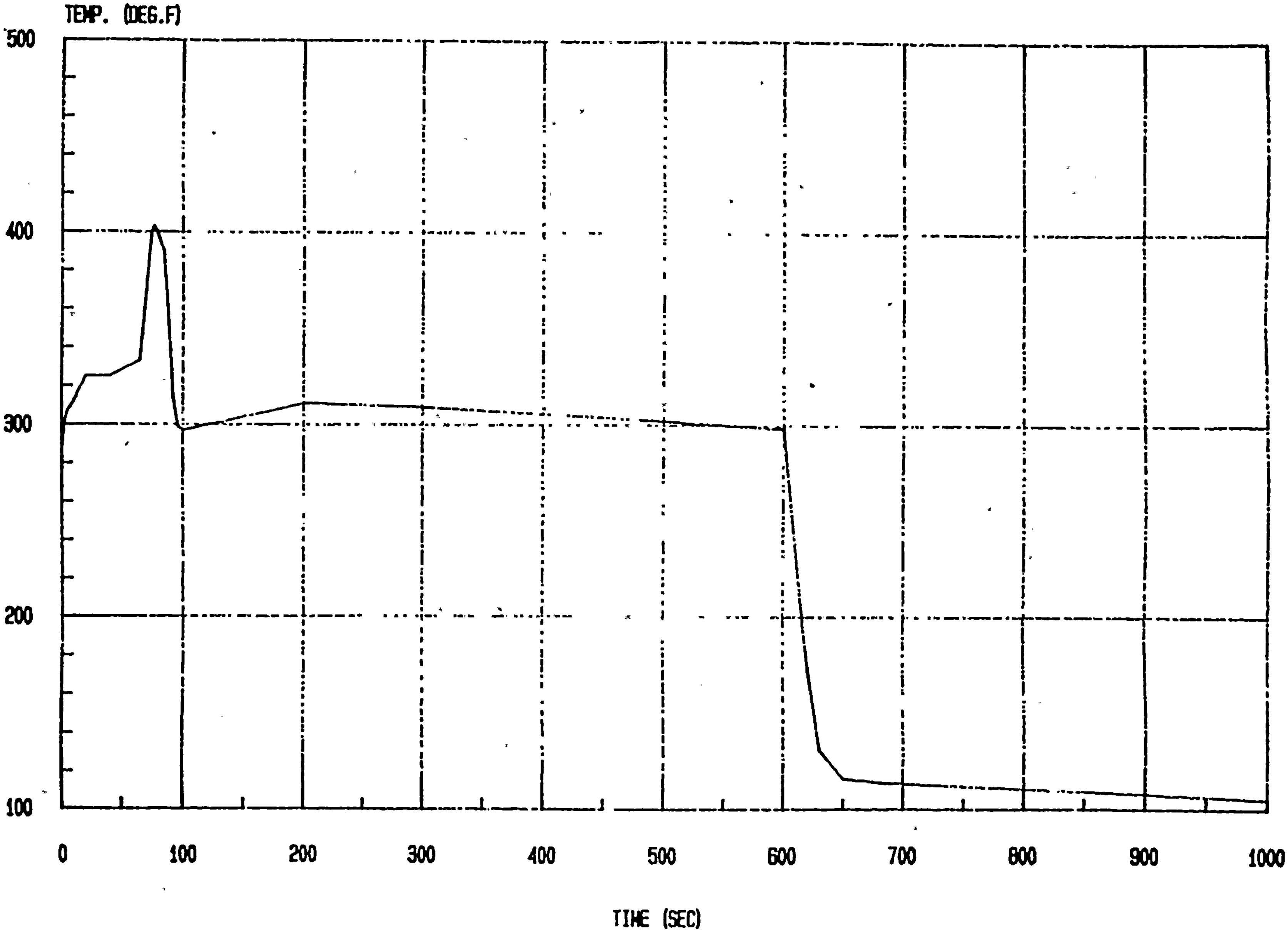

FIGURE 14.4.6-9 EAST MAIN STEAM ENCLOSURE
TEMPERATURE PROFILE IN ELEMENTS 2 AND 3

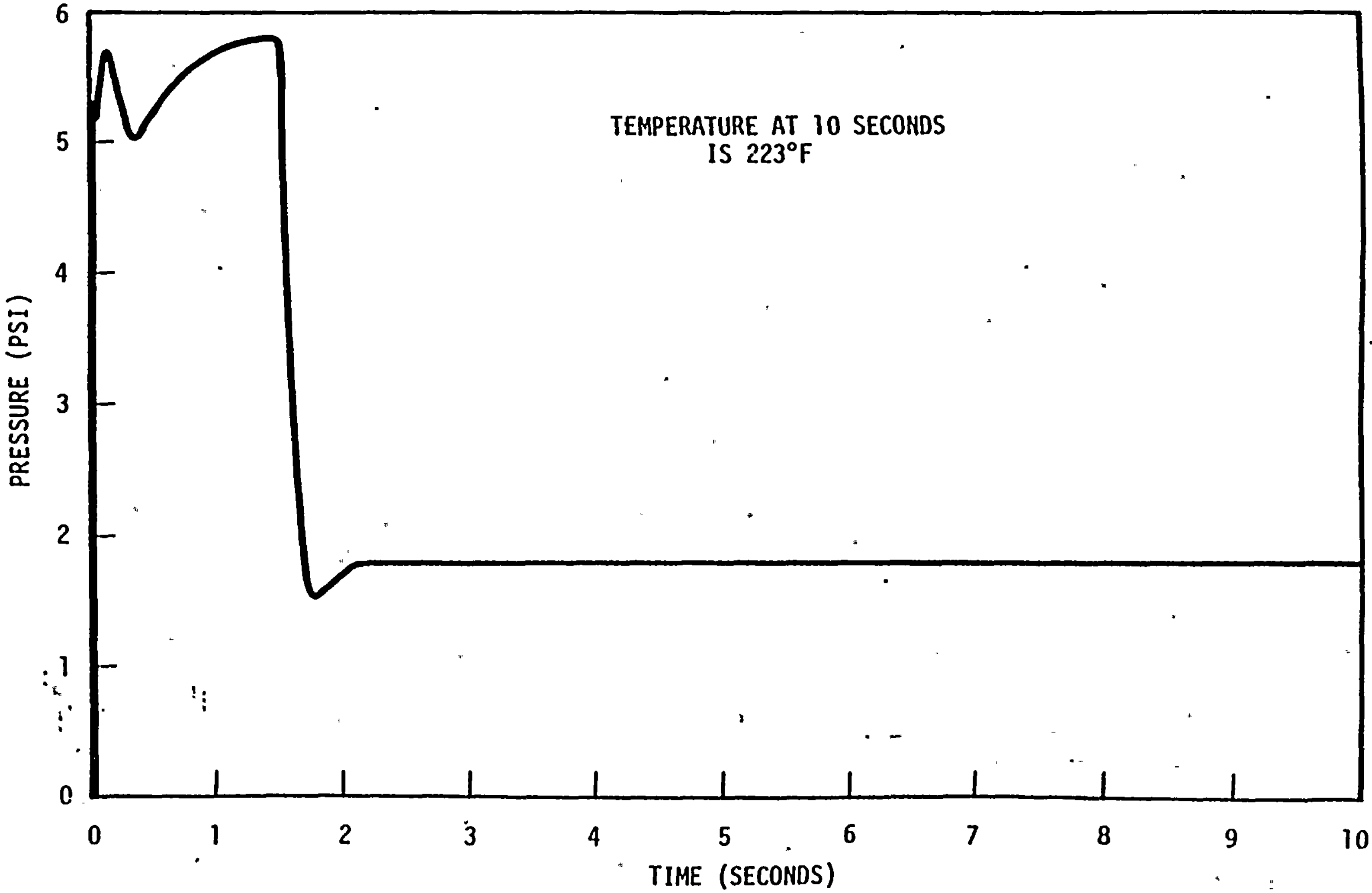

FIGURE 14.4.6-10 FEEDWATER LINE BREAK IN MAIN STEAM ACCESSWAY (ELEMENT 7), PRESSURE VS. TIME

AEPSC/DC COOK UNITS 1&2

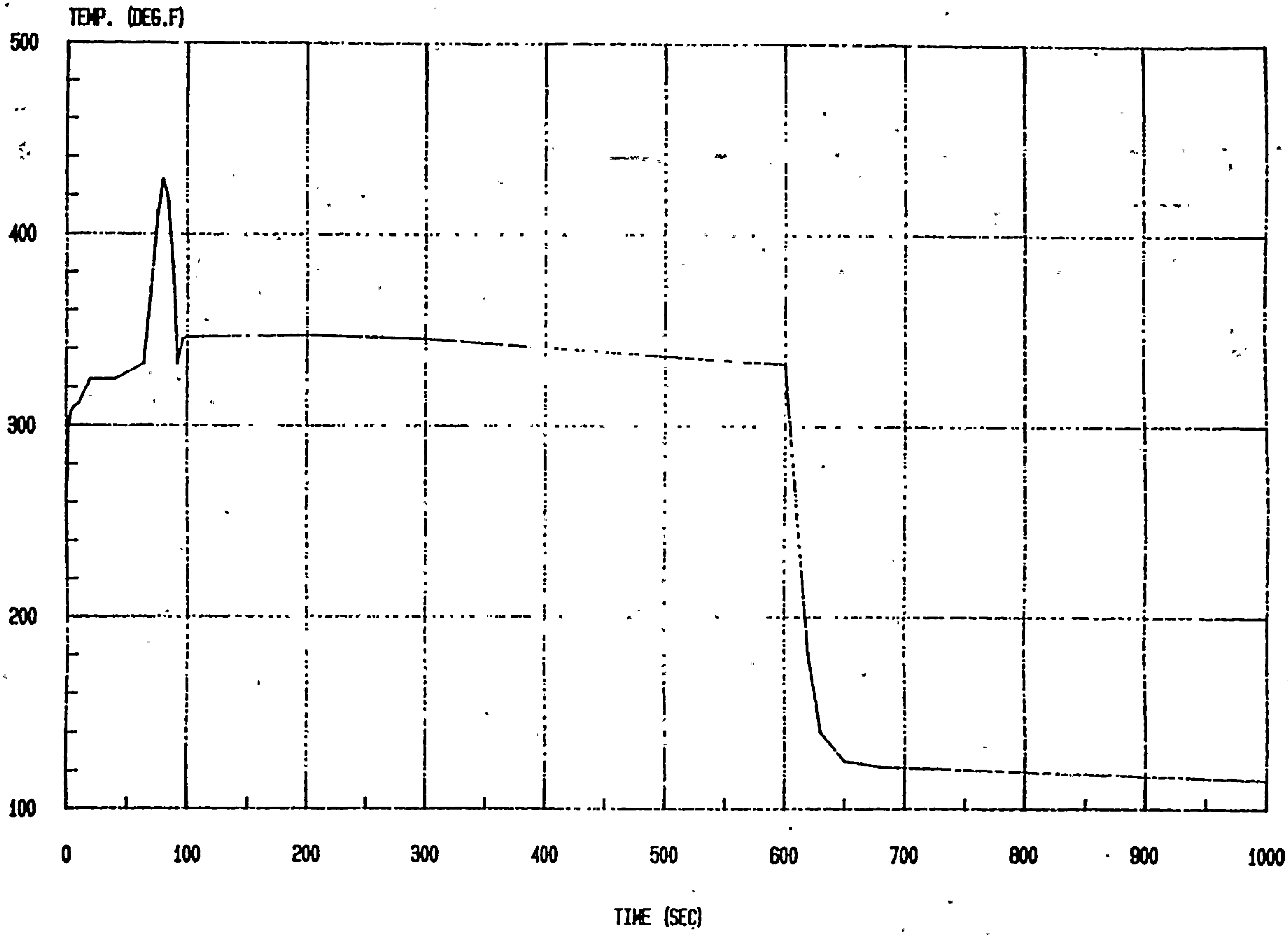

FIGURE 14.4.6-11 WEST MAIN STEAM ENCLOSURE
TEMPERATURE PROFILE IN ELEMENTS 2 AND 3

JULY, 1987

UNIT 1

TABLE 14.4.11-1

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

10 CFR 50.49 ENVIRONMENTAL QUALIFICATION EQUIPMENT LIST

Equipment Manufacturer	Model Or Item Number *	Equipment Description	Plant ID Number(s)
AEPSC Design & Conax Corp.	N/A	Cable Termination At Victoreen Radiation Monitoring System Detector	N/A
AEPSC Design & Conax Corp.	N/A	Cable Termination At Charge Converter (Acoustic Monitors)	N/A
Anaconda	#347	3 1/C #2 Cu Power Cable	Various
Anaconda	#3102	3 1/C #2 Al Power Cable	Various
Anaconda	#3103	3 1/C #2/0 Al Power Cable	Various
Anaconda	#3116	3 1/C #10 Cu Power Cable	Various
Anaconda	#3120	5/C #12 Cu Control Cable	Various
ASCO	206-381-2RVU	Solenoid Valve	XSO-291, 292, 293, 294, 295, 296, 297, 298
ASCO	NP-831654V NP-8316A54V	Solenoid Valve	XSO- 12, 21, 111, 113, 121, 122, 123, 124, 125, 126, 127, 320, 503, 505, 507
Boston Insulated Wire	#3075	2/C #16 Instrument Cable	Various

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Boston Insulated Wire	#3077	Instrument Cable	Various
Brand Rex	#3059	RG 59 B/U Cable	Various
Brand Rex	#3074	RG 11 U Cable	Various
Brand Rex	#3092	4/C #12 AWG Control Cable	Various
Brand Rex	#3093	7/C #12 AWG Control Cable	Various
Brand Rex	#3112	RG 11 AU Cable	Various
Cerro Wire & Cable	#3077	1 STQ #16 Cu Instrument Cable	Various
Champlain	N/A	Kapton Insulated Penetration Feed-through Extension Wire	Various
Conax Corporation	7H57-10000-01	Resistance Temp-erature Detector	NTR-110,120,130,140 210,220,230,240
Conax Corporation	7K 12-11000-01 7K 12-11000-02	Conax Seal Assembly for In-core T/C at Mineral Cable and Instrument Cable	N/A
Conax Corporation	7AD2-10000-01	Resistance Temperature Detector	ITI-310,320
Conax Corporation	7AD2-10000-01	Thermocouple Termination for ECCS	N/A
Conax Corporation	7CC1-10000-01	Resistance Temperature Detector	ETR-12, 14, 20
Conax Corporation	7CR0-10000-01	Resistance Temperature Detector	ITR-311, 321
Conax Corporation	N-11162-01	Seal Assembly for Target Rock Solenoid	FT-045

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Conax Corporation	EP-1	4-kV Electrical Penetration	Various
Conax Corporation	EP-2 Through EP-14	600-V and Below Electrical Penetrations	Various
Conax Corporation	N/A	NAMCO Limit Switch Cable Termination Seal Assembly	FT-37
Continental	#3069	1/C #12 Cu Instrument Cable	Various
Continental	#3075	1 STP #16 Cu Instrument Cable	Various
Continental	#3077	1 STQ #16 Cu Instrument Cable	Various
Continental	#3092	4/C #12 Cu (strnd) Control Cable	Various
Continental	#3093	7/C #12 Cu (Strnd) Control Cable	Various
Continental	#3119	2/C #12 Cu Control Cable	Various
Continental	#3120	4/C #12 Cu Control Cable	Various
Continental	#3121	7/C #12 Cu Control Cable	Various
Continental	#3122	12/C #12 Cu Control Cable	Various
Continental	#3123	15/C #12 Cu Control Cable	Various
Cyprus	#324	3TC #12 Cu Power Cable	Various

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Cyprus	#3102	3 1/C #2 Al Power Cable	Various
Eaton	#3190	2 TP #16 AWG Solid Chromel/Constanton Instrument Cable	Various
Eberline	DA1-6-HT-CC	Radiation Monitoring System Detector	VRS-1101, 1201 (Unit 1); VRS-2101, 2201 (Unit 2)
Essex	#324	3TC #12 Cu Power Cable	Various
Essex	#3116	3 1/C #10 Cu Power Cable	Various
FCI	CL 86 (Sensor Only)	Containment Water Level Sensor	NLI-320, 321
FCI	CL 86 (Sensor Only)	Containment Sump Water Sensor	NLA-310, NLI-311
Foxboro	N-E11GM-HIE2	Pressure Transmitter	MPP-210, 211, 220, 221, 230, 231, 240, 241
Foxboro	E13DM-HIH2	Differential Pressure Transmitter	FFC-210, 211, 220, 221, 230, 231, 240, 241
Foxboro	N-E13DM-HIM1	Differential Pressure Transmitter	FFI-210, 220, 230, 240
Foxboro	N-E11GM-HIE2	Pressure Transmitter	NPP-151,152,153 NPS-153
Foxboro	N-E11GH-HIM2	Pressure Transmitter	NPS-110,111,121,122
Foxboro	N-E13DM-HIM2	Differential Pressure Transmitter	BLP-110,111,112,120 121,122,130,131,132 140,141,142

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Foxboro	N-E13DM-HIH2	Differential Pressure Transmitter	MFC-110,111,120,121 130,131,140,141 NLP-151,152,153
Foxboro	N-E13DH-HIH2	Differential Pressure Transmitter	NLP-151, 152, 153
Foxboro	N-E13DH-HIM2	Differential Pressure Transmitter	IFI-51, 52, 53, 54
Foxboro	N/A	Cable Termination at Integral Junction Box (Foxboro Instruments)	N/A
Foxboro	N-E13DM-HIM2	Differential Pressure Transmitter	ILA-110, 120, 130, 140
Foxboro	N-E11GM-HID2	Pressure Transmitter	IPA-110, 120, 130, 140
Gamma Metrics	N/A	Neutron Flux Detector Cable and Junction Box	NE-21, 23
General Electric	#3119	2/C #12 Cu Control Cable	Various
General Electric	#3120	4/C #12 Cu Control Cable	Various
General Electric	#3121	7/C #12 Cu Control Cable	Various
General Electric	#3122	12/C #12 Cu Control Cable	Various
General Electric	#3123	15/C #12 Cu Control Cable	Various
HAVEG	N/A	Kapton Insulated Penetration Feed-through Extension Wire	N/A

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
ITT Barton	764	Differential Pressure Transmitter	NLA-310; NLI-311, 320, 321 (Unit 1 only) NLI-110, 111, 120, 121, 130, 131
Kerite	#3116	3 1/C #10 Cu Power Cable	Various
Kerite	#3127	3 1/C #2 Cu Power Cable	Various
Limitorque	SMB-00; SMB-000; SMB-1; SMB-2 SMB-00; SMB-000; SMB-1; SMB-2	Valve Motor Operators	CMO-419, 429 FMO-211, 212, 221, 222, 231, 232, 241, 242 ICM-250, 251, 260 265, 305, 306 IMO-210, 211, 212, 215, 220, 221, 222, 225, 255, 256, 262, 263, 270, 275, 310, 312, 314, 315, 316, 320, 322, 324, 325, 326, 330, 331, 340, 350, 360, 361, 362, 910, 911 MCM-221, 231 NMO-151, 152, 153 QCM-250 QMO-225, 226 VMO-101, 102 WMO-711, 712, 713, 714, 717, 718, 721, 722, 723, 724, 725, 726, 727, 728
Marathon	300 Series	Cable Termination at Valve Motor	N/A

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Marathon	1600 Series	Cable Termination at Terminal Block, Outside Containment Only	N/A
Minco	58809	Resistance Temperature Detector	NTQ-110A, 110B, 111A, 111B, 120A, 120B, 121B, 130A, 130C, 131A, 131C
NAMCO	EA180	Limit Switch	Limit Switches for NRV-151, 152, 153
NAMCO	EA180-XX602	Limit Switch With Receptacle and Plug-In Cable	33-VCR-11, 21
NAMCO	EA740-2002X	Limit Switch With Receptacle and Plug-In Cable	33-DCR-310, 320, 330, 340 33-XCR-100, 101, 102, 103
Okonite	#324	3TC #12 Cu Power Cable	Various
Okonite	#399	1/C #2 Cu Power Cable	Various
Okonite	#3102	3 1/C #2 Al Power Cable	Various
Penn Union	6000 Series	Cable Termination at Terminal Block, Outside Containment Only	N/A
Raychem	#3059	RG 59 B/U Cable	Various
Raychem	#3074	RG 11 U Cable	Various
Raychem	#3112	RG 11 AU Cable	Various
Raychem	WCSF	Barton Instrument Connection	N/A
Raychem	WCSF	Connection to RdF RTD Pigtails	N/A

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Raychem	WCSF	Connection to Conax RTD Pigtails	N/A
Raychem	WCSF	Field Cable Splice at Termination Near Valve Actuator or Solenoid	N/A
Raychem	NPKS,NPKX	Foxboro Instrument Connection	N/A
Raychem	WCSF	Instrument Field Cable Splice to Penetration Feed Through Wire	N/A
Raychem, Conax	WCSF, Various	Instrument Field Cable Splice to Penetration Feed-through Wire	N/A
Raychem	WCSF	Solid Kapton Spliced Inside Flood-Up Tube	N/A
Raychem	WCSF	Stranded Kapton Spliced to Field Cable at Flood-Up Terminal Box	N/A
Raychem	NPKS, NPKX	Raychem NPKS or NPKX Splice Kits for Instrument Cable Connection at Penetration, at Penetration Inside Flood-Up Tube, and at Flood-Up Box at Instruments	N/A
Raychem	NPKV, NMCK	Raychem NPKV or NMCK Splice Kits for Cable Connection at Valve Motor Operators, Hydrogen Recombiners and Fan Motors	N/A

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Raychem	WCSF-650-9-N	Gamma Metrics Neutron Flux Monitor Triax Cable Termination at Penetration Inside Flood-up Tubing	N/A
Raychem	NPKS-8-STP-21A NPKS-9-STP-21A NPKS-5-STP-21A NPKS-12-STP-21A	Raychem Splice Kit for In-Core T/C Splice Between Conax Seal Assembly and Instrument Cable and at the Flood-Up Box	N/A
Raychem	NPKC-6-7D	Raychem Splice Kit for splice of control cable of valve and Limit Switch to one cable at Junction Box near valve	N/A
Reliance	Frame #5810P	Pump Motor	PP-009
RdF	21204	Resistance Temperature Detector (RTD)	NTP-111, 121, 131, 141, 211, 221, 231, 241, 110, 120, 130, 140, 210, 220, 230, 240
Rockbestos	3075	ISTP #16 AWG Cu Instrument Cable	Various
Samuel Moore	#3008	1/C #16 Cu Control Cable	Various
Samuel Moore	#3075	1 STP #16 Cu Instrument Cable	Various
Samuel Moore	#3120	4/C #12 AWG 7/S Cable	Various
Samuel Moore	#3186	12 pr. 20 AWG Instrument Cable	Various
Samuel Moore	#3077	1 STQ #16 Cu Instrument Cable	Various

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
Target Rock Corporation	79AB-007	Solenoid Actuated Globe Valve	NSO-021, 022, 023, 024, 061, 062, 063, 064
Technology For Energy (TEC); Endevco (via TEC)	504A; 2273AM1; 3075M6-36;	Acoustic Valve Flow Monitoring System Components	QR-107A, 107B, 107C, 107D
Telemecanique	C2GVJK1-AEP-50	Limit Switch With Pre-wired Cable	33-DCR-301, 302, 303, 304
Victoreen	877-1	Radiation Monitoring System Detector	VRA-1310, 1410 (Unit 1); VRA-2310, 2410 (Unit 2)
Westinghouse	5808Z; 5009H; 5009-P24	Pump Motors	PP-026, 035, 050,
Westinghouse	TBDP	Fan Motor	HV-CEQ-1, HV-CEQ-2
Westinghouse	N/A	Hydrogen Recombiners	HR-1, HR-2
Whittaker Corp. (via C-E)	N/A	Mineral Insulated T/C Cable and Connectors for In-Core T/C System (Unit 1 only)	Various
N/A	N/A	Control Cable Termination at Valve Operator Limit and Torque Switches	N/A
N/A	N/A	Power Cable Termination At Pump Motor	N/A

TABLE 14.4.11-1 (Cont'd)

Equipment Manufacturer	Model Or Item Number	Equipment Description	Plant ID Number(s)
N/A	N/A	Termination At Valve Motors, Fan Motors, and Hydrogen Recombiners	N/A
N/A	N/A	Cable Termination at Eberline Radiation Monitoring System Detector	N/A
N/A	N/A	Triaxial Cable Termination Triaxial Penetration Feedthrough Wire	N/A

Table 14.4.11-2

PEAK ENVIRONMENTAL QUALIFICATION CONDITIONS FOR

LOCA, MSLB, AND FEEDWATER LINE BREAK INSIDE CONTAINMENT

LOCA

Location	Temp(°F)*	Press(psia)+
- Upper Comp.	160	26
- Lower Comp.	240	26
- Inst. and F/A Room	240	26

MSLB AND FEEDWATER LINE BREAK

Location	Temp(°F)	Press(psia)***
- Upper Comp.	156**	35.5
- Lower Comp.	328.2++	35.5
- Inst. and F/A Room	328.1++	35.5

* FSAR, Figure 14.3.4-17

\+ FSAR, Figure 14.3.5-3

** Unit 2 FSAR Figure 14.1.5-9

++ Unit 2 FSAR Table 14.3.4-38

*** Nuclear Safety & Licensing Calculation TH-86-04 (Energy Balance Calculation)

Table 14.4.11-9

PEAK ENVIRONMENTAL QUALIFICATION CONDITIONS FOR
HELB OUTSIDE CONTAINMENT

Compartment	Temp(°F)	Press(psia)
East Main Steam Enclosure	405*	26.2****
West Main Steam Enclosure	429*	26.2****
Main Steam Accessway	306*	26.2****
Diesel Generator Pipe Tunnel	240**	26.2****
Turbine Driven Pump Room	225***	16***
Vestibule	225***	16.0***
ESW Tunnel	225***	16.0***
F. W. Tunnel	230***	26.2***
W. Heat Exchanger	230***	26.2***
Turbine Room	230***	26.2***

*Impell Report, "MSLB Environmental Analysis, D. C. Cook Units 1 and 2," Figures 4.1 and 4.2 dated September 1986

**NS&L Calculation TH 86-04

***Letter dated February 27, 1980 from J. F. Etzweiler to L. F. Caso

****Letter, R. G. Vasey to K. J. Munson dated August 7, 1986

14.0 SAFETY ANALYSIS

This chapter presents an evaluation of the safety aspects of Unit 2 of Cook Nuclear Plant and demonstrates that the plant can be operated safely even if highly unlikely events are postulated. It also shows that radiation exposures resulting from occurrences of these highly unlikely accidents do not exceed the guidelines of 10 CFR 100.

Unit 2 of Cook Nuclear Plant was initially loaded with fuel fabricated by Westinghouse Electric Corporation for the first three cycles. From Cycle 4 through Cycle 7, reload fresh fuel was fabricated by Advanced Nuclear Fuel, previously known as Exxon Nuclear Company. Starting with Cycle 8, the fabrication of fresh reload fuel is again furnished by Westinghouse, this time using the 17 x 17 Vantage 5 fuel assembly design. To the extent that analyses in this chapter involve a particular fuel design, it is the Westinghouse Vantage 5 fuel that is considered, either as the only fuel design, or in conjunction with previously burned Advanced Nuclear Fuel design assemblies, in order to reflect a "transition core" between the two designs.

This chapter is divided into the three sections described below, each section dealing with a different category of fault conditions. The numbers identifying the Cook Nuclear Plant Unit 2 accidents are organized in the same way as in the original licensing basis FSAR.

<u>Core and Coolant Boundary Protection Analysis, Section 14.1</u>

The fault conditions discussed in this section may occur with moderate frequency during the life of the plant. They are accommodated with, at most, a reactor shutdown with the plant being capable of returning to operation after a corrective action. In addition, no fault in this category shall cause consequential loss of function of fuel cladding and reactor coolant system barriers.

Standby Safeguards Analysis, Section 14.2

The fault conditions discussed in this section are more severe but very infrequent and may lead to a breach of fission product barriers.

Reactor Coolant System Pipe Rupture (Loss of Coolant Accident), Section 14.3

The accident discussed in this section is a rupture of a reactor coolant pipe including the double ended severance of the largest pipe in the reactor coolant system, which is worst conceivable accident and therefore is used as a basis for the design of engineered safeguards.

14.0.1 Summary of Results

To support the Cook Nuclear Plant Unit 2 17 x 17 VANTAGE 5 fuel transition, analyses and evaluations are performed as appropriate. Table 14.0-1 presents the list of non-LOCA and LOCA events analyzed or evaluated for the VANTAGE 5 fuel transition. The only non-LOCA event that was not analyzed for the VANTAGE 5 fuel transition is the startup of an inactive loop event. This accident was not analyzed since the event can not occur above the P-7 permissive setpoint (11% power) as restricted by the Technical Specifications. The analysis presented in FSAR Section 14.1.7 remains bounding with respect to the restriction to 11% power for the operation of 3 reactor coolant pumps imposed by the Technical Specifications. The results of the analyses and evaluations presented in the following sections show that the transition from ANF to 17 x 17 VANTAGE 5 fuel for Cook Nuclear Plant Unit 2 can satisfy the applicable FSAR safety limits.

The safety analyses and evaluations presented in Sections 14.1, 14.2 and 14.3 support Cook Nuclear Plant Unit 2 operation in the transition cycles beginning from Cycle 8, with core power of 3411 MWt in the range of primary full power vessel average temperatures between 547°F and 576°F at primary system pressure of 2250 psia. For a full Westinghouse 17 x 17 VANTAGE 5 core, the non-LOCA safety analyses and evaluations support plant operation with uprated core power of 3588 Mwt in the range of primary full power vessel average temperatures between 547°F and 581.3°F at primary pressure values of 2100 psia <u>or</u> 2250 psia.

TABLE 14.0-1

ACCIDENTS CONSIDERED FOR VANTAGE 5 FUEL TRANSITION

EVENTS EVALUATED

FSAR Section	Accident
14.1.7	Startup of an Inactive Reactor Coolant Loop

EVENTS ANALYZED

FSAR Section	Accident
14.1.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition
14.1.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
14.1.3	Rod Cluster Control Assembly Misalignment
14.1.4	Rod Cluster Control Assembly Drop
14.1.5	Uncontrolled Boron Dilution
14.1.6	Loss of Reactor Coolant Flow (including Locked Rotor Analysis)
14.1.8	Loss of External Electric Load or Turbine Trip
14.1.9	Loss of Normal Feedwater

The differences between the limiting trip setpoint assumed for the analysis and the nominal trip setpoint in Table 14.1.0-4 represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications and are shown in Table 14.1.0-4 for completeness. The protection system channels are calibrated and instrument response times determined periodically in accordance with the Technical Specifications.

<u>Engineered Safety Features (ESF) Setpoints and Time Delays</u>

Table 14.1.0-5 presents the limiting ESF setpoints assumed in the accident analyses and the time delay assumed for each trip function. The nominal value of the low steamline pressure setpoint assumed was 500 psig. The revised low steamline pressure setpoint value provides operating margin for the potential reduced temperature operating conditions of Table 14.1.0-1 (cases 2, 5, and 6). The difference between the limiting ESF trip setpoint assumed for the analysis and the nominal trip setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal ESF trip setpoints are specified in plant Technical Specifications and are shown in Table 14.1.0-5 for completeness.

14.1.0.7 Plant Systems and Components Available for Mitigation of Accident Effects

Table 14.1.0-6 is a summary of reactor trip functions, engineered safety features, and other equipment available for mitigation of accident effects. The trips in the Table 14.1.0-6 include some that are anticipatory and/or backup functions. These trips are not necessarily taken credit for the safety analysis.

In the analysis of the Chapter 14.1 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operations if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

14.1.0.8 Residual Decay Heat

For the Non-LOCA analyses, conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS decay heat standard (Reference 3) plus uncertainty was used for calculation of residual decay heat levels. Figure 14.1.0-7 presents this curve as a function of time after shutdown.

Distribution of Decay Heat Following Loss of Coolant Accident

During a loss-of-coolant accident, the core is rapidly shutdown by void formation or rod cluster control assembly insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4% which represents the fraction of heat generated within the clad and pellet drops to 95% for the hot rod in a loss-of-coolant accident.

For example, consider the transient resulting from the postulated double ended break of the largest reactor coolant system pipe, 1/2 second after the rupture about 30% of the heat generated in the fuel rods is from gamma ray absorption. The power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma ray contribution or 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

14.1.0.9 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized

TABLE 14.1.0-2
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output(9) (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	See Section 14.1.1.2	NA	(11)	W-3 ANF WRB-2 and W-3 V-5	No	0	162,840	547.0	2037.0(6)
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power (2), (3)	LOFTRAN	NA* +5	.54 NA	Max (4) Min (1)	W-3 ANF WRB-2 V-5	Yes	3608 2165 361	366,400	576.0 564.4 549.9	2250.0
Rod Cluster Control Assembly Misalignment	LOFTRAN THINC	NA	NA	NA	W-3 ANF WRB-2 V-5	Yes	3600	366,400	581.3	2100.0(10)
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA	NA	3600 0	NA	NA	NA
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	NA	Max(4)	W-3 ANF WRB-2 V-5	Yes	3608	366,400	581.3(12)	2100.0(10)
Locked Rotor (Peak Pressure)	LOFTRAN	+5	NA	Max(4)	NA	NA	3608	354,000	585.4	2312.6

*NA - Not Applicable
(1) Minimum Doppler power coefficient (pcm/%power) = -9.55 + 0.3732Q, where Q is in % power (see Figure 14.1.0-1)
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Initial conditions for the separate analysis to bound assumed operating conditions for full V-5 core are shown in Table 14.1.0-3.
(4) Maximum Doppler power coefficient (pcm/%power) = -19.4 + 0.7176Q, where Q is in % power (see Figure 14.1.0-1)
(5) Minimum and maximum reactivity feedback cases were examined.
(6) Core Pressure.
(7) Full Power Doppler Power defect at BOL and EOL assumed to be -910 pcm and -840 pcm respectively.
(8) Core thermal power.
(9) Includes reactor coolant pump heat, if applicable.
(10) For transition cycles, pressurizer pressure is 2250 psia.
(11) Zero Power Doppler Power Defect at BOL assumed to be -1020 pcm.
(12) For Transition Cycles, Vessel Average Temperature is 576°F.

TABLE 14.1.0-2 (Continued)
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output(9) (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	NA	Max(4)	NA	NA	3608	354,000	585.4	2037.4
Loss of Electrical Load Turbine Trip (3), (5)	LOFTRAN	NA	.54	Max(4)	W-3 ANF WRB-2 V-5	Yes	3600	366,400	576.0	2250.0
		+5	NA	Min(1)						
Loss of Normal Feedwater	LOFTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	585.4	2312.6
Excessive Heat Removal Due to Feedwater System Malfunction (3)	LOFTRAN	NA	.54	Min(1)	W-3 ANF WRB-2 V-5	Yes	3600 0	366,400	576.0 547.0	2250.0
Excess Load Increase (3)	LOFTRAN	NA	0	Min(1)	W-3 ANF WRB-2 V-5	Yes	3600	366,400	576.0	2250.0
		NA	.54	Max(4)						
Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	+5	NA	Max(4)	NA	NA	3680	354,000	541.4	2312.6
Rupture of a Steam Pipe	LOFTRAN THINC	See Figure 14.2.5-1	NA	See Figure 14.2.5-2	W-3 ANF W-3 V-5	NO	0	354,000	547.0	2100.0

*NA - Not Applicable
(1) Minimum Doppler power coefficient (pcm/%power) = -9.55 + 0.03732Q, where Q is in % power (see Figure 14.1.0-1)
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Initial conditions for the separate analysis to bound assumed operating conditions for full V-5 core are shown in Table 14.1.0-3.
(4) Maximum Doppler power coefficient (pcm/%power) = -19.4 + 0.07176Q, where Q is in % power (see Figure 14.1.0.-1)
(5) Minimum and maximum reactivity feedback cases were examined.
(6) Core Pressure.
(7) Full Power Doppler Power defect at BOL and EOL assumed to be -910 pcm and -840 pcm respectively.
(8) Core thermal power.
(9) Includes reactor coolant pump heat, if applicable.
(10) For transition cycles, pressurizer pressure is 2250 psia.
(11) Zero Power Doppler Power Defect at BOL assumed to be -1020 pcm.
(12) For Transition Cycles, Vessel Average Temperature is 576°F.

TABLE 14.1.0-2 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (Δk/gm/cc)	Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output(9) (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 14.2.6	NA	(7)	NA 0	NA 162,840	3660(8) 547.0	354,000	585.4	2037.4(6)
Rupture of Feedwater Pipe	LOFTRAN	NA	.54	Max(4)	NA	NA	3680	354,000	585.4	2162.6

*NA - Not Applicable
(1) Minimum Doppler power coefficient (pcm/%power) = -9.55 + 0.03732Q, where Q is in % power (see Figure 14.1.0-1)
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Initial conditions for the separate analysis to bound assumed operating conditions for full V-5 core are shown in Table 14.1.0-3.
(4) Maximum Doppler power coefficient (pcm/%power) = -19.4 + 0.07176Q, where Q is in % power (see Figure 14.1.0.-1)
(5) Minimum and maximum reactivity feedback cases were examined.
(6) Core Pressure.
(7) Full Power Doppler Power defect at BOL and EOL assumed to be -910 pcm and -840 pcm respectively.
(8) Core thermal power.
(9) Includes reactor coolant pump heat, if applicable.
(10) For transition cycles, pressurizer pressure is 2250 psia.
(11) Zero Power Doppler Power Defect at BOL assumed to be -1020 pcm.
(12) For Transition Cycles, Vessel Average Temperature is 576°F.

TABLE 14.1.0-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED: SEPARATE FULL VANTAGE 5 CORE ANALYSES

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output(5) (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal At power (2),	LOFTRAN	NA*	.54	Max(3)	WRB-2	Yes	3680 2165 361	366,400	581.3 567.6 550.4	2100.0
		+5	NA	Min(1)						
Loss of Electrical Load or Turbine Trip (4)	LOFTRAN	NA	.54	Max(3)	WRB-2	Yes	3600	366,400	581.3	2100.0
		+5	NA	Min(1)						
Excessive Heat Removal Due to Feedwater System Malfunction	LOFTRAN	NA	.54	Min(1)	WRB-2	Yes	3600 0	366,400	581.3 547.0	2100.0
Excess Load Increase	LOFTRAN	NA	0	Min(1)	WRB-2	Yes	3600	366,400	581.3	2100.0
		NA	.54	Max(3)						

*NA - Not Applicable
(1) Minimum Doppler power coefficient (pcm/%power) = -9.55 + 0.03732Q, where Q is in % power (see Figure 14.1.0-1)
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler power coefficient (pcm/%power) = -19.4 + 0.07176Q, where Q is in % power (see Figure 14.1.0.-1)
(4) Minimum and maximum reactivity feedback cases were examined.
(5) Includes reactor coolant pump heat, if applicable.

14.1.2A Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal At Power (Mixed Core)

14.1.2A.1 Identification of Causes and Accident Description

An uncontrolled Rod Control Cluster Assembly (RCCA) withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator lags behind the power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to minimize the possibility of breaching the cladding, the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value.

The automatic features of the reactor protection system which minimize adverse effects to the core in an RCCA bank withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip on high neutron flux if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable fuel power rating is not exceeded.

4. A high pressure reactor trip, actuated from any two out of four pressure channels, is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is set at a fixed point.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

a. High neutron flux (one out of four)
b. Overpower ΔT (two out of four)
c. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower ΔT and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is illustrated in Figure 14.1.0-5. This figure represents the allowable conditions of reactor coolant loop average temperature and power with the design power capability in a two-dimensional plot.

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions. Reactivity insertion rates and initial conditions govern which protective function occurs first.

14.1.2A.2 Analysis of Effects and Consequences

<u>Method of Analysis</u>

This transient is analyzed using the LOFTRAN code[1]. The core limits as illustrated in Figure 14.1.0-5 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

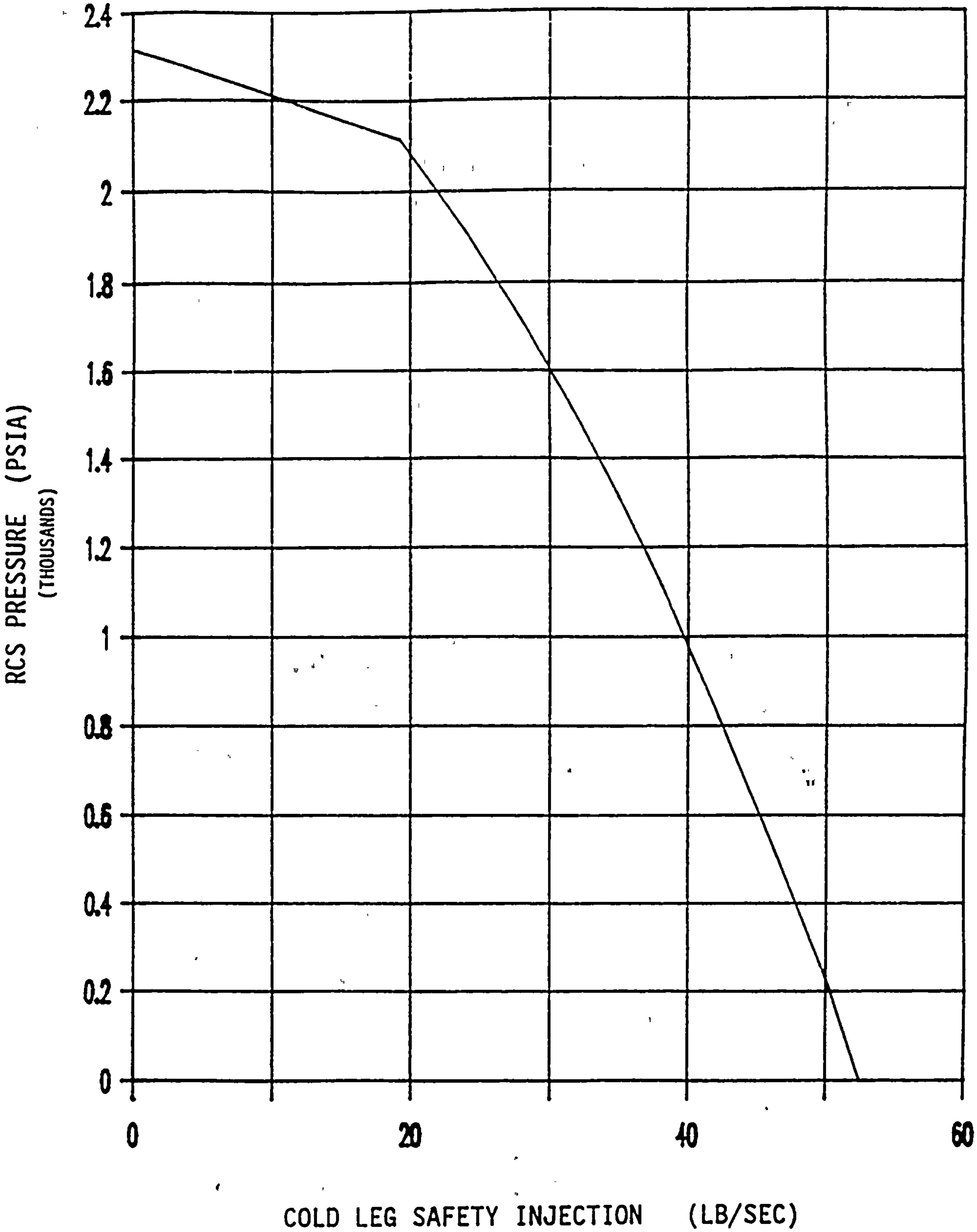

Figure 14.2.5-3 Safety Injection Flow Supplied by One Charging Pump

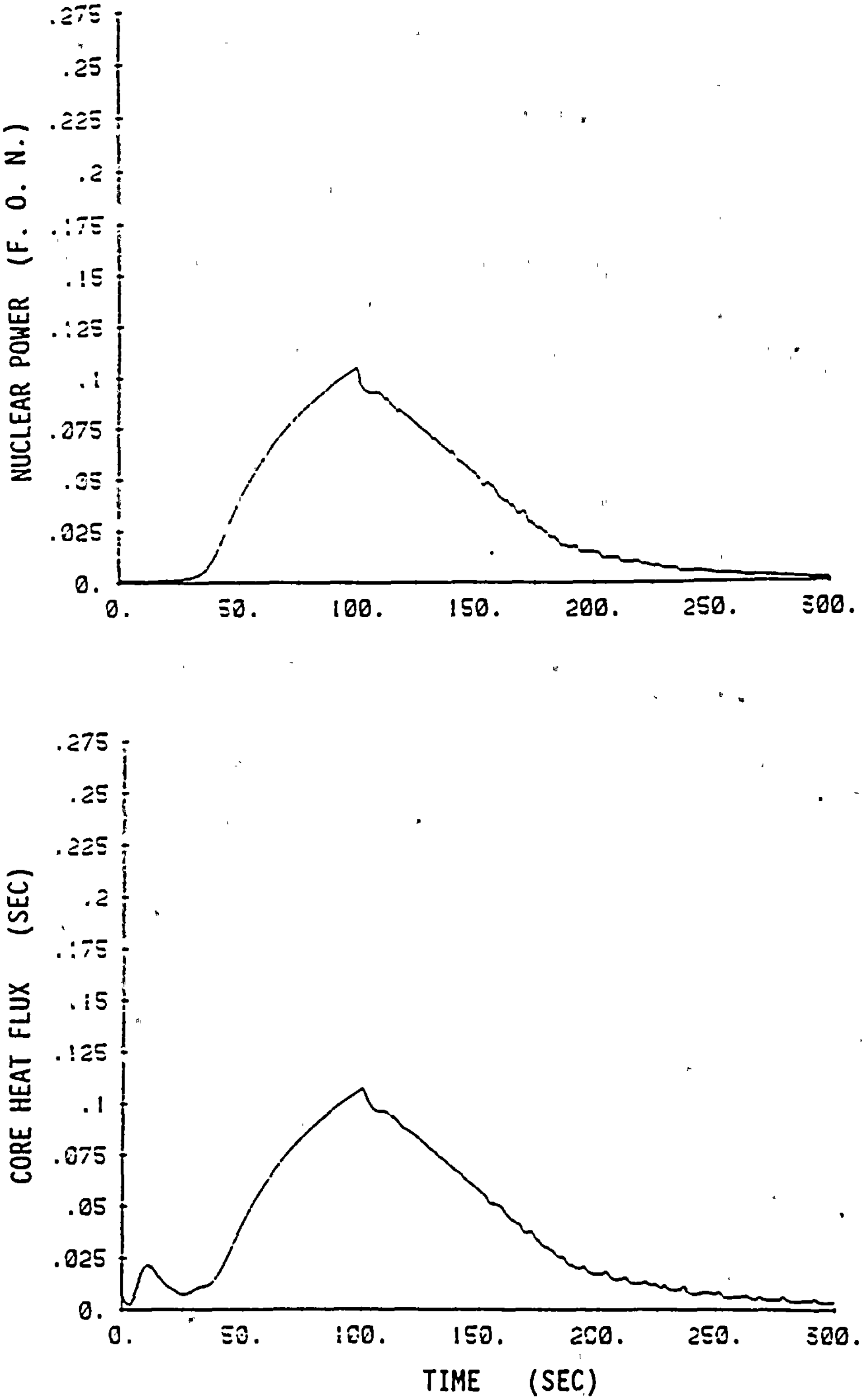

Figure 14.2.5-4 Steamline Break DER Inside Containment with Power
Nuclear Power and Core Heat Flux Versus Time

14.3 REACTOR COOLANT SYSTEM PIPE RUPTURE (LOSS OF COOLANT ACCIDENT)

Cook Nuclear Plant Unit 2 was originally supplied with fuel by Westinghouse Electric Co. It was later refueled with replacement fuel supplied by Exxon Nuclear Company (now Advanced Nuclear Fuels Corporation (ANF)). Most recently, Vantage 5 replacement fuel from Westinghouse is used for reload fresh fuel.

This section discusses loss-of-coolant accident analyses applicable to the current Westinghouse Vantage 5 fuel, and to the fuel supplied by ANF.

Loss-of-coolant accidents (LOCAs) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The Donald C. Cook Nuclear Plant Unit 2 emergency core cooling system (ECCS) has been designed to mitigate the effects of postulated LOCAs by providing a sufficient amount of borated water to protect the fuel in the reactor core.

In order to assure effective long-term core cooling, certain operator actions are assumed. These actions are principally (1) to switch the ECCS from the injection phase to the recirculation phase, (2) to place the reactor coolant pumps in a condition where they can most effectively aid core cooling, and (3) to switch the ECCS from cold leg recirculation to hot leg recirculation at the appropriate time to prevent boron precipitation. All of these items and other appropriate actions are specified in plant procedures. Long term cooling also requires long-term criticality control. Criticality control is achieved by determining the RWST and accumulator concentration necessary to maintain subcriticality without credit for RCCA insertion. The necessary RWST and accumulator concentration is a function of each core design. The current Technical Specification value is 2400 ppm to 2600 ppm boron [20].

14.3.1 LARGE BREAK LOSS-OF-COOLANT-ACCIDENT ANALYSES

14.3.1.1 MAJOR LOCA ANALYSES APPLICABLE TO WESTINGHOUSE FUEL

14.3.1.1.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the Donald C. Cook Nuclear Plant Unit 2, but is postulated as a conservative design basis.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50, 1974)[1] as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen, does not exceed 1 percent of the total amount of Zircaloy in the fuel rod cladding.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Therefore, the worst break for Cook Nuclear Plant Unit 2 (C_D=0.6) was reanalyzed, assuming maximum safeguards (Case A vs. Case F of Table 14.3.1-1). Examination of the LOCA analysis results in Table 14.3.1-6 demonstrates that minimum safeguards assumptions result in the highest peak clad temperature for Cook Nuclear Plant Unit 2.

14.3.1.1.3.4 Transition Core Effects

When assessing the effect of transition cores on the large break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than either a complete core of the 17x17 ANF assembly design or a complete core of the Westinghouse 17x17 VANTAGE 5 design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

The 17x17 ANF fuel assembly is nearly identical to the Westinghouse 17x17 OFA assembly in terms of hydraulic and geometric characteristics. Therefore, the analyses reported in Reference 19 which demonstrate that the 17x17 VANTAGE 5 fuel features result in a fuel assembly that is more limiting than a Westinghouse 17x17 OFA fuel assembly, with respect to large break LOCA ECCS performance, remain valid as applied at Cook Nuclear Plant Unit 2. The same large break LOCA transition core penalty reported in Section 5.2.3 of Reference 19 will be applied to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies.

Westinghouse transition core designs, including specific 17X17 OFA to 17x17 VANTAGE 5 transition core cases, were analyzed. The increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50°F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50°F (Reference 19). As stated earlier, this transition core penalty continues to apply to the transition from 17x17 ANF fuel assemblies to Westinghouse 17x17 VANTAGE 5 fuel assemblies due to the near identical design of 17x17 ANF and Westinghouse 17x17 OFA fuel assemblies. Once a full core of VANTAGE 5 fuel is achieved the large break LOCA analysis will apply without the transition core penalty.

14.3.1.1.3.5 Results

Based on the results of the LOCA sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]), the limiting large break was found to be the double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 14.3.1-5 and 14.3.1-6.

The containment data used to generate the LOTIC backpressure transient are shown in Table 14.3.1-4. The mass and energy release data for the minimum and maximum safeguards cases (Case A and F) are shown in Tables 14.3.1-7 and 14.3.1-8 respectively. In addition, mass and energy release data for Case G (3413 Mwt, RHR cross tie valve closed) are shown in Table 14.3.1-9. The mass releases for the remaining cases are not presented, since they do not vary significantly from the data shown in Table 14.3.1-7. Nitrogen release rates to the containment are given in Table 14.3.1-10.

Figures 14.3.1-3a through 14.3.1-30 present the results of the cases analyzed for the large break LOCA. The alphabetic designation in the figure number corresponds to the cases as described in Table 14.3.1-1.

Figures 14.3.1-3a-g The system pressure shown is the calculated core pressure.

Figures 14.3.1-4a-g	The flow rate from the break is plotted as the sum of both ends of the guillotine break.
Figures 14.3.1-5a-g	The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
Figures 14.3.1-6a-g	The core flow rate is shown during the blowdown phase of the transient.
Figures 14.3.1-7a-g	The accumulator flow rate during blowdown is plotted as the sum of that injected into the intact cold legs.
Figures 14.3.1-8a-g	The core and downcomer collapsed liquid water levels are plotted during the reflood phase of the transient.
Figures 14.3.1-9a-g	The core inlet flow rate is shown as it is calculated during the reflood phase.
Figures 14.3.1-10a-g	The total pumped ECCS flow rate injecting into the intact cold legs is shown.
Figures 14.3.1-11a-g	The integral of the core inlet flow rate as calculated with BASH is plotted.
Figures 14.3.1-12a-g	The mass flux is plotted at the hot spot (the node which produced the peak clad temperature) on the hot rod.
Figures 14.3.1-13a-g	The heat transfer coefficient is plotted at the hot spot on the hot rod.
Figures 14.3.1-14a-g	The fluid temperature at the hot spot on the hot rod is plotted.

Figures 14.3.1-15a-g	The clad temperature at the hot spot is shown for the hot rod.
Figures 14.3.1-16-18	The containment backpressure transient used in the analysis is provided for Cases A, F and G (the minimum and maximum SI flow cases, and the 3413 Mwt cross tie valve closed case).
Figures 14.3.1-19-27	These figures show the heat removal rates of the heat sinks found in the lower and upper compartment and the heat removal by the sump and lower compartment spray for Cases A, F and G.
Figures 14.3.1-28-30	These figures show the temperature transients in both the lower and upper compartments of containment and flow from the upper to lower compartments for Cases A, F and G.

The peak clad temperature calculated for a large break is 2140°F, which is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 6.80 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, corresponding to less than 0.3 percent hydrogen generation, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

20. Attachment 13 to letter, M. P. Alexich, I&M, to H. R. Denton, NRC, March 26, 1987, AEP:NRC:0916W.

TABLE 14.3.1-1

LARGE BREAK LOCA - CASES ANALYZED

CASE A - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, F^N_{DH}=1.620, Minimum SI with cross-tie valves open. Limiting break case, i.e., this case had highest PCT for all cases analyzed.

CASE B - C_D=0.4, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.240, F^N_{DH}=1.620, Minimum SI with cross-tie valves open.

CASE C - C_D=0.8, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.240, F^N_{DH}=1.620, Minimum SI with cross-tie valves open.

CASE D - C_D=0.6, 3588 Mwt Core Power, Low Temperature (T_{HOT}=582.3°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, F^N_{DH}=1.620, Minimum SI with cross-tie valves open.

CASE E - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.0°F), Low Pressure (P_{RCS}=2037 psia), F_Q=2.220, F^N_{DH}=1.620, Minimum SI with cross-tie valves open.

CASE F - C_D=0.6, 3588 Mwt Core Power, High Temperature (T_{HOT}=615.2°F), High Pressure (P_{RCS}=2313 psia), F_Q=2.220, F^N_{DH}=1.620, Maximum SI with cross-tie valves open.

DONALD C. COOK NUCLEAR PLANT UNIT 2

TABLE 14.3.1-4 (continued)
LARGE BREAK LOCA CONTAINMENT DATA
(ICE CONDENSER CONTAINMENT)

Structural Heat Sinks

Compartments	Area (ft2)	Thickness (ft)	Material
1. DE	12,105	0.0469/2.0	Steel/Concrete
2. LC/DE	11,700	2.00	Concrete
3. LC/DE	65,980	1.35	Concrete
4. LC	5,481	0.0833	Steel
5. LC	4,735	0.01147	Steel
6. LC	289	0.250	Lead
7. LC	14,690	0.0079	Steel
8. LC	3,439	0.1561	Steel
9. LC	5,775	0.009	Steel
10. LC	49,665	0.0096	Steel
11. LC	7,013	0.037	Steel
12. LC	2,457	0.0334	Stainless Steel
13. UC	378	0.0365/0.1667	Steel/Concrete
14. UC	29,772	0.0092	Steel
15. UC	8,033	0.0209	Steel
16. UC	420	0.0052	Steel
17. UC	29,330	1.47	Concrete
18. UC	34,125	0.0469/2.0	Steel/Concrete
19. UC	420	0.0052	Steel

KEY:

UC: Upper Compartment
LC: Lower Compartment
DE: Dead-Ended Compartment
IC: Ice Condenser Compartment

14.3.1.2 Major LOCA Analyses Applicable to ANF Fuel

Beginning with Cycle 4, Donald C. Cook Nuclear Plant Unit 2 contains fuel supplied by Exxon Nuclear Company (ENC), now Advanced Nuclear Fuels Corporation (ANF). The following is the current LOCA analysis associated with that fuel.

14.3.1.2.1 Introduction - Large break LOCA/ECCS analyses were performed in 1982[1,2] to support operation of the Cook Nuclear Plant Unit 2 reactor at 3425 MWt with ANF fuel. Reference 1 presented analytical results for a spectrum of postulated large break LOCAs. The limiting break was identified as the 1.0 double ended cold leg guillotine (DECLG) break. Reference 2 presented results for the previously identified limiting break using the EXEM/PWR[3] ECCS models, except GAPEX was used as the fuel performance model in place of RODEX2. The RODEX2[4] code was not approved by the NRC for use in ECCS analyses in 1982. Therefore, the NRC-approved GAPEX[5] code was used to calculate fuel properties at the initialization of the LOCA calculation. The Reference 2 report documented the results of calculations with one and two LPSI pumps operating. At equivalent core peaking limits, higher peak cladding temperatures (PCTs) were calculated in the LOCA analysis when two LPSI pumps were assumed operating. The Reference 2 analysis with two LPSI pumps operating was performed for Cycle 4 operation of Cook Nuclear Plant Unit 2.

This Section 14.3.1.2 documents the results of a LOCA/ECCS analysis to support operation of the Cook Nuclear Plant Unit 2 reactor for Cycle 6 and future cycles at a thermal power rating of 3425 MWt, with up to 10% of the steam generator tubes plugged. Results are also reported for the analysis performed to confirm the axial dependence of permissible limits on power peaking, i.e., the K(Z) curve. The K(Z) curve was established for a maximum total peaking (F^T_Q) of 2.10. The calculations were performed using the EXEM/PWR LOCA/ECCS models, including fuel properties calculated at the start of the LOCA transient with ANF's generically approved RODEX2 code.[4] The quench time, quench velocity and CRF correlations in REFLEX and the heat transfer correlation in TOODEE2 are based on ANF's 17x17 Fuel Cooling Test Facility (FCTF) data[6,7,8].

14.3.1.2.2 Summary - LOCA/ECCS calculations were performed to determine core power peaking limits and to confirm their axial dependence, K(Z), which permit operation of the Cook Nuclear Plant Unit 2 reactor within requirements of 10 CFR 50.46 and Appendix K.[9] The calculations assumed operation:

1) At a thermal power of 3425 MWt;
2) With 10% average steam generator tube plugging; and
3) With the entire core ANF fuel.

The calculations were performed for the previously identified limiting break: the 1.0 DECLG break, with full ECCS flow. The calculations were performed at three exposures using a center peaked cosine axial power shape to determine exposure dependence. The exposures range from 2 MWD/kg to 47 MWD/kg peak rod average burnup.

The axial dependence of the power peaking limit is denoted K(Z) and is defined as $K(Z) = F_Q(Z)/\max F_Q(Z)$, where $F_Q(Z)$ is the maximum peaking allowed at any elevation Z. The top-most segment of the K(Z) curve represents the small break LOCA, which is based upon the current linear heat generation rate (LHGR) limits presented in the Cook Nuclear Plant Unit 2 Technical Specifications.

The confirmation of the axial dependence of the limits on power peaking for the large break LOCA/ECCS analysis is based on three power distributions: a center peaked chopped cosine power distribution, and two conservative top skewed power shapes. The power distributions are analyzed at the limiting exposure, 2 MWD/kg, where the peak stored energy occurs. A summary of these results and the exposure study is presented in Table 14.3.1-11.

This analysis verifies the validity of the existing K(Z) limit reported in Reference 10 with the maximum total power peaking factor (F_Q) raised to 2.10 rather than 2.04 as used in the Reference 10 analysis. The axial power distributions used in this analysis have maximum peaking factor values that are equal to or conservatively greater than the limiting values for plant operation defined by the K(Z) curve shown in Figure 14.3.1-31. The K(Z) curve retains the upper portion of the curve which is defined by LHGR limits for small break LOCA events.

The analysis presented in this Section 14.3.1.2 supports operation of the Cook Nuclear Plant Unit 2 reactor for Cycle 6, and future cycles with ANF fuel, at a total power peaking factor limit (F_Q) of 2.10.

14.3.1.2.3 <u>Limiting Break LOCA Analysis</u> - The analysis described here follows previous LOCA/ECCS analyses performed and documented for Cook Nuclear Plant Unit 2. A spectrum of LOCA breaks was performed and reported in XN-NF-82-35.[1] The limiting LOCA break was determined to be the large double-ended guillotine break of the cold leg or reactor vessel inlet pipe with a discharge coefficient of 1.0 (1.0 DECLG). For this analysis, an additional calculation of the blowdown portion of the LOCA was performed which confirms the limiting break remains the 1.0 DECLG. Reference 2 established that for Cook Nuclear Plant Unit 2, it is more limiting in the LOCA analysis to assume no failure of a LPSI pump. The analysis performed and reported herein considers:

1) That an average of 10% of the steam generator tubes are plugged;
2) That the entire core is composed of ANF fuel; and
3) That both LPSI pumps are operational.

The K(Z) curve is the variation of the limit on power peaking with axial elevation in the core. The allowed power peaking is reduced at the top of the core to offset the effect on peak cladding temperature (PCT) of reduced coolant heat transfer from (1) the short uncovery periods at the top of the core during small break LOCAs, and (2) reduced cooling capacity at the top of the core during the reflood period of large break LOCAs. The analysis model and the results of the analysis are described below.

14.3.1.2.3.1 <u>LOCA Analysis Model</u> - The ANF EXEM/PWR-ECCS evaluation model was used to perform the analyses required. This model[3] consists of the following computer codes: RODEX2[4] code for initial stored energy; RELAP4-EM[3c] for the system blowdown and hot channel blowdown calculations; ICECON[11] for the computation of the ice condenser containment backpressure; REFLEX[3,12] for computation of system reflood; and TOODEE2[3,13,14] for the calculation of final fuel rod heatup. The quench time, quench velocity and CRF correlations in REFLEX and the heat transfer correlation in TOODEE2 are based on ANF's 17x17 fuel cooling test facility (FCTF) data,(6, 7, 8) as opposed to the FLECHT tests. A summary of the LOCA/ECCS models used for this analysis is shown in Table 14.3.1-12.

The Cook Nuclear Plant Unit 2 power plant is a 4-loop Westinghouse pressurized water reactor with ice condenser containment. The reactor coolant system is nodalized into control volumes representing reasonably homogeneous regions, interconnected by flow-paths or "junctions". The system nodalization is depicted in Figure 14.3.1-32. The unbroken loops were modeled as one intact loop with appropriately scaled input. Pump performance curves characteristic of a Westinghouse Series 93A pump were used in the analysis. The transient behavior was determined from the governing conservation equations for mass, energy, and momentum. Energy transport, flow rates, and heat transfer were determined from appropriate correlations.

This analysis used conditions for 10% average plugging of the steam generators' tubes. The plant was modeled assuming asymmetric steam generator tube plugging: an average of 8.33% of the tubes plugged per steam generator in the intact loops, and 15% of the tubes plugged in the broken loop. The larger plugging in the broken loop results in higher PCTs. System input parameters are given in Table 14.3.1-13. Additionally, the core was modeled as an all ANF fueled core.

The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and with decay heating as required by Appendix K of 10 CFR 50. The limits on the axial power distribution (APD) are defined by analyzing power distributions for which the acceptance criteria in 10 CFR 50.46 for LOCA/ECCS analysis are met.

The analysis of the loss-of-coolant accident is performed at 102% of rated power. The core power and other parameters used in the analysis are given in Tables 14.3.1-13 and 14.3.1-14.

14.3.1.2.3.2 LOCA Analysis Results - Table 14.3.1-15 presents the timing and sequence of events as determined for the large break guillotine configuration with a discharge coefficient of 1.0 for full ECCS operation. Table 14.3.1-6 presents the results of the exposure analysis performed with the cosine axial power distribution and the axial dependence study performed with skewed axial power distributions.

The results of this analysis for the cosine axial power shapes show that the BOC conditions produce the most limiting peak cladding temperatures. The skewed power shapes were analyzed conservatively at BOC conditions. Previous calculations have indicated that power shapes which peak at higher axial locations in the core will produce higher calculated peak clad temperatures. For this reason, two power shapes representative of the maximum power peaking in the core top were chosen to be analyzed for the determination of the limit on power peaking versus core height. The most limiting peak cladding temperature calculated is at 2.0 MWD/kg for the EOC axial power distribution.

Three axial power distributions were utilized in the analysis. As shown in Table 14.3.1-14, these axial power distributions result in conformance to the criteria of 10 CFR 50.46. These axial power distributions were used to confirm that the K(Z) curves reported in Reference 10 remain valid with maximum total peaking (F_Q^T) raised to 2.10.

Results of the analyses are given in Figures 14.3.1-33 to 14.3.1-87. Figures 14.3.1-33 to 14.3.1-40 provide plots of key system blowdown parameters versus times. Figures 14.3.1-41 to 14.3.1-70 provide plots of key core responses during the blowdown period. Figures 14.3.1-71 to 14.3.1-74 provide the ECCS flows in the broken and intact loop during the refill period. Figure 14.3.1-75 presents the containment pressure during the LOCA. Figures 14.3.1-76 to 14.3.1-78 present the normalized power during the LOCA for the three exposure cases analyzed. Figures 14.3.1-79 to 14.3.1-82 provide results from the reflood portion of the transient for the most top skewed axial power distribution. The reflood results shown are representative of the other four cases analyzed. Finally, Figures 14.3.1-83 to 14.3.1-87 provide the response of the fuel during the refill and reflood periods of the LOCA transient for the fuel burnup cases investigated.

14.3.1.2.4 Conclusions - For breaks up to and including the double-ended severance of a reactor coolant pipe, the Cook Nuclear Plant Unit 2 emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46 for operation with ANF 17x17 fuel operating in accordance with the LHGR limits noted in Table 14.3.1-11 and Figure 14.3.1-31[19]. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200°F limit.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of zircaloy in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

14.3.1.2.5 RHR Cross-Tie Closure Consideration

The Cook Nuclear Plant Unit 2 safety injection system consists of two RHR pumps, two CCPs, and two HHSI pumps. For both the RHR and the HHSI systems, each pump discharge line splits to deliver flow into two of the four cold legs, and a cross tie connects the two pump discharge lines, enabling one pump to deliver flow to all four cold legs. The licensing basis large break LOCA analysis assumes that flow delivery is available through all four lines from each pump in the safety injection system.

For the RHR system, an open cross-tie line has the potential for dead-heading the weaker of the two pumps as described in NRC Bulletin 88-04[21]. It has been determined that comparable pump dead-heading is precluded for the HHSI and CCP systems at the Cook Nuclear Plant. The corrective action is to normally operate with the RHR cross-tie valves closed [22]. In order to perform maintenance activities, it may at times be desirable to close other safety injection cross-tie valves. When another safety injection cross-tie line is closed, the RHR cross-tie valves will be opened such that only one cross-tie line will be closed at a time. Cross-tie line closure results in the flow from one pump being delivered to only two loops. This results in a reduction in the amount of total safety injection flow delivery to the core during a LOCA event when the single failure of an emergency diesel generator to start following the loss of offsite power is considered. Justification is provided herein demonstrating that for the large break LOCA event, isolating two safety injection lines is acceptable[23]. The limiting case of closing the RHR pump cross-tie line is evaluated for impact at full power. Full HHSI and CCP flow delivery will be available for the large break LOCA event even if two RHR delivery lines are unavailable due to cross-tie closure.

To support operation while one RHR train is out of service with two RCS injection points isolated, ANF investigated the effect of RHR isolation of two injection points on the large break LOCA transient.

The RHR isolation of two injection points case was evaluated with ANF's EXEM/PWR evaluation models to demonstrate conformance with 10 CFR 50.46 criteria.

The total RHR, SI and charging pump flow rate tables input to RELAP4 for this analysis were estimated from a previous single failure analysis (loss of 1 RHR pump) where flow was from 1 RHR pump to 3 intact loops and 1 broken loop. For the two point RHR injection case, the lower total RHR pump flow rate would allow the RHR pump to force more flow out of each of the two points than to each of the four points for the single RHR failure case. The new pump operating point would therefore be at a higher head than the single failure case on its operating curve, with more than half of the total pump flow rate of the single failure case. ANF conservatively assumed the total RHR flow table input to RELAP4 to be one-half of the single failure case values as a function of system or containment back pressure. The RHR flow rate at zero psig system pressure for the two-point RHR injection case is 272.7 lb/sec. It was also assumed that only one charging pump was operating with a maximum flow rate of 35 lb/sec and one SI pump with a maximum flow rate of 56 lb/sec.

Since the primary pressure is at approximately 150 psia at the end-of-bypass, an estimate of the depressurization of the primary system was included in the calculation of accumulator, SI, charging pump, and RHR flow rates in the intact loop during the refill period. The flow rates from the SI, charging pump, and RHR systems to the intact loop are less than the flow rates to the broken loop for the two-point RHR injection case until several seconds into the refill period when the primary system and containment pressures equilibrate.

The change in RHR flow rate to the intact and broken loops was the only difference between the two-point RHR injection and the full ECCS flow analysis described earlier in this section. The limiting break in the break spectrum can be identified by the break size with the highest fuel rod stored energy at the end-of-bypass. The change in RHR flow rate does not affect the system blowdown behavior or the fuel rod stored energy at the end-of-bypass. Therefore, the 1.0 DECLG remains the limiting break size.

The reduced total RHR flow rate for the two-point RHR injection case resulted in a slightly slower rate of filling the lower plenum during the refill period. The beginning-of-core-recovery time increased from 40.31 sec to 40.48 sec. The accumulators emptied at approximately the same time as in the full ECCS flow analysis. In both cases, the downcomer is filled to approximately the same level above the cold leg lip with accumulator flow. However, the two-point RHR injection case resulted in less condensation at the ECCS injection point of steam flowing around the intact loops. The reduced condensation caused more steam to flow out the broken loop stub from the intact loops. The larger steam flow required a larger pressure drop, resulting in a higher system pressure for the two point RHR injection case than for the full ECCS flow case. Although the reduced RHR flow to the intact loop caused some reduction in the downcomer liquid level compared to the full ECCS flow case, the higher absolute system pressure caused a net increase in reflood rate of up to 4% after the time the accumulators and lines emptied (~57 sec) due to increased steam density and a higher pressure drop from the core to the containment through the broken loop steam generator line. Since the full ECCS flow analysis conservatively modeled flows to the containment by including broken loop accumulator, RHR and SI system flows but not including break flows (steam flow out either side of the break), only the change in the broken loop RHR flow rate was included in calculating a revised containment pressure for the two-point RHR injection case. The reduced RHR flow to the containment resulted in a small increase in containment pressure compared to the full ECCS flow case (less condensation of steam in the containment). The higher system pressure had an accompanying higher saturation temperature, which also had an effect on fuel rod plenum temperature and pressure during reflood which can affect the timing and location of fuel rod cladding rupture.

The peak cladding temperature for the BOC power shape was 1823°F and 1790°F for the EOC power shape. Both the BOC and EOC shapes were considered in the two-point RHR injection analysis. For the BOC case, the increase in reflood rate resulted in a 10°F reduction in PCT. The PCT was predicted to be 1813°F as compared to the full ECCS flow case of 1823°F. The location of cladding rupture and time of rupture were unchanged for the BOC shape. For the EOC shape, a change in the timing and location of cladding rupture was observed. A node lower in the core was observed to rupture for the two point RHR injection case. This node was within a few degrees of rupture in the full ECCS flow case, but was turned over prior to rupturing. The ruptured node in the full ECCS flow case became the PCT node in the two-point RHR injection case and caused the PCT to occur later in time. This resulted in an increase in PCT to 1988°F as compared to a PCT of 1790°F for the full ECCS flow analysis. Thus, the EOC shape became the limiting shape for the two-point RHR injection case as opposed to the BOC shape being the limiting shape for the full ECCS flow case. Plots of cladding temperature versus time during the refill and reflood period are shown in Figures 14.3.1-86 through 14.3.1-89 for both the BOC and EOC shapes and for the full ECCS flow and the two-point RHR injection case.

For a condition when the SI and RHR cross-tie lines are closed simultaneously such that two-point SI injection (1 intact loop and 1 broken loop) occurs as well as two-point RHR injection, a small decrease in SI flow to the intact side and a small increase in SI flow to the broken loop side would be observed relative to the two-point RHR injection case. Since the SI flow rate is already a small portion of the total ECCS flow, a small change in the SI flow rate would not be expected to significantly alter the results of the two-point RHR injection analysis and would not preclude meeting the acceptance criteria of 10 CFR 50.46(b).

The results of the evaluation demonstrate that the plant will operate in conformance with 10 CFR 50.46(b) criteria with RHR injection to two points (1 intact loop and 1 broken loop) and SI injection to 4 points (3 intact loops and 1 broken loop).

14.3.1.2.6 REFERENCES FOR SECTION 14.3.1.2

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Table 14.3.1-11 Cook Nuclear Plant Unit 2 LOCA/ECCS Analysis Summary

Case					
o Peak Rod Avg. BU (MWD/kg)	2.0	10.0	47.0	2.0	2.0
o Axial Power Distribution Type	Cosine	Cosine	Cosine	BOC	EOC
o F_Q^T	2.10	2.10	2.10	2.10	2.015
o $F_{\Delta H}^T$	1.55	1.55	1.55	1.68	1.55
o F_Z	1.355	1.355	1.355	1.250	1.30
Peak Clad Temperature					
o Temperature (^oF)	1739	1739	1617	1823	2079
o Time (sec)	67.7	67.7	65.9	67.7	232.1
Hot Rod Burst					
o Time (sec)	58.2	59.0	65.2	56.8	69.2
o Channel Blockage Fraction	.22	.25	.48	.23	.30

Table 14.3.1-11 Cook Nuclear Plant Unit 2 LOCA/ECCS Analysis Summary (Cont.)

<u>Zr-H2O Reaction</u>					
Core Maximum* (%)	<1.0	<1.0	<1.0	<1.0	<1.0
Local Maximum* (%)	1.52	1.37	0.72	2.06	6.02

*Values at 450 sec into LOCA transient.

Table 14.3.1-12 Cook Nuclear Plant Unit 2 LOCA/ECCS Model Summary

	Model/Code	Reference Number
1) Fission Gas Release Model	EXEM/RODEX2	4
2) Stored Energy Model	EXEM/RELAP4	3c, 4
3) Blowdown Model	WREM/RELAP4	15,16,17
4) Containment Model	WREM-II/ICECON	18, 11
5) Clad Swelling and Rupture Model	EXEM/RUPTURE	13
6) Reflood Model		
	REFLEX. Code	
a) Carryout and Quench Correlation	EXEM/FCTF	6, 7, 8
b) Downcomer/Upper Plenum Leakage	Yes(EXEM)	3a
c) Break Model	Guillotine CD = 1.0	3a
d) Core Outlet Enthalpy Model	EXEM	3a
e) Z-Equivalent Model	EXEM	3d
7) Heatup Model		
	TOODEE2.Code	
a) Steam Cooling Model	EXEM	3
b) Heat Transfer Correlation	EXEM/FCTF	6, 7, 8
c) Mixing Vane Multiplier	1.0	-
d) Local Peaking Multiplier	1.0	-
e) Z-Equivalent Model	EXEM	3d
f) Radiation Model	On	3d

Table 14.3.1-13 Cook Nuclear Plant Unit 2 System Input Parameters

Thermal Power, MWt*	3425
Core, MWt	3411
Pump, MWt	14
Primary Coolant Flow, Mlbm/hr	141.3
Primary Coolant Volume, ft^3	11,613
Operating Pressure, psia	2250
Average Coolant Temperature, ^oF	574.1
Reactor Vessel Volume, ft^3	4631
Pressurizer Volume, Total, ft^3	1800
Pressurizer Volume, Liquid, ft^3	1080
Accumulator Volume, Total, ft^3 (each of four)	1350
Accumulator Volume, Liquid, ft^3 (each of four)	950
Accumulator Pressure, psia	636
Steam Generator Heat Transfer Area (Total for 4 loops), ft^2	46,352
Steam Generator Secondary Flow (Total for 4 Loops), lbm/hr	14.6×10^6
Steam Generator Secondary Pressure, psia	794
Reactor Coolant Pump Head ft	277
Reactor Coolant Pump Speed, rpm	1189
Moment of Inertia, lbm-ft^2	82,000
Cold Leg Pipe, I.D. in.	27.5
Hot Leg Pipe, I.D. in.	29.0
Pump Suction Pipe, I.D. In.	31
Fuel Assembly Rod Diameter, in.	0.360
Fuel Assembly Rod Pitch, in.	0.496
Fuel Assembly Pitch, in.	8,466
Fueled (Core) Height, in.	144.0
Fuel Heat Transfer Area, ft^2**	57,327
Fuel Total Flow Area, Bare Rod, ft^2**	53.703
Refueling Water Storage Tank Temperature, ^oF	80
Accumulator Water Temperature, ^oF	120

*Primary Heat Output used in RELAP4-EM Model = 1.02 x 3425 = 3493.5 MWt

**ANF Fuel Parameters

Table 14.3.1-14 1.0 DECLG Break Analysis Parameters

Peak Rod Average Burnup (MWD/kg)	2.0	10.0	47.0	2.0	2.0
Total Core Power (MWt)*	3411	3411	3411	3411	3411
Total Peaking (F_Q^T)	2.10	2.10	2.10	2.10	2.10
Fraction Energy Deposited in Fuel					
Fully Moderated Core	0.974	0.974	0.974	0.974	0.974
Voided Core	0.954	0.954	0.954	0.954	0.954
Axial Distribution Type	COSINE	COSINE	COSINE	BOC	EOC
Peaking					
Axial	1.355	1.355	1.355	1.250	1.30
Enthalpy rise ($F^T{}_H$)	1.55	1.55	1.55	1.680	1.55

Break Type: 1.0 DECLG

Steam generator tube plugging level: 10%

*2% power uncertainty is added to this value in the LOCA analysis.

Table 14.3.1-15 Cook Nuclear Plant Unit 2 1.0 DECLG Break Event times

Event	Time (sec.)
Start	0.00
Break Initiation	0.05
Safety Injection Signal	0.65
Accumulator Injection	
Broken Loop	3.1
Intact Loop	15.2
End of Bypass	24.20
Safety Pump Injection	25.65
Start of Reflood	40.31
Accumulator Empty	
Broken Loop	44.2
Intact Loop	52.7

Table 14.3.1-16 Cook Nuclear Plant Unit 2 1.0 DECLG Break Fuel Response

<u>Calculational Basis</u>					
o Peak Rod Avg. BU (MWD/kg)	2.0	10.0	47.0	2.0	2.0
o Axial Power Distribution Type	Cosine	Cosine	Cosine	BOC	EOC
o F_Q^T	2.10	2.10	2.10	2.10	2.015
o $F_{\Delta H}^T$	1.55	1.55	1.55	1.68	1.55
o F_Z	1.355	1.355	1.355	1.250	1.30
<u>Peak Clad Temperature</u>					
o Temperature (^oF)	1765	1739	1617	1823	2079
o Time (sec)	67.7	67.7	65.9	67.7	232.1

Table 14.3.1-16 Cook Nuclear Plant Unit 2 1.0 DECLG Break Fuel Response
(Continued)

<u>Hot Rod Burst</u>					
o Time (sec)	58.2	59.0	65.2	56.8	69.2
o Channel Blockage Fraction	.22	.25	.48	.23	.30
<u>Zr-H2O Reaction</u>					
o Core Maximum* (%)	<1.0	<1.0	<1.0	<1.0	<1.0
o Local Maximum* (%)	1.52	1.37	0.72	2.06	6.02

*Values at 450 sec into LOCA transient

the limiting break would be shifted from the 4-inch diameter cold leg break to the 3-inch diameter break size. To verify this conclusion, two calculations were performed which assumed break sizes of 3- and 4-inch diameters at the reduced pressure, high temperature initial conditions. Table 14.3.2-8 lists the results of the cross-tie closed cases, which show that with the reduced safety injection flow the 3-inch diameter break is limiting. The sequence of events for these calculations is listed in Table 14.3.2-7. Past small break LOCA analyses that were performed for plants which are similar to Cook Nuclear Plant Unit 2 but have power to safety injection flow rate ratios less than that of Cook Nuclear Plant Unit 2, have shown that an assumed break size of 2 inches did not result in the limiting peak clad temperature. Thus, based on the comparison of power to safety injection flow rate ratio, it was concluded that a 2-inch diameter break would not yield a peak clad temperature more limiting than that of the 3-inch diameter break size. Plots for the 3- and 4-inch break with the high head safety injection cross-tie valves closed are shown in Figures 14.3.2-43 through 14.3.2-50 and 14.3.2-51 through 14.3.2-58, respectively.

NUREG-0737[5], Section II.K.3.31, required plant-specific small break LOCA analysis using an evaluation model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-65[6], generic analyses using NOTRUMP[7,8] were performed and are presented in WCAP-11145[9]. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break location is limiting.

14.3.2.3 REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Water Cooled Nuclear Power Reactors" 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8201, (Proprietary), June 1974.

3. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600, June 1979.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants," NUREG-0611, January 1980.

5. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

6. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31", November 2, 1983.

7. Meyer, P. E., "NOTRUMP - A nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

8. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," WCAP-10054-P-4, August 1985.

9. Rupprecht, S. D., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code;" WCAP-11145-P-A, October 1986.

TABLE 14.3.2-3

TIME SEQUENCE OF EVENTS for CONDITION III EVENTS

Small-break Loss of Coolant Accident

Event	Time (sec) High Temp. High Pressure 4-Inch	Time (sec) Reduced Temp. High Pressure 4-Inch
Break occurs	0	0
Reactor trip signal	11.75	9.85
Safety injection signal	18.95	13.08
Start of safety injection delivery	45.95	40.08
Loop seal venting	333.7	346.9
Loop seal core uncovery	327.4	407.8
Loop seal core recovery	344.0	428.1
Boil-off core uncovery	673.1	664.7
Accumulator injection begins	878.7	874.3
Peak clad temperature occurs	943.6	942.0
Top of core covered	1722.	1658.
SI flow exceeds break flow	1515.	1516.

TABLE 14.3.2-4

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATION

RESULTS

PARAMETER	High Temperature, Reduced Pressure		
Break Size:	3-Inch	4-Inch	6-Inch
Peak clad temperature (F)	1133	1357	959
Elevation (ft)	11.50	11.50	10.50
Zr/H_2O cumulative reaction			
Maximum local (%)	0.07	0.15	0.03
Elevation (ft)	11.50	11.50	10.50
Total core (%)	< 0.3	< 0.3	< 0.3
Rod Burst	None	None	None

CALCULATION:

NSSS Power Mwt 102% of	3588*
Peak Linear Power kw/ft 102% of	12.825
Hot Rod Power Distribution (kw/ft)	See Figure 14.3.2-10
Accumulator Water Volume, cu. ft.	946

*Does not include pump heat.

TABLE 14.3.2-7

TIME SEQUENCE OF EVENTS for CONDITION III EVENTS

Small-break Loss of Coolant Accident

HHSI Cross-tie Valves Closed @ 3413 Mwt

Event	Time (sec) High Temp. Reduced Pressure 3-Inch	High Temp. Reduced Pressure 4-Inch
Break occurs	0	0
Reactor trip signal	10.88	6.74
Safety injection signal	20.36	13.46
Start of safety injection delivery	47.36	40.46
Loop seal venting	611.9	357.1
Loop seal core uncovery	N/A	359.2
Loop seal core recovery	N/A	368.3
Boil-off core uncovery	962.0	611.2
Accumulator injection begins	1566.	839.2
Peak clad temperature occurs	1640.	908.0
Top of core covered	N/A	2356.
SI flow exceeds break flow	1915.	N/A

TABLE 14.3.2-8

SMALL-BREAK LOSS OF COOLANT ACCIDENT CALCULATION

Results HHSI Cross Tie Valves Closed @ 3413 Mwt

PARAMETER	High Temp. Reduced Pressure 3-Inch	High Temp. Reduced Pressure 4-Inch
Peak clad temperature (F)	2124	1530
Elevation (ft)	12.00	11.50
Zr/H_2O cumulative reaction		
Maximum local (%)	8.64	0.37
Elevation (ft)	12.00	11.50
Total core (%)	< 0.3	< 0.3
Rod Burst	None	None

CALCULATION:

NSSS Power MWt 102% of	3413*
Peak Linear Power kw/ft 102% of	12.756
Hot Rod Power Distribution (kw/ft)	See Figure 14.3.2-59
Accumulator Water Volume, cu. ft.	946

*Does not include pump heat.

14.3.4 CONTAINMENT INTEGRITY EVALUATION

14.3.4.1 General Description of Containment Pressure Analysis

The time history of conditions within an ice condenser containment during a postulated loss-of-coolant accident can be divided into two periods for calculational purposes:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs in approximately 10 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

During the first few seconds of the blowdown period following a large rupture of the Reactor Coolant System, containment conditions are characterized by rapid pressure and temperature transients. To calculate these transients, a detailed spatial and short time increment analysis is necessary. This analysis is performed with the TMD code with the calculation time of interest extending up to a few seconds following the accident initiation.

Physically, tests at the ice condenser Waltz Mill test facility have shown that the blowdown phase represents that period of time in which the lower compartment air, and a portion of the ice condenser air, are displaced and compressed into the upper compartment and the remainder of the ice condenser. The containment pressure at or near the end of blowdown is governed by this air compression process.

Containment pressure during the post blowdown phase of the accident is calculated with the LOTIC Code, which models the containment structural heat sinks and containment safeguards systems.

14.3.4.2 Long Term Containment Pressure Analysis

Early in the ice condenser development program it was recognized that there was a need for modeling of long term ice condenser containment performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO Code) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC Code.[1]

The model of the containment consists of five distinct control volumes, as follows: the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartments. The ice condenser control volume with unmelted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the Waltz Mill ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the Reactor Coolant System, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas and the thermodynamic properties of steam are taken from the ASME steam tables.

Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Donald C. Cook Units 1 and 2 containments:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two residual heat removal pumps and one of two residual heat removal heat exchangers with cross-tie valves closed providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. 2.11×10^6 pounds of ice initially in the ice condenser which is at 14°F. This temperature assumption maximizes the air mass in the ice condenser and is conservative with respect to the 27°F Technical Specification limit.

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 14.3.4.5 were used.

4. Blowdown and post blowdown ice condenser drain temperatures of 190°F and 130°F are used. (These numbers are based on Reference 2.)

5. Nitrogen from the accumulators in the amount of 4510 pounds is included in the calculations.

6. Service water temperature of 81°F is used for the spray heat exchanger and the component cooling heat exchanger. (An evaluation of the acceptability of service water temperatures of up to 85°F has also been made as discussed in Reference 17).

7. The air return fan is effective 10 minutes after the transient is initiated.

8. No maldistribution of steam flow to the ice bed is assumed.

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are temperatures of 60°F in the upper, 60°F in the lower, 60°F in the dead ended and 14°F in the ice bed volumes. All volumes are at a pressure of 0.3 psig and 15 percent relative humidity, with the exception of the ice bed, which is at 100 percent relative humidity.

11. A spray pump flow of 1900 gpm is used for the upper compartment.

12. A residual spray (2000 gpm) is used after recirculation is initiated, but before 50 minutes has elapsed after the accident. The residual heat removal pump and spray pump take suction from the sump during recirculation. (Recirculation switchover is initiated at 2295 seconds.)

13. Containment structural heat sink data are assumed with conservatively low heat transfer rates, and may be found in Table 14.3.4-2.

14. The operation of one containment spray heat exchanger (UA = 2.92×10^6 BTU/hr-°F) for containment cooling and the operation of one residual heat removal heat exchanger (UA = 2.16×10^6 BTU/hr-°F) for core cooling. The component cooling heat exchanger was modeled at 3.87×10^6 BTU/hr-°F.

15. The air return fan returns air at a rate of 39,000 cfm from the upper to lower compartment.

16. An active sump volume of 40,600 ft^3 is used.

17. The refueling water storage tank is at a temperature of 100°F.

18. 102% of 3425 MWt power is used in the calculation.

19. Credit is taken for subcooling of the ECC water from the RHR heat exchanger.

20. Essential service water flow to the containment spray heat exchanger was modeled as 2400 gpm. The essential service water flow to the component cooling heat exchanger was modeled as 5500 gpm. The component cooling flow to the RHR heat exchanger was modeled as 5000 gpm.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 14.3.4-1, Containment pressure transient.

Figure 14.3.4-2, Upper compartment temperature transients.

Figure 14.3.4-3, Lower compartment temperature transients.

Figure 14.3.4-4, Active and inactive sump temperature transient.

Figure 14.3.4-5, Ice melt transient.

In addition, Table 14.3.4-1 gives energy accountings at various points in the transient.

The analysis results show that the maximum calculated containment pressure is 11.89 psig, for the double-ended pump suction minimum safeguards case. This pressure peak occurs at approximately 6955 seconds, with ice bed meltout at approximately 4443 seconds. An elevation of the containment pressure for a service water temperature of 85°F showed that the maximum containment pressure would also be less than the design pressure of 12 psig.

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference form accounts for transient conduction into and out of the node and temperature rise of the node. Table 14.3.4-2 is a summary of the containment structural heat sinks used in the analysis. The material property data used are found in Table 14.3.4-3.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami. An explanation of the manner of application is given in Reference (3).

When applying the Tagami correlation, a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to 72 $BTU/hr\text{-}ft^2$. This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

14.3.4.3. Short Term Blowdown Analysis

TMD Code - Short Term Analysis

1. Introduction

The basic performance of the Ice Condenser Reactor Containment System has been demonstrated for a wide range of conditions by the Waltz Mill Ice Condenser Test Program.[2] These results have clearly shown the capability and reliability of the ice condenser concept to limit the containment pressure rise subsequent to a hypothetical loss-of-coolant accident.

To supplement this experimental proof of performance, a mathematical model has been developed to simulate the ice condenser pressure transients. This model, encoded as computer program TMD (Transient Mass Distribution), provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. This model is used to compute pressure differences on various structures within the containment as well as the distribution of steam flow as the air is displaced from the lower compartment. Although the TMD Code can calculate the entire blowdown transient, the peak pressure differences on various structures occur within the first few seconds of the transient.

2. Analytical Models - (No Entrainment)

The mathematical modeling in TMD is similar to that of the SATAN blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum, and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are given in Reference (4).

The moisture entrainment modifications to the TMD Code are discussed in detail in Reference (4). These modifications comprise incorporating the additional entrainment effects into the momentum and energy equations.

As part of the review of the TMD Code, additional effects are considered. Changes to the analytical model required for these studies are described in Reference (4).

These studies consist of:

a. Spatial acceleration effects in ice bed.
b. Liquid entrainment in ice beds.
c. Upper limit on sonic velocity.
d. Variable ice bed loss coefficient.
e. Variable door response.
f. Wave propagation effects.

Experimental Verification

The performance of the TMD Code was verified against the 1/24 scale air tests and the 1968 Waltz Mill tests. For the 1/24 scale model the TMD Code was utilized to calculate flow rates to compare against experimental results. The effect of increased nodalization was also evaluated. The

Waltz Mill test comparisons involved a reexamination of test data. In conducting the reanalyses, representation of the 1968 Waltz Mill test was reviewed with regard to parameters such as loss coefficients and blowdown time history. The details of this information are given in Reference (4).

The Waltz Mill Ice Condenser Blowdown Test Facility was reactivated in 1973 to verify the ice condenser performance with the following redesigned plant hardware scaled to the test configuration:

1. Perforated metal ice baskets and new design couplings.

2. Lattice frames sized to provide the correct loss coefficient relative to plant design.

3. Lower support beamed structure and turning vanes sized to provide the correct turning loss relative to the plant design.

4. No ice baskets in the lower ice condenser plenum opposite the inlet doors.

The result of these tests was to confirm that conclusions derived from previous Waltz Mill tests had not been significantly changed by the redesign of plant hardware. The TMD Code has, as a result of the 1973 test series, been modified to match ice bed heat transfer performance. Detailed information on the 1973 Waltz Mill test series is found in Reference (5).

A number of analyses have been performed to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The analyses were performed using the following assumptions and correlations:

1. Flow was limited by the unaugmented critical flow correlation.

2. The TMD variable volume door model, which accounts for changes in the volumes of TMD elements as the door opens, was implemented.

3. The heat transfer calculation used was based on performance during the 1973-1974 Waltz Mill test series. A higher value of the ELJAC parameter has been used and an upper bound on calculated heat transfer coefficients has been imposed (see Reference [5]).

4. One hundred percent moisture entrainment was assumed.

Application to Plant Design

The Donald C. Cook Unit 2 containment has been divided into 45 elements or compartments as shown in Figures 14.3.4-6 through 14.3.4-9. The interconnection between containment elements in the TMD Code is shown schematically in Figure 14.3.4-10. Flow resistance and inertia are lumped together in the flow paths connecting the elements shown. The division of the lower compartment into 6 volumes occurs at the points of greatest flow resistance, i.e., the four steam generators, pressurizer, and refueling cavity.

Each of these lower compartment sections delivers flow through doors into a section behind the doors and below the ice bed. Each vertical section of the ice bed is, in turn, divided into elements. The upper plenum between the top of the ice bed and the top deck doors is represented by another element. Thus, a total of thirty elements (elements 7 through 24 and 34 through 45) are used to simulate the ice condenser. The six elements at the top of the ice bed between bed and top deck doors deliver to element number 25, the upper compartment. Note that cross flow in the ice bed is not accounted for in the analysis; this yields the most conservative results for the particular calculations described herein. The upper reactor cavity (element 33) is connected to the lower compartment volumes and provides cross flow for pressure equalization of the lower compartments. The less active compartments, called dead ended compartments (elements 26, 28, 29, 30 and 32), and the fan and accumulator compartments (elements 27 and 31) outside the crane wall are pressurized by ventilation openings through the crane wall into the fan compartments.

For each element in the TMD network the volume, initial pressure, and initial temperature conditions are specified. The ice condenser elements have additional inputs of mass of ice, heat transfer area, and condensate layer length. For each flow path between elements flow resistance is specified as a loss coefficient "K" or a friction loss " f_D " or a combination of the two based on the flow area specified between elements. Friction factor, friction factor length, and hydraulic diameter are specified for the friction loss. The code input for each flow path is the flow path length used in the momentum equation. In addition, the ice condenser loss coefficients have been based on the 1/4 scale tests representative of the current ice condenser geometry. The loss coefficient is based on removal of door port flow restrictors. To better represent short term transient effects, the opening characteristics of the lower, intermediate, and top deck ice condenser doors have also been modeled in the TMD Code. An initial containment pressure of 0.3 psig was assumed in the analysis. Initial containment pressure variation about the assumed 0.3 psig value had only a slight effect on the initial pressure peak and the compression ratio pressure peak. (Table 14.3.4-33* gives the flow path input data used.)

The reactor coolant blowdown rates used in these cases are based on the SATAN analysis of a double ended rupture of either a hot or cold leg reactor coolant pipe utilizing a discharge coefficient of 1.0. The blowdown analysis has been presented in Section 4.0 of Appendix N of the original FSAR.

Results of the analysis for Donald C. Cook Unit 2 are presented in Tables 14.3.4-4 through 14.3.4-7. The peak pressures and peak differential pressures occur within the first 3.0 seconds of the blowdown.

*These inputs are generic type inputs applicable to Donald C. Cook Nuclear Plant, Units 1 and 2. Later analyses were performed with Unit 2 specific input.

A number of analyses have been performed using 100 percent moisture entrainment to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The maximum peak pressure and differential pressure for all cases have been determined for each compartment element. Figure 14.3.4-11 is representative of the upper and lower compartment pressure transients that result from a hypothetical double ended rupture of a reactor coolant pipe for the worst possible location in the lower compartment of the containment, a hot leg break (DEHL) in element 6.

Initial Pressure Peaks

Table 14.3.4-4 presents the maximum calculated peak pressure in each of the lower compartment elements resulting from a hot leg and cold leg pipe break. Generally, a pipe break within an element results in the maximum peak pressure for the element. A break located in element 1 or 6 results in the highest pressure peak (14.4 psig) in that element of the lower compartment, because of the limited vent area from these locations in the lower compartment; a break in element 3 results in a peak lower compartment pressure of only 9.2 psig. It should be noted that these pressures exist only inside the crane wall and not on the containment shell itself.

Table 14.3.4-5 presents the maximum calculated peak pressure in each of the ice condenser compartment elements resulting from any pipe break location. The maximum value calculated anywhere in the ice condenser compartment is 10.8 psig, and this value is also conservative because of blowdown rate and heat transfer assumptions.

Table 14.3.4-6 presents the maximum calculated differential pressure across the operating deck (divider barrier) between the lower compartment elements and the upper compartment. These values are also approximately the same as the maximum calculated differential pressure across the lower crane wall between the lower compartment elements and the dead ended volumes surrounding the lower compartment. Based on the highest of these values, the maximum calculated differential pressure across the operating deck or the lower crane wall is 14.1 psi.

Table 14.3.4-7 presents the maximum calculated differential pressures across the upper crane wall between the ice condenser elements and the upper compartment. Because the steam generator enclosures are common with sections of the upper crane wall, each section of the crane wall is designed for different loadings. Based on the values shown in Table 14.3.4-7 the end sections of the crane wall enclosing elements 7-8-9 experience a maximum differential pressure of 8.2 psig. The sections in common with the steam generator enclosures are designed for the higher pressure conditions inside the enclosure which would occur if a steam generator steam line breaks within the enclosure; therefore, the differential pressures in Table 14.3.4-7 for elements 10-11-12 and 19-20-21 are not the limiting values for those locations. The remaining sections of the crane wall, enclosing elements 13-14-15 and 16-17-18, experience a maximum differential pressure of 6.0 psig and 5.9 psig, respectively.

Careful consideration is given to the design of those containment internal structures where a pipe break could cause localized compartment pressure to be higher than for the design basis double ended reactor coolant pipe rupture. These subcompartments include the steam generator enclosure, fan room, pressurizer enclosure, and upper and lower reactor cavity. The results of these subcompartment analyses are discussed in Subsection 14.3.4.7.

Detailed TMD Results - Loss-of-Coolant Accident

The TMD analysis of the containment response to assumed RCS pipe ruptures in the RCS loop compartments using non-augmented critical flow, the ice condenser heat transfer coefficient determined by the 1974 Waltz Mill full scale test program, and the compressibility factor Y for the subcritical flow correlation are presented in this Section.

The TMD input is presented in Table 14.3.4-33. Mass and energy transients for a double ended cold leg guillotine (DECLG) break in a loop compartment are presented in Figures 14.3.4-21 through 14.3.4-24. Mass and energy transients for a doubled ended hot leg guillotine (DEHLG) break in a loop compartment are presented in Figures 14.3.4-25 and 14.3.4-26. Compartment pressure transients for a DECLG in compartment No. 1 appear in Figures 14.3.4-27 through 14.3.4-71.

Compartment pressure transients for a DEHLG in compartment No. 1 appear in Figures 14.3.4-72 through 14.3.4-116. Compartment pressure transients for the additional loop compartment breaks are presented as described below:

Break & Location	Figures
DEHLG in Compartment No. 2	14.3.4-117 through 14.3.4-161
DEHLG in Compartment No. 3	14.3.4-162 through 14.3.4-206
DEHLG in Compartment No. 4	14.3.4-207 through 14.3.4-251
DEHLG in Compartment No. 5	14.3.4-252 through 14.3.4-296
DEHLG in Compartment No. 6	14.3.4-297 through 14.3.4-341
DECLG in Compartment No. 3	14.3.4-342 through 14.3.4-386
DECLG in Compartment No. 4	14.3.4-387 through 14.3.4-431
DECLG in Compartment No. 6	14.3.4-432 through 14.3.4-476

Analyses and evaluations of the loop compartments were performed to support operation for reduced temperature and pressure operation. This analysis is presented in Unit 1 FSAR Section 14.3.4.9.4.

Sensitivity Studies

A series of TMD runs for Cook Nuclear Plant Unit 2 investigated the sensitivity of peak pressures to variations in individual input parameters for the design basis blowdown rate and 100 percent entrainment. This analysis used a DEHL break in element 6 of Cook Nuclear Plant Unit 2. Table 14.3.4-8 gives these results.

Choked Flow Characteristics

The data in Figure 14.3.4-12 illustrate the behavior of mass flow rate as a function of upstream and downstream pressures, including the effects of flow choking. The upper plot shows mass flow rate as a function of upstream pressure for various assumed values of downstream pressure. For zero back pressure (P_D = 0), the entire curve represents choked flow conditions with the flow rate approximately proportional to upstream pressure, P_U. For higher back pressure, the flow rates are lower until the upstream pressure is high enough to provide choked flow.

14.3.4.4 Compression Ratio Analysis

As blowdown continues following the initial pressure peak from a double-ended cold leg break, the pressure in the lower compartment again increases, reaching a peak at or before the end of blowdown. The pressure in the upper compartment continues to rise from beginning of blowdown and reaches a peak which is approximately equal to the lower compartment pressure. After blowdown is complete, the steam in the lower compartment continues to flow through the doors into the ice bed compartment and is condensed.

The primary factor in producing this upper containment pressure peak, and, therefore, in determining design pressure, is the displacement of air from the lower compartment into the upper compartment. The ice condenser quite effectively performs its function of condensing virtually all the steam that enters the ice beds. Essentially, the only source of steam entering the upper containment is from leakage through the drain holes and other leakage around crack openings in hatches in the operating deck, which separate the lower and upper portions of the containment building.

A method of analysis of the compression peak pressure was developed based on the results of full scale section tests. This method consists of the calculation of the air mass compression ratio, the polytropic exponent for the compression process, and the effect of steam bypass through the operating deck on this compression.

The compression peak pressure in the upper compartment for the Donald C. Cook Units 1 and 2 design is calculated to be 7.25 psig. This compression pressure includes the effect of a pressure increase of 0.4 psi from steam bypass. The nitrogen partial pressure from the accumulators is not included since this nitrogen is not added to the containment until after the compression peak pressure has been reduced, which is after blowdown is completed. This nitrogen is considered in the analysis of pressure decay following blowdown as presented in the long term performance analysis using the LOTIC Code. In the following sections, a discussion of the major parameters affecting the compression peak will be discussed. Specifically they are: air compression, steam bypass, blowdown rate, and blowdown energy.

Air Compression Process Description

The volumes of the various containment compartments determine directly the air volume compression ratio. This is basically the ratio of the total active containment air volume to the compressed air volume during blowdown. During blowdown, air is displaced from the lower compartment and compressed into the ice condenser beds and into the upper containment above the operating deck. It is this air compression process which primarily determines the peak in containment pressure following the initial blowdown release. A peak compression pressure of 7.25 psig is based on the Cook Nuclear Plant Units 1 and 2 compartment volumes shown in Table 14.3.4-9.

Methods of Calculation and Results

The actual Waltz Mill test compression ratios were found by performing air mass balances before the blowdown and at the time of the compression peak pressure, using the results of three full scale special section tests. These three tests were conducted with an energy input representative of the plant design.

In the calculation of the mass balance for the ice condenser, the compartment is divided into two subvolumes; one volume representing the flow channels and one volume representing the ice baskets. The flow channel volume is further divided into four subvolumes, and the partial air pressure and mass in each subvolume are found from thermocouple readings, assuming that the air is saturated with steam at the measured temperature. From these results, the average temperature of the air in the ice condenser compartment is found, and the volume occupied by the air at the total condenser pressure is found from the equation of state as follows:

$$V = \frac{MaRaTa}{P} \qquad (14.3.4\text{-}1)$$

Where:

- Va = Volume of ice condenser occupied by air (ft^3).
- Ma = Mass of air in ice condenser compartment (lb).
- Ta = Average temperature of air in ice condenser (^oF).
- P = Total ice condenser pressure (lb/ft^2).

The partial pressure and mass of air in the lower compartment are found by averaging the temperatures indicated by the thermocouples during the test located in that compartment and assuming saturation conditions. For these three tests, it was found that the partial pressure, and hence the mass of air in the lower compartment, were zero at the time of the compression peak pressure.

The actual Waltz Mill test compression ratio is then found from the following:

$$C_r = \frac{V_1 + V_2 + V_3}{V_3 + V_{a\ 2}} \qquad (14.3.4\text{-}2)$$

Where:

- V_1 = Lower compartment volume (ft^3).
- V_2 = Ice condenser compartment volume (ft^3).
- V_3 = Upper compartment volume (ft^3).

The polytropic exponent for these tests is then found from the measured compression pressure and the compression ratio calculated above. Also considered is the pressure increase that results from the leakage of steam through the deck into the upper compartment.

The compression peak pressure in the upper compartment for the tests for containment design is then given by:

$$P = P_0\ (C_r)^n + \Delta P_{deck} \qquad (14.3.4\text{-}3)$$

Where:

P_0 = Initial pressure (psia).

P = Compression peak pressure (psia).

C_r = Volume compression ratio.

n = Polytropic exponent.

ΔP_{deck} = Pressure increase caused by deck leakage (psi).

Using the method of calculation described above, the compression ratio was calculated for the three full scale section tests. From the results of the air mass balances, it was found that air occupied 0.645 of the ice condenser compartment volume at the time of peak compression, or

$$V_{a\,2} = 0.645\, V_2 \qquad (14.3.4\text{-}4)$$

The final compression volume includes the volume of the upper compartment as well as part of the volume of air in the ice condenser. The results of the full scale section tests (Figure 14.3.4-13) show a variation in steam partial pressure from 100% near the bottom of the ice condenser to essentially zero near the top. The thermocouples and pressure detectors confirm that at the time when the compression peak pressure is reached steam occupies less than half of the volume of the ice condenser. The analytical model used in defining the containment pressure peak uses the upper compartment volume plus 64.5 percent of the ice condenser air volume as the final volume. This 64.5 percent value was determined from appropriate test results.

The calculated volume compression ratios are shown in Figure 14.3.4-14, along with the compression peak pressures for these tests. The compression peak pressure is determined from the measured pressure, after accounting for the deck leakage contribution. From the results shown in Figure 14.3.4-14, the polytropic exponent for these tests is found to be 1.13.

Plant Case

For the Donald C. Cook Unit 2 design the volume compression ratio, not accounting for the dead ended volume effect, is calculated using Equation 14.3.4-2 and Table 14.3.4-9 as:

$$C_r = \frac{1,179,636}{745,896 + 0.645 \times 126,940}$$

$C_r = 1.42$

The peak compression pressure, based on an initial containment pressure of 15.0 psia, is then given by Equation 14.3.4-3 as:

$$P_3 = 15.0\ (1.42)^{1.13} + 0.4$$

$$P_3 = 22.7 \text{ psia or } 8.0 \text{ psig}$$

This peak compression pressure includes a pressure increase of 0.4 psi from steam bypass through the deck.

Effect of Steam Bypass

The method of analysis used to obtain the maximum allowable deck leakage capacity as a function of the primary system break size is presented below. This analysis demonstrates the margin between the design leakage of 5 ft^2 and the maximum allowable.

During the blowdown transient, steam and air will flow through the ice condenser doors and also through the deck bypass area into the upper compartment. For the containment the bypass area is composed of two parts, a known leakage area of 2.2 ft^2 with a geometric loss coefficient of 1.5 through the deck drainage holes location at the bottom of the refueling canal, and an undefined deck leakage area with a conservatively small loss coefficient of 2.5. A resistance network similar to that used in TMD is used to represent 6 lower compartment volumes, each with a representative portion of the deck leakage and the lower inlet door flow resistance adjacent to the lower compartment element. The inlet door flow resistance and flow area are calculated for small breaks that would only partially open these doors.

The coolant blowdown rate as a function of time is used with this flow network to calculate the differential pressures on the lower inlet doors and across the operating deck. The resultant deck leakage rate and integrated steam leakage into the upper compartment are then calculated. The lower inlet doors are initially held shut by the cold head of air behind the doors (approximately one pound per square foot). The initial blowdown from a small break opens the doors and removes the cold head on the doors. With the door differential pressure removed the door position is slightly open. An additional pressure differential of one pound per square foot is then sufficient to fully open the doors. The nominal door opening characteristic as shown in Appendix N of the Original FSAR were used in the analysis.

One analysis conservatively assumed that flow through the postulated leakage paths is pure steam. During the actual blowdown transient, steam and air representative of the lower compartment mixture would leak through the holes; thus less steam would enter the upper compartment. If flow were considered to be a mixture of liquid and vapor, the total leakage mass would increase but the steam flow rate would decrease. The analysis also assumed that no condensing of the flow occurs due to structural heat sinks. The peak air compression in the upper compartment for the various break sizes is assumed with steam mass added to this value to obtain the total containment pressure. Air compression for the various break sizes is obtained from previous full scale section tests conducted at Waltz Mill.

The allowable leakage area for the following Reactor Coolant System break sizes was determined: DE, 0.6 DE, 3 ft^2, 8 inch diameter, 6 inch diameter, 2.5 inch diameter, and 0.5 inch diameter. For break sizes 3 ft^2 and above a series of deck leakage sensitivity studies was made to establish the total steam leakage to the upper compartment over the blowdown transient. This steam was added to the air in the upper compartment to establish a peak pressure. Air and steam were assumed to be in thermal equilibrium, with the air partial pressure increased over the air compression value to account for heating effects. For these breaks, sprays were neglected. Reduction in compression ratio by return of air to the lower compartment was conservatively neglected. The results of this analysis are shown in Table 14.3.4-10. This analysis is confirmed by Waltz Mill tests conducted with various deck leaks equivalent to over 50 ft^2 of deck leakage for the double ended blowdown rate.

For breaks 8 inches in diameter and smaller, the effect of containment sprays was included. The method used is as follows: For each time step of the blowdown the amount of steam leaking into the upper compartment was calculated to obtain the steam mass in the upper compartment. This steam was mixed with the air in the upper compartment, assuming thermal equilibrium with air. The air partial pressure was increased to account for air heating effects. After sprays were initiated, the pressure was calculated based on the rate of accumulation of steam in the upper compartment. Reduction in pressure due to operation of the air recirculation fans has been conservatively neglected.

This analysis was conducted for the 8 inch, 6 inch and 2-1/2 inch break sizes assuming two spray pumps were operating (4000 gpm at 80°F). As shown in Table 14.3.4-10, the 8 inch break is the limiting case for this range of break sizes although the 0.6 DE is the limiting case for the entire spectrum of break sizes. With one spray pump operating (2000 gpm at 80°F) the limiting case for the entire spectrum of break sizes is the 8 inch case and results in an allowable deck leakage area of approximately 35 ft^2.

A second, more realistic, method was used to analyze this limiting case. This analysis assumed a 30 percent air, 70 percent steam mixture flowing through the deck leakage area. This is conservative considering the amount of air in the lower compartment during this portion of the transient. Operation of the deck fan would increase the air content of the lower compartment, thus increasing the allowable deck leakage area. Based on the LOTIC Code analysis a structural heat removal rate of over 8000 BTU/sec from the upper compartment is indicated. Therefore a steam condensation rate of 8 lb/sec was used for the upper compartment. The results indicated that with one spray pump operating and a deck leakage area of 56 ft^2, the peak containment pressure will be below design for the 8 inch case.

The 1/2 inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, either the lower compartment or the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area and that which can be tolerated without exceeding containment design pressure.

14.3.4.5 LONG TERM MASS AND ENERGY RELEASE ANALYSIS

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS reaches initial equilibration with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

Break Size and Location

Generic studies have been performed with respect to the effect on the LOCA mass and energy releases relative to postulated break size. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture.

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed is the double-ended pump suction guillotine break (10.48 ft^2). Pump suction break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA. The following information provides a discussion on each break location.

The double-ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore the reflood (and subsequent post-reflood) releases are not calculated for a hot leg break. The mass and energy releases for the hot leg break have not been included in the scope of this containment integrity analysis because for this break only the blowdown phase of the transient is of any significance. Since there are no reflood and post-reflood phases to consider, the limiting peak pressure calculated would be the compression peak pressure and not the peak pressure following ice bed meltout.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient is, in general, less limiting than the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this analysis.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the reactor coolant system in calculating the releases to containment. This break location has been determined to be the limiting break for all ice condenser plants. The analysis of this break location for Cook Nuclear Plant as the limiting break is consistent with other ice condenser plants.

In summary then, the analysis of the limiting break location for an ice condenser containment has been performed. The double-ended pump suction guillotine break has historically been considered to be the limiting break location, by virtue of its consideration of all energy sources present in the RCS. This break location provides a mechanism for the release of the available energy in the RCS, including both the broken and intact loop steam generators. Inclusion of these energy sources conservatively results in the maximum amount of ice being melted in the event of a LOCA.

Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the double-ended pump suction (DEPS) break. For the DEPS results presented in this section, an inherent assumption in the generation of the mass and energy releases is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system.

Two cases have been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, thereby minimizing the safety injection flow. This assumption also results in the loss of one containment spray pump. An additional conservatism has been included in that the RHR cross-tie valve has been assumed to be closed during safety injection. Closure of the RHR cross-tie valve has been considered over the closure of the HHSI cross-tie valve because it results in a greater reduction of safety injection flow. Thus, the consideration of having the RHR cross-tie valve closed will result in a more limiting analysis than having the cross-tie valve open. The analysis further considers the safety injection pump head curves to be degraded by 10%. This results in the greatest SI flow reduction possible for the minimum safeguards case.

For the case of maximum safeguards, no failure is postulated to occur in the generation of the mass and energy release data. This results in the maximum safety injection flow possible. The RHR cross-tie valve is assumed to be open in this instance, with no pump head degradation considered. The single failure considered is the failure of a containment spray pump. The analysis of both maximum and minimum safeguards cases ensure that the effect of all credible single failures is bounded.

MASS AND ENERGY RELEASE DATA

Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient, and is the same as that used for the February 1978 ECCS calculation. The methodology for the use of this model is described in Reference 6.

Tables 14.3.4-11 and 14.3.4-12 present the calculated mass and energy releases for the blowdown phase of the break analyzed.

The mass and energy releases for the double-ended pump suction break, given in Tables 14.3.4-11 and 14.3.4-12 terminate 29.0 seconds after the postulated accident. Since safety injection is not considered during the blowdown phase, these releases are the same for both minimum and maximum safety injection.

Reflood and Mass Energy Release Data

The WREFLOOD code is used for computing the reflood transient, and is a modified version of that used in the ECCS calculation. The methodology for the use of this model is described in Reference 6.

An exception to the mass and energy evaluation model described in Reference 6 is taken, in that steam/water mixing in the broken loop has been included in this analysis. This assumption is justified and is supported by test data, and is summarized as follows.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered.

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 16), which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 6. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis is the double-ended pump suction break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECC injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECC injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECC injection nozzle. A description of the test and test results is contained in References 6 and 16.

Tables 14.3.4-13 and 14.3.4-14 present the calculated mass and energy releases for the reflood phase of the double-ended pump suction break with minimum and maximum safety injection, respectively.

Transients of the principal parameters during reflood are given in Tables 14.3.4-15 and 14.3.4-16 for the double-ended pump suction break with minimum and maximum safety injection.

<u>Post-Reflood Mass and Energy Release Data</u>

The FROTH code is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference 6. The mass and energy release rates calculated by FROTH are used in the containment analysis to the time of steam generator cooldown.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference 15 and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Tables 14.3.4-17 and 14.3.4-18 present the two phase (froth) mass and energy release data for the double-ended pump suction break with minimum and maximum safety injection. Data for these tables are terminated at the end of froth time, after which the LOTIC code performs its own core boiloff calculation.

SOURCES OF MASS AND ENERGY

The sources of mass and energy considered in the LOCA mass and energy release analysis are given in Tables 14.3.4-19 and 14.3.4-20 for the double-ended pump suction break with minimum and maximum safety injection, respectively.

The mass sources are the reactor coolant system, accumulators, and pumped safety injection. The energy sources include:

1. Reactor coolant system water

2. Accumulator water

3. Pumped injection water

4. Decay Heat

5. Core stored energy

6. Reactor coolant system metal

7. Steam generator metal

8. Steam generator secondary energy

9. Secondary transfer of energy (feedwater into and steam out of the steam generator secondary).

In the mass and energy release data presented, no zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the zirc-water reaction heat to be of any significance.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in the analysis.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)

2. End of blowdown time

3. End of refill time

4. End of reflood time

5. Time of steam generator depressurizations

6. End of analysis

The methods and assumptions used to release the various energy sources are given in Reference 6, except as noted in the reflood mass and energy section, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

SIGNIFICANT MODELING ASSUMPTIONS

The following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

1. Maximum expected operating temperature of the reactor coolant system

2. Allowance in temperature for instrument error and dead band (+5^{o}F)

3. Margin in volume of +3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty).

4. Allowance for calorimetric error (+2 percent of 3425 MWt).

5. Conservatively modified coefficients of heat transfer.

6. Allowance in core stored energy for effect of fuel densification.

7. Margin in core stored energy (+15 percent).

14.3.4.6 Containment Analysis for Steam Line Break

A series of steamline breaks were analyzed to determine the most severe break condition for the containment temperature and pressure response. The assumptions on the initial conditions are taken to maximize the mass and total energy released. The higher primary temperatures along with the higher uprated power level associated with the Unit 2 rerating parameters are conservative for the mass/energy release calculations. The upper bound temperature case of the potential Unit 2 rerating of Table 14.3.4-38 was used.

PIPE BREAK BLOWDOWNS

1. Spectra and Assumptions

A bounding analysis was performed to address the range of conditions possible for the Unit 1 rerating and the future Unit 2 rerating. The following assumptions were used in the analysis:

a. Double ended pipe breaks were assumed to occur at the nozzle of one steam generator and also downstream of the flow restrictor. Split pipe ruptures were assumed to occur at the nozzle of one steam generator.

b. The blowdown is assumed to be dry saturated steam.

c. The Unit 1 steamline break protection system design is assumed and is conservative for the calculation of mass/energy releases with respect to the Unit 2 steamline break protection design. However, credit was not taken for safeguards actuation on high steam line differential pressure or high-high steam flow coincident with low-low Tavg.

d. Steamline isolation is assumed complete 11.0 seconds after the setpoint is reached for either high-high steam flow coincident with low steam pressure or hi-hi containment pressure. The isolation time allows 8 seconds for valve closure plus 3 seconds for electronic delays and signal processing. The total delay time for steamline isolation of 11 seconds is assumed to support the relaxation of the main steam isolation valve (MSIV) closure time.

e. 4.6 and 1.4 square foot double ended pipe breaks were evaluated at 102, 70, 30 and zero percent power levels.

f. Split pipe ruptures were evaluated at 0.86 square foot 102 percent power, 0.908 square foot 70 percent power, 0.942 square foot 30 percent power, and 0.4 square foot hot shutdown.

g. Failure of a main steam isolation valve, failure of a feedwater isolation valve or main feed pump trip, and failure of auxiliary feedwater runout control were considered. Two cases for each break size and power level scenario were evaluated with one case modeling the MSIV failure and the other case modeling the AFW runout control failure. Each case assumed conservative main feedwater addition to bound the feedwater isolation valve or main feed pump trip failure.

h. The auxiliary feedwater system is manually realigned by the operator after 10 minutes.

i. A shutdown margin of 1.3% $\Delta k/k$ is assumed. This assumption includes added conservatism with respect to the end-of-life shutdown margin requirement of 1.6% $\Delta k/k$ at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position.

j. A moderator density coefficient of 0.54 $\Delta k/gm/cc$ is assumed to support the relaxation of the most negative moderator temperature coefficient limit.

k. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system (ECCS) consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the intermediate head safety injection system, and 4) the high head safety injection (charging) system. Only the high head safety injection (charging) system and the passive accumulators are modeled for the steam line break accident analysis.

The modeling of the safety injection system in LOFTRAN is described in Reference 22. Figure 3.3-52 of WCAP-11902 presents the safety injection flow rates as a function of RCS pressure assumed in the analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold legs. The safety injection flows assumed in this analysis take into account the degradation of the ECCS charging pump performance. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the boron injection tank isolation valves prior to the delivery of boric acid to the reactor coolant loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed.

After the generation of the safety injection signal (appropriate delays for instrumentation, logic and signal transport included), the appropriate valves begin to operate and the safety injection charging pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into the core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

l. For the at-power cases, reactor trip is available by safety injection signal, overpower protection signal (high neutron flux reactor trip or OPΔT reactor trip), and low pressurizer pressure reactor trip signal.

m. Offsite power is assumed available. Continued operation of the reactor coolant pumps maximizes the energy transferred from the reactor coolant system to the steam generators.

n. No steam generator tube plugging is assumed to maximize the heat transfer characteristics.

2. Break Flow Calculations

a. Steam Generator Blowdown

The LOFTRAN computer code (Reference 22) was used to calculate the break flows and enthalpies of the release through the steam line break. Blowdown mass/energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant thick metal heat storage, and reverse steam generator heat transfer.

b. Steam Plant Piping Blowdown

The calculated mass and energy releases include the contribution from the secondary steam piping. For all ruptures, the steam piping volume blowdown begins at the time of the break and continues until the entire piping inventory is released. The flow rate is determined using the Moody correlation and the pipe cross sectional area.

3. Single Failure Effects

a. Failure of a main steam isolation valve (MSIV) increases the volume of steam piping which is not isolated from the break. When all valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and the isolation valves in the other steam lines, including safety and relief valve headers and other connecting lines, will feed the break. For the cases which modeled a failure of a MSIV, the steam line volumes associated with Unit 2 were assumed since the volume available for blowdown for this scenario is greater than Unit 1. For the cases which did not model a failure of a

MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than Unit 2.

b. Failure of a diesel generator would result in the loss of one containment safeguards train, resulting in minimum heat removal capability.

c. Failure of a feedwater isolation valve would result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flash into steam and exit through the break. For consistency with the FSAR steamline break mass/energy release analysis, all cases conservatively assumed failure of the feedwater isolation valve, which resulted in the additional inventory available for release through the steambreak and in higher than normal main feedwater flows.

d. Failure of the auxiliary feedwater runout control equipment could result in higher auxiliary feedwater flows entering the steam generator prior to realignment of the auxiliary feedwater system. For cases where the runout control operates properly, a constant auxiliary feedwater flow of 670 gpm to the faulted steam generator was assumed. This value was increased to 1325 gpm to simulate a failure of the runout control.

CONTAINMENT ANALYSIS

Following a steam line break in the lower compartment of an ice condenser plant, two distinct analyses must be performed. The first analysis, the short term pressure analysis, has been performed with the TMD Code. The second analysis, the long term analysis, does not require the large number of nodes which the TMD analysis requires. The computer code which performs this analysis is the LOTIC[20] Code.

The LOTIC Code includes the capability to calculate the superheat conditions, and has the ability to begin calculations from time zero.[7, 8, 9] The major thermodynamic assumption which is used in the steam break analysis is complete re-evaporation of the condensate under superheated conditions for large breaks. For the most limiting small breaks, no re-evaporation is assumed; however, convective heat transfer as detailed in Reference (8) is used. The version of the LOTIC Code which incorporates the above is the LOTIC3 Code.[18] This code was used to perform the steam line break analyses for Cook Nuclear Plant Unit 2 and is the version which has been accepted for this use.[10,11]

Containment Transient Calculations

The following are the major input assumptions used in the LOTIC3 steam break analysis for the Cook Nuclear Plant Unit 2:

1. Minimum safeguards are employed, e.g., one of two spray pumps (this includes a 10% degradation in the spray pump flow) and one of two air return fans.

2. The air return fan is effective 10 minutes after the high-high containment pressure signal is read.

3. A uniform distribution of steam flow into the ice bed is assumed.

4. The total initial ice mass used water 2.11 x 106 lbs.

5. The initial conditions in the containment are a temperature of 120^{o}F in the lower and dead ended volumes, a temperature of 57^{o}F in the upper volume, and a temperature of 27^{o}F in the ice condenser. All volumes are at a pressure of 0.3 psig and a relative humidity of 15%.

6. A spray pump flow of 1900 gpm is used in the upper compartment and 900 gpm in the lower compartment. The spray initiation time assumed was 45 sec. after reaching the high-high setpoint.

7. The refueling water storage tank temperature is assumed to be 100oF.

8. The essential service water used on the spray heat exchanger and the component cooling water heat exchanger is modeled at a temperature of 81oF.

9. Containment structural heat sinks as presented in Table 14.3.4-2 were used.

10. The air return fan empties air at a rate of 40,000 cfm from the upper to the lower compartments.

11. The material property data given in Table 14.3.4-3 were used.

12. The mass and energy releases given in Tables 14.3.4-23 and 14.3.4-24 were used. Since these rates are considerably less than the RCS double ended breaks, and their total integrated energy is not sufficient to cause ice bed meltout, the containment pressure transients generated for the previously presented double ended pump suction RCS break is considerably more severe.

13. The heat transfer coefficients to the containment structures are based on the work of Tagami. An explanation of their manner of application is given in References (3, 7, and 8).

<u>Results</u>

The results of the analysis are presented in Table 14.3.4-25. The worst case of the double ended steam line breaks was a 4.6 ft^2 break, occurring at 102% power with main steam line isolation valve failure. This temperature transient is shown in Figure 14.3.4-477.

The results from the steam line split ruptures (or small breaks) are presented in Table 14.3.4-26. The worst case for these cases was a 0.86 ft^2 small break, occurring at 102% power, with failure of auxiliary feed runout protection. A temperature transient of this case is presented in Figure 14.3.4-478.

Parameter studies have been performed as part of previous analyses, varying the ice mass between 2.0 and 2.45 million pounds. These previous ice mass parameter studies have shown that the maximum containment calculated temperatures are not sensitive (less than $1^{o}F$ change) to these ice mass changes.

Sensitivity of the Results

The previous section pertains to the steam line break analysis and its subsequent response in identifying the limiting small break. The following evaluation describes additional sensitivity studies of a generic nature, done for breaks smaller and up to 0.942 ft^2 at 30% power[19].

The LOTIC-3 computer code was employed in the generic analysis. The LOTIC-3 computer code[18] was found to be acceptable for the analysis of steam line breaks with the following restrictions:

a. Mass and energy release rates are calculated with an approved model.

b. Complete break spectrums are analyzed.

c. Convective heat flux calculations are performed for all break sizes.

A detailed comparison of the Cook Nuclear Plant characteristics with those of the generic plant can be found in Reference (18).

Reference 23 also shows that use of generic parameters is conservative with respect to Cook Nuclear Plant. Figure 14.3.4-497 contains a comparison of the limiting small break cases, 0.942 ft^2, from the Cook Nuclear Plant Unit 2 and generic plant's previous small break submittals. Figure 14.3.4-497 illustrates that the small steam line break temperature transients result in very similar peaks with any differences being incidental to the results. In addition, elevated containment temperatures for Cook Nuclear Plant last for a shorter duration in the transient.

Further, the containment pressure Hi-2 setpoint which provides the actuation signal for the containment spray and fan systems was assumed to be 3.5 psig in the generic analysis. The Cook Nuclear Plant Unit 2 Hi-2 setpoint is 2.9 psig. Therefore, the actuation setpoint would have been reached sooner in Cook Nuclear Plant Unit 2 and therefore the containment transient would have been mitigated more rapidly.

Therefore, a generic LOTIC3 spectrum of small breaks analysis is provided here for Cook Nuclear Plant Unit 2 instead of plant specific analysis. The generic analysis provides the containment responses for a spectrum of small breaks at the 30% power level with assumed failure of the auxiliary feedwater runout protection system. The analyses studied a spectrum of breaks ranging in size from 0.1 ft^2 up to the break identified as the most severe small split break, 0.942 ft^2. The lower bound break size was established in discussions held between the NRC staff and Westinghouse Electric Corporation.

This spectrum included breaks of 0.6, 0.35 and 0.10 ft^2. Figures 14.3.4-498 and 14.3.4-499 provide the upper compartment temperature and lower compartment pressure transients. As Figure 14.3.4-500 shows, similar lower compartment temperature transients were calculated for the spectrum of breaks analyzed. However, the 0.6 ft^2 break resulted in a slightly higher maximum lower compartment temperature (See Table 14.3.4-34). When this transient was compared to the transient identified as the most severe small break at 30% power in the previous analysis, it was found to result in very similar peaks, with the difference being incidental to the results (See Figure 14.3.4-501).

In the analysis, spray and fan initiation are automatic after reaching the containment Hi-2 setpoint. Associated times are included in table 14.3.4-34. As described above, these times are conservative in regard to Cook Nuclear Plant Unit 2. Tables 14.3.4-35 and 14.3.4-36 provide the mass and energy release rates for the transients analyzed. These results demonstrate the conservatism of the results previously discussed and also the somewhat insensitive nature of the ice condenser plant containment response to break size.

Table 14.3.4-37 further demonstrates the conservatism of the generic analysis discussed above. The actual plant specific analysis results for the smaller breaks would be similar to the Cook Nuclear Plant Unit 2 results in Figure 14.3.4-497. The temperature would peak, then sharply fall off when the sprays come on, and finally settle to a much lower temperature level for the remainder of the transient.

14.3.4.7 Subcompartment Analysis

A reevaluation of the pressure response of containment subcompartment analyses was made as part of the Unit 2 licensing process. The purpose of the reevaluation was to determine maximum dynamic pressure loads that could act on the structures forming subcompartment enclosure boundaries and equipment supports. These reevaluations are discussed below.

Pressurizer Enclosure*

This analysis was done by comparison of the design of the Donald C. Cook Nuclear Plant pressurizer enclosure to the McGuire Nuclear Station, Units 1 and 2, and Watts Bar Nuclear Plant, Units 1 and 2, pressurizer enclosure designs. Since the designs are similar, the same nodalization scheme was used. The transient response of the pressurizer enclosure was performed assuming a double ended rupture of the 4-inch spray line (at the piping-to-pressurizer vessel nozzle weld) using the Westinghouse TMD subcompartment Code to perform the calculations.

Figure 14.3.4-479 presents a schematic illustration for the nodalization used in performing the pressurizer subcompartment analysis for these three units. Table 14.3.4-27 presents the node volumes for the three units compared here. Table 14.3.4-28 provides the Cook Nuclear Plant Unit 2 pressurizer enclosure

*The current licensing basis for the pressurizer enclosure is located in Section 14.3.4.9.2, Unit 1 UFSAR. The following material represents previous licensing basis for the pressurizer enclosure of Unit 2.

nodalization hydraulic data. Table 6.2.1-29 from the Watts Bar FSAR and Table 6.2.1-34 of the McGuire FSAR (reproduced as part of Table 14.3.4-28) provide a direct comparison of flow paths, flow areas, coefficients, and associated data with those for Cook Nuclear Plant Unit 2. As can be seen from this table, a certain degree of similarity exists between all three designs.

The results of a TMD analysis for the Cook Nuclear Plant Unit 2 pressurizer subcompartment are given in Table 14.3.4-29. Table 14.3.4-30 provides the mass and energy releases for the Cook Nuclear Plant Unit 2 TMD analyses. Figure 14.3.4-480 gives the pressurizer enclosure noding and flowpaths, Figure 14.3.4-481 the TMD code network, and Figures 14.3.4-482 through 14.3.4-493 the pressure transient resulting from this analysis.

The location of the largest break possible in the pressurizer enclosure, a double-ended break of the spray line from the reactor coolant system, is assumed at the top of the enclosure for each of the three plants compared here. The Cook Nuclear Plant Unit 2 spray line within the enclosure is a 4-inch pipe and is therefore the maximum break possible. For Watts Bar and McGuire, a reduction from the 6-inch line from the RCS to the 4-inch spray connection is made within the enclosure and 6-inch line breaks have been selected for these plants as the largest possible break size.

Table 14.3.4-31 tabulates the differential pressures calculated for Cook Nuclear Plant Unit 2 and compares these with the referenced plants FSAR data across the enclosure walls and the pressurizer vessel.

The design of the pressurizer supports for the differential pressure of 0.27 psi resulting from a double ended break of the spray line from the reactor coolant system was reviewed. This differential pressure was combined with the SSE load and LOCA load associated with the double ended break of the spray line. The results of the analysis indicate that the pressurizer supports can accommodate these combined loads within the allowable design elements. This analysis shows the adequacy of the pressurizer supports if this differential pressure were as high as 1.3 psi. The differential pressure load to the upper and lower pressurizer support were conservatively distributed, utilizing simple static equilibrium equations.

Steam Generator Subcompartment Analysis

The steam generator subcompartment analysis was submitted to the NRC by letters dated January 23, 1978[12] and February 27, 1978.[13] The analysis confirmed the adequacy of the steam generator subcompartment and steam generator supports design to withstand the consequences of a steam line break inside the subcompartment.

Figures 14.3.4-494 and 14.3.4-495 show the details of the steam generator subcompartment used in the analysis. Figure 14.3.4-496 shows the 9 node TMD model which was used in the analytical model.

Two breaks were analyzed, a break at the outlet nozzle and a break at the side of the vessel. The peak differential pressures for both of these breaks across the structures and the steam generator vessel are shown in Tables 14.3.4-32 and 14.3.4-33. Detailed analyses showing the adequacy of the subcompartment structures and steam generator supports are presented in References 12 and 13.

It should be noted that the analyses discussed above are only for short term pressure peaks and are not applicable to long term type analyses. The design differential pressures are exceeded for a short duration only. The dynamic analysis of the affected structures and of the steam generator vessel supports has shown that the effects of the short duration peak pressures will not result in consequences that will adversely affect the public health and safety.

REFERENCES, SECTION 14.3.4, UNIT 2

1. Grimm, N.P. and Colenbrander, H. G. C., "Long Term Ice Condenser, Containment Code - LOTIC Code," WCAP-8354-P-A, April, 1976 (Proprietary) and WCAP-8355-A, April, 1976 (Non-Proprietary).

2. "Final Report Ice Condenser Full Scale Section Test at the Waltz Mill Facility," WCAP-8282, February, 1974 (Proprietary) and WCAP-8110, Supplement 6, May, 1974 (Non-Proprietary).

3. Hsich, T. and Raymond, M., "Long Term Ice Condenser Containment Code - LOTIC Code," WCAP-8354-P-A, Supplement 1, April, 1976 (Proprietary) and WCAP-8355, Supplement 1, April, 1976 (Non-Proprietary).

4. Salvatori, R. (approved), "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8077, March, 1973 (Proprietary) and WCAP-8078, March, 1973 (Non-Proprietary).

5. Salvatori, R. (approved), "Ice Condenser Full Scale Section Test at the Waltz Mill Facility," WCAP-8110, Supplement 6, May, 1974.

6. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-Proprietary).

7. NS-CE-1250, 10/22/76, C. Eicheldinger Letter to J. F. Stolz, NRC, Supplemental Information on LOTIC3 questions.

8. NS-CE-1453, 6/14/77, C. Eicheldinger letter to J. F. Stolz, NRC, Responses to LOTIC3 questions.

9. NS-CE-1626, 12/7/77, C. Eicheldinger letter to J. F. Stolz, NRC, Responses to LOTIC3 questions.

10. J. F. Stolz, NRC, to C. Eicheldinger, 5/3/78, "Evaluation of Supplement to WCAP-8354 (LOTIC3)."

11. J. F. Stolz, NRC, to C. Eicheldinger, 5/10/78, "Staff Approval of LOTIC3 Code."

12. Letter from John Tillinghast, Indiana & Michigan Power Co., to Edson G. Case, U.S. Nuclear Regulatory Commission, dated January 23, 1978.

13. Letter from G.P. Maloney, Indiana & Michigan Power Co., to Edson G. Case, U.S. Nuclear Regulatory Commission, dated February 27, 1978.

14. Amendment No. 83 to the original FSAR - Letter No. AEP:NRC:0117, dated April 4, 1979, John E. Dolan of Indiana & Michigan Power Company to H. R. Denton of NRC. See also letter No. AEP:NRC:0131, dated April 1, 1980 from G. P. Maloney of Indiana and Michigan Electric Company to H. R. Denton of NRC.

15. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1974.

16. EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

17. Letter from D. B. Black to B. A. Svensson, August 3, 1988, "Operation at Essential Service Water Temperatures Above 81°F."

18. Hsich, T. and Liparulo, N. J., "Westinghouse Long Term Ice Condenser Containment Code -LOTIC3 Code", WCAP-8354-P-A, Supplement 2, February 1979.

19. Letter No. AEP-80-525, March 10, 1980 (F. Noon of Westinghouse to D. V. Shaller of AEPSC).

20. Grimm, N. P. and Colenbrander, H. G. C., "Long Term Ice Condenser Containment Code - LOTIC Code," WCAP-8354 (Proprietary) and WCAP-8355 (Non-Proprietary), July 1974.

21. WCAP-12135, "Rerating Engineering Report," September 1989.

22. Burnett, T. W. T., et. al., "LOFTRAN Code Description", WCAP-7907-A, April 1, 1984.

23. Letter, G. P. Maloney (AEP) to H. R. Denton (NRC), dated April 1, 1980 (Letter No. AEP:NRC:0131).

TABLE 14.3.4-1

ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. End of Blowdown (t=10.0 sec) (BTU)	Approx. End of Reflood (t=294.7 sec) (BTU)
*Ice Heat Removal	210.5	261.7
*Structural Heat Sinks	17.34	46.10
*RHR Heat Exchanger Heat Removal	0	0
*Spray Heat Exchanger Heat Removal	0	0
Energy Content of Sump	190.1	263.34
Ice Melted (Pounds)	0.68 (10^6)	0.884 (10^6)

*Integrated Energies

TABLE 14.3.4-1 (continued)

ENERGY ACCOUNTING IN MILLIONS OF BTU

	Approx. Time of Ice Melt Out (t-4443 sec) (BTU)	Approx. Time of Peak Pressure (t-6955 sec) (BTU)
*Ice Heat Removal	567.75	567.75
*Structural Heat Sinks	69.44	101.0
*RHR Heat Exchanger Heat Removal	75.52	148.73
*Spray Heat Exchanger Heat Removal	88.59	174.63
Energy Content of Sump	622.8	625.6
Ice Melted (Pounds)	2.11	2.11

*Integrated Energies

TABLE 14.3.4-2

STRUCTURAL HEAT SINK TABLE

	SURFACES	AREA (FT2)	THICKNESS (FT)
Upper Compartment Material			
1.	Paint	32500.	0.001083
	Carbon Steel	32500.	0.0469
	Concrete	32500.	2.0
2.	Paint	10086.	0.001083
	Concrete	10086.	2.0
3.	Paint	5880.	0.001250
	Concrete	5880.	1.5
4.	Paint	11970.	0.00125
	Concrete	11970	1.0
Lower Compartment Material			
5.	Paint	5069.	0.00125
	Concrete	5069.	2.0
6.	Paint	13660.	0.00125
	Concrete	13660	1.5
7.	Paint	16730.	0.00125
	Concrete	16730.	1.0
8.	Paint	8665.	0.00125
	Concrete	8665.	2.0
Ice Condenser			
9.	Steel	180600.	0.00663
10.	Steel	76650.	0.0217
11.	Steel	28670.	0.0267
12.	Paint	3336.	0.000833
	Concrete	3336.	0.333
13.	Steel and Insulation	19100.	1.0
	Steel	19100.	0.0625
14.	Steel and Insulation	13055.	1.0
	Concrete	13055.	1.0

TABLE 14.3.4-3

MATERIAL PROPERTY DATA

Material	Thermal Conductivity BTU/hr-ft-°F	Volumetric Heat Capacity BTU/ft^3-°F
Paint	0.0833	28.4
Concrete	0.8	28.8
Steel	26.0	56.4
Steel and Insulation	0.2	3.663

TABLE 14.3.4-4

CALCULATED MAXIMUM PEAK PRESSURES IN LOWER COMPARTMENT ELEMENTS ASSUMING UNAUGMENTED FLOW

Element	1	2	3	4	5	6	
Peak Pressure (psig)	13.6	11.6	10.5	10.6	11.5	13.0	DECL-100% Ent.
Peak Pressure (psig)	14.4	11.0	9.2	9.1	10.8	14.4	DEHL-100% Ent.

TABLE 14.3.4-5

CALCULATED MAXIMUM PEAK PRESSURES IN THE ICE CONDENSER COMPARTMENT ASSUMING UNAUGMENTED FLOW

Element	40	41	42	43	44	45	
Peak Pressure (psig)	10.4	8.8	7.9	7.9	8.7	9.9	DECL-100% Ent.
Peak Pressure (psig)	10.8	8.3	7.2	7.5	8.3	10.6	DEHL-100% Ent.

TABLE 14.3.4-6

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES ACROSS THE OPERATING DECK OF LOWER CRANE WALL ASSUMING UNAUGMENTED FLOW

Element	1	2	3	4	5	6	
Peak Δp (psi)	12.1	8.9	7.4	7.2	8.7	11.7	DECL - 100% Ent.
Peak Δp (psi)	14.1	10.6	8.1	8.3	10.5	14.1	DEHL - 100% Ent.

TABLE 14.3.4-7

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES ACROSS THE UPPER CRANE WALL ASSUMING UNAUGMENTED FLOW

Element	7-8-9	10-11-12	13-14-15	16-17-18	19-20-21	22-23-24	
Peak Δp (psi)	6.9	5.5	5.1	5.1	5.6	6.8	DECL - 100% Ent.
Peak Δp (psi)	8.2	6.8	6.0	5.9	6.7	8.2	DEHL - 100% Ent.

TABLE 14.3.4-8

SENSITIVITY STUDIES FOR THE DONALD C. COOK UNIT 2 PLANT

Parameter	Change Made From Base Value	Change In Operating Deck P	Change In Peak Pressure Against The Shell
Blowdown	+ 10%	+ 11%	+ 12%
Blowdown	- 10%	- 10%	- 12%
Blowdown	- 20%	- 20%	- 23%
Blowdown	- 50%	- 50%	- 53%
Break Compartment Inertial Length	+ 10%	+ 4%	+ 1%
Break Compartment Inertial Length	- 10%	- 4%	- 1%
Break Compartment Volume	+ 10%	- 2%	- 1%
Break Compartment Volume	- 10%	+ 2%	+ 1%
Break Compartment Vent Areas	+ 10%	- 6%	- 5%
Break Compartment Vent Areas	- 10%	+ 8%	+ 5%
Door Port Failure in Break Compartment	one door port fails to open	+ 1%	- 1%
Ice Mass	+ 10%	0	0
Ice Mass	- 10%	0	0
Door Inertia	+ 10%	+ 1%	0
Door Inertia	- 10%	- 1%	0
All Inertial Lengths	+ 10%	+ 5%	+ 4%
All Inertial Lengths	- 10%	- 5%	- 3%
Ice Bed Loss Coefficients	+ 10%	0	0
Ice Bed Loss Coefficients	- 10%	0	0
Entrainment Level	0% Ent.	- 27%	-

TABLE 14.3.4-8 (Con't)

SENSITIVITY STUDIES FOR THE DONALD C. COOK UNIT 2 PLANT

Parameter	Change Made From Base Value	Change In Operating Deck ΔP	Change In Peak Pressure Against The Shell
11%Entrainment Level	30% Ent.	- 19%	- 15%
Entrainment Level	50% Ent.	- 13%	- 12%
Entrainment Level	75% Ent.	- 6%	- 6%
Lower Compartment Loss Coefficients	+ 10%	0	0
Lower Compartment Loss Coefficients	- 10%	0	0
Cross Flow in Lower Plenum	Low estimate of resistance	0	- 7%
Cross Flow in Lower Plenum	High estimate of resistance	0	- 3%
Ice Condenser Flow Area	+ 10%	0	- 3%
Ice Condenser Flow Area	- 10%	0	+ 4%
Ice Condenser Flow Area	+ 20%	0	- 6%
Ice Condenser Flow Area	- 50%	0	+ 8%
Initial Pressure in Containment	+ 0.3 psi	+ 2%	+ 2%
Initial Pressure in Containment	- 0.3 psi	- 2%	- 2%
Reactor Coolant Break Enthalpy	- 13.0%	+ 6%	+ 3%
Compressibility Factor	Addition of the compressibility factor	+ 4%	0

All values shown are to the nearest percent.

TABLE 14.3.4-9

DONALD C. COOK UNIT 2 ICE CONDENSER ANALYSIS PARAMETERS

Reactor Containment Volume (net free volume)	
Upper Compartment, ft^3	774,481
Ice Condenser, ft^3	110,520
Lower Compartment (active), ft^3	301,583
Total Active Volume, ft^3	1,186,584
Lower Compartment (dead ended), ft^3	60,727
Total Containment Volume, ft^3	Not Applicable
Reactor Containment Air Compression Ratio	1.403
NSSS Power, MWt	3425
Design Energy Release to Containment	
Initial blowdown mass release, lb	541,135
Initial blowdown energy release, BTU	337.1×10^6
Ice Condenser Parameters	
Weight of ice in condenser, lb	2.11×10^6

Additional System Parameters

Core Inlet Temperature (+5°F), °F	552.3
Initial Steam Generator Steam Pressure, psia	836.3
Assumed Maximum Containment Back Pressure, psia	26.7

TABLE 14.3.4-10

DECK LEAKAGE SENSITIVITY

Break Size	5 ft^2 Deck Leak Air Compression Peak (psig)	Deck Leakage Area (ft^2)	Spray Flow Rate (gpm)	Resultant Peak Containment Pressure (psig)
Double ended	7.8	54	0	12.0
0.6 double ended	6.6	46	0	12.0
3 ft^2	6.25	50	0	12.0
8-inch diameter	5.5	56	4000	12.2
8-inch diameter	5.5	35	2000	12.0
8-inch diameter*	5.5	56	2000	11.3
6-inch diameter	5.0	56	4000	10.4
2 1/2-inch diameter	4.0	56	4000	8.5
1/2-inch diameter	3.0	50	4000	3.0

*This case assumes upper compartment structural heat sink steam condensation of 8 lb/sec and 30 percent of deck leakage is air.

TABLE 14.3.4-11

BLOWDOWN MASS AND ENERGY RELEASES***

TIME	BREAK PATH NO. 1 FLOW*		BREAK PATH NO. 2 FLOW**	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	40909.1	22356.0	21689.4	11814.3
0.300	46769.9	25936.7	23551.7	12846.4
0.801	45096.3	26426.0	19952.6	10912.8
1.10	41851.9	25127.4	19001.7	10399.8
2.20	33356.1	21693.8	18305.9	10022.6
2.80	26606.5	18112.7	17189.2	9413.8
3.00	22059.7	15198.6	16819.0	9212.9
3.30	19874.4	13886.0	16260.3	8909.3
4.60	15322.1	10780.9	14204.2	7791.8
6.60	12200.9	8502.2	12540.0	6886.8
7.20	12072.2	8342.8	13152.0	7227.8
8.20	12104.4	8425.4	12712.0	6996.2
8.60	9875.1	7695.5	12495.5	6879.6
11.0	9252.2	6913.6	11003.8	6060.8
15.8	5727.5	4971.9	8155.7	4503.7
18.4	4276.5	3862.8	6120.7	3120.1
18.8	4052.9	3750.6	10461.9	5463.2
19.0	3962.9	3747.5	5052.9	2623.5
19.2	3839.1	3695.6	8734.7	4432.0
19.6	3594.3	3706.8	6452.8	3128.6
19.8	3382.6	3640.6	8549.4	4256.0
20.0	3149.6	3552.8	4200.7	2079.1
20.2	2903.4	3394.6	6584.7	3029.2
20.4	2695.2	3258.3	4440.9	2063.2
20.8	2274.8	2799.2	5497.9	2392.1
21.0	2132.6	2636.4	3691.6	1605.3
22.0	1468.8	1835.2	4981.3	1974.0
22.6	1241.0	1556.4	2335.8	865.6
22.8	1141.8	1433.7	4845.3	1636.2
23.0	1058.4	1330.9	3545.6	1191.7
24.4	436.5	552.1	2262.7	670.2
25.4	193.0	245.7	2361.9	656.2
27.2	40.0	51.5	547.5	160.4
29.0	0.0	0.0	118.3	46.3

*Break Path No. 1 is steam generator side of break.

**Break Path No. 2 is pump side of break.

***Min SI

TABLE 14.3.4-12

BLOWDOWN MASS AND ENERGY RELEASE***

TIME	BREAK PATH NO. 1 FLOW*		BREAK PATH NO. 2 FLOW**	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	40909.1	22356.0	21689.4	11814.3
0.300	46769.9	25936.7	23551.7	12846.4
0.801	45096.3	26426.0	19952.6	10912.8
1.10	41851.9	25127.4	19001.7	10399.8
2.20	33356.1	21693.8	18305.9	10022.6
2.80	26606.5	18112.7	17189.2	9413.8
3.00	22059.7	15198.6	16819.0	9212.9
3.30	19874.4	13886.0	16260.3	8909.3
4.60	15322.1	10780.9	14204.2	7791.8
6.60	12200.9	8502.2	12540.0	6886.8
7.20	12072.2	8342.8	13152.0	7227.8
8.20	12104.4	8425.4	12712.0	6996.2
8.60	9875.1	7695.5	12495.5	6879.6
11.0	9252.2	6913.6	11003.8	6060.8
15.8	5727.5	4971.9	8155.7	4503.7
18.4	4276.5	3862.8	6120.7	3120.1
18.8	4052.9	3750.6	10461.9	5463.2
19.0	3962.9	3747.5	5052.9	2623.5
19.2	3839.1	3695.6	8734.7	4432.0
19.6	3594.3	3706.8	6452.8	3128.6
19.8	3382.6	3640.6	8549.4	4256.0
20.0	3149.6	3552.8	4200.7	2079.1
20.2	2903.4	3394.6	6584.7	3029.2
20.4	2695.2	3258.3	4440.9	2063.2
20.8	2274.8	2799.2	5497.9	2392.1
21.0	2132.6	2636.4	3691.6	1605.3
22.0	1468.8	1835.2	4981.3	1974.0
22.6	1241.0	1556.4	2335.8	865.6
22.8	1141.8	1433.7	4845.3	1636.2
23.0	1058.4	1330.9	3545.6	1191.7
24.4	436.5	552.1.	2262.7	670.2
25.4	193.0	245.7	2361.9	656.2
27.2	40.0	51.5	547.5	160.4
29.0	0.0	0.0	118.3	46.3

*Break Path No. 1 is steam generator side of break.

**Break Path No. 2 is pump side of break.

***Max SI

TABLE 14.3.4-13

3425 MWT / DEPS - MIN SI / RHR X-TIE CLOSED
REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
29.0	0.0	0.0	0.0	0.0
29.3	104.3	121.6	2003.2	176.7
29.8	92.5	107.8	1979.7	174.6
30.6	90.1	104.9	1924.8	169.7
31.1	102.6	119.5	1895.1	176.0
36.1	131.6	153.5	1681.0	147.9
38.1	141.5	165.2	1614.0	141.9
38.8	249.6	292.5	3473.6	399.2
39.1	302.0	354.7	4220.2	511.8
40.1	348.5	410.3	4855.7	636.6
41.1	348.2	409.9	4850.6	640.6
42.1	343.7	404.6	4790.9	634.1
46.1	325.9	383.4	4548.0	605.4
47.1	321.9	378.6	4491.4	598.6
51.1	307.6	361.5	4282.1	573.3
53.1	300.4	352.9	4187.9	562.1
54.1	296.9	348.7	4143.0	556.8
56.1	290.3	340.8	4057.1	546.6
60.1	278.3	326.6	3899.5	527.9
61.2	383.3	451.8	276.3	206.0
62.2	394.2	465.1	281.7	214.0
71.2	339.6	399.7	254.8	177.8
76.2	317.8	373.6	243.7	162.9
77.2	313.9	369.0	241.6	160.2
85.2	284.5	334.0	228.2	142.4
86.2	281.2	330.1	226.7	140.5
94.2	257.8	302.3	216.6	127.2
112.2	221.0	258.8	201.1	106.9
114.2	218.0	255.2	199.9	105.3
132.2	197.1	230.6	191.4	94.2
158.2	180.5	211.0	184.6	85.5
190.2	172.4	201.4	181.2	81.2
268.2	169.5	198.0	179.6	79.1
274.2	170.6	199.3	181.0	79.7
291.2	175.0	204.5	191.4	81.8

TABLE 14.3.4-14

3425 MWT / DEPS - MAX SI / RHR X-TIE OPEN
REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
29.0	0.0	0.0	0.0	0.0
29.3	113.5	132.4	2160.0	187.3
29.7	100.7	117.4	2144.5	185.9
30.4	98.4	114.6	2096.0	181.5
31.4	116.2	135.4	2037.8	176.3
35.1	138.2	161.2	1874.2	161.7
37.1	148.9	173.8	1803.0	155.3
38.1	325.8	383.1	4523.6	533.7
38.2	333.6	392.3	4643.3	548.7
39.2	386.3	455.5	5357.6	683.1
40.2	385.9	455.2	5352.5	687.0
41.2	380.9	449.2	5292.1	680.3
43.2	370.7	437.0	5167.3	665.3
44.2	365.9	431.2	5106.8	658.0
46.2	356.8	420.3	4990.9	643.8
47.2	352.6	415.2	4935.7	637.1
49.2	344.6	405.7	4830.3	624.2
53.2	330.7	389.1	4638.8	600.8
57.2	318.9	375.0	4469.3	580.1
61.2	308.8	362.9	4318.0	561.7
62.2	241.5	283.0	751.9	179.5
63.2	238.2	279.1	758.0	180.0
67.2	222.8	260.9	784.1	181.9
73.2	159.4	186.2	892.5	192.2
74.2	149.8	174.9	909.9	193.1
90.2	146.0	170.4	919.3	191.6
118.2	138.6	161.7	925.7	186.9
150.2	131.1	153.0	938.3	183.2
160.2	128.9	150.3	943.2	182.3
192.2	121.5	141.6	958.6	179.7
210.2	116.7	136.1	964.0	177.4
244.2	107.5	125.2	978.6	174.0
270.2	100.2	116.7	990.4	171.7
296.2	98.0	114.1	991.8	169.2
393.8	90.6	105.5	1002.2	160.7

TABLE 14.3.4-15
REFLOOD TRANSIENT PARAMETERS
DOUBLE ENDED PUMP SUCTION

3425 MWT / DEPS - MIN SI / RHR X-TIE CLOSED

TIME SECONDS	FLOODING TEMP DEGREE F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (POUNDS MASS PER SECOND)	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY BTU/LBM
29.0	227.8	0.000	0.000	0.000	0.000	0.250	0.0	0.0	0.0	0.00
29.5	224.3	22.533	0.000	0.62	0.37	1.000	7981.5	7501.1	0.0	88.21
29.8	222.3	13.274	0.000	1.09	0.48	1.000	7907.7	7427.1	0.0	88.19
30.1	221.7	1.776	0.055	1.24	0.93	1.000	7824.6	7343.8	0.0	88.18
30.4	221.8	2.971	0.096	1.29	1.48	1.000	7744.1	7263.2	0.0	88.16
31.7	222.2	2.120	0.334	1.50	3.77	0.589	7458.1	6976.6	0.0	88.11
31.8	222.2	1.854	0.340	1.51	3.93	0.585	7450.6	6969.2	0.0	88.11
31.9	222.3	2.053	0.358	1.52	4.12	0.574	7419.1	6937.6	0.0	88.10
35.1	224.1	1.824	0.571	1.78	9.79	0.473	6865.4	6382.9	0.0	87.99
38.8	226.2	2.735	0.663	2.00	15.70	0.540	6090.8	5627.2	0.0	87.86
40.1	226.5	3.397	0.682	2.12	15.99	0.586	5574.8	5136.3	0.0	87.81
42.1	227.1	3.216	0.711	2.29	16.00	0.585	5380.9	4941.1	0.0	87.74
45.1	228.2	3.003	0.735	2.50	16.00	0.583	5147.2	4703.8	0.0	87.65
53.8	232.5	2.642	0.765	3.00	16.00	0.575	4618.9	4167.2	0.0	87.40
60.1	235.9	2.466	0.775	3.31	16.00	0.568	4324.6	3868.4	0.0	87.23
61.2	236.5	3.210	0.766	3.37	15.92	0.616	429.6	0.0	0.0	68.00
62.2	237.0	3.261	0.765	3.43	15.76	0.616	424.2	0.0	0.0	68.00
63.4	237.7	3.185	0.767	3.51	15.58	0.615	426.4	0.0	0.0	68.00
72.3	242.8	2.729	0.778	4.00	14.48	0.614	440.5	0.0	0.0	68.00
83.2	244.3	2.364	0.786	4.50	13.56	0.612	451.5	0.0	0.0	68.00
95.9	241.7	2.072	0.788	5.00	12.90	0.606	459.5	0.0	0.0	68.00
112.2	243.1	1.817	0.793	5.55	12.49	0.599	465.7	0.0	0.0	68.00
127.4	244.2	1.667	0.797	6.00	12.39	0.593	469.2	0.0	0.0	68.00
146.2	242.7	1.555	0.796	6.52	12.49	0.588	471.8	0.0	0.0	68.00
165.3	244.3	1.485	0.800	7.00	12.76	0.584	473.3	0.0	0.0	68.00
186.2	243.2	1.452	0.799	7.51	13.15	0.582	474.1	0.0	0.0	68.00
206.5	244.3	1.430	0.801	8.00	13.58	0.582	474.5	0.0	0.0	68.00
228.2	243.6	1.423	0.800	8.51	14.08	0.581	474.7	0.0	0.0	68.00
248.9	244.3	1.418	0.802	9.00	14.56	0.581	474.8	0.0	0.0	68.00
258.2	244.3	1.418	0.802	9.22	14.77	0.582	474.8	0.0	0.0	68.00
270.2	243.8	1.422	0.801	9.50	15.05	0.582	474.8	0.0	0.0	68.00
291.2	244.3	1.448	0.801	10.00	15.47	0.589	473.9	0.0	0.0	68.00

TABLE 14.3.4-16

REFLOOD TRANSIENT PARAMETERS

DOUBLE ENDED PUMP SUCTION

3425 MWT / DEPS - MAX SI / RHR X-TIE OPEN

TIME SECONDS	FLOODING TEMP DEGREE F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWNCOMER HEIGHT FT	FLOW FRACTION	INJECTION TOTAL (POUNDS MASS PER SECOND)	INJECTION ACCUMULATOR (POUNDS MASS PER SECOND)	INJECTION SPILL (POUNDS MASS PER SECOND)	ENTHALPY BTU/LBM
29.0	227.0	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
29.5	223.9	24.524	0.000	0.68	0.38	1.000	8616.3	7493.5	0.0	86.70
29.7	222.2	18.860	0.000	1.05	0.43	1.000	8566.9	7443.9	0.0	86.68
30.1	221.3	1.698	0.071	1.26	1.08	1.000	8448.0	7324.5	0.0	86.64
30.3	221.4	2.910	0.104	1.30	1.48	1.000	8402.0	7278.3	0.0	86.62
30.4	221.4	2.896	0.125	1.32	1.68	1.000	8373.8	7250.0	0.0	86.61
30.5	221.4	2.949	0.157	1.35	1.91	0.838	8338.9	7215.0	0.0	86.60
31.5	221.7	2.247	0.333	1.50	3.78	0.606	8137.2	7012.5	0.0	86.53
34.1	223.0	1.915	0.547	1.74	9.00	0.494	7647.7	6521.3	0.0	86.33
38.1	225.0	3.354	0.657	2.00	15.82	0.579	6552.7	5488.7	0.0	86.01
39.2	225.1	3.720	0.676	2.11	15.99	0.598	6148.2	5116.2	0.0	85.89
41.2	225.5	3.504	0.708	2.29	16.00	0.598	5945.3	4910.8	0.0	85.76
43.9	226.3	3.287	0.731	2.51	16.00	0.596	5727.4	4686.4	0.0	85.59
51.6	229.5	2.925	0.759	3.01	16.00	0.590	5236.6	4180.5	0.0	85.16
60.8	234.2	2.670	0.774	3.50	16.00	0.585	4803.5	3734.5	0.0	84.72
62.2	235.0	2.208	0.782	3.57	16.00	0.545	1097.6	0.0	0.0	68.00
75.0	242.0	1.566	0.807	4.00	16.00	0.462	1129.2	0.0	0.0	68.00
96.2	242.4	1.510	0.810	4.51	16.00	0.460	1129.9	0.0	0.0	68.00
116.5	238.0	1.474	0.804	5.00	16.00	0.458	1130.6	0.0	0.0	68.00
138.2	237.6	1.425	0.804	5.52	16.00	0.456	1131.3	0.0	0.0	68.00
159.6	240.2	1.370	0.809	6.00	16.00	0.454	1132.0	0.0	0.0	68.00
184.2	244.3	1.302	0.820	6.51	16.00	0.451	1132.7	0.0	0.0	68.00
209.7	242.7	1.251	0.816	7.00	16.00	0.447	1133.5	0.0	0.0	68.00
238.2	242.9	1.184	0.818	7.53	16.00	0.440	1134.4	0.0	0.0	68.00
265.6	244.3	1.113	0.824	8.00	16.00	0.432	1135.2	0.0	0.0	68.00
296.2	243.0	1.080	0.818	8.50	16.00	0.433	1135.4	0.0	0.0	68.00
327.6	244.3	1.040	0.823	9.00	16.00	0.436	1135.4	0.0	0.0	68.00
362.2	243.4	1.007	0.819	9.53	16.00	0.440	1135.4	0.0	0.0	68.00
393.8	243.7	0.972	0.819	10.00	16.00	0.444	1135.4	0.0	0.0	68.00

TABLE 14.3.4-17

3425 MWT / DEPS - MIN SI / RHR X-TIE CLOSED
POST REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
291.2	207.0	258.3	281.3	99.7
296.2	207.3	258.6	281.1	99.5
311.2	205.8	256.8	282.5	99.5
336.2	205.7	256.7	282.6	99.0
356.2	204.1	254.7	284.2	98.9
376.2	204.3	254.9	284.1	98.5
381.2	202.3	253.6	285.1	98.6
406.2	302.0	253.3	285.3	98.1
471.2	199.8	249.3	288.5	97.5
476.2	200.3	249.9	288.0	97.3
486.2	199.3	248.7	289.0	97.3
506.2	199.3	248.7	289.0	96.9
536.2	197.5	246.4	290.8	96.7
541.2	197.7	246.7	290.6	96.5
571.2	196.1	244.6	292.3	96.3
576.2	196.4	245.1	291.9	96.1
616.2	194.4	242.6	293.9	95.7
851.2	194.4	242.6	293.9	95.7
851.3	80.6	100.2	407.7	115.1
881.2	80.0	99.4	408.3	114.6
1046.2	76.9	95.5	411.4	111.7
1121.2	75.8	94.2	412.5	115.2
1421.2	71.3	88.6	417.0	112.7
1506.2	70.1	87.1	418.2	110.3
1811.2	66.9	83.0	421.5	107.9
1816.2	66.8	83.0	421.5	107.7
2041.2	64.6	80.2	423.7	106.8
2201.2	63.7	79.0	424.7	150.1
2327.0	63.7	79.0	424.5	150.1

TABLE 14.3.4-18

3425 MWT / DEPS - MAX SI / RHR X-TIE OPEN
POST REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
393.9	100.4	126.4	1038.7	162.7
463.9	98.9	124.5	1040.2	161.7
468.9	100.0	125.9	1039.1	161.4
503.9	98.7	124.3	1040.4	161.0
508.9	99.8	125.6	1039.4	160.7
548.9	99.5	125.3	1039.6	160.0
633.9	97.8	123.2	1041.3	158.7
638.9	98.9	124.5	1040.2	158.4
683.9	97.6	122.9	1041.5	157.8
688.9	98.7	124.3	1040.4	157.5
738.9	97.2	122.4	1041.9	156.8
743.9	98.3	123.8	1040.8	156.5
788.9	97.0	121.1	1042.2	155.9
793.9	98.0	123.4	1041.1	155.6
843.9	96.7	121.8	1042.4	154.9
848.9	97.8	123.1	1041.4	154.6
933.9	96.7	121.8	1042.4	153.1
938.9	260.1	327.5	879.0	217.4
948.9	259.9	327.2	879.2	217.3
998.9	256.2	322.6	882.9	217.0
1003.9	271.6	341.9	867.6	218.7
1033.9	269.8	339.6	869.4	218.4
1078.9	265.9	334.8	873.2	218.2
1208.9	265.9	334.8	873.2	218.2
1209.0	75.1	94.0	1064.0	255.7
1223.9	74.9	93.8	1064.3	255.4
1468.9	71.2	89.2	1067.9	254.3
2083.9	65.0	81.3	1074.2	249.5
2388.1	63.9	80.0	1075.2	250.6

TABLE 14.3.4-19

MASS AND ENERGY BALANCES - MINIMUM SAFEGUARDS

3425 MWT / DEPS - MINS SI / RHR X-TIE CLOSED

MASS BALANCE

	TIME (SECONDS)	0.00	29.00	29.00	291.19	856.20	2326.97
	MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	769.41	769.41	769.41	769.41	769.41	769.41
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	122.42	398.32	1116.54
	TOTAL ADDED	0.00	0.00	0.00	122.42	398.32	1116.54
***	TOTAL AVAILABLE ***	769.41	769.41	769.41	891.83	1167.74	1885.95
DISTRIBUTION	REACTOR COOLANT	535.41	64.69	67.13	138.74	138.74	138.74
	ACCUMULATOR	234.00	163.58	161.14	0.00	0.00	0.00
	TOTAL CONTENTS	769.41	228.27	228.27	138.74	138.74	138.74
EFFLUENT	BREAK FLOW	0.00	541.14	541.14	753.09	1028.99	1747.20
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	541.14	541.14	753.09	1028.99	1747.20
***	TOTAL ACCOUNTABLE ***	769.41	769.41	769.41	891.83	1167.73	1885.94

ENERGY BALANCE

	TIME (SECONDS)	0.00	29.00	29.00	291.19	856.20	2326.97
	ENERGY (MILLION BTU)						
INITIAL ENERGY	IN RCS, ACC, S GEN	927.19	927.19	927.19	927.19	927.19	927.19
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	8.32	27.09	86.51
	DECAY HEAT	0.00	9.41	9.41	38.65	86.43	183.70
	HEAT FROM SECONDARY	0.00	-4.76	-4.76	-4.76	4.80	8.77
	TOTAL ADDED	0.00	4.65	4.65	42.22	118.31	278.97
***	TOTAL AVAILABLE ***	927.19	931.84	931.84	969.40	1045.50	1206.16

TABLE 14.3.4-19 (Continued)
MASS AND ENERGY BALANCES - MINIMUM SAFEGUARDS
3425 MWT / DEPS - MINS SI / RHR X-TIE CLOSED

DISTRIBUTION	REACTOR COOLANT	316.73	13.71	13.92	30.61	30.61	30.61
	ACCUMULATOR	20.94	14.64	14.42	0.00	0.00	0.00
	CORE STORED	34.87	16.98	16.98	3.19	3.16	2.92
	PRIMARY METAL	178.90	168.26	168.26	141.99	94.92	66.24
	SECONDARY METAL	103.29	103.36	103.36	95.23	75.05	46.66
	STEAM GENERATOR	272.46	277.77	277.77	252.45	203.14	128.85
	TOTAL CONTENTS	927.19	594.72	594.72	523.47	406.88	275.27
EFFLUENT	BREAK FLOW	0.00	337.12	337.12	438.11	630.79	923.06
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	337.12	337.12	438.11	630.79	923.06
***	TOTAL ACCOUNTABLE ***	927.19	931.84	931.84	961.58	1037.67	1198.33

TABLE 14.3.4-20
MASS AND ENERGY BALANCES - MAXIMUM SAFEGUARDS
3425 MWT / DEPS - MAX SI / RHR X-TIE OPEN

MASS BALANCE

	TIME (SECONDS)	0.00	29.00	29.00	393.83	1213.90	2388.05
	MASS (THOUSAND LBM)						
INITIAL	IN RCS AND ACC	769.41	769.41	769.41	769.41	769.41	769.41
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	411.13	1345.22	2682.73
	TOTAL ADDED	0.00	0.00	0.00	411.13	1345.22	2682.73
***	TOTAL AVAILABLE ***	769.41	769.41	769.41	1180.54	2114.63	3452.14
DISTRIBUTION	REACTOR COOLANT	535.41	64.69	66.97	140.95	140.95	140.95
	ACCUMULATOR	234.00	163.58	161.30	0.00	0.00	0.00
	TOTAL CONTENTS	769.41	228.27	228.27	140.95	140.95	140.95
EFFLUENT	BREAK FLOW	0.00	541.14	541.14	1039.59	1973.68	3311.19
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	541.14	541.14	1039.59	1973.68	3311.19
***	TOTAL ACCOUNTABLE ***	769.41	769.41	769.41	1180.54	2114.62	3452.14

ENERGY BALANCE

	TIME (SECONDS)	0.00	29.00	29.00	393.83	1213.90	2388.05
	ENERGY (MILLION BTU)						
INITIAL ENERGY	IN RCS, ACC, S GEN	927.19	927.19	927.19	927.19	927.19	927.19
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	27.96	118.73	326.04
	DECAY HEAT	0.00	9.41	9.41	48.24	112.56	187.25
	HEAT FROM SECONDARY	0.00	-4.76	-4.76	-4.76	8.77	8.77
	TOTAL ADDED	0.00	4.65	4.65	71.44	240.26	522.06
***	TOTAL AVAILABLE ***	927.19	931.84	931.84	998.63	1167.25	1449.25

TABLE 14.3.4-20 (Continued)

MASS AND ENERGY BALANCES - MAXIMUM SAFEGUARDS

3425 MWT / DEPS - MAX SI / RHR X-TIE OPEN

DISTRIBUTION	REACTOR COOLANT	316.73	13.71	13.91	30.75	30.75	30.75
	ACCUMULATOR	20.94	14.64	14.44	0.00	0.00	0.00
	CORE STORED	34.87	16.98	16.98	3.19	3.16	2.94
	PRIMARY METAL	178.90	168.26	168.26	140.34	90.37	67.21
	SECONDARY METAL	103.29	103.36	103.36	95.52	70.25	47.54
	STEAM GENERATOR	272.46	277.77	277.77	252.80	197.47	131.01
	TOTAL CONTENTS	927.19	594.72	594.72	522.61	387.01	279.46
EFFLUENT	BREAK FLOW	0.00	337.12	337.12	468.17	772.40	1161.95
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	337.12	337.12	468.17	772.40	1161.95
***	TOTAL ACCOUNTABLE ***	927.19	931.84	931.84	990.78	1159.41	1441.41

TABLE 14.3.4-21

TMD INPUT

Element	VOLM	ELJAC	M ICE	ARA HT	P STM	P AIR	TEMP	STATE
1	2.7250E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
2	3.8000E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
3	5.5000E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
4	3.5000E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
5	3.8000E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
6	2.2500E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
7	3.2950E+03	2.2000E-02	9.3576E+04	1.1290E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
8	3.2950E+03	2.2000E-02	9.3576E+04	1.1290E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
9	3.2950E+03	2.2000E-02	4.6788E+04	5.6450E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
10	3.8950E+03	2.2000E-02	1.1060E+05	1.3342E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
11	3.8950E+03	2.2000E-02	1.1060E+05	1.3342E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
12	3.8950E+03	2.2000E-02	5.5295E+04	6.6710E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
13	7.7890E+03	2.2000E-02	2.2118E+05	2.6685E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
14	7.7890E+03	2.2000E-02	2.2118E+05	2.6685E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
15	7.7890E+03	2.2000E-02	1.1059E+05	1.3343E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
16	5.3930E+03	2.2000E-02	1.5313E+05	1.8474E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
17	5.3930E+03	2.2000E-02	1.5313E+05	1.8474E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
18	5.3930E+03	2.2000E-02	7.6562E+04	9.2370E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
19	4.1940E+03	2.2000E-02	1.1910E+05	1.4369E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
20	4.1940E+03	2.2000E-02	1.1910E+05	1.4369E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
21	4.1940E+03	2.2000E-02	5.9549E+04	7.1850E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
22	4.1940E+03	2.2000E-02	1.1910E+05	1.4369E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
23	4.1940E+03	2.2000E-02	1.1910E+05	1.4369E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
24	4.1940E+03	2.2000E-02	5.9549E+04	7.1850E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
25	7.4239E+05	0.	0.	0.	7.0000E-02	1.4930E+01	7.5000E+01	1
26	1.0602E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
27	2.6423E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
28	1.0602E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
29	1.6317E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
30	1.0602E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
31	2.6423E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1

TABLE 14.3.4-21 (Cont'd)

TMD INPUT

Element	VOLM	ELJAC	M ICE	ARA HT	P STM	P AIR	TEMP	STATE
32	1.0602E+04	0.	0.	0.	1.4000E-01	1.4860E+01	9.8000E+01	1
33	1.9731E+04	0.	0.	0.	2.0600E-01	1.4794E+01	1.1000E+02	1
34	5.3850E+03	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
35	6.3650E+03	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
36	1.2729E+04	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
37	8.8130E+03	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
38	6.8570E+03	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
39	6.8540E+03	0.	0.	0.	7.0000E-02	1.4930E+01	3.0000E+01	1
40	2.7780E+03	2.2000E-02	4.6788E+04	5.6450E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
41	3.2830E+03	2.2000E-02	5.5295E+04	6.6710E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
42	6.5650E+03	2.2000E-02	1.1059E+05	1.3343E+04	7.0000E-02	1.4930E+01	3.0000E+01	1
43	4.5450E+03	2.2000E-02	7.6562E+04	9.2370E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
44	3.5350E+03	2.2000E-02	5.9549E+04	7.1850E+03	7.0000E-02	1.4930E+01	3.0000E+01	1
45	3.5350E+03	2.2000E-02	5.9549E+04	7.1850E+03	7.0000E-02	1.4930E+01	3.0000E+01	1

TABLE 14.3.4-22

TMD FLOW PATH INPUT DATA

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
1 to 2	16.7	635	0.3		.529
2 to 3	21.2	585	0.34		.488
3 to 4	26.2	740	0.22		.617
4 to 5	17.3	585	0.34		.488
5 to 6	16.7	635	0.3		.529
6 to 1	30.0	72	1.45		.060
26 to 32	34.81	20	1.6		.134
27 to 26	18.4	43	2.7		.125
28 to 3	29	40	4.2		.0349
29 to 30	21.81	11.09	1.5		.0322
30 to 28	47	55	1.6		.368
31 to 30	18.41	43	2.7		.125
32 to 30	70	100	0.5		.669
33 to 2	5.5	24	1.5		.038
40 to 1	10.36	121.9	0.89		.225
41 to 2	10.36	144	0.89		.225
42 to 3	10.36	288	0.89		.225

TABLE 14.3.4-22 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
43 to 4	10.36	199.4	0.89		.225
44 to 5	10.36	155.1	0.89		.225
45 to 6	10.36	155.1	0.89		.225
1 to 33	5.5	20	1.5		.038
2 to 27	15	154	4.2		.0349
3 to 33	8	56	1.5		.038
4 to 33	6.5	32.6	1.5		.038
5 to 31	15	154	4.2		.0349
6 to 33	5.2	18	1.5		.038
7 to 8	12.28	112.8		0.516	.727
8 to 9	12.28	112.8		0.516	.727
9 to 34	8.86	112.8	0.812	0.258	.727
10 to 11	12.28	131.3		0.516	.727
11 to 12	12.28	131.3		0.516	.727
12 to 35	8.86	131.3	0.812	0.258	.727
13 to 14	12.28	266.6		0.516	.727
14 to 15	12.28	266.6		0.516	.727
15 to 36	8.86	266.6	0.812	0.258	.727
16 to 17	12.28	184.6		0.516	.727

TABLE 14.3.4-22 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio $^{a}t/A$
32 to 31	18.4	43	2.7		.125
33 to 5	5.5	24	1.5		.038
34 to 25	2.8	233.8	1.45		.659
35 to 25	2.8	267.6	1.43		.659
36 to 25	2.8	539.5	1.43		.625
37 to 25	2.8	376.5	1.41		.636
38 to 25	2.8	289.4	1.44		.646
39 to 25	2.8	296.3	1.43		.249
40 to 7	8.222	106.7	0.227	0.142	.33
41 to 10	8.222	126.1	0.227	0.142	.33
42 to 13	8.222	252.2	0.227	0.142	.33
43 to 16	8.222	174.6	0.227	0.142	.33
44 to 19	8.222	135.8	0.227	0.142	.33
45 to 22	8.222	135.8	0.227	0.142	.33
40 to 41	13.8	24.7	7.5		.075
41 to 42	22.4	24.7	12.5		.046
42 to 43	25.3	24.7	12.5		.041
43 to 44	18.4	24.7	10.0		.056
44 to 45	16.1	24.7	10.0		.064

TABLE 14.3.4-22 (Cont'd)

Flow Path Element to Element	Flow Path Length (Ft)	Flow Area (Ft)	Loss Coefficient K	Flow Resistance fl/D	Area Ratio a_t/A
17 to 18	12.28	184.6		0.516	.727
18 o 37	8.86	184.6	0.812	0.258	.727
19 to 20	12.28	143.6		0.516	.727
20 to 21	12.28	143.6		0.516	.727
21 to 38	8.86	143.6	0.812	0.258	.727
22 to 23	12.28	143.6		0.516	.727
23 to 24	12.28	143.6		0.516	.727
24 to 39	8.86	143.6	0.812	0.258	.727
26 to 28	70.3	100	0.5		.669
27 to 29	14.3	10	3.0		.0291
28 to 27	18.4	43	2.7		.125
29 to 28	21.81	11.09	1.5		.0322
30 to 4	29	40	4.2		.0349
31 to 29	14.3	10	3.0		.0291

TABLE 14.3.4-23

UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER DER (4.6 FT^2) BREAK
FAILURE - MSIV

Time (sec)	Mass (lb/sec)	Energy (BTU x 106/SEC)
0.00	0.00	0.0
0.20	10430.00	1.250
3.60	6552.00	7.883
6.60	5612.00	6.748
12.80	4978.00	5.974
13.00	4913.00	5.895
13.20	4847.00	5.816
13.40	4781.00	5.737
13.60	4716.00	5.660
14.00	4587.00	5.504
14.40	4458.00	5.350
14.80	4332.00	5.198
15.00	4269.00	5.123
15.20	4206.00	5.047
15.60	4083.00	4.899
15.80	4022.00	4.826
16.00	3961.00	4.753
16.60	3782.00	4.538
17.20	3606.00	4.328
17.60	3492.00	4.190
17.80	3435.00	4.122
18.40	3268.00	3.921
18.60	3213.00	3.856
18.80	3158.00	3.790
19.20	3050.00	3.660
23.80	1876.00	2.251
28.80	1623.00	1.421
30.40	1575.00	1.883
36.40	1461.00	1.746
39.20	1431.00	1.708
50.70	1369.00	1.634
57.20	1356.00	1.618
106.20	1331.00	1.588
109.20	1331.00	1.587
111.20	1184.00	1.409
118.20	308.70	0.358
125.20	188.10	0.217
136.20	98.97	0.1139
602.70	93.24	0.1073

TABLE 14.3.4-24

UNIT 1/UNIT 2 STEAMLINE BREAK MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER SPLIT (0.86 FT^2) BREAK
FAILURE - AUXILIARY FEEDWATER RUNOUT PROTECTION

Time (sec)	Mass (lb/sec)	Energy (BTU x 106/SEC)
0.00	0.00	0.0000
0.20	1394.00	1.6690
1.60	1366.00	1.6370
2.00	1358.00	1.6270
2.40	1350.00	1.6170
2.80	1342.00	1.6080
4.20	1316.00	1.5770
4.40	1312.00	1.5730
8.60	1550.00	1.8540
9.40	1575.00	1.8840
12.00	1632.00	1.9500
12.60	1638.00	1.9570
15.80	1635.00	1.9530
18.00	1618.00	1.9340
21.40	1458.00	1.7460
22.60	1400.00	1.6790
23.60	1357.00	1.6280
23.80	1349.00	1.6180
25.00	1302.00	1.5630
32.00	1103.00	1.3260
32.20	1098.00	1.3210
33.80	1064.00	1.2810
42.00	928.70	1.1180
42.60	920.80	1.1090
43.20	913.10	1.1000
43.80	905.70	1.0910
44.40	898.40	1.0820
55.20	799.10	0.9625
67.20	732.60	0.8823
80.20	691.30	0.9625
82.20	686.60	0.8269
96.20	662.50	0.7977
98.70	659.50	0.7941
118.20	645.70	0.7775
124.20	643.60	0.7749
282.70	633.20	0.7623
285.20	633.10	0.7622
290.70	615.00	0.7402
292.70	579.70	0.6977
297.70	556.60	0.6695
302.70	490.40	0.5896
320.20	304.70	0.3643
330.20	238.70	0.2845
340.20	206.50	0.2456
352.70	190.20	0.2259
525.20	181.90	0.2160
535.20	182.00	0.2160
600.20	182.10	0.2162
605.20	190.70	0.2258

TABLE 14.3.4-25

1.4 FT2 DOUBLE-ENDED STEAMLINE BREAKS

Operating Power, %	102	102	70	30	0
Aux. Feed Failure	w/o	w	w/o	w/o	w/o
MSIV Failure	w	w/o	w	w	w
T_{max}, °F	324.15	323.05	323.47	322.52	322.48
Time of T_{max}, sec	6.87	6.67	7.77	10.11	10.11

4.6 FT2 DOUBLE-ENDED STEAMLINE BREAKS

Operating Power, %	102	102	70	70	30	0
Aux. Feed Failure	w/o	w	w	w	w/o	w/o
MSIV Failure	w	w/o	w/o	w/o	w	w
T_{max}, °F	324.89	323.20	323.88	322.20	322.28	321.15
Time of T_{max}, sec	6.39	6.07	6.15	5.93	6.23	5.95

TABLE 14.3.4-26

STEAMLINE RUPTURES

Size of Break, ft^2	0.86	0.86	0.908	0.908	0.942	0.942	0.4	0.4
Operating Power %	102	102	70	70	30	30	Hot Shutdown	Hot Shutdown
Aux. Feed Failure	w/o	w	w/o	w	w/o	w	w/o	w
MSIV Failure	w	w/o	w	w/o	w	w/o	w	w/o
T_{max}, ^oF	323.49	324.38	323.88	323.76	324.21	324.25	313.28	311.21
Time of T_{max}, sec	51.49	50.72	50.11	50.43	49.84	49.28	56.28	57.15

TABLE 14.3.4-27

PRESSURIZER ENCLOSURE NODALIZATION VOLUMES

NODE				VOLUME (ft3)		
D. C. Cook	Watts Bar	McGuire		D. C. Cook	Watts Bar	McGuire
46	51	1		2,005	2,508	1,763
47	52	3		391	438	477
48	53	4		557	843	906
49	54	2		457	848	886
			Total	3,410	4,637	4,032

TABLE 14.3.4-28

PRESSURIZER ENCLOSURE NODALIZATION HYDRAULIC DATA

Donald C. Cook Unit 2

	Flow Path	K	f	L_I (ft)	D_H (ft)	L_{eq} (ft)	A $(ft)^2$	a_T (ft^2)	T/..
48A	48-47	0.888	.025	10.302	2.50	10.302	111.06	29.23	0.26
49R	49-47	1.064	.025	11.793	2.50	14.99	111.06	29.23	0.26
48R	48-49	1.268	.025	7.883	4.50	7.883	111.06	52.61	0.47
46H	46-47	1.264	.025	15.54	2.50	13.86	44.23	17.93	0.41
46R	46-48	0.856	.025	17.86	5.19	15.80	41.93	27.24	0.65
46A	46-49	0.788	.025	17.43	3.66	15.40	41.62	25.14	0.60
47H	47-Lower(4)	1.91	.025	11.69	2.50	11.69	17.93	17.93	1.00
48H	48-Lower(4)	1.665	.025	8.87	3.84	9.22	25.93	18.93	0.73
49H	49-Lower(4)	1.291	.025	8.80	2.66	9.05	24.35	17.60	0.72

Watts Bar FSAR

Flow Path	K	F	L_I (ft)	D_H (ft)	A (ft^2)	L_{eq} (ft)	a/A
51 to 52	0.50	0.02	13.3	2.96	18.6	11.8	0.12
51 to 53	0.50	0.02	15.1	5.82	40.3	12.2	0.26
51 to 54	0.50	0.02	15.1	5.82	40.3	12.2	0.26
53 to 52	0.04	0.02	8.1	3.00	31.6	6.6	0.25
54 to 52	0.04	0.02	8.1	3.00	31.6	6.6	0.25
52 to lower compartment	1.50	0.02	11.8	2.96	18.6	11.8	1.00
53 to lower compartment	1.50	0.02	10.2	3.93	28.6	6.9	0.73
54 to lower compartment	1.50	0.02	10.3	5.64	30.7	7.3	0.74

McGuire FSAR

Flow Path (Element to Element)	K	fl/d	Inertia Length (ft)	Flow Area (ft^2)	Contraction a_t/A_u
1-2	0.324	----	14.3	34.5	0.233
1-3	0.439	----	13.55	18.2	0.123
1-4	0.381	----	14.3	35.25	0.239
2-5	1.59	----	11.0	29.5	0.355
3-5	1.66	----	6.16	8.2	0.451
4-5	1.58	----	11.04	30.25	0.358
2-3	0.416	----	9.68	29.8	0.336
3-4	0.534	----	9.64	29.8	0.333
4-2	0.531	----	7.51	61.7	0.631

TABLE 14.3.4-29

DIFFERENTIAL PRESSURE

$\Delta P(46-25)$	$\Delta P(47-25)$	$\Delta P(48-25)$	$\Delta P(49-25)$
5.82 psi	4.41 psi	4.65 psi	4.63 psi
@ .058 sec.	@ .070 sec.	@ .063 sec.	@ .063 sec.

$\Delta P(47-48)$	$\Delta P(47-49)$	$\Delta P(48-49)$
-.27 psi	-.24 psi	-.12 psi
@ .057 sec.	@ .059 sec.	@ .025 sec.

PEAK PRESSURE

P(46)	P(47)	P(48)	P(49)
20.82 psia	19.41 psia	19.65 psia	19.63 psia
@ .055 sec.	@ .066 sec.	@ .061 sec.	@ .063 sec.

TABLE 14.3.4-30

Summary - Break Mass Flow and Energy Flow

Time (S)	Mass Flow (LB/S)	Energy Flow (BTU/S)	Avg Enthalpy (BTU/LB)
.00000	0.	0.	0.00
.00101	1.8429935E+03	1.2012554E+06	651.80
.00201	2.0768869E+03	1.3312080E+06	640.96
.00302	2.0954232E+03	1.3410507E+06	639.99
.00401	2.0943466E+03	1.3398926E+06	639.77
.00502	2.0906198E+03	1.3372569E+06	639.65
.00600	2.0847010E+03	1.3334349E+06	639.63
.00703	2.0776894E+03	1.3290038E+06	639.65
.00800	2.0718446E+03	1.3252736E+06	639.66
.00901	2.0684832E+03	1.3229372E+06	639.57
.01000	2.0667746E+03	1.3215515E+06	639.43
.01102	2.0657651E+03	1.3205642E+06	639.26
.01200	2.0657312E+03	1.3202011E+06	639.07
.01301	2.0710642E+03	1.3227266E+06	638.67
.01400	2.0895916E+03	1.3326883E+06	637.77
.01505	2.1237904E+03	1.3513713E+06	636.30
.01605	2.1428041E+03	1.3615888E+06	635.42
.01705	2.1327342E+03	1.3556010E+06	635.62
.01804	2.1286790E+03	1.3530305E+06	635.62
.01905	2.1408255E+03	1.3594780E+06	635.03
.02004	2.1330134E+03	1.3547994E+06	635.16
.02108	2.1170413E+03	1.3456154E+06	635.61
.02207	2.1196063E+03	1.3467887E+06	635.40
.02308	2.1314002E+03	1.3530903E+06	634.84
.02400	2.1400977E+03	1.3576929E+06	634.41
.02501	2.1475536E+03	1.3615802E+06	634.01
.02602	2.1559827E+03	1.3660339E+06	633.60
.02707	2.1678910E+03	1.3724086E+06	633.06
.02804	2.1768474E+03	1.3771613E+06	632.64
.02907	2.1816274E+03	1.3795673E+06	632.36
.03009	2.1827097E+03	1.3799305E+06	632.21
.03107	2.1841837E+03	1.3805245E+06	632.06
.03212	2.1872679E+03	1.3820107E+06	631.84
.03301	2.1894643E+03	1.3830366E+06	631.68
.03411	2.1903771E+03	1.3833149E+06	631.54
.03504	2.1897285E+03	1.3827485E+06	631.47
.03607	2.1861418E+03	1.3805329E+06	631.49
.03703	2.1786958E+03	1.3761844E+06	631.66
.03807	2.1711481E+03	1.3717868E+06	631.83
.03906	2.1650758E+03	1.3682259E+06	631.95
.04009	2.1587371E+03	1.3645103E+06	632.09
.04102	2.1527706E+03	1.3610346E+06	632.22
.04212	2.1468107E+03	1.3575467E+06	632.36
.04305	2.1432206E+03	1.3554088E+06	632.42
.04406	2.1404771E+03	1.3537320E+06	632.44

TABLE 14.3.4-30 (Continued)

.04510	2.1384998E+03	1.3524847E+06	632.45
.04601	2.1374125E+03	1.3517571E+06	632.43
.04705	2.1369632E+03	1.3513688E+06	632.38
.04809	2.1372622E+03	1.3514036E+06	632.31
.04911	2.1379671E+03	1.3516676E+06	632.22
.05006	2.1388546E+03	1.3520428E+06	632.13
.05108	2.1400630E+03	1.3525916E+06	632.03
.05207	2.1411605E+03	1.3530872E+06	631.94
.05318	2.1418610E+03	1.3533545E+06	631.86
.05404	2.1420277E+03	1.3533548E+06	631.81
.05515	2.1419234E+03	1.3531771E+06	631.76
.05609	2.1415416E+03	1.3528611E+06	631.72
.05711	2.1410383E+03	1.3524782E+06	631.69
.05812	2.1409677E+03	1.3523442E+06	631.65
.05901	2.1414214E+03	1.3525173E+06	631.60
.06004	2.1427085E+03	1.3531458E+06	631.51
.06104	2.1448478E+03	1.3542536E+06	631.40
.06207	2.1481439E+03	1.3559989E+06	631.24
.06303	2.1514351E+03	1.3577536E+06	631.09
.06402	2.1550387E+03	1.3596806E+06	630.93
.06501	2.1585740E+03	1.3615691E+06	630.77
.06604	2.1618098E+03	1.3632912E+06	630.62
.06710	2.1647229E+03	1.3648345E+06	630.49
.06814	2.1673799E+03	1.3662365E+06	630.36
.06905	2.1694828E+03	1.3673439E+06	630.26
.07008	2.1716796E+03	1.3684953E+06	630.16
.07111	2.1735789E+03	1.3694836E+06	630.06
.07213	2.1750499E+03	1.3702354E+06	629.98
.07317	2.1759218E+03	1.3706558E+06	629.92
.07412	2.1760897E+03	1.3706871E+06	629.89
.07523	2.1755041E+03	1.3702969E+06	629.88
.07602	2.1743802E+03	1.3696191E+06	629.89
.07705	2.1726900E+03	1.3686222E+06	629.92
.07801	2.1708169E+03	1.3675296E+06	629.96
.07902	2.1688239E+03	1.3663702E+06	630.01
.08006	2.1667850E+03	1.3651854E+06	630.05
.08100	2.1651864E+03	1.3642490E+06	630.08
.08204	2.1632686E+03	1.3631342E+06	630.13
.08306	2.1613432E+03	1.3620171E+06	630.17
.08410	2.1593726E+03	1.3608757E+06	630.22
.08504	2.1576149E+03	1.3598670E+06	630.26
.08603	2.1556593E+03	1.3587375E+06	630.31
.08708	2.1538655E+03	1.3576993E+06	630.35
.08814	2.1523767E+03	1.3568314E+06	630.39
.08915	2.1513646E+03	1.3562298E+06	630.40
.09005	2.1510544E+03	1.3560230E+06	630.40
.09115	2.1514210E+03	1.3561899E+06	630.37
.09214	2.1524508E+03	1.3567285E+06	630.32

TABLE 14.3.4-30 (Continued)

.09307	2.1540139E+03	1.3575665E+06	630.25
.09405	2.1562098E+03	1.3587565E+06	630.16
.09504	2.1589196E+03	1.3602332E+06	630.05
.09601	2.1619926E+03	1.3619135E+06	629.93
.09712	2.1660659E+03	1.3641459E+06	629.78
.09811	2.1700593E+03	1.3663392E+06	629.63
.09910	2.1743398E+03	1.3686932E+06	629.48
.10011	2.1788810E+03	1.3711929E+06	629.31
.10505	2.1997980E+03	1.3827080E+06	628.56
.11017	2.2084042E+03	1.3873725E+06	628.22
.11511	2.1979605E+03	1.3814517E+06	628.52
.12010	2.1755220E+03	1.3688892E+06	629.22
.12501	2.1516732E+03	1.3555705E+06	630.01
.13001	2.1350205E+03	1.3462681E+06	630.56
.13501	2.1319480E+03	1.3445234E+06	630.65
.14009	2.1415298E+03	1.3498034E+06	630.30
.14510	2.1540075E+03	1.3566900E+06	629.84
.15015	2.1608228E+03	1.3604378E+06	629.59
.15500	2.1603593E+03	1.3601520E+06	629.60
.16007	2.1547404E+03	1.3570065E+06	629.78
.16515	2.1446526E+03	1.3513889E+06	630.12
.17002	2.1327374E+03	1.3447619E+06	630.53
.17509	2.1250311E+03	1.3404793E+06	630.80
.18010	2.1255323E+03	1.3407503E+06	630.78
.18500	2.1307501E+03	1.3436315E+06	630.59
.19010	2.1359241E+03	1.3464861E+06	630.40
.19507	2.1377315E+03	1.3474736E+06	630.33
.20008	2.1357354E+03	1.3463523E+06	630.39
.21002	2.1305966E+03	1.3434798E+06	630.57
.22009	2.1455142E+03	1.3517412E+06	630.03
.23010	2.1609519E+03	1.3602792E+06	629.48
.24007	2.1500994E+03	1.3542251E+06	629.84
.25001	2.1319210E+03	1.3441210E+06	630.47
.26002	2.1312847E+03	1.3437534E+06	630.49
.27006	2.1408201E+03	1.3490219E+06	630.14
.28015	2.1382240E+03	1.3475602E+06	630.22
.29011	2.1219655E+03	1.3385282E+06	630.80
.30028	2.1125694E+03	1.3333092E+06	631.13
.31011	2.1146636E+03	1.3344534E+06	631.05
.32004	2.1134582E+03	1.3337664E+06	631.08
.33005	2.1102424E+03	1.3319670E+06	631.19
.34002	2.1162685E+03	1.3352858E+06	630.96
.35002	2.1252674E+03	1.3402496E+06	630.63
.36002	2.1249488E+03	1.3400427E+06	630.62
.37007	2.1182970E+03	1.3363273E+06	630.85
.38010	2.1179299E+03	1.3360961E+06	630.85
.39027	2.1240125E+03	1.3394390E+06	630.62
.40003	2.1244819E+03	1.3396673E+06	630.59
.41001	2.1157197E+03	1.3347842E+06	630.89
.42004	2.1079031E+03	1.3304251E+06	631.16

TABLE 14.3.4-30 (Continued)

.43006	2.1057949E+03	1.3292307E+06	631.23
.44011	2.1038434E+03	1.3281221E+06	631.28
.45008	2.1005501E+03	1.3262686E+06	631.39
.46002	2.1008990E+03	1.3264325E+06	631.36
.47013	2.1046381E+03	1.3284716E+06	631.21
.48001	2.1065466E+03	1.3294937E+06	631.12
.49011	2.1055653E+03	1.3289117E+06	631.14
.50010	2.1065499E+03	1.3294210E+06	631.09
.51005	2.1109307E+03	1.3318103E+06	630.91
.52014	2.1132353E+03	1.3330479E+06	630.81
.53006	2.1104929E+03	1.3314882E+06	630.89
.54010	2.1067324E+03	1.3293690E+06	631.01
.55002	2.1047031E+03	1.3282077E+06	631.07
.56003	2.1025984E+03	1.3270031E+06	631.13
.57000	2.0994619E+03	1.3252280E+06	631.22
.58000	2.0974864E+03	1.3240966E+06	631.28
.59023	2.0978998E+03	1.3242862E+06	631.24
.60010	2.0987566E+03	1.3247206E+06	631.19
.61005	2.0988030E+03	1.3247041E+06	631.17
.62019	2.0998437E+03	1.3252383E+06	631.11
.63007	2.1029363E+03	1.3269102E+06	630.98
.64009	2.1057724E+03	1.3284358E+06	630.85
.65011	2.1063355E+03	1.3287028E+06	630.81
.66008	2.1058521E+03	1.3283898E+06	630.81
.67006	2.1056166E+03	1.3282154E+06	630.80
.68003	2.1048258E+03	1.3277315E+06	630.80
.69009	2.1027606E+03	1.3265436E+06	630.86
.70006	2.1004766E+03	1.3252341E+06	630.92
.71021	2.0994186E+03	1.3246041E+06	630.94
.72013	2.0991217E+03	1.3243955E+06	630.93
.73018	2.0986674E+03	1.3240994E+06	630.92
.74006	2.0987148E+03	1.3240815E+06	630.90
.75009	2.1001279E+03	1.3248191E+06	630.83
.76008	2.1022067E+03	1.3259240E+06	630.73
.77007	2.1038110E+03	1.3267654E+06	630.65
.78001	2.1049477E+03	1.3273479E+06	630.58
.79005	2.1060244E+03	1.3278975E+06	630.52
.80008	2.1065644E+03	1.3281484E+06	630.48
.81002	2.1059659E+03	1.3277697E+06	630.48
.82003	2.1046670E+03	1.3270043E+06	630.51
.83006	2.1036100E+03	1.3263731E+06	630.52
.84009	2.1028557E+03	1.3259083E+06	630.53
.85013	2.1019975E+03	1.3253876E+06	630.54
.86002	2.1012207E+03	1.3249123E+06	630.54
.87004	2.1011387E+03	1.3248209E+06	630.53
.88001	2.1018523E+03	1.3251709E+06	630.48
.89009	2.1029599E+03	1.3257378E+06	630.42
.90001	2.1041960E+03	1.3263762E+06	630.35
.91007	2.1055640E+03	1.3270867E+06	630.28

TABLE 14.3.4-30 (Continued)

.92003	2.1066767E+03	1.3276561E+06	630.21
.93004	2.1070790E+03	1.3278333E+06	630.18
.94009	2.1068436E+03	1.3276548E+06	630.16
.95015	2.1064606E+03	1.3273966E+06	630.15
.96022	2.1060345E+03	1.3271155E+06	630.15
.97001	2.1052992E+03	1.3266634E+06	630.15
.98005	2.1042714E+03	1.3260489E+06	630.17
.99010	2.1033853E+03	1.3255134E+06	630.18
1.00003	2.1030192E+03	1.3252670E+06	630.17
1.00015	2.1030185E+03	1.3252660E+06	630.17
1.01010	2.1031598E+03	1.3252999E+06	630.15
1.02001	2.1036738E+03	1.3255407E+06	630.11
1.03002	2.1044850E+03	1.3259460E+06	630.06
1.04010	2.1053738E+03	1.3263933E+06	630.00
1.05011	2.1060053E+03	1.3266987E+06	629.96
1.06006	2.1063215E+03	1.3268299E+06	629.93
1.07003	2.1065103E+03	1.3268901E+06	629.90
1.08001	2.1065562E+03	1.3268717E+06	629.88
1.09002	2.1062535E+03	1.3266603E+06	629.87
1.10005	2.1054973E+03	1.3261987E+06	629.87
1.11001	2.1045159E+03	1.3256120E+06	629.89
1.12004	2.1036583E+03	1.3250948E+06	629.90
1.13020	2.1030886E+03	1.3247376E+06	629.90
1.14007	2.1028399E+03	1.3245581E+06	629.89
1.15004	2.1028392E+03	1.3245158E+06	629.87
1.16002	2.1030452E+03	1.3245891E+06	629.84
1.17008	2.1033358E+03	1.3247086E+06	629.81
1.18009	2.1036084E+03	1.3248175E+06	629.78
1.19008	2.1038945E+03	1.3249345E+06	629.75
1.20005	2.1041572E+03	1.3250391E+06	629.72
1.21005	2.1041987E+03	1.3250213E+06	629.70
1.22004	2.1038425E+03	1.3247830E+06	629.70
1.23001	2.1031332E+03	1.3243498E+06	629.70
1.24003	2.1022976E+03	1.3238470E+06	629.71
1.25007	2.1014833E+03	1.3233569E+06	629.73
1.26010	2.1007977E+03	1.3229378E+06	629.73
1.27007	2.1002438E+03	1.3225929E+06	629.73
1.28000	2.0998158E+03	1.3223173E+06	629.73
1.29006	2.0995386E+03	1.3221250E+06	629.72
1.30006	2.0993915E+03	1.3220048E+06	629.71
1.31013	2.0994058E+03	1.3219736E+06	629.69
1.32023	2.0995298E+03	1.3220039E+06	629.67
1.33005	2.0995986E+03	1.3220040E+06	629.65
1.34003	2.0994684E+03	1.3218931E+06	629.63
1.35003	2.0990790E+03	1.3216402E+06	629.63
1.36003	2.0984545E+03	1.3212564E+06	629.63
1.37021	2.0977429E+03	1.3208245E+06	629.64
1.38008	2.0969880E+03	1.3203695E+06	629.65
1.39003	2.0962009E+03	1.3198970E+06	629.66
1.40007	2.0954355E+03	1.3194364E+06	629.67

TABLE 14.3.4-30 (Continued)

1.41000	2.0947605E+03	1.3190272E+06	629.68
1.42008	2.0941729E+03	1.3186648E+06	629.68
1.43000	2.0937703E+03	1.3184063E+06	629.68
1.44009	2.0935338E+03	1.3182384E+06	629.67
1.45002	2.0933865E+03	1.3181211E+06	629.66
1.46007	2.0932071E+03	1.3179859E+06	629.65
1.47003	2.0929094E+03	1.3177850E+06	629.64
1.48002	2.0924851E+03	1.3175144E+06	629.64
1.49013	2.0919562E+03	1.3171856E+06	629.64
1.50005	2.0913287E+03	1.3168027E+06	629.65
1.51010	2.0906026E+03	1.3163655E+06	629.66
1.52003	2.0897881E+03	1.3158794E+06	629.67
1.53000	2.0889400E+03	1.3153753E+06	629.69
1.54004	2.0881300E+03	1.3148919E+06	629.70
1.55003	2.0874180E+03	1.3144632E+06	629.71
1.56002	2.0868363E+03	1.3141071E+06	629.71
1.57009	2.0863657E+03	1.3138119E+06	629.71
1.58003	2.0859830E+03	1.3135658E+06	629.71
1.59008	2.0856058E+03	1.3133222E+06	629.71
1.60014	2.0852041E+03	1.3130654E+06	629.71
1.61003	2.0847664E+03	1.3127889E+06	629.71
1.62002	2.0842650E+03	1.3124767E+06	629.71
1.63000	2.0836788E+03	1.3121187E+06	629.71
1.64000	2.0829808E+03	1.3116980E+06	629.72
1.65005	2.0822069E+03	1.3112361E+06	629.73
1.66001	2.0813530E+03	1.3107288E+06	629.75
1.67010	2.0805367E+03	1.3102429E+06	629.76
1.68013	2.0797667E+03	1.3097828E+06	629.77
1.69008	2.0790890E+03	1.3093745E+06	629.78
1.70008	2.0784817E+03	1.3090042E+06	629.79
1.71005	2.0779442E+03	1.3086733E+06	629.79
1.72006	2.0774536E+03	1.3083681E+06	629.79
1.73003	2.0769919E+03	1.3080787E+06	629.79
1.74005	2.0765358E+03	1.3077923E+06	629.80
1.75017	2.0760313E+03	1.3074793E+06	629.80
1.76021	2.0754782E+03	1.3071398E+06	629.80
1.77003	2.0748364E+03	1.3067506E+06	629.81
1.78009	2.0741177E+03	1.3063189E+06	629.82
1.79004	2.0733817E+03	1.3058784E+06	629.83
1.80001	2.0725989E+03	1.3054116E+06	629.84
1.81000	2.0718428E+03	1.3049592E+06	629.85
1.82003	2.0711342E+03	1.3045334E+06	629.86
1.83004	2.0704825E+03	1.3041393E+06	629.87
1.84005	2.0698712E+03	1.3037673E+06	629.88
1.85006	2.0693256E+03	1.3034315E+06	629.88
1.86007	2.0688391E+03	1.3031290E+06	629.88
1.87004	2.0683537E+03	1.3028266E+06	629.89
1.88007	2.0678600E+03	1.3025183E+06	629.89
1.89005	2.0673398E+03	1.3021967E+06	629.89

TABLE 14.3.4-30 (Continued)

1.90017	2.0667719E+03	1.3018481E+06	629.89
1.91001	2.0661520E+03	1.3014713E+06	629.90
1.92008	2.0654916E+03	1.3010717E+06	629.91
1.93007	2.0647986E+03	1.3006537E+06	629.92
1.94002	2.0641131E+03	1.3002406E+06	629.93
1.95002	2.0634306E+03	1.2998286E+06	629.94
1.96004	2.0627835E+03	1.2994362E+06	629.94
1.97001	2.0621900E+03	1.2990740E+06	629.95
1.98001	2.0616378E+03	1.2987343E+06	629.95
1.99001	2.0611133E+03	1.2984091E+06	629.96
2.00006	2.0606351E+03	1.2981098E+06	629.96
2.01017	2.0601755E+03	1.2978210E+06	629.96
2.02002	2.0596908E+03	1.2975184E+06	629.96
2.03010	2.0591909E+03	1.2972067E+06	629.96
2.04002	2.0586654E+03	1.2968812E+06	629.96
2.05018	2.0581050E+03	1.2965364E+06	629.97
2.06005	2.0575012E+03	1.2961667E+06	629.97
2.07004	2.0568982E+03	1.2957985E+06	629.98
2.08009	2.0562820E+03	1.2954219E+06	629.98
2.09012	2.0556887E+03	1.2950588E+06	629.99
2.10002	2.0551138E+03	1.2947052E+06	629.99
2.11010	2.0545765E+03	1.2943725E+06	629.99
2.12013	2.0540837E+03	1.2940645E+06	630.00
2.13010	2.0536205E+03	1.2937729E+06	630.00
2.14019	2.0531798E+03	1.2934939E+06	630.00
2.15007	2.0527452E+03	1.2932172E+06	630.00
2.16008	2.0523014E+03	1.2929356E+06	629.99
2.17003	2.0518411E+03	1.2926450E+06	629.99
2.18009	2.0513588E+03	1.2923422E+06	629.99
2.19004	2.0508552E+03	1.2920279E+06	629.99
2.20009	2.0503249E+03	1.2916980E+06	630.00
2.21011	2.0497884E+03	1.2913649E+06	630.00
2.22017	2.0492528E+03	1.2910321E+06	630.00
2.23014	2.0487253E+03	1.2907039E+06	630.00
2.24005	2.0482384E+03	1.2903987E+06	630.00
2.25001	2.0477626E+03	1.2900986E+06	630.00
2.26009	2.0473225E+03	1.2898187E+06	630.00
2.27004	2.0468923E+03	1.2895433E+06	630.00
2.28001	2.0465016E+03	1.2892908E+06	630.00
2.29005	2.0460991E+03	1.2890308E+06	629.99
2.30001	2.0456898E+03	1.2887674E+06	629.99
2.31012	2.0452463E+03	1.2884848E+06	629.99
2.32007	2.0448340E+03	1.2882196E+06	629.99
2.33014	2.0443954E+03	1.2879401E+06	629.99
2.34004	2.0439367E+03	1.2876484E+06	629.98
2.35001	2.0434889E+03	1.2873633E+06	629.98
2.36011	2.0430542E+03	1.2870850E+06	629.98
2.37008	2.0426322E+03	1.2868139E+06	629.98
2.38012	2.0422433E+03	1.2865609E+06	629.97
2.39005	2.0418693E+03	1.2863161E+06	629.97
2.40004	2.0415246E+03	1.2860877E+06	629.96

TABLE 14.3.4-30 (Continued)

2.41008	2.0412113E+03	1.2858761E+06	629.96
2.42005	2.0408975E+03	1.2856645E+06	629.95
2.43008	2.0406149E+03	1.2854698E+06	629.94
2.44013	2.0403182E+03	1.2852674E+06	629.93
2.45001	2.0400192E+03	1.2850640E+06	629.93
2.46020	2.0397042E+03	1.2848514E+06	629.92
2.47007	2.0393890E+03	1.2846382E+06	629.91
2.48001	2.0390787E+03	1.2844277E+06	629.91
2.49005	2.0387565E+03	1.2842108E+06	629.90
2.50017	2.0384506E+03	1.2840029E+06	629.89
2.51008	2.0381551E+03	1.2838008E+06	629.88
2.52000	2.0378823E+03	1.2836107E+06	629.87
2.53003	2.0376188E+03	1.2834258E+06	629.87
2.54003	2.0373717E+03	1.2832502E+06	629.86
2.55004	2.0371414E+03	1.2830827E+06	629.84
2.56007	2.0369174E+03	1.2829200E+06	629.83
2.57007	2.0367206E+03	1.2827722E+06	629.82
2.58005	2.0365020E+03	1.2826116E+06	629.81
2.59008	2.0362817E+03	1.2824507E+06	629.80
2.60014	2.0360644E+03	1.2822909E+06	629.79
2.61006	2.0358300E+03	1.2821214E+06	629.78
2.62001	2.0355880E+03	1.2819481E+06	629.77
2.63008	2.0353598E+03	1.2817817E+06	629.76
2.64006	2.0351206E+03	1.2816095E+06	629.75
2.65009	2.0349045E+03	1.2814499E+06	629.73
2.66003	2.0346896E+03	1.2812913E+06	629.72
2.67013	2.0344844E+03	1.2811382E+06	629.71
2.68004	2.0342912E+03	1.2809913E+06	629.70
2.69000	2.0341169E+03	1.2808550E+06	629.69
2.70009	2.0339484E+03	1.2807220E+06	629.67
2.71006	2.0337700E+03	1.2805830E+06	629.66
2.72003	2.0335952E+03	1.2804464E+06	629.65
2.73003	2.0334197E+03	1.2803093E+06	629.63
2.74018	2.0332272E+03	1.2801628E+06	629.62
2.75014	2.0330332E+03	1.2800152E+06	629.61
2.76007	2.0328279E+03	1.2798616E+06	629.60
2.77002	2.0326244E+03	1.2797091E+06	629.58
2.78002	2.0324222E+03	1.2795567E+06	629.57
2.79008	2.0322290E+03	1.2794097E+06	629.56
2.80006	2.0320428E+03	1.2792666E+06	629.55
2.81009	2.0318584E+03	1.2791244E+06	629.53
2.82006	2.0316917E+03	1.2789921E+06	629.52
2.83006	2.0315113E+03	1.2788520E+06	629.51
2.84015	2.0313383E+03	1.2787161E+06	629.49
2.85005	2.0311662E+03	1.2785814E+06	629.48
2.86003	2.0309809E+03	1.2784384E+06	629.47
2.87004	2.0307972E+03	1.2782965E+06	629.46
2.88006	2.0306091E+03	1.2781518E+06	629.44
2.89010	2.0304103E+03	1.2780018E+06	629.43
2.90019	2.0302106E+03	1.2778513E+06	629.42
2.91001	2.0299968E+03	1.2776931E+06	629.41

TABLE 14.3.4-30 (Continued)

2.92007	2.0297956E+03	1.2775412E+06	629.39
2.93007	2.0295874E+03	1.2773863E+06	629.38
2.94014	2.0293708E+03	1.2772257E+06	629.37
2.95003	2.0291890E+03	1.2770853E+06	629.36
2.96002	2.0289876E+03	1.2769341E+06	629.35
2.97013	2.0287920E+03	1.2767858E+06	629.33
2.98010	2.0285960E+03	1.2766370E+06	629.32
2.99010	2.0283918E+03	1.2764839E+06	629.31
3.00017	2.0281887E+03	1.2763315E+06	629.30

TABLE 14.3.4-31

COMPARISON OF PEAK DIFFERENTIAL PRESSURES

1. Across Enclosure Walls

Nodes			Differential Press. (PSI)		
D. C. Cook	Watts Bar	McGuire	D. C. Cook*	Watts Bar	McGuire
46	51	1	5.82	13.2	18.4
47	52	3	4.41	12.1	17.6
48	53	4	4.65	12.1	17.6
49	54	2	4.63	12.1	17.6

*The enclosure design pressure is 80 psig.

2. Across Pressurizer Vessel

Nodes			Differential Press. (PSI)		
D. C. Cook	Watts Bar	McGuire	D. C. Cook**	Watts Bar	McGuire
47-48	52-53	3-4	0.27	0.49	<0.2
47-49	52-54	3-2	0.24	0.38	<0.2
48-49	53-54	4-2	0.12	0.12	<0.2

**Design of pressure vessel supports accounts for a differential pressure across the pressurizer vessel of as high as 1.3 psi.

TABLE 14.3.4-32

PEAK DIFFERENTIAL PRESSURE (PSI)

BREAK AT OUTLET NOZZLE

Across Structures

DP46-25	- 34.19	PSI	@	1.219	SEC
DP47-25	- 28.88	"	"	1.245	"
DP48-25	- 28.91	"	"	1.245	"
DP49-25	- 29.18	"	"	1.241	"
DP50-25	- 29.51	"	"	1.237	"
DP51-25	- 13.88	"	"	.045	"
DP52-25	- 12.58	"	"	1.267	"
DP53-25	- 12.67	"	"	.059	"
DP54-25	- 14.13	"	"	.050	"
DP46-55	- 26.05	"	"	.018	"
DP47-56	- 17.14	"	"	.029	"
DP48-57	- 17.17	"	"	.031	"
DP51-60	- 9.31	"	"	.038	"
DP52-61	- 7.96	"	"	.035	"

Across Steam Generator Vessel:

DP47-49	- -2.74	PSI	@	.026	SEC
DP48-50	- -4.74	"	"	.022	"
DP51-53	- 1.88	"	"	.044	"
DP52-54	- -2.98	"	"	.029	"

TABLE 14.3.4-33

PEAK DIFFERENTIAL PRESSURE (PSI)

BREAK AT SIDE OF VESSEL

Across Structures:

DP46-25	- 12.20	PSI	@	.044 SEC
DP47-25	- 12.16	"	"	.015 "
DP48-25	- 11.60	"	"	.062 "
DP49-25	- 13.52	"	"	.029 "
DP50-25	- 11.40	"	"	.064 "
DP51-25	- 38.87	"	"	.008 "
DP52-25	- 21.05	"	"	.014 "
DP53-25	- 15.34	"	"	.021 "
DP54-25	- 13.82	"	"	.016 "
DP46-55	- 8.03	"	"	.038 "
DP47-56	- 10.71	"	"	.013 "
DP48-57	- 9.06	"	"	.020 "
DP51-60	- 38.81	"	"	.008 "
DP52-61	- 20.66	"	"	.013 "

Acros Steam Generator Vessel

DP47-49	- 10.41	PSI	@	.013 SEC
DP48-50	- 3.26	"	"	.016 "
DP51-53	- 37.67	"	"	.007 "
DP52-54	- 8.06	"	"	.012 "

TABLE 14.3.4-34

LOWER COMPARTMENT TEMPERATURE TRANSIENT CALCULATION RESULTS

CASE	MAXIMUM LC TEMP °F	TIME Tmax SEC.	TIME OF CONTAINMENT* SPRAY	FAN
0.6 ft^2	326.1	151.39	53.	605.
0.35 ft^2	325.8	322.8	59.	617.
0.1 ft^2	320.7	651.	106.	663.

*Hi-2 Pressure Setpoint used was 3.5 psig.
Relay time used for spray actuation after Hi-2 signal was 45 sec.
Relay time used for fan actuation after Hi-2 signal was 600 sec.

TABLE 14.3.4-35

0.35 FT2 SPLIT 30% POWER

Time (sec)	Mass (lb/sec)	Energy (BTU/sec)
.1000E-01	.7970E+03	.9480E+06
.1000E+01	.7970E+03	.9480E+06
.3000E+01	.7890E+03	.9388E+06
.5000E+01	.7820E+03	.9308E+06
.7000E+01	.7760E+03	.7239E+06
.9000E+01	.7700E+03	.9169E+06
.1000E+02	.7680E+03	.9145E+06
.1300E+02	.7760E+03	.9237E+06
.1500E+02	.7800E+03	.9284E+06
.1600E+02	.8960E+03	.1066E+07
.1900E+02	.1240E+03	.1476E+07
.2000E+02	.7720E+03	.9195E+06
.2500E+02	.7090E+03	.8466E+06
.3000E+02	.6630E+03	.7930E+06
.3500E+02	.6280E+03	.7520E+06
.4000E+02	.6010E+03	.7203E+06
.5000E+02	.5630E+03	.6756E+06
.6000E+02	.5350E+03	.6425E+06
.7000E+02	.5140E+03	.6176E+06
.8000E+02	.4970E+03	.5974E+06
.9000E+02	.4830E+03	.5808E+06
.1000E+03	.4700E+03	.5653E+06
.1200E+03	.4500E+03	.5415E+06
.1400E+03	.4320E+03	.5200E+06
.1600E+03	.4160E+03	.5008E+06
.1800E+03	.4020E+03	.4841E+06
.2000E+03	.3890E+03	.4685E+06
.2400E+03	.3650E+03	.4397E+06
.2800E+03	.3440E+03	.4144E+06
.3200E+03	.3240E+03	.3904E+06
.3600E+03	.3060E+03	.3687E+06
.4000E+03	.2890E+03	.3481E+06
.5000E+03	.2530E+03	.3046E+06
.6000E+03	.2230E+03	.2683E+06
.7000E+03	.1990E+03	.2392E+06
.8000E+03	.1790E+03	.2150E+06
.9000E+03	.1620E+03	.1944E+06
.1000E+04	.1480E+03	.1774E+06

TABLE 14.3.4-36

0.6 FT2 SPLIT 30% POWER

Time (sec)	Mass (lb/sec)	Energy (BTU/sec)
.1000E-01	.1365E+04	.1624E+07
.1000E+01	.1365E+04	.1624E+07
.3000E+01	.1341E+04	.1596E+07
.5000E+01	.1320E+04	.1572E+07
.7000E+01	.1302E+04	.1551E+07
.8000E+01	.1293E+04	.1541E+07
.1000E+02	.1297E+04	.1545E+07
.1200E+02	.1298E+04	.1546E+07
.1300E+02	.1297E+04	.1545E+07
.1400E+02	.1268E+04	.1513E+07
.1600E+02	.1196E+04	.1429E+07
.1800E+02	.1133E+04	.1355E+07
.2000E+02	.1079E+04	.1292E+07
.2200E+02	.1033E+04	.1238E+07
.2400E+02	.9940E+03	.1192E+07
.2700E+02	.9440E+03	.1133E+07
.3200E+02	.8800E+03	.1057E+07
.3600E+02	.8420E+03	.1012E+07
.4000E+02	.8110E+03	.9754E+06
.4600E+02	.7740E+03	.9313E+06
.5000E+02	.7540E+03	.9074E+06
.6000E+02	.7130E+03	.8584E+06
.7500E+02	.6680E+03	.8045E+06
.9500E+02	.6250E+03	.7529E+06
.1200E+03	.5840E+03	.7036E+06
.1400E+03	.5570E+03	.6711E+06
.1800E+03	.5110E+03	.6156E+06
.2200E+03	.4720E+03	.5685E+06
.2400E+03	.4530E+03	.5455E+06
.2600E+03	.4350E+03	.5238E+06
.3000E+03	.4020E+03	.4838E+06
.3600E+03	.3600E+03	.4330E+06
.4200E+03	.3250E+03	.3905E+06
.5000E+03	.2870E+03	.3445E+06
.5600E+03	.2680E+03	.3154E+06
.6000E+03	.2480E+03	.2972E+06
.8600E+03	.1790E+03	.2136E+06
.9600E+03	.1610E+03	.1918E+06
.9800E+03	.1580E+03	.1882E+06
.1000E+04	.1550E+03	.1846E+06

TABLE 14.3.4-37

KEY PARAMETERS AFFECTING SPLIT STEAM LINE BREAKS

Variable	Values Used In LOTIC-3 Report	Values for D. C. Cook
Fall Load Steam Pressure (psia)	1000	820
Plant Power (Mwt)	3425	3403
Time Delay to Feedline Isolation (sec)	15	$\leq$ 9.0
Time Delay to Steam Line Isolation (sec)	15	$\leq$ 9.0

TABLE 14.3.4-38

UNIT 2 RERATING PARAMETERS

Used in Bounding Steamline Break
Mass/Energy Releases for Unit 1 and Unit 2

Parameter	(Unit 2 Rerating) Lower Bound Temperatures	Upper Bound Temperatures
NSSS Power, MWt	3600	3600
Core Power, MWt	3588	3588
RCS Flow, gpm/loop	88500	88500
Minimum Measure Flow, gpm/loop	91600	91600
RCS Temperature, °F		
Core Outlet	585.4	618.0
Vessel Outlet	582.3	615.2
Core Average	549.9	584.6
Vessel Average	547.0	581.3
Vessel/Core Inlet	511.7	547.3
Steam Generator Outlet	511.4	547.1
Zero Load	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100
Steam Pressure, psia	587	820
Steam Flow (10^6 lb/hr total)	15.90	16.00
Feedwater Temp., °F	449.0	449.0
SG Tube Plugging, %	10	10

14.3.5 RADIOLOGICAL CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT AND OTHER EVENTS CONSIDERED IN SAFETY ANALYSIS

14.3.5.1 Introduction

This section addresses the effect of increased fuel exposure for Cook Nuclear Plant Unit 2 on radiological consequences. The radiological consequences of postulated accidents were assessed to verify compliance with 10 CFR 100 limits.

14.3.5.2 Summary

An assessment of radiological consequences was performed for Cook Nuclear Plant Unit 2 to conservatively envelope operation to a core average exposure of 40,000 MWD/MTU and a peak assembly exposure of 50,000 MWD/MTU. The radiological release methodology discussed in Reference 1 was used. The whole body and thyroid doses received from the postulated loss-of-coolant accident (LOCA), fuel handling accident (FHA), and other events remain a small fraction of the 10 CFR 100 limits.

14.3.5.3 Radiological Consequences

The analysis supports reactor operation at 3411 MWt to a core average exposure of 40,000 MWD/MTU and a peak assembly exposure to 50,000 MWD/MTU. This assessment supersedes previous Cook Nuclear Plant Unit 2 radiological analyses performed in support of lower exposure limits. The analysis was performed with Advanced Nuclear Fuels' (ANF) methodology[1] for radiological consequences. The approved RODEX[2,6] fuel performance code with the ANS 5.4 fission gas release model was used to calculate fission product inventories in the free volume within the fuel rod.

The results of the radiological analysis indicate that in comparison to the original FSAR some predicted doses increase while others decrease. This behavior is a result of several factors: 15 x 15 versus 17 x 17 fuel arrays, the use of more isotopes in the current analysis compared to the original FSAR analysis, and the use of fixed fission gas release fractions versus RODEX2 calculated release fractions.

The extended burnup fuel is a 17 x 17 fuel array and the base case fuel is a 15 x 15 fuel array. The 15 x 15 fuel has a higher linear heat generation rate, larger rod diameter and higher temperatures than the 17 x 17 fuel. The higher temperatures in the 15 x 15 rods result in greater fission gas release fractions being predicted by the RODEX2 code. For the dose calculations which use fixed fission gas release fractions this effect is not present. Thus, the dose predicted using the RODEX2 release fractions would decrease and the dose predicted using the fixed release fractions would increase or remain unchanged.

The use of more isotopes in the ANF radiological methodology than are used in the FSAR 15 x 15 base case also has a direct impact on the calculated dose. This is especially true in the calculation of the whole body doses where the number of isotopes included in the analysis shows the largest variation. In some dose calculations the integrated energies of this larger number of isotopes overpowers the smaller number of isotopes considered in the 15 x 15 FSAR base case. When this occurs the predicted doses increase over those given in the FSAR 15 x 15 base case irregardless of whether RODEX2 or fixed fission gas release fractions are used.

14.3.5.3.1 Summary of Methodology

The calculation procedure used in this analysis is described in Sections 3 and 4 of Reference 1. The procedure requires calculation of radioactive source terms for both the original FSAR case (base case) and the ANF high exposure fuel. A source term or dose ratio is then calculated based on these source terms. The revised offsite dose, accounting for the change in fuel burnup, is calculated by multiplying the FSAR reported dose by this dose ratio.

This scaling is valid since no changes in the assumptions relating to transport mechanisms are made. The fission gas release fractions used in the analysis are either NRC approved fixed fractions or computed fractions from the NRC approved version of RODEX2[2,6] with the ANS 5.4 fission gas release model.

The radiological assessment for Cook Nuclear Plant Unit 2 high burnup fuel followed the procedure presented above. The fuel and core fission product inventories were calculated for the base case and the extended burnup fuel using the ORIGEN code[3]. Consistent bounding power histories are used for the base case and high burnup fuel. The operating conditions assumed for both the LOCA and fuel handling accidents are summarized in Table 14.3.5-1. Fixed fractions of the fission product inventory were assumed to be released in the maximum hypothetical LOCA and the FHA using Atomic Energy Commission (AEC) Guideline 25. In the design base LOCA and FHA analyses, the release fractions for the isotopes were computed using the approved version of RODEX2 with the ANS 5.4 gas release model. RODEX2 was run using the code logic switches set to calculate maximum fission product inventory in the fuel rod free volume of the fuel. The results of the 17 x 17 fuel inventory and release calculations are presented in Tables 14.3.5-2 and 14.3.5-3[7].

Whole body doses are calculated based on an energy weighted summation of all fission product isotopes over the period of exposure. The thyroid doses, however, are calculated based only on the iodine isotopes. Whole body and thyroid doses were evaluated for 2 hour and 30 day exposure times for the LOCA case and 2 hour exposure for the fuel handling accidents. In both accidents, the 2 hour dose was calculated at the nearest site boundary while the 30 day dose resulting from a LOCA was evaluated at the low population zone boundary (5 miles) in accordance with 10 CFR 100 guidelines.

14.3.5.3.2 Fuel Handling Accident

The fuel handling accident is assumed to occur for the assembly experiencing the highest burnup, 50,000 MWD/MTU. The analysis assumes the accident occurs either inside the auxiliary building or inside the containment. In each case the accident is assumed to occur 100 hours after shutdown. The assumptions inherent in these calculations are presented in the Cook Nuclear Plant Unit 1 FSAR, Sections 14.2.1.1 and 14.2.1.2.

Auxiliary Building

For this analysis, it was assumed that the gap activities in all of the rods in the worst case fuel assembly were released. The gap activities used in the analysis are based on the RODEX2 computed fission gas release fractions. The calculated whole body and thyroid doses for both the conservative case (Regulatory Guide 1.25) and the realistic case (Regulatory Guide 4.2) are shown in Table 14.3.5-4 for the previous base case calculation and this calculation. The potential doses at the site boundary remain only a small fraction (<10%) of the limiting values specified in 10 CFR 100.

Inside Containment

The reference calculation assumed that the release of activity for this accident is a fixed fraction (AEC Guideline 25) of the inventory of fission products in the fuel. These doses, for both the conservative case (Regulatory Guide 1.25) and the realistic case (Regulatory Guide 4.2), are presented in Table 14.3.5-5. As can be seen, these potential site boundary doses are a small fraction of the limits specified in 10 CFR 100.

14.3.5.3.3 Locked Rotor

The analysis of the locked rotor event, submitted with the Cycle 6 documentation, was performed with the code PTSPWR2[4]. This analysis indicated adequate margin in the MDNBR so that no fuel failures were predicted. However, even if this analysis had indicated failures, the offsite radiological consequences could be conservatively estimated as a small fraction of the doses calculated for the double-ended coolant pipe break (LOCA) event discussed in Section 14.3.5.3.7 and therefore would be well below the 10 CFR 100 limits.

14.3.5.3.4 Steam Generator Tube and Main Steamline Ruptures

The Cook Nuclear Plant Unit 2 FSAR analysis of the steam generator tube rupture, and the ANF analysis of the main steamline rupture indicated that there would be no fuel failures for these transients. Thus, as with the previous vendor's fuel in the core the Technical Specification limits on coolant activity will be the controlling factor for offsite doses. This limit has not been changed so the FSAR dose results remain bounding.

As with the locked rotor transient, if fuel failures were projected to occur, they would be bounded by the assumptions in the loss of coolant analysis presented in Section 14.3.5.3.7 and therefore only a small fraction of the 10 CFR 100 limits.

14.3.5.3.5 RCC Assembly Ejection Incident

An analysis of the RCC assembly ejection incident has been performed by ANF using current methodology based on MDNBR. The failure of fuel based on this methodology, however, is less than 10.5%. Release of offsite fission products will occur through two mediums: 1) through containment structure; and 2) through leakage of primary coolant to the steam generator secondary side.

Release of offsite fission products through the containment structure is only a small fraction of that which would occur under LOCA conditions and therefore would be bounded by that accident. Release due to leakage to the secondary side is based on the amount of fission products relative to the amount of fuel failed. The gap activities are used so that the current dose is calculated on a consistent basis with that given in Reference 5. The gap activities used in the analysis are based on the RODEX2 computed fission gas release fractions. Tables 14.3.5-6 and 14.3.5-7 report the doses calculated for both site boundary and a low population zone. These potential doses are well below the 10 CFR 100 guidelines.

14.3.5.3.6 Single RCC Assembly Withdrawal Incident

ANF has performed an analysis of the single RCCA withdrawal incident and predicted that up to 3.7% of the core will experience fuel failure. This failure is assumed to result in an instantaneous release of fission gas to the primary coolant, where it mixes throughout the volume. The gap activities used in the analysis are based on the RODEX2 computed fission gas release fractions. The transport path to the environment is the primary-to-secondary side leakage through the steam generator safety relief valves. The resulting doses for both cases at site boundary and low population zone are reported in Tables 14.3.5-8 and 14.3.5-9. These results show that the probable doses are well below the 10 CFR 100 limits.

14.3.5.3.7 LOCA Event

The design base accident (DBA) assumes that the entire inventory of volatile fission products contained in the pellet-cladding gap is released. That inventory, as given in Table 14.3.5-2 and Table 14A.2-1 of the Unit 1 UFSAR, is based on an end-of-cycle core averaged exposure of 24,560 MWD/MTU with a rated reactor power of 3391 MWt. The offsite dose for Unit 2 due to the different operating power and increased end-of-cycle average fuel exposure is proportional to the change in fission product inventory within the pellet-clad gap.

For the maximum hypothetical accident, the fission product source is assumed to be a fixed fraction of the total core inventory of volatile fission products. Consequently, for the hypothetical accident assumptions, the offsite dose resulting from operations at the proposed higher power and end-of-cycle fuel exposure is proportional to the change in the total core inventory of volatile fission products. As for the DBA, the inventory is based on the RODEX2 fission gas release fractions and the end-of-cycle core average fission product source.

Bounding offsite doses which could result from a loss-of-coolant accident involving fuel operating at the present Unit 2 power level and increased exposure levels considered in this analysis are given in Table 14.3.5-10 and compared with values determined in the FSAR. As can be seen, the estimated offsite radiological consequences of a LOCA are within the limits specified in 10 CFR 100.

Control Room Habitability

Analyses associated with control room habitability are presented in Section 14.3 of the Unit 1 FSAR. These analyses bond both Units 1 and 2.

Table 14.3.5-6 RCCA Ejection Accident
Two Hour Site Boundary Doses

	Base Case (Rem)	Extended Burnup (Rem)	10 CFR 100 Limit-Rem
Whole Body	.005	.021	25
Thyroid	.38	1.69	300

Table 14.3.5-7 RCCA Ejection Accident
Two Hour Low Population Doses

	Base Case (Rem)	Extended Burnup (Rem)	10 CFR 100 Limit-Rem
Whole Body	.0012	.005	25
Thyroid	.09	.40	300

Table 14.3.5-8 Single RCCA Withdrawal Accident
Two Hour Site Boundary Doses

	Base Case (Rem)	Extended Burnup (Rem)	10 CFR 100 Limit-Rem
Whole Body	.005	.007	25
Thyroid	.38	.60	300

Table 14.3.5-9 Single RCCA Withdrawal Accident
Two Hour Low Population Doses

	Base Case (Rem)	Extended Burnup (Rem)	10 CFR 100 Limit-Rem
Whole Body	.0012	.0018	25
Thyroid	.09	.14	300

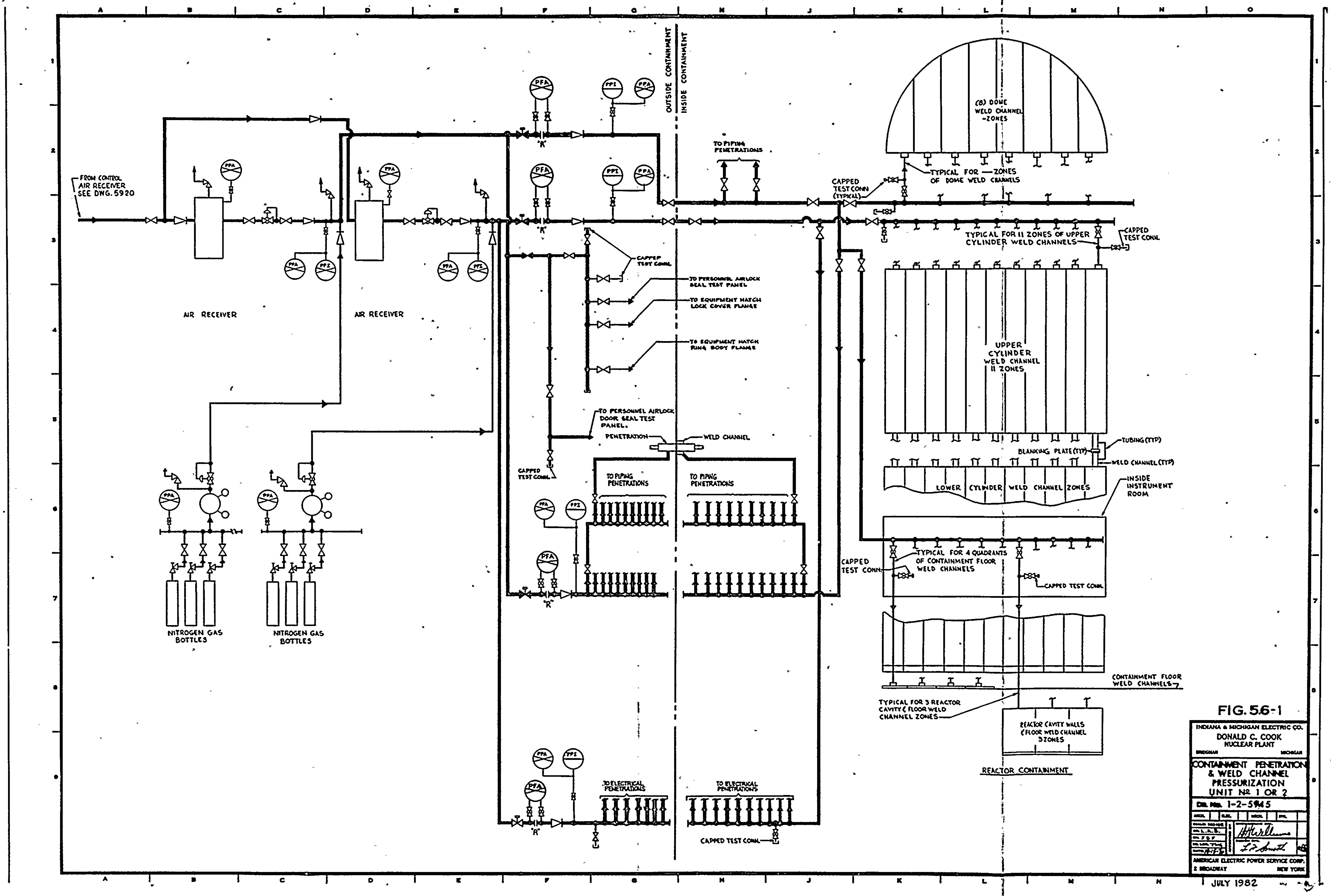
FROM CONTROL AIR RECEIVER SEE DWG. 5920
AIR RECEIVER
AIR RECEIVER
PFA
PPS
NITROGEN GAS BOTTLES
NITROGEN GAS BOTTLES
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
"R"
CAPPED TEST CONN.
TO PERSONNEL AIRLOCK SEAL TEST PANEL
TO EQUIPMENT HATCH LOCK COVER FLANGE
TO EQUIPMENT HATCH RING BODY FLANGE
TO PERSONNEL AIRLOCK DOOR SEAL TEST PANEL.
PENETRATION
WELD CHANNEL
TO PIPING PENETRATIONS
TO PIPING PENETRATIONS
TO PIPING PENETRATIONS
TO ELECTRICAL PENETRATIONS
TO ELECTRICAL PENETRATIONS
CAPPED TEST CONN.
CAPPED TEST CONN (TYPICAL)
(8) DOME WELD CHANNEL ZONES
TYPICAL FOR — ZONES OF DOME WELD CHANNELS
TYPICAL FOR 11 ZONES OF UPPER CYLINDER WELD CHANNELS
CAPPED TEST CONN.
UPPER CYLINDER WELD CHANNEL 11 ZONES
TUBING (TYP)
BLANKING PLATE (TYP)
WELD CHANNEL (TYP)
LOWER CYLINDER WELD CHANNEL ZONES
INSIDE INSTRUMENT ROOM
CAPPED TEST CONN.
TYPICAL FOR 4 QUADRANTS OF CONTAINMENT FLOOR WELD CHANNELS
CAPPED TEST CONN.
CONTAINMENT FLOOR WELD CHANNELS
TYPICAL FOR 3 REACTOR CAVITY & FLOOR WELD CHANNEL ZONES
REACTOR CAVITY WALLS & FLOOR WELD CHANNEL 3 ZONES
REACTOR CONTAINMENT
INDIANA & MICHIGAN ELECTRIC CO.
DONALD C. COOK NUCLEAR PLANT
BRIDGMAN MICHIGAN
CONTAINMENT PENETRATION & WELD CHANNEL PRESSURIZATION UNIT No 1 OR 2
DR. No. 1-2-5945
AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY NEW YORK
JULY 1982

FIG. 5.6-1