1.0 INTRODUCTION AND SUMMARY

1.5 PLANT OPERATION

The Facility Operating License and Plant Technical Specifications define administrative, environmental and technical operating limits in the interest of the health and safety of the public. Procedures have been developed to ensure operation is in conformity with the Technical Specifications and the facility Operating License.

1.6 RESEARCH AND DEVELOPMENT REQUIREMENTS

A number of Research and Development programs germane to the Operating License review were summarized in Sub-Chapter 1.6 of the original FSAR. For historical record purposes, major portions of Sub-Chapter 1.6 from the Original FSAR are attached without modification. An updated list of references in the areas of Emergency Core Cooling System, Ice Condenser, and Nuclear, Thermal-Hydraulic and Mechanical Design Parameters is contained in Table 1.6-1.

1.6.0 [Historical] Research And Development Requirements (From Original FSAR)

Each Research and Development program is briefly summarized for identification and its relationship to the Donald C. Cook Nuclear Plant is discussed. Detailed discussions of each R&D program are available in a more expanded summary form in Westinghouse reports (WCAPs) which have been submitted to the AEC (NRC) staff (see References 1, 2, 3 and 28). Refer to Section 1.6.3 for a discussion of the 17 x 17 test programs for Unit 2.

1.6.1 [Historical] Programs Required For Plant Operation

In the PSAR, five programs were identified as required for plant design and operation.

1. Development of the design of the Emergency Core Cooling System.
2. Development of the final core thermal-hydraulic, nuclear and mechanical design parameters.
3. Further evaluation of core stability.
4. Development of the design details of the Containment Spray System.
5. Development of the Ice Condenser System.

A discussion of these programs and the applicability of the results to the Donald C. Cook Nuclear Plant follows.

1.6.1.1 [Historical] Development Of The Design Of The Emergency Core Cooling System (ECCS)

A detail design of the ECCS has been developed, and details of the design are presented in Chapter 6. As discussed in Section 1.2.9 (Original FSAR) above, the design of the ECCS has been substantially modified to improve its ability to meet single active failure during the injection phase or single active or passive failure during recirculation phase and to deliver

dissolved chemical poison more rapidly to the reactor. The ECCS has been designed to prevent clad melting. The basic design criteria for loss-of-coolant accident (LOCA) evaluations are given in Section 14.3. Satisfaction of these criteria ensures that the core remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

The effect of rod bursting, swelling or shattering has been considered in the LOCA evaluations. In the blowdown phase of the accident, core geometry distortion may be due to clad bursting or swelling. The clad temperature may get sufficiently high (1200 to 2000°F) so that bursting or swelling of the clad would occur by virtue of the internal gas pressure and a significant reduction of clad strength. Clad bursting or swelling is of concern because of the potential of blocking the flow channel to the extent that core-cooling flow would be insufficient to meet the LOCA design criteria.

To demonstrate that effective core cooling will not be impaired during the reflooding phase of a loss-of-coolant accident, Westinghouse undertook a rod burst research and development program. (Item 2 in Reference 1.) The program to investigate the performance of fuel rods during a simulated LOCA has been completed. It has supplied empirical data on the above safety related problems from which the amount and kinds of geometry distortion on the ability of the ECCS to meet the LOCA design criteria has been determined using present analytical design techniques.

a. Single Rod Burst Tests (SRBT)

The performance of the fuel rods during a simulated loss-of-coolant accident has been evaluated in a test program, which is described in Reference 4.

Volume I of the reference describes burst, quench and eutectic formation tests with unirradiated tubes and an evaluation of the data from both Volume I and II. An interpretation with regard to the postulated sequence during the loss-of-coolant accident is given.

Volume II reports the results of work under AEC Contract AT-(30-1)-3017 and describes burst and quench tests on irradiated tubes.

The single rod tests indicated that rod-to-rod interference might occur following rod burst and must be considered. The quantitative evaluation of the influence of adjacent rods in a fuel assembly would be difficult, if not impossible, to determine analytically. Therefore, the rod burst program was extended to include multi-rod burst tests. Multi-rod burst tests (MRBT) were performed to demonstrate that the

rods in a PWR rod bundle burst randomly so that a minimal flow channel area, for core cooling purposes, is maintained.

b. Multi-Rod Burst Tests (MRBT)

The results of this phase of the Rod Burst Program are reported in Reference 5.

Volume I describes test apparatus and conditions along with an evaluation of the test results. Volume II presents the application of the MRBT results to the LOCA core thermal analysis.

Results of the MRBT show that the burst locations are staggered axially along the fuel rods and that, to some degree, rod to rod contact does occur. However, the remaining flow area is always sufficient to ensure adequate core cooling. Analytical evaluations for a typical double-ended cold leg break, considering flow redistribution due to the geometry distortion and rod-to-rod contact, have shown that the peak clad temperature increases less than 100°F over the peak temperature without geometry distortion.

The program is complete and results are satisfactory. No backup measures are considered necessary.

1.6.1.2 [Historical] Development Of The Final Core Thermal-Hydraulic Nuclear And Mechanical Design Parameters

In the course of plant design, further engineering information than presented in the PSAR has been developed for those thermal-hydraulic, nuclear and mechanical design parameters for which design criteria have been established.

The engineering information demonstrates that the systems, as designed, will meet the established criteria. This demonstration consists principally of analyses, calculations, and evaluations as presented in Chapters 3 and 14. Tests will be made prior to initial startup, during initial startup, and during initial approach to power to check plant operation relative to design objectives (Refer to Chapter 13).

1.6.1.3 [Historical] Further Evaluation Of Core Stability

The purpose of this program was to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance. This program has been completed in two areas: (a) confirmation of the ability of the out-of-core detector system to indicate gross core power distribution sufficient to permit

control of xenon oscillation within specified operating limits; and (b) development of a control system utilizing the out-of-core detector system and control rods. The third part of this program, verification through start-up testing that the control system can control the core power distribution and that adequate margins exist to operate the plant will be evaluated on a continuing basis for Westinghouse reactors going into operation prior to D. C. Cook. Safe operation at the design power level depends upon experimental demonstration, at the time of Donald C. Cook startup, that the actual power shapes at full power are no worse than those used in the calculation of core integrity. Further, the analytical model used to predict these power shapes will have been justified by these and earlier measurements so that a calculation of margin to design limits in a transient or accident situation can be made conservatively. However, it is clear that very similar conditions will exist on earlier plants and that very little, if any, extrapolation will be required.

In the unlikely event that the development program described above does not show that margins for operation at the proposed power levels are adequate, the margins designed for Donald C. Cook could be achieved by systems modifications or restrictions on operation.

1.6.1.4 [Historical] Development Of The Design Details Of The Containment Spray System

A Containment Spray System is provided to remove post-accident decay heat and to remove iodine from the containment atmosphere. A description of the Containment Spray System is given in Chapter 6.

The spray additive that will be used for iodine removal is sodium hydroxide (NaOH). This selection was based on completed research and development work done by Westinghouse and others, notably that of Oak Ridge National Laboratory (ORNL) and the Battelle Northwest Laboratories. The areas investigated before the selection was made included studies of chemical characteristics, material compatibility and radiolysis. A report on the original research and development work is presented in the Preliminary Safety Analysis Report Section 1.6.4.

Additional research and development work on the containment spray system has been performed by AEP and above-mentioned laboratories. It is emphasized here that this additional work was not performed to prove the ability of the containment spray system to meet the requirements of 10 CFR 100 guidelines. This goal was attained even with conservative assumptions for iodine removal by the sprays.

The additional work was performed to justify the use of larger elemental iodine removal constant in assessing the potential off-site doses resulting from the design basis accident. The additional research and development program completed is described below.

[Historical] Containment Spray Research And Development

The addition of the reactive chemical, sodium hydroxide (NaOH), to the containment sprays will be employed as a means of reducing the iodine concentration of the containment atmosphere under postulated accident conditions. Data which have already been obtained in engineering scale tests at the Nuclear Safety Pilot Plant (NSPP) and the Containment Systems Experiment (CSE) confirm the absorptive capacity of the chemically modified sprays. Further refinement has been pursued by American Electrical Power Service Corporation in order to justify additional performance of the sprays and to evaluate non-ideal factors in extrapolating to large containment structures. It has been established that in no way does the use of proposed additive jeopardize the performance or integrity of the containment or Emergency Core Cooling System. The discussion below describes the research and development program in these areas for the Donald C. Cook units.

The following areas were investigated with regard to their effect on spray performance in order to demonstrate the full capability of the Containment Spray System:

a. Droplet coalescence

b. Non-uniformity of spray droplet size and coverage

c. Liquid phase mass transfer resistance

The work in these three areas was performed jointly by American Electric Power Service Corporation and Battelle Memorial Institute's Columbus Laboratories. The study included an evaluation of the significance of liquid phase mass transfer resistance to iodine removal, and the development of methods to include the effects of spray drop coalescence, and of changes in the cross-sectional area of spray coverage within the containment vessel. The study concluded with construction of a digital computer program, which incorporates an analytical treatment of each of the above factors in a simultaneous multi-region model. The program therefore provides a coupled analysis of iodine cleanup in each of three connected regions which may contain both a sprayed and unsprayed volume.

A summary of the effects of these phenomena on spray performance is given on the following page.

a. Droplet Coalescence

The basic model conservatively assumes that any collision between two sprays drops results in coalescence. The containment atmosphere is divided into three spray regions, containing 17 sections in each region, with individual droplet trajectories and flux densities calculated for each region. Quantitative evaluation of this effect is presented in the original FSAR Section 14.3.5.

b. Non-Uniformity of Spray Droplet Size and Coverage

To account for the distribution of droplet sizes as verified in tests performed by Westinghouse, the spray was divided into a lognormal distribution of eleven discrete size intervals with a constant geometric standard deviation. All droplet sizes were assumed to interact as described in the previous paragraph. Non-uniform containment coverage was accounted for by isolating those areas of the containment which are not directly sprayed and by using a conservative estimate of the mass transfer rate between these areas and the sprayed regions.

c. Liquid Phase Mass Transfer Resistance

In order to account for the buildup of iodine in the spray solution and the possibility of increases resistance to mass transfer in the liquid phase, the partition coefficients are calculated continuously throughout the entire analysis. Quantitative evaluation of this effect is presented in the original FSAR Section 14.3.5.

[Historical] Applicability To The Donald C. Cook Plant

The results obtained from this study in addition to the experimental results from NSPP and CSE verify the ability to the spray system to perform as an extremely effective means of reducing iodine leakage from the containment in event of a loss-of-coolant accident to below the guideline limits of 10 CFR Part 100. The original FSAR Section 14.3.5 presents a quantitative evaluation of all the different parameters that influence spray performance for iodine removal.

1.6.1.5 [Historical] Development Of The Ice Condenser System

The design of the Ice Condenser Reactor Containment is based on proven and tested concepts for heat removal by the ice contained in the system. Sufficient test results, along with continuing design evaluation, have proven the feasibility, practicality and advantages of this type of containment. Proprietary Westinghouse Reports (References 6, 7, 8, 9, and 10) which describe this work in detail, have been submitted to the Division of Reactor Licensing of the Atomic Energy Commission for their review of the ice condenser concept. Additional proprietary documents (References 11, and 12) submitted to the Commission describe additional full-scale section tests and present the complete analysis of the ice condenser design, performance and sensitivity to variations in important parameters specifically related to the Donald C. Cook Nuclear Plant.

Westinghouse has prepared and submitted to the Commission an additional proprietary report (Reference 13) which describes additional analyses and experimentation not covered in the previous reports. The following is an outline of the material presented in this new report.

a. The final series of full-scale section ice condenser tests, completed in December 1968.

b. The analytical models developed from the final series of ice condenser tests.

c. Description of a digital computer code to calculate the transient pressures in the subcompartments of the reactor containment due to the loss-of-coolant accident.

d. Current status and conclusions drawn from the results of long-term ice storage tests. Description of additions to the program to determine effects of long-term storage on ice having a sodium tetraborate additive, which is compatible with the containment spray solution used for iodine absorption.

As stated in Reference 13, the additional experimental information and analyses identify the degree of conservatism in the ice condenser design presented in the previous reports.

Westinghouse has been proceeding with the development of the Ice Condenser concept since 1965, and this development is essentially complete, with performance capabilities supported by analytical work backed up by a significant amount of experimental evidence. The development programs are described below and more fully in the reference reports.

[Historical] Small-Scale Ice Bed Performance Tests

These tests consisted typically of about a 3-foot-high ice bed within a 10-foot-high by 10-inch-diameter autoclave acting as the reactor containment containing an ice condenser. It received steam blow-down from another, heated autoclave. A total of approximately 50 tests were run to search out key phenomena and develop sufficient understanding of ice condenser system performance. This understanding served to guide the design of the reactor containment and the design of the full-scale condenser section test facility.

[Historical] Full-Scale Ice Bed Performance Tests

These tests were performed in the test facility located at the Westinghouse Waltz Mill Site. The facility consisted of a 104 cu. ft. boiler to simulate reactor coolant system mass and energy; a 56-foot-high by 11-foot-diameter receiver vessel to simulate the containment; an instrumentation and control building; and ice making and long-term ice storage facilities. The tests have utilized an ice condenser, approximately 40 feet high, located inside the receiver vessel. The facility is divided into compartments to nearly duplicate a section of an actual Ice Condenser System. The test arrangement was designed to provide the volume ratios and scale factors equivalent to the containment design. The ice bed was made to closely duplicate the containment design in ice loading and airflow channels through the bed, as well as entrance and exit openings for the steam and airflows. These tests were extensively instrumented so that the maximum amount of information could be obtained from each test. Results of these tests have shown that the ice condenser is relatively insensitive to changes in such parameters as blowdown rate, blowdown energy, ice heat-transfer area and flow area, and that the performance change is predictable. All testing in the Waltz Mill facility needed for the Cook Plant design has been completed, and is reported in References 11, 12 and 13.

[Historical] Ice Handling And Ice Basket Loading Tests (Reference 2)

Ice Handling Tests and Ice Basket Loading Tests were conducted at the Westinghouse Waltz Mill Site to demonstrate the adequacy of a system of pneumatic fluidized transport of chemical flake ice and methods of loading ice baskets inside the ice condenser. These trials satisfactorily demonstrated that ice can be manufactured outside of the containment and charged into ice

baskets in the ice condenser. The ice machine and all major pneumatic equipment is located outside the containment.

[Historical] Ice Storage Test

These tests have consisted of storing ice for long periods of time under a number of different conditions (e.g., relative humidity, temperature, and mechanical loads), employing borated and non-borated and sodium tetraborate additive. Tests results indicate that the addition of boron or sodium tetraborate to the ice has no unfavorable effect on long-term storage performance. In addition, test results have demonstrated that the ice condenser design is adequate to preserve ice condenser integrity for long periods of time (at least a number of years). In particular, the use of a refrigerated ice storage arrangement with separate air compartments for the ice bed and for the cooling system will reduce sublimation and frosting effects to a negligible amount. Furthermore, the ice bed compaction rate is very small and the effect of this amount of reduction in heat transfer surface on containment design pressure is negligible. Tests of ice bed support by expanded metal screens or gratings indicate no significant amount of extrusion through horizontal or shear along vertical ice bed walls. Further, the tests indicate that the storage characteristics of borated ice are just as satisfactory as non-borated ice.

[Historical] Ice Condenser Door Gasket Tests (Reference 1)

Tests have been run to validate the sealing capability of the ice condenser inlet door gasket design. Tests consisted of two parts, (1) a leakage test and (2) a load deflection test. The leakage test was run simulating actual gasket service conditions, such as, pressure and gasket load. The load deflection test was used to determine the effect of surface irregularities on sealing.

The gasket leak tests have shown that the maximum total leakage from the ice condenser through the lower inlet doors that could be expected in service is 5 scfm compared with the design criterion of 50 scfm.

[Historical] Ice Condenser Inlet Door Tests (Reference 1)

Manufacture and assembly of a lower inlet door unit having two door panels have been completed and testing is in progress at the Westinghouse Research and Development Laboratories.

The test program includes evaluation of the following aspects:

- Tolerance of flatness of the seal surface of door panel
- Spring Rate
- Hinge bearing friction
- Door position load characteristics
- Hinge Bearing and panel strength tests

Results now being analyzed will be included in the design evaluation and test report on the doors to be submitted to the AEC.

[Historical] Ice Condenser Floor Drains(Reference 14)

Consistent with the Preliminary Safety Analysis Report, and proprietary technical submissions, floor drains have been implemented in the design of the plant.

To obviate any prolonged resistance to reopening of the lower inlet doors after a postulated design basis accident, the design basis of 15 square feet of flow area has been met by the inclusion of 20 drains each of 12-inch diameter.

It should be noted, however, that this criterion is based on there being no flow of condensate water through the doors during blowdown, and all the condensate must flow through the drain.

It has been established by analysis and tests that this is not the case, making the criterion itself very conservative.

[Historical] Iodine Removal In The Ice Condenser(Reference 15)

Tests have been conducted which illustrate the capability of the Ice Condenser System to remove fission product iodine released to a reactor containment during the Design Basis Accident (DBA). Iodine is condensed along with steam condensation by the ice and is collected in the ice melt, thus becoming unavailable for leakage from the containment to the environment. An alkaline additive in the ice, sodium tetraborate, enhances the dissolution and retention of iodine by the ice melt through hydrolysis reactions. The efficiency of iodine removal in the tests has

been evaluated in terms of a number of parameters and at a number of conditions relating to the Design Basis Accident. These results indicate that the Ice Condenser System serves as an additional passive safety feature for control of volatile fission products.

[Historical] Hydrogen Control Program

American Electric Power Service Corporation (AEPSC) and Battelle Columbus Laboratories (BCL) initiated a program for final resolution of the hydrogen control issue (10 CFR 50.44(c)). This program has two phases. In Phase I the AEPSC provided the following information to the NRC staff for review and approval:

- MARCH-2 computer code (Reference 198 Table 1.6-1) input deck(s) applicable to the D. C. Cook containment and systems;
- A hydrogen combustion model inferred from available Nevada Test Site (NTS) data; and
- Justification of scenario selection.

Phase II would consist of actual analysis to be performed by BCL after the above submittals are approved by the NRC.

1.6.2 [Historical] Other Areas Of Research and Development Not Required For Plant Operation

Other areas of research and development, as outlined below, are those, which give, added confirmation that the designs are conservative.

1. [Historical] Burnable Poison Program (Item 7 in Reference 1)

Burnable poison rod development is complete. The burnable poison rods are borosilicate glass encased in stainless steel tubes. The fixed rods are used in the first core only to reduce the concentration of boric acid poison in the moderator, thereby ensuring that the moderator coefficient of reactivity is always negative at operating temperature. The rods are now in use in the R. E. Ginna plant. An evaluation of these rods is expected to be available prior to operation of the Donald C. Cook Nuclear Plant.

2. [Historical] Fuel Development Program For Operation At High Power Densities (Item 8 in Reference 1)

As part of the program to demonstrate satisfactory operation of fuel at high burnup and power densities, fuel is being tested in both the Saxton and Zorita reactors. The Saxton loose-lattice

irradiation program will demonstrate fuel performance at conditions significantly in excess of current PWR design limits, and will establish power burnup limits for the fuel. The Zorita reactor is the first PWR with a Zircaloy core to operate at similar core conditions as the current design units. Because of the timely manner in which fuel can be irradiated in Zorita, four fuel assemblies are being tested there to demonstrate satisfactory operation of the fuel in a commercial PWR environment.

Sustained successful operation of special Zorita fuel rods at peak design power levels, in excess of those planned for these units, will increase assurance that the fuel has adequate performance margins to accommodate transient overpower operation. This program is further discussed in Chapter 3.

3. [Historical] In-Core Detector Program (Item 9 in Reference 1)

The purpose of this program is to develop fixed in-core neutron detectors suitable for continuous monitoring of power distribution in a PWR core.

Testing at San Onofre, the Western New York Research Reactor, and the Brookhaven High Flux Beam Reactor, has been completed. Tests at the Union Carbide reactor (Tuxedo) are being performed for detectors to be installed.

The present status of this program permits fixed in-core flux detectors to be installed. These detectors will serve as an operational convenience to the plant operator, and as tests to evaluate the need for and suitability of in-core detectors for power distribution monitoring and control. The in-core detector development program will be continued in these early large plants with the principal aims of demonstrating design lifetime, in a PWR, and of optimizing detector parameters. Since out-of-core detectors, particularly long ion chambers, have been found effective for monitoring both axial and radial gross power distribution there is at present no intention of installing the incore system in this plant.

4. [Historical] ESADA DNB Program (Item 11 in Reference 1)

This program provides experimental rod bundle DNB data with non-uniform rod axial flux distributions. The program has been conducted at Columbia University under the direction of WNES, Pittsburgh, Pennsylvania. Reference 16 details the results of this program.

The experimental rod bundle data with non-uniform rod axial flux distributions is directly applicable to the design of this unit. The results of the program show that the W-3 DNB correlation applied in this design is conservative.

5. [Historical] FLECHT (Full Length Emergency Core Cooling Heat Transfer Test) (Item 12 in Reference 1)

The purpose of the FLECHT program is to investigate experimentally the thermal behavior of a simulated pressurized water reactor core during the core recovery period, which follows a loss-of-coolant accident. The results of the first series of tests (Group I) are reported in Reference 17. The results of the second series of tests (Group II) are reported in Reference 20.

The loss-of-coolant evaluation presented in this application used conservative design assumptions in the heat transfer models for analyses of the reflooding phase of the accident. The FLECHT program will assist in developing new analytical models to describe the core recovery phenomena. The results to date have been favorable, and the program is essentially complete.

6. [Historical] Flashing Heat Transfer Program (Item 13 in Reference 1)

The program is completed, and it concluded that the present core thermal design analysis used for evaluating the loss-of-coolant accident results in a conservative prediction of the peak clad temperature. The results from the program are in the loss-of-coolant analysis presented in Chapter 14. The program and results are summarized in Reference 2.

7. [Historical] Loss-of-Coolant Analysis Program (Item 14 in Reference 1)

The loss-of-coolant analysis program was established to integrate, as appropriate, the more realistic heat transfer models obtained 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. [Historical] Blowdown Forces Program (Item 15 in Reference 1)

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. [Historical] Gross Failed Fuel Detector Program

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

Failed fuel detection at Cook Nuclear Plant is accomplished by periodic analysis of reactor coolant grab samples. This method has been found acceptable by the NRC (Reference 29 and 30).

10. [Historical] Reactor Vessel Thermal Shock (Item 16 in Reference 1)

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 Advisory Committee for Reactor Safety (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.

It is not anticipated that the continuing HSST Program will lead to any new conclusions about reactor vessel integrity under LOCA conditions. Several backup positions are available if the

results of the HSST program do not conclusively indicate that vessel integrity could be assured for the full plant life with the operating modes presently planned. One solution would be to anneal the reactor vessel such that material properties approach the original value. This solution is already feasible, in principle, and could be performed with the vessel in place.

11. [Historical] BART Program

Westinghouse has developed a model to calculate the fluid and heat transfer conditions in the core during reflood. This model is contained in the BART computer code (191). BART model has two-phase fluid conservation equations. The empirical constants used in BART model are determined by comparing prediction with a selected number of FLECHT Tests.

The behavior of the quench front, which is of crucial importance in determining the core heat transfer during reflood, is determined from data and the overall heat release supplied to the BART program. The thermal-hydraulic model used to determine quench front progression is a two-dimensional heat transfer equation.

The core heat transfer model is used to calculate the peak-clad temperature in the core during a postulated LOCA. A design procedure has been developed which utilizes BART in conjunction with other ECCS codes. Forced flooding tests from several different experiments are used to verify BART, while the design procedure is accomplished by comparison with several FLECHT-SET tests.

1.6.3 [Historical] 17 X 17 Fuel Assembly Verification

The test program for the 17 x 17 fuel assembly has been successfully completed. The tests verified that the 17 x 17 fuel assembly meets the design criteria and requirements as specified in References 21 through 26. Plans for in-service surveillance of fuel assembly performance are given in Section 7 of Reference 27. This performance will be monitored and reported in the periodic updates of WCAP-8183 "Operational Experience with Westinghouse Cores."

1.6.4 [Historical] References For Section 1.6

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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).

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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.

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.

29. Alexich, M. P. (I&M), letter to T. E. Murley (NRC), October 5, 1988, AEP:NRC:0773AB.

30. Colburn, T. G. (NRC), letter to M. P. Alexich (I&M), "Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3, for the D. C. Cook Nuclear Plant, Units 1 and 2," December 14, 1990.

1.7 QUALITY ASSURANCE

In accordance with 10 CFR 50.54, the quality assurance program for Donald C. Cook Nuclear Plant and Independent Spent Fuel Storage Installation (ISFSI) is described in a separate document entitled "Quality Assurance Program Description."

1.8 IDENTIFICATION OF CONTRACTORS

The plant was designed and constructed by the American Electric Power Service Corporation (AEPSC) which performed the function of Architect-Engineer and Constructor for Indiana Michigan Power Company (I&M). Westinghouse Electric Corporation designed and supplied the Nuclear Steam Supply Systems including the initial fuel assemblies for both Units 1 and 2 of the Donald C. Cook Nuclear Plant. In 2000, the Unit 1 Westinghouse Model 51 lower steam generator assembly and upper internals and feedrings were replaced with Babcock and Wilcox (BWI) replacement steam generators Model 51R. Installation was performed by Bechtel. Subsequent reload fuel assemblies for these units have been and will be procured from qualified suppliers such as Westinghouse.

In the design and construction of these units, AEPSC employed various contractors and sub-contractors; however, the ultimate responsibility for all work performed was assumed by AEPSC. AEPSC and I&M are responsible for the implementation of all functions associated with the operation, maintenance, modification and control of the Donald C. Cook Nuclear Plant.

1.9 FACILITY SAFETY CONCLUSIONS

The safety of the public and plant operating personnel, and reliability of plant equipment and systems have been the primary considerations in the plant design. The approach taken in fulfilling the safety consideration is three-fold. First, careful attention has been given to the design so as to prevent the release of radioactivity to the environment under conditions which could be hazardous to the health and safety of the public. Second, the plant has been designed so as to provide adequate protection for plant personnel wherever a potential radiation hazard exists. Third, Engineered Safety Features have been designed with redundancy and diversity, and to stringent quality standards.

Based on the overall design of the plant including its safety features and the analyses of the possible incidents and hypothetical accidents, it is concluded that Donald C. Cook Nuclear Plant Units No. 1 and No. 2 can be operated without undue hazard to the health and safety of the public.