APPENDIX H

STATION RESEARCH, DEVELOPMENT, AND FURTHER INFORMATION, REQUIREMENTS AND RESOLUTIONS

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APPENDIX H

STATION RESEARCH, DEVELOPMENT, AND FURTHER INFORMATION, REQUIREMENTS AND RESOLUTIONS

1.0 SUMMARY DESCRIPTION

This USAR appendix contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of the purpose of highlighting certain text with Italics. The factual information being presented on the Advisory Committee on Reactor Safeguards issues has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR. The current resolutions to these issues, including references to appropriate USAR sections and/or GE Topical Reports, are contained in USAR Section I-11.

The design of the boiling water reactor for Cooper Nuclear Station is based upon proven technological concepts evolved during the development, design, and operation of numerous similar reactors. The AEC, in reviewing the Browns Ferry and Cooper Nuclear Station dockets at the construction permit stage, identified several areas where further research and development efforts were required to more definitely assure safe operation of this station. Also, both the AEC Staff and the Advisory Committee on Reactor Safeguards (ACRS) in their review of other reactor projects identified several additional technical areas for which further detailed support information was to be obtained. All of these development efforts thus are of three general types - (a) those which pertain to the broad category of water-cooled reactors, (b) those which pertain specifically to boiling water reactors, and (c) those which have been noted particularly for a facility during the construction permit licensing activities by the AEC Staff and ACRS reviews.

The scope of many of the areas of technology for items in a, b, and c below is discussed in detail as part of an official response[1] by the General Electric Company to the various ACRS concern subjects.

The following discussion is a comprehensive examination of each of these concern areas raised during the original Licensing of CNS, indicating the planned or accomplished resolution. The discussion has been subdivided as follows:

a. Areas specified in the Cooper Nuclear Station-ACRS construction permit letter.

b. Areas specified in the Cooper Nuclear Station-AEC Staff construction permit safety evaluation report.

c. Areas specified in other related AEC-ACRS construction and operating permit letters.

General Electric Company submitted many topical reports to the AEC in support of the CNS application and those of other GE-BWR facilities. Refer to Table I-11-1 in Section I-11 of the CNS-FSAR.

The results of this appendix are also presented in summary form in Tables I-11-2 through I-11-4 in CNS-FSAR Section I-11.

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[1] *Bray, A.P., et al, APED-5608, "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns," APED-5608, April 1968.*

2.0 *AREAS SPECIFIED IN THE COOPER NUCLEAR STATION ACRS CONSTRUCTION PERMIT LETTER*

2.1 *Introduction*

At its ninety-fifth meeting, on March 7-9, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Consumers Public Power District of Nebraska for authorization to construct its Cooper Nuclear Station. The project was previously considered at ACRS Subcommittee meetings held on October 10, 1967 at the plant site, and on February 29, 1968 in Washington, D.C. During its review, the Committee had the benefit of discussions with representatives of Consumers Public Power District, General Electric Company, Burns and Roe, Inc., and the AEC Regulatory Staff. (Cooper, ACRS Letter, 3/12/68, AEC Docket No. 50-298.) (As a result of merger January 1, 1970, Consumers Public Power District is now known as Nebraska Public Power District.)

This letter contains several items of concern to the ACRS. These concerns and the applicant's resolutions are tabulated below.

2.2 *Station Foundation Support*

Concern

"The soils near the river at this location consist primarily of saturated, poorly compacted sands which may be susceptible to the phenomenon of liquefaction if shaken by a sufficiently severe earthquake. The applicant will improve the foundation support of the plant by excavation to within a few feet of bedrock, dewatering, replacing, and compacting the sand to a density sufficient to prevent liquefaction if such an earthquake should occur. The reactor containment will be placed upon a concrete mat which, in turn, will rest upon the compacted sand." (Cooper, ACRS Letter, 3/12/68, AEC Docket No. 50-298.)

Resolution

The in-situ material was excavated to within nine feet of bedrock and compacted in place. The remaining soil was also compacted to a density that would preclude liquefaction under the worst earthquake. Sand was spread evenly and thoroughly mixed to obtain uniformity of material and water content. Lift thicknesses were limited to 12 inches and compaction achieved with several passes of a vibratory compactor. (See Section XII-2-4, "Foundation Analysis" for further information relative to this matter.)

2.3 *Emergency On-Site Power System*

Concern

"The Committee recommends that the applicant give further consideration to the design of the emergency on-site power system to avoid the need for synchronization of the diesel-driven generators." (Cooper, ACRS Letter, 3/12/68, AEC Docket No. 50-298.)

Resolution

The standby power system is designed for automatic dead load pickup and requires no synchronization. The 4160 V emergency buses are isolated from each other and the diesel generators will not be paralleled. Refer to Section VIII-5, "Standby A-C Power Source."

2.4 *AEC General Design Criteria Number 35 Intent Design*

Concern

"Discussion of General Design Criterion Number 35 (10 CFR 50.34 proposed) has occurred in connection with this review. The manner in which the intent of this criterion will be met for the Cooper Nuclear Station

should be resolved between the applicant and the AEC Regulatory Staff." (Cooper, ACRS Letter, 3/12/68, AEC Docket No. 50-298.)

Resolution

The piping and pressure containing parts of the reactor coolant pressure boundary will conform to the NDTT requirements of Criterion 35 as follows:

a. *Piping and pressure containing parts with a wall thickness greater than 1/2 inch will have a nil-ductility transition temperature, by test, 60°F below anticipated minimum operating temperature when the system has a potential for being pressurized to above 20% of the reactor design pressure.*

b. *Those pipes and pressure containing parts with a wall thickness 1/2 inch or less need not have material property tests (i.e., Charpy V-notch) if: 1) Fabricated from austenitic stainless steel; 2) The material has been normalized (heat treated); 3) The material has been fabricated to "fine-grain practice."*

2.5 *Browns Ferry ACRS Comments (3/14/67) Applicable to Cooper Nuclear Station*

Concern

"The Committee, in its report to you of March 14, 1967 on the Browns Ferry Nuclear Power Station, called attention to a number of matters that warrant careful consideration with regard to reactors of this type and power density. These matters apply similarly to the Cooper Nuclear Station." (Cooper, ACRS Letter, 3/12/68, AEC Docket No. 50-298.)

Resolution

Areas of concern which were expressed in the Browns Ferry ACRS Report and which apply to Cooper Nuclear Station are discussed in the following subsections.

2.5.1 *Effects of Fuel Failure on ECCS Performance*

Concern

"Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for the power density and fuel burnup proposed." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket Nos. 50-259 and 50-260.)

Resolution

The proposed experimental investigation program of fuel failure mode is presented in a GE topical report[1] submitted to the AEC in April, 1968.

The objective of this test program is to demonstrate the ability of the station ECCS to prevent fuel cladding melting as a result of perforation and swelling in the cladding under the combination of temperature and internal pressure which prevail from the pre-accident fuel performance. The general plan of action is to simulate as closely as possible all of the significant aspects of the problem in out-of-pile tests, starting with single-rod tests,

1 Bray, A.P. "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns," APED-5608, April, 1968.

expanding to multi-rod Zircaloy clad simulated fuel assembly tests in air and under emergency core cooling conditions, and culminating with full-size assembly tests. This general plan is supplemented by individual phenomenon tests as might be required to corroborate specific points of the experiment or related analysis work.

Fuel clad perforation will occur when the gas pressure within the fuel rod exceeds the pressure the clad can withstand for that particular clad temperature. The mode of this failure is known. The perforation will be local in that a given fuel rod will perforate at a particular location, the extent of which will be random in that it will occur at a particular, even a very slight, weak point along the fuel rod length - probably at a point of clad flaw, pellet oversize or pellet chip, or point of slightly increased clad oxidation. Such weak points will be randomly distributed. However, the location of failures will be clustered about the point where peak heat flux is located, probably in a two or three foot region.

The position that the perforation will be random and local has been supported by experiments observed on failed irradiated fuel. It has also been demonstrated in test loops by placing single Zircaloy tubes containing UO_2 pellets with internal pressurization in an electric induction heating facility and observing the failure mode. The observed failures in this single rod test were always localized, of the order of one inch in the axial direction and random along the length of the heated rod. Furthermore, the analysis of the perforation test results showed good agreement of clad stress at failure with ultimate stress at failure temperature. Additional research and development testing has been performed with a nine-rod section consisting of nine Zircaloy tubes with internal pressurization. These rods were heated internally by electrical means. These observed failures were again localized and did not block the flow passage enough to preclude effective cooling.

Since the fuel perforation will have the characteristics identified above, the overall geometry of the 49 rod fuel bundles, which are 12 feet long, will essentially remain the same and analytical investigation based upon the preceding experimental observations indicate that the core flooding would not be adversely affected. A full length, internally pressurized, 9 rod Zircaloy clad heater assembly was tested under postulated design basis loss-of-coolant conditions with core spray cooling. A full-scale Zircaloy clad heater bundle with collars welded to the cladding to simulate actual perforations has been tested in both spray and flooding cooling modes. Also, a single full scale test was conducted with internally pressurized Zircaloy clad heater rods to approximate as closely as possible a postulated design basis loss of coolant accident in terms of heatup perforations and spray cooling.

The test program results have been submitted to the AEC as a GE topical report.[1]

2.5.2 Effects of Cladding Temperature and Materials on ECCS Performance

Concern

"In a loss-of-coolant accident, the core spray and flooding systems are required to function effectively under circumstances in which some areas of fuel clad may have attained temperatures higher than those at which such cooling mechanisms have been tested to date. The applicant is conducting tests of these devices at increased temperatures and have reported preliminary results which are promising. The Committee again urges that these tests be extended to temperatures as high as practicable. The use of stainless steel in these tests for simulation of the Zircaloy clad appears suitable, but some corroborating tests employing Zircaloy should be included." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260.)

Resolution

The resolution of the above item is presented in a GE topical report[2] submitted to the AEC in April, 1968.

[1] Liffengren, D.J., "Effects of Fuel Rod Failure on ECCS Performance", NEDO-10208, August, 1970.
[2] Bray, A. P., "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns," APED-5608, April, 1968.

The General Electric Company experimental program on reactor core spray cooling effectiveness is currently in progress and extensive data and analysis of its results have been reported in a GE topical report[1] submitted to the AEC. Experimental full scale fuel assemblies exactly like the one being used in this reactor as well as in all the current General Electric 1965 and 1967 product line boiling water reactors are being employed in this test program. These simulated fuel assemblies contain Calrod units inside the fuel rod cladding instead of nuclear fuel, and complete simulation of the hardware (nose piece, spacers, handle, channel box, etc.) is incorporated. The power in the assembly is also simulated (axial cosine heater Calrods, corner fuel rod peaking, decay heat variation in time).

Tests already conducted as of this date have encompassed fuel assembly powers in excess of those which will occur at either this station or Browns Ferry, and flows which are lower than those being provided for these plants. The results of those experiments confirm that the design basis of the reactor core spray cooling system is firmly established as adequate.

The general approach being followed is to develop high-temperature Zircaloy-clad electrically heated fuel rod simulators and to use these to conduct full-size zircaloy-clad assembly tests. Testing conditions will be selected (1) to duplicate cooling modes, initial temperatures, coolant flow rate, power transients, subcooling temperatures, and time of cooling initiation representative of the multitude of tests performed with stainless steel clad heaters, and (2) to investigate core standby cooling effectiveness at peak temperatures in excess of 2500°F, to the highest temperatures the heaters will permit. The first area of testing will be to corroborate use of models based on the wealth of stainless steel data obtained in the past while the latter area of testing will be to extend the knowledge to high temperatures as closely approaching the cladding melting limit temperature as possible.

A series of "low temperature" spray tests are being conducted to provide information on the correlation between stainless steel and Zircaloy assemblies. "High temperature" effects will also be investigated in the spray mode. In a manner similar to the spray tests, flooding-only tests will be conducted to provide correlation information with "low temperature" tests and to investigate high temperature effects. A single test was conducted early in this program to obtain scoping results under realistic high temperature conditions with combined spray flooding mode of cooling.

Several core spray distribution tests have recently been performed using simulated reactor core spray cooling system spargers and "top-of-reactors" fuel assembly geometry which would be exposed to the spray action. These tests, which measure the water entering each fuel assembly, show that the design flow distribution can be attained. Tests also included experiment with air updraft to simulate potential steam upflow. The data obtained to date and the forthcoming data will make it possible to further refine the understanding of the core spray and core flood phenomena as well as increase the wealth of information now available to confirm that core spray is an effective means of accomplishing core cooling.

A GE topical report on the heated rod-core cooling aspects of this test program has been submitted to the AEC.[2]

2.5.3 *Control Systems for Emergency Power*

Concern

"The applicant stated that the control systems for emergency power will be designed and tested in accordance with standards for reactor protection systems." (Browns Ferry, ACRS Report, 3/14/67, AEC Docket Nos. 50-529 and 50-260.)

Resolution

In common with all reactor protection features, the station standby diesel generator system was designed and is capable of being tested in accordance with the intent of the proposed IEEE-279 standards. The design

[1] *Ianni, P.W., "Effectiveness of Core Standby Cooling Systems for General Electric BWR", APED-5458, March, 1968.*
[2] *Liffengren, D.J., "Effects of Cladding Temperature and Material on ECCS Performance", NEDO-10179, June, 1970.*

basis of the system includes the requirement that no single component failure shall prevent the subject system from operating with sufficient capacity to supply the required emergency loads when necessary.

2.5.4 Diversification of ECCS Initiation Signals

Concern

"Also, he will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates emergency core cooling, to provide additional assurance against delay of this vital function." (Browns Ferry, ACRS Report, 3/14/67, AEC Docket Nos. 50-259 and 50-260.)

Resolution

The preliminary design of sensors for the emergency core cooling systems (ECCS) equipment consisted of a reactor vessel low water signal from either of two independent instrumentation sources to activate the pumping equipment. Further studies were conducted to ascertain whether reliability could be improved by utilizing alternate or improved sensors. As a result of these studies, instrumentation which detects high pressure in the drywell has been incorporated in addition to the reactor low water level instruments to actuate reactor core spray cooling, HPCI, and LPCI and the standby diesel-generator systems.

Diversity of sensors which perform interlocking action of the ECCS pump have also been incorporated into the design. That is, two different types of pressure interlock sensors - bellow-type and bourdon-type - are used for this function in order to circumvent any unknown phenomenological uncertainties associated with pressure parameter measurements.

2.5.5 Main Steam Line Isolation Valve Testing Under Simulated Accident Conditions.

Concern

"Steam line isolation valves are provided which constitute an important safeguard in the event of failure of a steam line external to the containment. One or more valves identical to these will be tested under simulated accident conditions prior to a request for an operating license." (Browns Ferry Units 1 and 2, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260.)

Resolution

General Electric Company implemented a program to test a full size main steam line isolation valve under simulated accident conditions. This research and development program involved (1) testing of valves on a small scale to permit evaluation of hydro-dynamics of the blowdown under prototypical conditions: (2) testing of a valve essentially identical in design to those to be used in this plant simulating as closely as feasible the accident conditions; and (3) testing the main steam line isolation valves of this plant during the pre-operational test phase to verify that the valves as installed will meet functional requirements.

The detailed description of the program was presented in a General Electric Topical Report[1] submitted to the AEC in April, 1968. The testing programs have been successfully completed and reported in a GE Topical Report[2] submitted to the AEC in March, 1969. Analysis of the accident event is discussed in a GE Topical Report[3] submitted to the AEC in July, 1969.

Refer also to FSAR Subsection IV-6.

[1] *Bray, A.P., et at., APED-5608, "General Electric Company, Analytical and Experimental Program for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.*
[2] *"Design and Performance of GE-BWR Main Steam Line Isolation Valves", APED-5750, March, 1969 by D.A. Rockwell and E.H. Van Zylstra.*
[3] *"Consequences of a Steam Line Break for a GE-BWR", NEDO-10045, July, 1969, by J.E. Hench.*

2.5.6 Misorientation of Fuel Assemblies

Concern

"Operation with a fuel assembly having an improper angular orientation could result in local thermal conditions that exceed by a substantial margin the design thermal operating limits. The applicant stated that he is continuing to investigate more positive means for precluding possible misorientation of fuel assemblies." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket Nos. 50-259 and 50-260.)

Resolution

Operation with a misoriented fuel assembly would be an economic rather than a safety concern. Analyses have shown that less than 10 fuel rods in a misoriented assembly would experience a MCHFR less than 1.9. Under normal operating conditions these 10 fuel rods would, even in the peak power position, remain at a MCHFR greater than 1.0 and peak linear heat generation rate less than 28 kW/ft.

Studies into means of precluding possible fuel misorientation have been completed. It is concluded that the present method of procedural controls is the most desirable of the alternates. Fuel handling operations at operating GE-BWR's have shown this to be an efficient, effective method.

Various mechanical devices to prevent inserting a misoriented fuel assembly were also studied and eventually discarded. These devices tended to provide greater potentials for fuel damage during loading and storage operations than the misorientation they were designed to prevent.

Visual identification has been successfully used in all BWR's operated to date to provide assurance of fuel location and orientation. Photos taken of the KRB core after the initial fuel loading clearly showed four different means of identifying a misoriented fuel assembly: (1) All the assembly numbers point towards the center of the cell, (2) The spring-clip assemblies all face the control rod, (3) The lugs on the handles point towards the control rods, (4) Cell to cell symmetry. Experience has shown that the distinguishing features will be visible during the design lifetime of the fuel. In all cases, fueling procedures require that the fuel assembly number be verified. As a result of this study and the accumulated fuel handling experience, no further work with respect to providing an alternate means of preventing fuel assembly misorientation is planned.

Refer to CNS-FSAR Section III for further details.

2.5.7 Effects of Fuel Bundle Flow Blockage

Concern

"The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly, by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260.)

Resolution

The resolution of the above concerned item is presented in a GE topical report[1] submitted to the AEC in April, 1968.

Experience with fuel performance in operating reactors similar in design to this station, together with appropriate core mechanical analysis, has indicated that flow blockage during normal operation could only be local in nature and could not propagate to the extent that the remainder of the reactor core would be affected.

[1] Bray, A.P., "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.

Calculation of hydraulic forces under flow blockage conditions has indicated that the fuel channels would remain intact.

Analytical study of data derived from experimental work with induced melting of UO_2 at Argonne National Laboratory and Oak Ridge National Laboratory has indicated that melting of a portion of fuel assembly would not lead to unacceptable results in terms of fission product release, local high pressure production or initiation of failure in adjacent assemblies.

The nature of potential flow blockages is being examined and analyses are being performed to determine possible sequence of events and consequences. From these analyses an experimental program will proceed with test conditions approximating those from analysis as closely as possible. The experimental measurements will be used in conjunction with an analytical model to apply the results to the reactor situation or the results will be used directly to show that a safety concern does not exist.

The test program results have been submitted to the AEC as a GE topical report.[1]

2.5.8 Verification of Fuel Damage Limit Criterion

Concern

"A linear heat generation rate of 28 kW/Ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260.)

Resolution

The resolution of the above concerned item is presented in a GE topical report[2] *submitted to the AEC in April, 1968.*

General Electric Company believes that the data presently available is adequate to support the validity of the 28 kW/ft damage limit for the 1965 (low power density) and 1967 (high power density) product line BWR fuel. The fuel design and the associated linear heat generation rate have been selected as a result of development programs and experience over the past six to seven years. These programs, combined with extensive, large boiling water reactor operating experience, have demonstrated with a high degree of confidence that fuel integrity can be maintained in 1965 and 1967 product line BWR cores even for the worst anticipated transients.

General Electric Company has conducted fuel rod tests over a range of conditions to obtain data applicable to the design of the plant. Test fuel rods have been operated at various levels, including 17.5 kw/ft, and even high range 18.5 kw/ft, 21 to 22 kw/ft, and 28 kw/ft. These tests have verified that the calculational methods adequately predict the clad strain associated with a particular linear heat generation rate. In addition to tests performed by General Electric Company, tests in the range of 12 to 24 kw/ft have been performed by others.

Additional fuel tests are in progress as a development effort primarily to provide a basis for possible extensions in fuel technology. These data, as well as the operational history of BWR's placed in service prior to the operation of 1965 and 1967 product line plants, will provide additional confirmation of the present design bases and will demonstrate operation at heat generation rates comparable to the worst anticipated transients for both the 1965 and 1967 product lines.

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[1] *Scateau, G.J., "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor", NEDO-10174, July, 1970.*

[2] *Bray, A.P. "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.*

A summary of the fuel test programs and their results is given in Amendment 14/15 of Dresden Nuclear Power Station, Units 2/3 (AEC Docket Nos. 50-237 and 50-249).

A GE topical report[1] was submitted to the AEC on the final results of these test programs in June, 1970.

2.5.9 Control Rod Block Monitor Design

Concern

"The Rod Block Monitor system should be designed so that if bypassing is employed for purposes other than brief testing no single failure will impair the safety function." (Browns Ferry Units 1 and 2, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260).

Resolution

The rod block monitor (RBM) system was incorporated for operational reasons for the purpose of backing up the reactor operator in preventing a single operator error or a single equipment malfunction from causing fuel damage, and it is felt that the level or reliability provided by the system is consistent with this application. The applicant and General Electric Company do not consider this a safety system.

The control rod block action of the RBM system is not to be confused with the nuclear instrumentation system -- APRM rod block function. The APRM rod block is a bulk power control system. The RBM is a local power control system.

Refer to Section VII of the CNS-FSAR for further details and description on this system.

The analyses detailed in Appendix G support the non-safety status of this system by indicating that no unacceptable results are encountered because of a single operator error or single equipment failures associated with the control rod system assuming that the RBM system was completely unavailable.

An operational analysis was performed on this system. It demonstrated that an improbable event (a worst case rod pattern) plus five to seven operator and equipment malfunctions concurrently would only lead to an improbable failure of 150 fuel rods. This would not constitute a 10CFR20 dose event. The details of this analysis are presented in Dresden Amendment 19/20 (AEC Docket No. 50-237 and 50-245). Details of operator error analysis is given in Brunswick, Unit 1 and 2, Supplement 5 (AEC Docket No. 50-324 and 50-325).

2.5.10 Quality Assurance and Inspection of the Reactor Primary System

Concern

"The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system and of inspection during service life. The Committee recommends that the applicant implement those improvements in primary system quality which are practical with current technology." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260).

Resolution

Design and fabrication of the reactor primary system was of the highest quality practical with current technology. The reactor vessel is designed, fabricated, and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A for nuclear vessels. The design basis for other primary systems components will incorporate a quality level which produces equal serviceability to that of the reactor vessel. This is accomplished by specifying these components to meet applicable codes (ASME Section III, Class C or USAS B31.1.0). Installation procedures have been developed by the piping contractor and have been approved by all concerned parties. Quality

[1] "Current State of Knowledge High Performance BWR Zircaloy – Clad UO_2 Fuel", NEDO-10173, May, 1970.

assurance measures have been taken such as surveillance and auditing by NPPD and their contractors during the fabrication and installation of the primary system components, to assure that the established quality requirements for these components have been met.

Provisions are being made to the maximum extent considered feasible for inspection of primary system components during service life.

Refer to CNS-FSAR Section IV and Appendix D and Appendix J for further details.

2.5.11 Formulation of an In-Service Inspection Program

Concern

"The Committee will wish to review the detailed in-service inspection program at the time of request for an operating license." (Browns Ferry, ACRS Report, 3/14/67, AEC Docket Nos. 50-259 and 50-260).

Resolution

The in-service inspection program is based on Section XI of the ASME Boiler and Pressure Vessel Code, In-Service Inspection of Nuclear Reactor Coolant Systems, and follows the code to the maximum extent considered feasible. The program is described in CNS-FSAR Appendix J, "In-service Inspection Program."

2.5.12 Station Start-up Program

Concern

"In view of the high design power density of the core, an especially careful and extensive start-up program will be required for this plant. If the startup program or the additional information on fuel behavior referred to above should fail to confirm adequately the designer's expectations, plant modifications or restrictions on operation may be appropriate." (Browns Ferry, ACRS Letter, 3/14/67, AEC Docket No. 50-259 and 50-260.)

Resolution

The extent and scope of the start-up program for this station will reflect consideration appropriate for the size of the reactor and the thermal characteristics, service or transient conditions which might affect fuel integrity, reactor control and response characteristics, and functional performance of safeguard features contained in the station design.

In particular, extensive surveys of reactor core power distribution will be performed during the initial approach to reactor power. It is expected that this program will demonstrate that power distributions, as good or better than predicted, will be realized. Appropriate steps will be taken to ensure that safety margins are maintained under operational conditions, including any power limitation adjustments needed to maintain the thermal margin limits of 18.5 kW/ft maximum and MCHFR $\leq$ 1.9 of power (see Section III of the FSAR).

A GE topical report[1] was submitted to the AEC on a summary of results obtained from a typical start-up and power test program for a GE-BWR in February, 1969. The start-up test program for Cooper Nuclear Station is outlined in Subsection XIII-5 of the FSAR.

[1] "Summary of Results Obtained from a Typical Start-up and Power Test Program for a GE-BWR," APED-5698, February, 1969.

3.0 *AREAS SPECIFIED IN THE COOPER NUCLEAR STATION AEC STAFF CONSTRUCTION PERMIT SAFETY EVALUATION REPORT*

3.1 *Introduction*

The AEC Staff Construction Permit Safety Evaluation Report, dated April 4, 1968 identified three general areas of specific concerns in Section V:

a. *Development Program of Significance for all Large Water Cooled Power Reactors*
b. *Development Program of Significance for Boiling Water Reactors*
c. *Areas Requiring Further Technical Information*

3.2 *General*

"A number of areas requiring further analytical, experimental, design development, or testing efforts to substantiate the adequacy of system design and safety features of the present generation of boiling water reactors similar to Cooper Station have been identified. Some of these matters are pertinent not only to boiling water reactors, but to all large water-cooled power reactors, particularly those employing high performance cores similar to the Cooper reactor. There are other development areas which are of significance solely to boiling water reactors. In the following sections, we have differentiated between these two categories. In addition, we have identified four areas in which further information will be required as the facility design progresses but which we feel are now adequate." (Cooper, AEC Staff Safety Evaluation Report, 4/4/68, AEC Docket No. 50-298.)

3.3 *Development Program of Significance for all Large Water-Cooled Power Reactors*

3.3.1 *Linear Heat Generation Rate Fuel Damage Limit*

Statement

"A linear heat generation rate of 28 kilowatts per foot is used by the applicant as a fuel element damage limit. In Amendment 2 of the PSAR, the applicant outlined a fuel element test program which will cover the worst anticipated transient heat generation rates, and maximum expected fuel burnup. Test fuel rods have been operated at various linear heat generation levels, and have verified calculational models. Additional work is planned which includes experience with high burnup of fuel (20,000 to 30,000 MWD/T) and long-term operation at high linear heat generation rates, of capsules as well as complete fuel assemblies. These tests cover the spectrum of anticipated operating conditions of Cooper Station and thus we believe that the work done to date and anticipated will solve outstanding questions in this area. The results of this test program are expected to become available in 1969, prior to the date proposed for initial operation of Cooper Station."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.8.

3.3.2 *Local Fuel Melting Resulting from Inlet Coolant Orifice Blockage*

Statement

"The applicant has undertaken a research and development effort in this area. Preliminary results have indicated that flow blockage during normal operation is local in nature, and cannot propagate to affect the remainder of the core. Additional analytical and experimental work will be conducted to confirm the results of these preliminary studies. Program completion is scheduled for early 1969."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.7.

3.3.3 Effect of Fuel Clad Failure on Emergency Core Cooling

Statement

"Based upon analytical and experimental work done to date, clad perforation occurs at a localized area of a fuel rod. Perforation is caused by high internal pressure and the point at which perforation occurs is random, depending upon a weak point in the rod. Further experimental and analytical work will be continued in order to confirm and further refine the understanding of this fuel damage model. This work will include further perforation tests of fuel cladding under various conditions of temperature, pressure and metal ductility, further heat transfer analysis of fuel bundles under accident conditions, and other tests as appropriate. A report of the results of these tests is scheduled for the end of 1968. Based upon the work done to date and the scope and schedule for the test programs, we believe there is reasonable assurance that this area will be satisfactorily resolved prior to the date proposed for initial operation of Cooper Nuclear Station."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.1.

3.4 Development Program of Significance for Boiling Water Reactors

3.4.1 Core Spray Effectiveness

Statement

"Analytical and test work is currently underway to optimize the core spray systems for the General Electric boiling water reactors. Application of the core spray system to the Cooper Station reactors will be based on the results of this development work. Appendix E of the PSAR provides a summary of the core spray test program.

"To date, core spray tests have been conducted at fuel bundle powers greater than expected in Cooper, and at water flow rates lower than that which will be provided. The latest results on core spray tests have been reported in Amendment 2 of the PSAR. These latest 49-rod fuel bundle tests indicate that wetting of both sides of a fuel bundle channel by the core spray flow can reduce peak clad temperatures significantly. Future experimental work will include testing at higher fuel temperatures and using Zircaloy rather than stainless steel clad so as to more closely simulate actual reactor condition. Tests have also been done and will continue on spray distribution over a simulated reactor core. In view of the effort expended on this matter to date and the plans for continued work scheduled through 1968, we believe that this matter will be resolved prior to the date proposed for initial operation of Cooper Station."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.2.

3.4.2 Steam Line Isolation Valve Testing

Statement

"General Electric Company is currently developing a program to test the function and closure time of main steam line isolation valves under simulated accident conditions. Three specific programs have been planned (Amendment 2). The first includes small scale tests to observe the phenomenon of high speed steam being stopped by valve closure; the second is a full scale test where the steam flow rate through the valve is increased over normal flow; and the third program will simulate accident situations where the valve is subjected to conditions of high steam flow with water entrainment. Success in these programs is the ability of the valves to close in a required period of time.

"The results of these tests scheduled for the end of 1968 are expected to satisfactorily demonstrate the performance characteristics of the steam isolation valves. We expect that this matter will be satisfactorily resolved prior to the date proposed for initial operation of Cooper Nuclear Station."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.5.

3.4.3 Adequacy of HPCI System as a Depressurizer

Statement

"The principal function of the HPCI system is to maintain water inventories sufficient to assure core cooling for postulated small breaks. For intermediate breaks around 0.3 square feet, it serves to depressurize the pressure vessel to a low enough level so that the core spray system or low pressure injection system can reach rated flow. The HPCI system is designed to pump 4250 gpm into the reactor pressure vessel within a reactor pressure range of about 1100 psig to 150 psig.

"We are continuing to follow the analytical techniques for predicting vessel depressurization using the HPCI system. In Amendment 2 of the PSAR, it was shown that over the spectrum of liquid break sizes, for the HPCI-LPCI combination, a minimum mixing efficiency of about 85% was required to prevent clad melting. However, in our opinion, the small difference between the calculated peak clad temperature for the HPCI-LPCI combination and the melting temperature of 3370°F does not allow adequate margin for uncertainties.

"The applicant has stated that the required mixing efficiency for the HPCI system will be obtained by optimizing the feedwater sparger design; i.e., by using a large number of small sparger holes to maximize the HPCI exit velocity and jet surface area, and by aiming the sparger holes to maximize wetted film area. This proposed optimization of the feedwater sparger design would be based on an analytical model which has not been verified. Therefore, the General Electric Company has formulated an experimental test program to determine HPCI mixing efficiencies. It is planned that the proposed tests will be completed in 1968. We believe that the principle of using the HPCI to depressurize the reactor has been adequately demonstrated. The experimental program should demonstrate the feasibility."

Resolution

The resolution of the above concern item is presented in a GE topical report[1] submitted to the AEC in April, 1968.

The primary function of the high pressure coolant injection system (HPCIS) is to provide a coolant makeup to the reactor vessel to keep the reactor core covered and cooled for small system breaks. The secondary function is to depressurize the reactor so that the low pressure coolant injection system or the reactor core spray cooling system in the ECCS network can become effective for somewhat larger breaks than can be handled entirely by HPCIS inventory makeup. An analytical model based upon solution to the mass and energy balances for the system assuming thermodynamic equilibrium is used to predict the depressurization characteristics due to HPCIS operation. Because equilibrium does not actually exist, a calculated "mixing efficiency" is used to represent how nearly the injected subcooled water is raised to the temperature of the reactor vessel fluids.

Engineering tests were conducted in which subcooled water was injected into a constant volume, high pressure steam-water system designed to simulate reactor conditions and geometry. Depressurization rate, inlet and fluid temperatures were measured. An overall mixing efficiency was evaluated. A sufficient range of variables were included in the tests such as to determine a mixing efficiency for each reactor primary system. Please refer to CNSFSAR Section VI for further details.

The results and successful completion of this test program were submitted to the AEC in a GE topical report[2] in June, 1969.

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1 Bray, A.P., "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.

2 Rogers, A.E., and Torbeck, J.E., "Depressurization Performance of the GE-BWR HPCIS", APED-5447 (June, 1969).

3.4.4 Engineered Safety Features -- Electrical Equipment Inside Containment

Statement

"Electrical equipment which must operate inside the primary containment in an accident environment is limited to isolation valve operators and cables. Where practical, the valves are designed to fail "as is" or closed (safe failure). A circuit failure after the valve has closed will be a safe failure. In addition to designing the equipment to withstand the accident environment long enough to operate the valves, the applicant will perform environmental testing. In Amendment 2 of the PSAR, the applicant stated that the manufacturers will test a sample of cable and a motor operator of the type to be installed in the Cooper Station primary containment. The tests will demonstrate that the material and equipment will survive the accident conditions of simultaneous pressure, temperature, and humidity for a period of time essential for their operation."

Resolution

Tests have been completed on cables and operators for isolation valves, recirculation line valves and relief valves. The test results indicate the capability of the equipment to meet or exceed the design requirements and to function while under postulated accident conditions.

3.4.5 Control Rod Worth Minimizer

Statement

"The applicant has stated that the basic system will have been tested, installed, and operated on a number of General Electric boiling water reactors prior to use at Cooper Station. A prototype system was installed in early 1965 in Dresden Unit 1 for test purposes.

"We expect that the operating data that will be forthcoming from these reactor plants will be sufficient to determine the adequacy of the rod worth minimizer for Cooper Station."

Resolution

The design of the Control Rod Worth Minimizer (CRWM) is complete as reported in a GE topical report[1] submitted to the AEC in March, 1967. Refer to CNS-FSAR Section VII for further details.

3.4.6 Jet Pump Development

Statement

"Considerable analytical and test work has been completed on the jet pump system for reactor coolant recirculation to establish its basic design characteristics. Additional development programs in progress, and planned, are summarized in Appendix D of the PSAR.

"This development program, and the fact that this device will have been operated on other reactors prior to its application on Cooper Station will be adequate to determine its capability."

Resolution

The design and test program of the Jet Pump Assemblies is complete and is reported in a GE topical report[2] submitted to the AEC in September, 1968.

[1] Stanley, L., Staff, J.D., and Thompson, O.A., "Control Rod Worth Minimizer", APED-5449, March, 1967.
[2] Holland, K., "Design and Performance of GE-BWR Jet Pumps", APED-5460 September, 1968.

3.4.7 Rod Velocity Limiter

Statement

"The rod velocity limiter, which is designed to limit the free-fall velocity of a control rod is being tested. This device will also have been tested during the pre-operational test phase in other boiling water reactors prior to its application in Cooper Station."

Resolution

The design and final test program of the Control Rod Velocity Limiter (CRVL) is complete as reported in a GE topical report[1] submitted to the AEC in March, 1967. Refer to CNS-FSAR Section III for further details.

3.4.8 In-Core Neutron Monitor System

Statement

"In-core startup and power neutron detectors have been developed to reduce neutron source requirements and to improve neutron flux monitoring capability in the startup and power ranges. Testing of these devices is presently being done in the Consumers Power Company's Big Rock Point reactor. The applicant has stated that the in-core detectors in this reactor have given excellent results and demonstrated satisfactory sensitivities in repeated counting cycles through subcritical and critical operation and have demonstrated counting ability during hot startup after a scram. A life test will also be conducted to demonstrate the feasibility of leaving the chambers in the high flux regions continuously. Identical chambers will have been in operation in boiling water reactors for several years before the first unit of this plant is operational. A complete report on this item is scheduled in 1968.

"Because of the experience described, satisfactory in-core testing will have been conducted to demonstrate the adequacy of these monitors prior to operation of Cooper Station."

Resolution

The design and adequate performance demonstration of the In-Core Nuclear Instrumentation System is complete and is reported in a set of GE topical reports[2,3] submitted to the AEC in August, 1968 and November, 1968 respectively. Refer to CNS-FSAR Section VII for further details.

3.5 Areas Requiring Further Technical Information

3.5.1 Failure of Passive Components of ECCS

Statement

"The torus water distribution system and ECCS pumps are located on the lowest level of the reactor building. To assure that a leak in the torus or the distribution system to the pumps would not flood other ECCS equipment, each ECCS pump will have a separate suction line from the suppression chamber. The pumps of the individual systems will be spatially and structurally isolated from one another. In the event of leakage from an active component of a system, the compartment housing the components will serve to segregate and isolate the leakage and provide means of detection. Passive failures could potentially flood an individual pump compartment or could result in

1 "Control Rod Velocity Limiter," APED-5446, March, 1967.

2 Morgan, W., "In-Core Nuclear Instrumentation Systems for Oyster Creek, Unit 1 and Nine Mile Point, Unit 1 Reactors", APED-5456, August, 1968.

3 Morgan, W., "In-Core Nuclear Monitoring Systems for GE-BWR", APED-5706, November, 1968, revised April, 1969.

flooding of the concrete cell in which the torus is situated. The design will be such that only a single pump would be lost due to flooding and also that NPSH for the remaining pumps will not be lost due to loss of torus water.

"We believe that this design reasonably protects the ECCS from passive failures which could negate ECCS function."

Resolution

Standby coolant supply connections and RHR crossties have been provided as discussed in CNS-FSAR Subsection IV-8.

3.5.2 Thermal Shock

Statement

"The effect of thermal shock on the reactor vessel and its appurtenances induced by injection of emergency core cooling water into the higher temperature reactor system has not yet been completely analyzed. The applicant has partially responded in Amendment 2. He will perform a complete analysis to demonstrate that the primary and secondary stresses from thermal shock through operation of the emergency core cooling system will meet the criterion outlined in their response, which are:

1. *Primary stress shall be within the applicable limits per the ASME Code Section III.*

2. *Secondary stress shall be determined and the associated strains shall be limited so that the capability to safely shut the plant down is not compromised."*

Resolution

A detailed reactor vessel thermal shock analysis was performed on a representative GE-BWR reactor vessel. The thermal shock analysis simulating ECCS-LOCA operation was performed on similar reactor vessel design to the Cooper Nuclear Station vessel and is reported in a GE topical report[1] *submitted to the AEC in July, 1969.*

The thermal shock analysis simulating ECCS-LOCA conditions was made on the reactor internals, including the core spray sparger and the reactor vessel shroud, and is described in the CNS-FSAR Subsections III-3 and IV-2.

3.5.3 Interchannel Flow Stability

Statement

"The General Electric Company has indicated that they are continuing their studies on interchannel flow stability and will keep us informed of their findings as they become available. We intend to continue our consideration of this matter. Additional analytical results and reactor operating data are expected to become available prior to the date proposed for initial operation of Cooper Nuclear Station."

Resolution

The development of a BWR Stability Model which would predict the onset of instabilities in the reactor core in this station has been completed and the excellent agreement between model predictions and experimental data has been reported in the following GE Topical Report[2 3] *submitted to the AEC and in GE*

[1] *Hsu, L.C., "An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (LOCA)", NEDO-10029, July, 1969.*

[2] *Holland, K., & Ianni, P.W., "Stability and Dynamic Performance of the GE-BWR", APED-5652, April, 1969.*

[3] *Crowther, R., "Xenon Considerations in Design of Large BWR", APED 5640, June, 1968.*

memorandum[1]*, submitted on Peach Bottom Atomic Power Station, Units 2 and 3, AEC Docket Nos. 50-277 and 50-278. Refer to Subsection VII-6 of this CNS-FSAR for further details.*

3.5.4 Inservice Inspection

Statement

"Provisions are being incorporated in the design of Cooper Station to facilitate inspection of selected areas of the interior of the reactor vessel and its components, as recommended in the report, APED-5450. This report, titled "Design Provisions for In-Service Inspection," was submitted by the General Electric Company to the regulatory staff on May 4, 1967. We will review the applicant's in-service inspection program at the operating license stage."

Resolution

Refer to CNS-FSAR Appendix H, Subsection H-2.5.11.

3.5.5 Primary System Leak Detection

Statement

"Detection of leaks in the primary system will be accomplished by monitoring the sump level in the containment vessel, by monitoring the ΔT of the cooling water of the containment vessel heat exchangers, and by monitoring the containment vessel pressure. By these means, a range of leaks of less than 1-gpm up to 40-gpm can be detected. Selected areas in the reactor building in the vicinity of the RCIC and RHR equipment will also be monitored."

Resolution

Leakage is collected in the sumps. Sump levels are monitored and sump pump discharges are metered to detect abnormal leakage rates. (Refer to Section IV, "Nuclear System Leakage Rate Limits".)

[1] *"Technology of BWR Stability Analysis" – GE Memorandum SC ER-60, July, 1967.*

4.0 *AREAS SPECIFIED IN OTHER RELATED AEC-ACRS CONSTRUCTION AND OPERATING PERMIT LETTERS (REFER TO TABLE I-11-4 OF CNS-FSAR SUBSECTION I-11)*

4.1 *General*

Development, testing and analysis programs are continuing in several other areas of related interest.[1] Other study programs which are related directly to the high power density reactor core design of Cooper Nuclear Station and indirectly to other low power density reactor core boiling water reactors are now near completion. Study programs related to reactor designs which have been reviewed since the Cooper Nuclear Station construction permit issuance are being pursued and will be issued soon. The information developed in these programs will be addressed to several of the technical concerns which have been voiced by the AEC-Advisory Committee on Reactor Safeguards (ACRS) recently with respect to the General Electric BWR product lines. The ACRS issues on the following facilities are identified and the Cooper Nuclear Station design capabilities relative to them are discussed in this section.

a. *PECO - Peach Bottom Units 2 and 3, ACRS Letter, 10/12/67, AEC Docket Nos. 50-277 and 50-278*

b. *VYNPS-Vermont Yankee, ACRS Letter, 6/15/67, AEC Docket No. 50-271*

c. *BECO - Pilgrim Unit 1, ACRS Letter, 4/12/68, AEC Docket No. 50-293*

d. *TVA - Browns Ferry Unit 3, ACRS Letter, 5/15/68, AEC Docket No. 50-296*

e. *GPC - Hatch Unit 1, ACRS Letter, 5/15/69, AEC Docket No. 50-321*

f. *CP&L - Brunswick Units 1 and 2, ACRS Letter, 5/15/69, AEC Docket Nos. 50-324 and 50-325*

g. *JCPLC - Oyster Creek Unit 1, ACRS Letter, 12/12/68, AEC Docket No. 50-218*

h. *NMPC - Nine Mile Point Unit 1, ACRS Letter, 4/17/69, AEC Docket No. 50-219*

i. *CECO - Dresden Unit 2, ACRS Letter, 9/10/69, AEC Docket No. 50-237*

Additional water-cooled reactor design ACRS concern items were documented in the PG&E - Diablo Canyon Unit 1 (AEC Docket No. 50-275) ACRS letter of 12/20/67. These items are also identified and the Cooper Nuclear Station design capability is discussed.

Thus, although these items have not been addressed as requirements for this station, a detailed comprehensive review of each item and the Cooper Nuclear Station design conformance to it is analyzed in the following subsections.

4.2 *LPCIS-Logic Control System Design*

Concern

"The applicant proposes to use sensing devices in the recirculation loops of the reactor to detect the location of a pipe break. Signals from these devices would be used automatically to select various valve actions that are essential to the proper operation of the emergency core cooling systems. In view of the importance of the proper valve actions in the unlikely event of a major pipe break, the Committee recommends that the sensing instrumentation and valve control system be designed to full reactor protection system standards, and that consideration be given to providing more than one type of sensing device in the system." (Vermont Yankee-ACRS Letter, June 15, 1967, AEC Docket No. 50-271.)

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[1] *Bray, A.P., "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.*

Resolution

The engineered safeguards with respect to core standby cooling systems includes a low pressure coolant injection system which is capable of reflooding the reactor core following a design basis loss-of-coolant accident. This system is equipped with sensing and initiating equipment which is capable of reliably detecting which of the two reactor coolant recirculation system loop lines is not associated with the reactor primary system rupture so that the coolant injection can occur in the proper loop line. The current technique for sensing this information by means of flow direction sensing devices and a logic control system has been found to have sufficient reliability and sensitivity to be an acceptable system. This system is described in Subsection VII-4 of the CNS-FSAR

4.3 *Re-Evaluation of Main Steam Line Break Accident*

Concern

"Fuel clad temperatures following a steam line break should be further evaluated during detailed design, with due attention to using conservative assumptions and methods in calculating these temperatures. Steam line isolation valve closure time as short as three seconds may be required to maintain acceptable low fuel clad temperatures in this accident. This applicant has stated that isolation valves with closure time adjustable from 3 to 10 seconds will be obtained for the plant." (Vermont Yankee-ACRS Letter, June 15, 1967, AEC Docket No. 59-271.)

Resolution

The resolution plan for the above concern item is presented in a GE topical report[1] submitted to the AEC in April, 1968. Subsequently, a more extensive study of this phenomena was undertaken. The program has been completed and a GE topical report[2] has been submitted to the AEC (July, 1969). Verification that the assumption of 10 seconds for the main steam line isolation valve closure time does not result in excessive clad temperatures or release of radioactive contaminants is given in Section XIV of the FSAR.

4.4 *Design of Piping Systems to Withstand Earthquake Forces*

Concern

"The Committee recommends that the applicant give special attention to the design of the critical elements of the plant piping, including the drywell torus connections, to ensure that these elements are not overstressed under maximum earthquake forces." (Vermont Yankee-ARCS Letter, June 15, 1967, AEC Docket No. 50-271.)

Resolution

Critical elements of the station piping, including the connections of that piping to the drywell and torus of the primary containment, are designed to withstand, without overstress, the maximum forces resulting from the maximum possible earthquake which is approximately two times the maximum probable earthquake to be expected at the site. This was accomplished by the performance of appropriate static or dynamic analyses of the piping in systems critical to reactor safety or to safe shutdown of the station. The stresses resulting from these earthquake forces have been calculated and are within the limits for the piping materials and other associated components involved, according to appropriate ASA and ASME Codes. Refer to Subsection XII-2, Appendix A and Appendix C, of the FSAR for further information. A detailed analysis of a typical GE-BWR and CB&I primary containment construction may be found in Dresden 2/3, Amendments 13/14, (AEC Docket No. 50-237 and 50-245).

[1] *Bray, A.P., "The General Electric Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.*

[2] *"Consequences of a Steam Line Break for a GE-BWR", NEDO-10045, by J.E. Hench, NEDO-1004 July, 1969.*

4.5 <u>Fuel Clad Disintegration Limitations</u>

Concern

"In connection with postulated loss-of-coolant accidents, the applicant stated that, using conservative assumptions and allowing appropriately for fuel element distortion from the original core geometry, the emergency core cooling systems will be designed to keep fuel-clad temperatures below the point at which the clad may disintegrate upon subsequent cooling." (Pilgrim-ACRS Letter, April 12, 1968, AEC Docket No. 50-293.)

Resolution

With respect to this overall concern of emergency core cooling systems (ECCS) effectiveness to cool overheated fuel rods, the industry had selected a maximum allowable temperature criterion of 2700 °F. This selection was based on a desire to keep the fuel bundle geometry intact. Please see the referenced GE topical report submitted to the AEC.[1]

Even though this criteria has been adopted for emergency core cooling systems (ECCS) equipment design, experimental effort continues at both General Electric Company and elsewhere to further refine our knowledge with respect to a proper tolerable maximum fuel temperature during loss-of-coolant accidents.

Some preliminary data from Argonne[2] tends to indicate that possible clad shattering rather than clad melting might be a more conservative criteria for loss of geometry intactness. This shattering might occur at temperatures as much as 400 °F below the Zircaloy melt temperature. Current testing programs at GE for special fuel bundle designs are fully investigating this possibility.

The current conservative core cooling evaluation techniques used on this station indicate that the maximum predicted fuel temperatures for the BWR reactor core postulated design basis loss-of-coolant accidents (less than 2000 °F) are sufficiently below any temperatures of clad shattering. These evaluation techniques indicate that there is no concern that a potential modification in the maximum allowable temperature criteria would adversely influence the present sizes of the core standby cooling systems equipment.

General Electric is developing high-temperature Zircaloy-clad electrically heated fuel rod simulators for use in full-size Zircaloy-clad bundle tests. Testing conditions have been selected (1) to duplicate cooling modes, initial temperatures, coolant flow rate, power transients, subcooling temperature, and time of cooling initiation representative of the multitude of tests performed with stainless steel clad heaters, and (2) to investigate ECCS effectiveness at peak temperatures in excess of 2500 °F, to the highest temperatures the heaters will permit. Refer to CNS-FSAR Section VI for further details.

The results of this and related program are reported in a GE Topical Report[3] and Amendment 14 to the Pilgrim Unit 1 Docket, Question 8.0.

4.6 <u>Automatic Pressure Relief System - Initiation Interlock</u>

Concern

"The applicant stated that he would give further consideration to a suitable interlock to ensure that low-pressure cooling capability would be available before the auto-relief depressurization could be initiated." (Pilgrim-ACRS Letter, April 12, 1968, AEC Docket No. 50-293.)

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[1] Bray, A. P., et al, "The GE Company, Analytical and Experimental Programs for Resolution of ACRS Safety Concerns", APED-5608, April, 1968.
[2] ANL 7438, Progress Report, March, 1968.
[3] "Effects of Cladding Temperature and Material on ECCS performance", NEDO-10179, June, 1970.

Resolution

A system is being installed to sense pressure downstream from the core spray and the LPCI mode of RHR pumps and to prevent auto-depressurization unless the pressure is above a value assuring the capability of low pressure core cooling.

Pressure sensors are being installed downstream of each of the ECCS pumps. The signals from these pressure sensors feed to a logic matrix unit. If the logic matrix unit determines from the sensed pressure that adequate core cooling is available, autoblowdown will be permitted to proceed upon receipt of its own initiation signals of low reactor water level, and high drywell pressure, and if timer elapsed time has run out.

Please refer to CNS-FSAR Section VI and Subsection VII-4 for further details.

4.7 *Effects of Blowdown Forces on Reactor Primary System Components*

Concern

"The effects of blowdown forces on core and other primary system components should be analyzed more fully as detailed design proceeds." (Diablo Canyon ACRS Letter, 12/20/67, AEC Docket No. 50-275.)

Resolution

The reactor core structural components are designed to accommodate the loadings applied during normal operation and maneuvering transients. Deflections are limited so that the normal functioning of the components under these conditions is not impaired. Where deflection is not the limiting factor, the ASME Boiler and Pressure Vessel Code, Section III, was used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure.

The loading conditions which occur during excursions or design basis loss-of-coolant accidents were examined. The reactor core shroud, shroud support, and jet pump body, which comprise the inner vessel around the core within the reactor vessel, are designed to maintain a reflooding capability following a design basis loss-of-coolant accident. Reflooding the reactor core to the top of the jet pump inlets provides adequate cooling of the fuel.

The design of the jet pump parts takes into account the pressure loading both in normal and accident conditions and the reactions at the supporting brackets due to differential thermal expansion of the pump and reactor vessel.

The reactor internals were designed to preclude failure which would result in any part being discharged through the main steam line, in the event of a steam line break, which might cause a blocking of a main steam line isolation valve.

The structural components which guide the control rods were analyzed to determine the loadings which would occur in a design basis loss-of-coolant accident. The reactor core structural components are designed so that deformations produced by accident loadings do not prevent insertion of control rods.

Further details on this analysis are described in the CNS-FSAR Sections III and IV and Appendix C.

4.8 *Separation of Control and Protection System Functions Concern*

Concern

"The applicant has proposed using signals from protection instruments for control purposes. The Committee believes that control and protection instrumentation should be separated to the fullest extent practicable. The Committee believes that the present design is unsatisfactory in this respect but that a satisfactory protection system

can be designed during the construction of this reactor. The Committee wishes to review an improved design prior to installation of the protection system." (Diablo Canyon, ACRS Report, 12/20/67, AEC Docket No. 50-275.)

Resolution

The reactor protection system, independent from the station process control and indication systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically and independently initiates appropriate action whenever the conditions approach pre-established operational limits.

All redundant instrumentation provided for safe reactor shutdown are powered from separate sources. The connections to this redundant instrumentation and controls are routed in separate wireways (conduit, trays, etc.) via independent paths to reduce the possibility of loss of both cables in the event of an accident condition or fire.

Please refer to CNS-FSAR Section VI and Subsections VII-2, VII-3 and VII-4 for further details.

4.9 *<u>Instrumentation for Prompt Detection of Gross Fuel Failure</u>*

Concern

"Considerations should also be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element." (Diablo Canyon, AEC Letter, 12/20/67, AEC Docket No. 50-275.)

Resolution

Gross failure is detected by the main steam line radiation monitors. Please refer to Responses 7.5 and 9.4.2 of Supplement No. 3 and to Response 7.5 of Supplement No. 4 of the Brunswick Steam Electric Plant Units 1 and 2 (AEC Docket Nos. 50-324 and 50-325). It is shown that the GE-BWR failed fuel element detection capability for gross failure is available and sufficient with a considerable margin of conservatism.

The referenced Brunswick submittals discuss the design criteria for the instrumentation for prompt detection of gross failure of a fuel element which is also applicable for this facility as well as all other GE-BWR projects.

In essence the GE-BWR detection system instantaneously detects and takes the necessary corrective action for not only gross, immediate fuel failures, but also local minor, long term fuel failures.

Please refer to CNS-FSAR Section VII for further details.

4.10 *<u>Scram Reliability Study</u>*

Concern

"The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. In the event of a turbine trip, reliance is placed on prompt control-rod scram to prevent large rises in primary system pressure. The applicant and his contractors have devoted considerable effort to providing a reliable protective system. However, systematic failures due to improper design, operation or maintenance could obviate the scram reliability. The Committee recommends that a study be made of further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients." (Brunswick Units 1 and 2, ACRS Letter, 5/15/69, AEC Docket No. 50-324 and 50-325.)

Resolution

Studies are being performed by General Electric Company:

a. To evaluate common mode failures which could negate scram action; and

b. Of design features to make tolerable the consequences of failure to scram during anticipated transients.

A description of the intended study programs is described in Brunswick Steam Electric Plant, Units 1 and 2 - Supplement No. 6, C/R 8.0 (AEC Docket No. 50-324 and 50-325).

A topical report on common mode failures which could negate scram action is to be submitted to the AEC. A second report on features to make tolerable the consequences of failure to scram is to be submitted to the AEC as an amendment to Hatch I.

4.11 Design Basis of Engineered Safety Features

Concerns

"For purposes of design of the engineered safety features, the applicant has proposed using a fission-product source term smaller than that suggested in TID-14844, and a treatment of this source within the containment different from that recommended in the same document. The Committee believes that the assumptions of TID-14844 should be used as a design basis for the engineered safety features of the Brunswick plant, unless and until the use of a different set of assumptions has been justified to the satisfaction of the Regulatory Staff and the ACRS." (Brunswick Units 1 and 2, ACRS Letter, 5/15/69, AEC Docket No. 50-324 and 50-325.)

Resolution

The engineered safety features were analyzed using the TID-14844 source terms. An assessment of the capability of the ECCS equipment to perform their intended functions is given in CNS-FSAR Subsection XIV-9, along with the off-site radiological effects of design basis accidents for such source assumptions.

4.12 Hydrogen Generation Study

Concern

"Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. The Committee believes that the applicant should evaluate all problems which may arise from hydrogen generation, including various levels of Zircaloy-water reactions which could occur if the effectiveness of the emergency core cooling system were significantly less than that predicted. The matter should be resolved between the applicant and the AEC Regulatory Staff." (Brunswick Units 1 and 2, ACRS Letter, 5/15/69, AEC Docket No. 50-324 and 50-325.)

Resolution

Studies are continuing on the possible effect of radiolysis of water in the unlikely event of a loss-of-coolant accident. The studies will evaluate all problems which may arise from credible hydrogen generation. Studies are also intended to show possible methods of handling postulated quantities of hydrogen generated by radiolysis. Details on the proposed studies have been submitted on Supplement No. 4, Brunswick Steam Electric Plant, Units 1 and 2 (AEC Docket No. 50-324 and 50-325).

Two GE Topical Reports[1,2] were submitted to the AEC in which it is clearly established that very little hydrogen is evolved as the result of the DBA-LOCA assuming the minimum ECCS equipment being available for operation under all required failure modes. Even with further ECCS degradations, the modeled design clad

[1] *"Metal-Water Reactions-Effects on Core Cooling and Containment", APED-5454, March, 1968.*
[2] *"Considerations Pertaining to Containment Inerting", APED-5654, August, 1968.*

temperatures (of approximately 2000°F) would not increase to levels (2800°F) where clad shattering or 1 percent metal-water reactions could take place. The containment metal-water reaction capability is 50 to 100 times the ECCS hydrogen levels.

A detailed discussion and analysis of the hydrogen problem in general is contained in Dresden Nuclear Power Station, Unit 3, Docket No. 50-238, Amendment 23 "Hydrogen Generation in a Boiling Water Reactor." Submitted 7/31/70.

4.13 *Seismic Design and Analysis Models*

Concern

"The applicant is reviewing the seismic design of Class I structural and mechanical components of the plant and will complete his analysis before the reactor goes into operation. In the event that changes to the plant should be found necessary, such changes will be made on a time scale to be agreed upon between the applicant and the Regulatory Staff." (Dresden Unit 2, ACRS Letter, 9/10/69, AEC Docket No. 50-237.)

Resolution

The piping systems were dynamically analyzed using the "response spectrum method" of analysis. For each of the piping systems, a mathematical model consisting of lumped masses at discrete joints connected together by weightless elastic elements was constructed. Valves were also considered as lumped masses in the pipe, and valve operators as lumped masses acting through the operator center of gravity. Where practical, a support is located on the pipe at or near each valve. Stiffness matrix and mass matrix were generated and natural periods of vibration and corresponding mode shapes were determined. Input to the dynamic analyses were the 0.5 percent damped acceleration response spectra for the applicable floor elevation. The increased flexibility of the curved segments of the piping systems was also considered. The results for earthquakes acting in the X and Y (vertical) directions simultaneously, and Z and Y directions simultaneously were computed separately. The maximum responses of each mode are calculated and combined by the root-mean-square method to give the maximum response quantities resulting from all modes. The response thus obtained was combined with the responses produced by other loading conditions to compute the resultant stresses.

4.14 *Automatic Pressure Relief System - Single Component Failure Capability - Manual Operation*

Concern

"The automatic pressure relief subsystem should be modified so that at least the manual actuation of the subsystem would not be prevented by any single failure in the subsystem." (Dresden Unit 2, ACRS Letter, 9/10/69, AEC Docket No. 50-237.)

Resolution

In order to provide an additional level of single component failure capability, the Automatic Depressurization System of the ECCS is designed to provide the system with the ability to sustain a d-c power failure in any of its d-c battery feeds. The system is designed and installed such that either of the redundant, independent 125 volt d-c battery system networks is available, automatically, for the required system action. This modification, along with the air reservoirs on the ADS valves, provides the system (when manually operated) with the single component failure criteria application capability.

4.15 *Flow Reference Scram*

Concern

"In the area of reactor instrumentation, the Committee believes: that the flux scram point should be automatically reduced to an appropriate level as the reactor recirculation flow is reduced below the normal full-power flow." (Brunswick Units 1 and 2, ACRS Letter, 5/15/69, AEC Docket No. 50-324 and 50-325.)

Resolution

An actual BWR facility test (Dresden, Unit 2) will be conducted in the near future to demonstrate that this feature restricts expected unit operational maneuvering and that its omission does not result in unacceptable consequences or dilute reactor protection within the facility when it is subjected to the design abnormal transients.

The flow reference scram system is being provided and will be connected and placed in service if the results of the Dresden 2 tests demonstrate its need.

The flow reference scram system will sum the flow sensed in each of the reactor coolant recirculation loops and provide a flow reference signal to vary the neutron flux scram set point. Flow will be sensed from one flow measurement venturi in each of the two reactor coolant recirculation loops.

The station safety analyses (Section XIV) demonstrated that, for all transients considered, the core is adequately protected with a fixed APRM scram trip setting at 120 percent of rated neutron flux and the high-pressure scram setting of 1070 psig. Therefore, it is intended to ultimately replace the automatic flow referenced scram with a fixed 120 percent scram setting, providing that initial power operation confirms the nuclear behavior characteristics used in these transient analyses.

4.16 *Matters of Current Regulatory Staff-Applicant Discussion*

Concern

"Several matters are still under discussion between the applicant and the Regulatory Staff. These include review of the need for separation of redundant components of the standby gas treatment system, and final revisions to the technical specifications. The ACRS believes these matters can be resolved by the applicant and the Regulatory Staff." (Dresden 2, ACRS Letter, 9/10/69, AEC Docket No. 50-237.)

Resolution

Standby Gas Treatment System

The standby gas treatment system design will provide both electrical and physical isolation capability to each of the two treatment trains. This design will provide highest possible degree of independence, isolation, and redundancy between the two full capacity treatment trains. Please refer to FSAR Sections V, VII, and VIII for further details.

Technical Specifications

The proposed facility technical specifications are submitted herewithin (Appendix B) to reflect current AEC operational requirements for similar facilities (Monticello Unit 1, Dresden, Pilgrim, and Browns Ferry.)

The applicant (NPPD) believes that all technical specification matters can be resolved with the Regulatory Staff

4.17 *Future Items of Considerations for Incorporation*

Concern

"Continuing research is expected to enhance safety of water-cooled reactors in other areas than those mentioned, for example, by the determination of the extent of radiolytic decomposition of cooling water in the unlikely event of a loss-of-coolant accident, development of instrumentation for in-service monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system, and evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As

solutions to the problems develop and are evaluated by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale." (Dresden 2, ACRS Letter, 9/10/69, AEC Docket No. 50-237.)

Resolution

Radiolytic Decomposition of Cooling Water

Please see Subsection H-4.12.

Development of Instrumentation - Vibration and Loose Parts Detection Studies

Please see Subsection H-4.19.

Consequences of Water Contamination - Structural Materials - LOCA

The applicant (NPPD) and the General Electric Company are presently unaware of any potential consequences or mechanisms of water contamination by structural materials and coatings in the design basis loss-of-coolant accident which could effect the health and safety of the public.

4.18 Development of Instrumentation - Primary Containment Leakage Detection System - Increased Sensitivity Studies

Concern

"It is recommended that supplemental and potentially more sensitive methods of primary system leak detection be studied, evaluated, and implemented if they provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The study and evaluation should be completed within a year." (Oyster Creek, ACRS Letter, 12/12/68, AEC Docket No. 50-218.)

Resolution

The Cooper Nuclear Station primary containment leakage detection system as described in FSAR Subsection IV-10 is both sensitive and reliable for its intended safety considerations. This system is based on the sump-pump technique approach to leakage detection phenomena. Other supplemental techniques as described in the FSAR are useful alternates and are considered as supplemental off-line specialty approaches versus the operational primary on-line process approach (the automatic sump-pump system).

The technical specifications justify the leakage rate sensitivity and operational readiness requirements of the system.

The results of studies committed by others (Jersey Central Power and Light Company, refer to ACRS Letter, Oyster Creek Nuclear Power Plant Unit 1, 12/12/68, AEC Docket No. 50-218) will be examined for appropriate disposition in regard to the Cooper Nuclear Station facility when such data is made available.

4.19 Development of Instrumentation - Vibration and Loose Parts Detection Studies

Concern

"The applicant has stated that he plans to study possible means of instrumenting and monitoring for vibration or for the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system and, by the time of the first refueling outage, to implement such means as are found practical and appropriate." (Nine Mile Point, ACRS Letter, 4/17/69, AEC Docket No. 50-219.)

Resolution

The applicant (NPPD) and the General Electric Company have no plans to provide instrumentation for monitoring for vibration or for the presence of loose parts in the reactor primary system. The results of studies committed by others (Niagara-Mohawk Power Corporation - refer to ACRS Letter - Nine Mile Point Nuclear Station Unit 1, 4/17/69, AEC Docket No. 50-219) will be examined by the applicant (NPPD) for appropriate disposition in regards to the Cooper Nuclear Station facility when such data is made available to it.

4.20 <u>ECCS Leakage Detection, Protection, and Isolation Capability</u>

Concern

"Engineered safety systems that are required to recirculate water after a loss-of-coolant accident should be designed so that a gross system leak will not result in critical loss of recirculation or in loss of isolation capability. The Committee believes that exception to this general rule may be made in respect to a very short run of pipe from the torus to the first valve, if extremely conservative design of the pipe (and its connection to the torus) is used and suitably remote operability of the valve is provided. The design of these systems also should provide adequate leak detection and surveillance capability." (Brunswick, ACRS Letter, 5/15/69, AEC Docket Nos. 50-324 and 50-325.)

Resolution

The design of the Cooper Nuclear Station although granted a construction permit under the AEC 27 Design Criteria meets and conforms to the intent of the recently proposed AEC 70 General Design Criteria (Refer to FSAR Appendix F). An examination of these criteria as given in the FSAR shows that this facility design meets the intent of all AEC Design Criteria with regards to ECCS and the station containment systems (Refer to Appendix F, Criterion 37 through 65). Examination of each of the AEC design Criterion requirements individually establishes that:

a. No AEC Design Criteria requires a Class I passive component failure(s) protection. That is, failure of pipes, valves, pumps, etc., is not required.

b. It does require the design . . . .

(1) to provide safety function assuming a failure of a single active component (Criteria 41);

(2) to provide safety systems which shall not share active components and shall not share other features of components unless it can be demonstrated that (i) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (ii) failure of the shared feature or component does not initiate a loss-of-coolant accident (LOCA), and (iii) capability of the shared feature or component to perform its required function is not impaired by the effects of a LOCA and is available during the entire period that this function is required following the accident (Criterion 44);

(3) to perform its required function and not be impaired by the effects of a LOCA (Criterion 42);

(4) to provide heat removal systems which prevent the containment from exceeding its design pressure. (Criterion 52)

The Cooper Nuclear Station design meets all the above (b.1 through b.4) but only under the single active component failure criteria. Provisions are also provided for mitigating the consequences of a non-Class I structure, system, component, or equipment failure on the Class I equipment required for safety functions.

With regard to leakage, the Cooper Nuclear Station design includes a station leakage detection, isolation, processing and makeup system. This system (consisting of many normal station operational subsystems) provides for leakage control capability by means of:

a. identifying the reactor building (or reactor primary system) leakage sources *b. efficiently isolating and controlling the sources*

c. effectively removing residual leakage water (before and after isolation)

d. conveniently and capably replacing the leakage liquid and/or restoring the source system function.

The above is done under normal operation or post-accident conditions in a manner in which normal (10CFR20) or accident (10CFR100) off-site dose limits do not exceed established values and in a manner in which core and containment cooling continuity is not impaired or negated.

Refer to CNS-FSAR Subsection IV-10 for additional details on this system. The normal or credible accident induced leakage cited above is the leakage from active components in the ECCS. These sources represent a maximum value of from 10 to 50 gpm in leakage rates.

The thorough examination of the AEC General Design Criteria reference above appears in Brunswick Steam Electric Plant Units 1 and 2 - Supplement 4, C/R 6.4 (AEC Docket Nos. 50-324 and 50-325). The analysis of the events applied to the Cooper Nuclear Station facility provide similar results.

4.21 Main Steam Lines - Standards for Fabrication, Q.C. and Inspection

Concern

"The Committee has reviewed the applicant's proposal concerning standards of design, fabrication, and inspection of the steam lines downstream of the second isolation valves. The Committee concurs with the approach used in analyzing the stresses in the piping during an Operating Basis Earthquake. The Committee recommends that a program of spot radiography of the field butt welds be employed by the applicant as a quality control measure. Consideration should be given to an appropriate program of inservice inspection." (Brunswick, ACRS Letter, 5/15/69, AEC Docket Nos. 50-324 and 50-325.)

Resolution

The main steam piping downstream of the isolation valves is designed in accordance with USAS B.31.1, 1967. The inspection criteria for all shop and field joints in these main steam lines require that they be random radiographed. Liquid penetrant inspections were also conducted on all root welds and completed welds of both shop and field joints.

Provisions are being made to permit inservice inspection of the main steam pipe and components which form a reactor vessel pressure boundary as discussed in Appendix J.

4.22 Conclusions

The Cooper Nuclear Station has been designed with sufficient flexibility and capability to accommodate the ACRS concern items above, as stated, although their original construction permit approval did not include or identify them as being required for safe operation.

5.0 SUMMARY CONCLUSIONS

The necessary research and development programs, additional information, or special analysis to support the application for a provisional operating permit for this station is discussed and justified in the preceding sections. Resolution of Cooper Nuclear Station AEC-ACRS and AEC Staff concern items at the construction permit phase have been examined and ample support for their complete satisfaction is presented.

Thus, it is concluded that no further research and development or related activities are necessary for this facility in order to comply with the construction permit cited concern and requirements.