Safety Evaluation Report

NUREG-0422
Supp. No. 2

U. S. Nuclear Regulatory Commission

related to operation of

McGuire Nuclear Station, Units 1 and 2

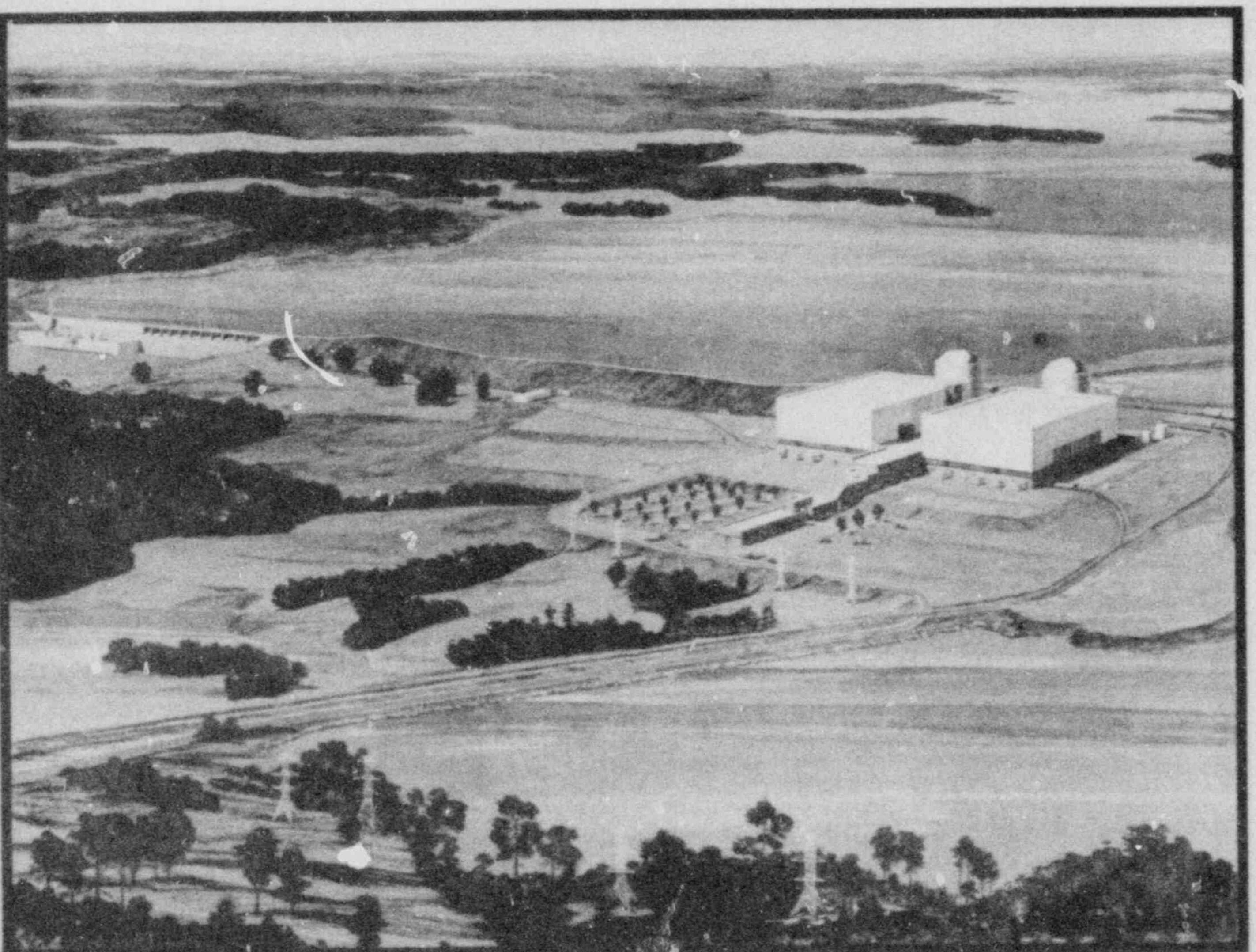
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Duke Power Company

March 1979

Supplement No. 2



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Supplement No. 2 to
NUREG-0422

March 1, 1979

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

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1.0 INTRODUCTION AND GENERAL DESCRIPTION

1.1 Introduction

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report in the matter of the application by the Duke Power Company to construct and operate its proposed McGuire Nuclear Stations, Units 1 and 2 was published in March 1978. Supplement No. 1 to that report was published in May 1978. Since publication of Supplement No. 1, we have received and reviewed seven amendments to the Final Safety Analysis Report (through Amendment 58) and held meetings with the applicant. These events are identified in the Chronology, Appendix A to this supplement. As a result of these actions, many of the issues identified as outstanding issues in Section 1.6 of the Safety Evaluation Report and Supplement No. 1 have been resolved. The remaining issues will be addressed in a forthcoming supplement. Also to be included in a forthcoming supplement will be a discussion of staff activities regarding generic safety issues. These longer term generic studies were the subject of a decision by the Atomic Safety and Licensing Appeal Board in its decision dated November 23, 1977 (ALAB-444).

This supplement provides our eval ation of additional information received from the applicant since preparation of Supplement No. 1 of the Safety Evaluation Report, including the resolution of previously identified outstanding issues. Our conclusions regarding the individual issues are found in the appropriate sections of this supplement. As stated in Section 22.0 of the Safety Evaluation Report, we will be able to reach the conclusions required in accordance with the provisions of 10 CFR 50.35a upon satisfactory resolution of the remaining four outstanding issues.

Except for the Appendices, each of the following sections of this supplement is numbered the same as the section of the Safety Evaluation Report that is being updated, and the discussions are supplementary to and not in lieu of the discussion in the Safety Evaluation Report. Appendix A to this supplement is a continuation of the chronology of the regulatory radiological review; Appendix B is our evaluation supporting exemptions from certain requirements of Appendices G and H of 10 CFR Part 50; and Appendix C is our evaluation supporting an exemption from a requirement of Appendix J to 10 CFR Part 50; and Appendix D is our evaluation of the McGuire fire protection program.

1.6 Outstanding Issues

Those outstanding issues delineated in Section 1.6 of the Safety Evaluation Report that have been resolved since the issuance of Supplement No. 1 to the Safety Evaluation Report are discussed in the following sections of this supplement. The resolution of the remaining four issues will be reported in a future supplement to the Safety Evaluation Report. These issues are (1) augmented inservice inspection (Section 3.6), (2) dynamic piping analysis (Section 3.6), (3) qualification of Class IE equipment (Section 7.8) and (4) emergency core cooling system analysis (Section 6.3).

	Section in this Supplement
(1) At the time the Safety Evaluation Report was issued we had not completed our review of fracture toughness compliance with code requirements. This review has now been completed. We find the applicant's design to be acceptable and this item is resolved.	5.2.3
(2) At the time the Safety Evaluation Report was issued we had not completed our review of the design of the combustible gas control system pending the acceptance of an emergency core cooling system performance analysis which would predict a maximum total	6.2.6

	Section in this Supplement
metal-water reaction of less than 0.2 percent of the zircaloy cladding in the core. This review has now been completed. We find the applicant's analysis to be acceptable and this item is resolved.	
(3) At the time the Safety Evaluation Report was issued we had not completed our review of the emergency core cooling system operation during off-design conditions and required additional information. The applicant submitted additional information and our review has now been completed. We find the applicant's design to be acceptable and this item is resolved.	6.3.4
(4) At the time the Safety Evaluation Report was issued we had not completed our review of the upper head injection isolation valves. This review has now been completed. We find the applicant's design to be acceptable and this item is resolved.	7.3.5
(5) At the time the Safety Evaluation Report was issued we had not completed our review of the fire protection program. This	7.9, 9.5.1, 13.2 and 13.3

Section in this Supplement

review has now been completed. We find the applicant's fire protection program to be acceptable and this item is resolved.

(6) At the time the Safety Evaluation Report was issued we had not completed our review of the design of electrical penetrations. 8.3.3

This review has now been completed. We find the applicant's design to be acceptable and this item is resolved.

(7) At the time the Safety Evaluation Report was issued we had not completed our review of the steam generator water hammer potential and required additional information. The applicant has submitted additional information and our review has now been completed. We find the applicant's design to be acceptable and this item is resolved. 10.3

(8) At the time the Safety Evaluation Report was issued we had not completed our review of the Modified Amended Security Plan and required additional information and upgrading. The applicant has submitted additional information. The applicant has submitted additional information and our review has now been completed. We find the applicant's plan to be acceptable and this item is resolved. 13.7

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEM

3.9.1 Dynamic System Analysis and Testing

Preoperational Vibration Assurance Program for Reactor Internals.

Originally, it had been planned that Sequoyah Unit No. 1 (Docket No. 50-327) was to be prototype plant for upper head injection. However, the construction of Sequoyah Unit No. 1 experienced delays. Therefore in the Safety Evaluation Report we stated that we would require that McGuire be designated as a prototype plant for the upper head injection system internals package and that it be instrumented and tested in accordance with the requirements for prototype Category I reactor internals, as detailed in Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

Since the issuance of the Safety Evaluation Report, the construction schedule for the McGuire station has experienced delays to the extent that the Sequoyah Unit No. 1 will be the propotype plant. In addition the applicant has provided additional justification for maintaining the initial designation of Sequoyah (rather than McGuire) as the upper head injection prototype plant. The justification includes the following items:

1. The upper head injection internals on the McGuire reactor are structurally similar to those of the Westinghouse designed Japanese plant, OHI, which has been operating safely for more than a year. The Japanese OHI upper internals have been subjected to full flow loads during hot functional testing and during operation at various power levels, up to and including operation at 100 percent power. Although the Japanese OHI reactor internals have not been instrumented, no problems or indications of excessive flow induced vibration have been encountered, nor has any deterioration of scram times been observed. For McGuire, the flow patterns are expected to be similar to the Japanese OHI so that the flow induced vibration response should also be similar. Therefore, no flow induced vibration problems are expected.

2. Comparisons have also been made between the vibration levels of the most heavily loaded guide tubes in the Japanese OHI and those of the prototype plant for these tubes, Indian Point Unit No. 2. The factor of safety for the Japanese OHI was considerably greater than for Indian Point Unit No. 2. The staff expects levels of McGuire will not exceed those of the prototype plant for reactor internals, Indian Point Unit No. 2.

3. The preoperational internal tests of the Sequoyah plant for which the upper head injection internals have been instrumented are currently scheduled to be performed within six months of the startup of McGuire Unit 1. The safe operation of the Japanese OHI plant provides assuranc that any possible degradation of McGuire internals resulting from flow induced vibration will not be of sufficient magnitude to have an effect on the safe operation of McGuire 1 for the relatively short time period between start up and completion of the Sequoyah testing.

Based on our evaluation as well as the redesignation of Sequoyah as the prototype we conclude that the requirements of Regulatory Guide 1.20 are satisfied, that the McGuire upper head injection internals need not be instrumented, and that testing may be conducted as originally proposed in the Final Safety Analysis Report. We consider this matter resolved.

4.0 REACTOR

4.4 Thermal and Hydraulic Design

In the Safety Evaluation Report we stated that the effect of a core exit radial pressure gradient on the thermal-hydraulic design was a generic matter and was being pursued on all recent Westinghouse pressurized water reactor reviews, the results of which would be applied to the McGuire Station.

Since the issuance of the Safety Evaluation Report, the applicant has provided additional analytical information that showed that the radial pressure gradient in open lattice cores like McGuire Units 1 and 2 caused a negligible redistribution of flow. This confirmed that the effect of the expected core exit radial pressure gradient is small and need not be specifically included in the design calculations.

We have reviewed this information and conclude that the effect on the departure from nucleate boiling is small and that the previous sensitivity studies are valid for the McGuire Station thermal-hydraulic design and are acceptable. We consider this matter resolved.

5.0 REACTOR COOLANT SYSTEM

5.2.1 Design of Reactor Coolant Pressure Boundary Components

As a result of additional information requested since the issuance of the Safety Evaluation Report, the applicant has performed additional analysis to confirm the structural integrity of the pressurizer and steam generator supports when subjected to asymmetric loads resulting from postulated reactor coolant pipe breaks in the vessel subcompartment. These loads were determined from subcompartment analysis using the Transient Mass Distribution (TMD) computer code developed by Westinghoue Electric Corporation. We have reviewed this code and have found it to be acceptable. The design requirements were satisfied when subjected to all load cases involving these loss-of-coolant accident loads. Based upon our review and the use of the Transient Mass Distribution code we conclude that the design of these supports satisfy the applicable portions of the General Design Criteria 1, 2 and 4 of Appendix A to 10 CFR Part 50 and are acceptable. We consider this matter resolved.

5.2.3 General Materials Considerations Reactor Coolant Pressure Boundary

Fracture Toughness Compliance With Code Requirements

In the Safety Evaluation Report we stated that we were reviewing fracture toughness information for the McGuire reactor vessel ferritic materials. We have now completed that review and have determined that exemptions from certain requirements of Appendix G to 10 CFR Part 50 are required for both Unit Nos. 1 and 2. We conclude that there is an acceptable basis for granting specific exemptions at the time that we issue the Operating License.

Our evaluation supporting the granting of these exemptions is included as Appendix B to this supplement. We have determined that the ferritic materials used for the pressure retaining components of the reactor coolant boundary are in compliance with the intent (safety objective) of Appendix G to 10 CFR Part 50.

For Unit Nos. 1 and 2 we conclude that the fracture toughness tests and procedure required by Section III of the American Society of Mechanical Engineers Code, as augmented by Appendix G of 10 CFR Part 50, for the reactor vessel provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for the pressure retaining components of the reactor coolant boundary. The portion of the McGuire Technical Specifications which limit the maximum reactor coolant system pressure for low temperature operation reflect these safety margins.

Reactor Vessel Material Surveillance Program

We have determined that exemptions from certain requirements of Appendix H to 10 CFR Part 50 are required for both Unit Nos. 1 and 2. We conclude that there is an acceptable basis for granting specific exemptions at the time we issue the Operating License. Our evaluation supporting the granting of these exemptions is included as Appendix B to this supplement. We have determined that the reactor vessel material surveillance program is in compliance with the intent (safety objective) of the requirements of Appendix H to 10 CFR Part 50. We consider this matter resolved.

6.0 ENGINEERED SAFETY FEATURES

6.2.1 Containment Functional Design

In the Safety Evaluation Report, we stated that we had not completed our review of the applicant's containment minimum pressure response analysis. We have now completed our review of the information presented in Amendments No. 47 and 52 to the Final Safety Analysis Report.

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the effect of c.⁻rating all the containment installed pressure reducing systems and processes be included in the emergency core cooling system performance evaluation. For the purpose of the emergency core cooling system evaluation, it is conservative to minimize the containment pressure. The reflood rate in the core will then be reduced because of the higher resistance to steam flow in the reactor coolant loop.

Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system to the containment atmosphere. After initial blowdown, heat transfer from the core, the primary system's metal structure, and the steam generators to the emergency core cooling system water will produce additional steam. This steam, together with any emergency core cooling system water spilled from the primary system, will flow through the postulated break into the containment. This energy will be released to the containment during both the blowdown and later emergency core cooling system operational phases, i.e., the reflood and post-reflood phases.

Energy removal from the containment atmosphere occurs by several means. Steam condensation on the containment walls and on other internal structures serves as a passive energy heat sink that is effective early in the blowdown transients. Subsequently, the operation of the containment heat removal systems such as containment sprays will remove steam from the containment atmosphere by condensing the steam. In an ice-condenser-type containment, energy is removed from the containment atmosphere as the mixture of steam, air, and water passes through the ice condenser, as it is forced from the containment's lower compartment to the upper compartment.

The emergency core cooling system containment pressure calculations for the McGuire Nuclear Station, Units 1 and 2, were based on the Westinghouse emergency core cooling system evaluation model. The Westinghouse Electric Corporation's LOTIC-2 containment code described in Report WCAP-8354, Supplement 1, "Longterm Ice Condenser Containment Code - LOTIC Code," was used. We have reviewed the LOTIC-2 code and have concluded that it is acceptable for use in calculating the min'mum containment pressure response for ice condenser plants.

Although we have accepted the methods used to calculate the containment pressure response, we required that justification of the plant dependent input parameters used in the analysis of containment pressure response be submitted for our review on a plant-by-plant basis. This information was submitted in Amendments 47 and 52 to the McGuire Final Safety Analysis Report. The applicant has reevaluated the containment net-free volume, the passive heat sink, operations of the containment heat removal systems and containment initial conditions with regard to the conservatism for the emergency core cooling system analysis. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

Based on our review we conclude that the plant dependent information used for the emergency core cooling system containment pressure analysis is conservative. We, therefore, find that the calculated containment pressures presented in Amendments 47 and 52 to the Final Safety Analysis Report are in accordance with Appendix K to 10 CFR Part 50, and are therefore acceptable. We consider this matter resolved.

6.2.5 Containment Isolation Systems

In the Safety Evaluation Report we stated that the applicant had not demonstrated the operability of the upper compartment purge system containment isolation valves; i.e., the capability of the valves to close while experiencing the pressure and temperature build-up within the containment upper compartment during the postulated loss-of-coolant accident. The applicant has confirmed that the design of the upper containment purge system's containment isolation valves meets the requirements of the valve operability program for active valves. Based on our review we conclude that the applicant has demonstrated the operability of the upper compartment purge system's containment isolation valves.

Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," states that the need for purging should be minimized. The applicant has proposed to limit the use only to the upper compartment purge system during the plant operating modes of start-up, power, hot-standby and hot-shutdown, and to limit the use of the upper compartment purge system to less than 90 hours per year (approximately one percent of the time). These limitations will be included in the technical specifications governing the operation of the plant, as well as the requirement that the containment isolation valves in the containment upper compartment purge system be local leak rate tested (Type C) following each use of the system.

Two other purge systems in the lower compartment and in the instrument room have been provided in the plant; but these systems will not be used during the plant operating modes mentioned above. We will, however, require that the isolation valves in these two containment purge systems be leak tested prior to plant operation in the above modes following each use of the system(s).

These actions are consistent with Branch Technical Position CSB 6-4, the implementation criteria in Standard Review Plan 6.2.4 - Containment Isolation System, Revision 1, and the action taken regarding leak testing of the containment purge system isolation valves for the Donald C. Cook Nuclear Plant, Unit 2. We consider this matter resolved.

6.2.6 Combustible Gas Control

In the Safety Evaluation Report, we concluded that the design of the combustible gas control system was acceptable pending the acceptance of an emergency core cooling system performance analysis which predicts a maximum total metal-water reaction of less than 0.2 percent of the Zircaloy cladding in the core. The emergency core cooling system performance analysis has been completed, and predicts a maximum of about 0.3 percent of the core cladd ng could be reacted. The appli ant has, therefore, recalculated the containment post-accident hydrogen concentrations assuming the appropriate 1.5 percent (five times the predicted maximum of 0.3 percent) metal-water reaction.

The results of the new analysis indicate that: (1) a four percent concentration of hydrogen (lower flammable limit) would not be reached in the lower compartment prior to the initiation of return air fan operation; and (2) following operation of the return air fan system, the containment volume would not reach the four percent limit until about 13 days after the accident. The hydrogen recombiners will be placed into operation when the hydrogen concentration is well below the four percent limit.

Based on our review we conclude that the combustible gas control system satisfies the design and performance requirements of Subpart 50.44 to 10 CFR Part 50, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors," Regulatory Guide 1.7, Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," and Criteria 41, 42 and 43 of the General Design Criteria, and is acceptable. We consider this matter resolved.

6.2.7 Containment Leak Testing Program

We have determined that an exemption from a requirement of Appendix J to 10 CFR Part 50 is required for both Units Nos. 1 and 2. We conclude that there is an acceptable basis for granting a specific exemption at the time we issue the Operating License. Our evaluation supporting the granting of this exemption is included as Appendix C to this supplement. We have determined that the McGuire containment leak testing program is in compliance with the intent (safety objective) of Appendix J of 10 CFR Part 50. We consider this matter resolved.

6.3.3 Emergency Core Cooling System - Testing

In the Safety Evaluation Report we required performance of preoperational tests including demonstration that recirculation from the containment sump with the low-pressure coolant injection system would assume that adverse hydraulic phenomena would not impede long-term cooling capability following a loss of-coolant accident. Subsequently, the applicant performed out-of-plant scale model tests at the Worcester Polytechnic Institute Alden Research Laboratory. These tests are described in a report by M. Padmanabhan, "Assessment of Flow Characteristics Within a Reactor Containment Recirculation Sump Using a Scale Model," May 1978.

The test facility contained a one-third scale model of the McGuire sump. The tests determined the original sump, which was inside the crane wall, to be unsuitable because of possible break flow impingement on the water surface with resultant air entrainment into the sump piping and thence to the emergency core cooling system pumps. The applicant subsequently relocated the sump between the crane wall and containment vessel to prevent direct jet impingement from a postulated pipe break upon the water surface. The new design permits water inside the crane wall to flow through several holes in the wall into the sump outside.

The new sump contains a trash rack to remove large debris, a fine screen to remove smaller particulate matter, and gratings to serve as vortex suppressors, with solid covers over the top of the sump pipes. All of this was modeled in the Alden tests as were parts of the crane wall, containment wall, and all pipes and holes two inches or larger in diameter.

Staff representatives witnessed a portion of the test program that used the new sump design. The tests witnessed raised and lowered sump levels, down to within inches of the pipe covers, and water velocities above those expected in the prototype. In addition, various screen blockage schemes were used with blockages as great as 50 percent of the total area. It was observed that surface dimple and/or dye core vortices would form, on vertical pipes adjacent to the sump, but would soon disintegrate. In no case was the vortex sustained sufficiently to draw water or air even to the screens, much less to the sump inlet pipes.

In addition, the test program conducted included:

(1) Screen blockage schemes;

(2) Temperature variation with water at 48,100 125 and 150 degrees Fahrenheit; and

(3) Froude number variation - tests at 0.5, 1.0, 1.4, and 1.73 N_{FR} at each of the above temperatures.

Observations were made of vortex generation, inlet loss coefficient, and pipe swirl. In the case of pipe swirl, a vortimeter (a set of vanes) was installed in one of the two sump pipes to determine whether the system was causing vortices to form at the entrance to the sump pipes. Neither excessive swirl nor vortex generation was noted.

Based on the results of the out-of-plant test program, we conclude that the new McGuire sump design to be acceptable with regard to vortex formation and air entrainment. Prior to the issuance of the operating license we will require the applicant to provide us with the net positive suction head available for the emergency core cooling system pumps in the most limiting alignment under most severe loss-of-coolant accident conditions using pump characteristics and hydraulic loss data obtained in these preoperational tests. We consider this matter resolved.

6.3.4 Emergency Core Cooling System - Performance Evaluation

In the Safety Evaluation Report we stated that operationally the applicant blocks the safety injection signal during plant cooldown and closes and locks out power to the cold leg accumulator valves during shutdown operations, and that the performance of the emergency core cooling system under these off design conditions was being reviewed. The applicant reported that off-design conditions could be divided into four phases: (1) operating pressure to 1900 pounds per square inch gauge, (2) 1900 to 1000 pounds per square inch gauge, (3) 1000 to 400 pounds per square inch gauge and (4) 400 pounds per square inch gauge to cold shutdown.

The full emergency core cooling system is available to mitigate the consequences of a loss-of-coolant accident in Phase 1; in Phase 2 the upper head injection isolation valves are closed and gagged; only this system is unavailable for emergency core cooling system operation in the event of a loss-of-coolant accident. Below 1900 pounds per square inch gauge the safety injection signal is blocked so that the emergency core cooling system is initiated automatically by high containment pressure or by manual action.

In Phase 3 the cold leg accumulator tanks isolation valves are locked closed. The breakers for these valves and those for both high head injection pumps and one charging pump are racked out and locked so that only one charging pump and both residual heat removal pumps are available for emergency core cooling system operation; in Phase 4 the plant is operating in the residual heat removal cooling mode.

In all phases, a line break may be detected by the following:

(1) Low pressurizer level

(2) Primary system pressure decrease

(3) Containment pressure increase

(4) Radiation alarms inside containment

(5) Sump Water level increase

In Phase 4 an alarm will alert the operator to low flow through the residual heat removal system.

Calculations performed by the applicant show that, in Phase 3, the maximum peak clad temperature attained, with single failure of a residual heat removal pump, would be substantially below the criterion specified in 10 CFR 50.46; in Phase 4, the operator has sufficient time (13 minutes) in which to initiate action to keep the core covered thereby preventing fuel clad heat up to excessive levels.

In Phases 1 and 2, the available emergency core cooling system equipment is considered sufficient so that the criteria specified in 10 CFR 50.46 are satisifed in the unlikely event of a loss-of-coolant accident.

Based on our evaluation we conclude that the available emergency core cooling system can cool the core under off-design conditions so that the applicable criteria are not violated. The applicant will be required to provide a plant procedure for operator action in the event of a residual heat removal pipe crack when operating in the residual heating removal cooling mode. The procedure must provide sufficient actions to keep the core covered at all times and remove decay heat. The Office of Inspection and Enforcement will verify this requirement. We consider this matter resolved.

7.0 INSTRUMENTATION AND CONTROL

7.3.5 Upper Head Injection Isolation Valves

In the Safety Evaluation Report we stated that we were reviewing the controls for the upper head injection isolation valves. We have now completed the review of the provisions and operation of the instrumentation that senses the water level in the accumulators, initiates closure and gagging of the hydraulically actuated isolation valves and the interlocks that control the reopening of those valves. Spurious action that closes one isolation valve will not interfere with the performance of the system safety function because two redundant 100 percent capacity injection lines are provided. Two valves in series are provided in each injection line so that failure of one isolation valve to close will not prevent the required isolation of the accumulator after delivery of its charge is accomplished. The isolation valves gag automatically only during the accident sequence when the safety injection is present. This is a required action during the accident migitation sequence. The safety injection signal must not be reset before this action has taken place since resetting effectively removes the safety injection signal and prevents automatic initiation of isolation valve closure.

During normal cooldown, closure of these valves must be manually initiated before the reactor coolant system pressure falls below the upper head injection accumulator pressure (1200 pounds per square inch). The gag insertion must then be manually initiated when the valve reaches its fully closed position. The staff requires that these valves remain closed during the time the reactor coolant system is cooled and depressurized to prevent the inadvertent discharge of the upper head injection accumulator into the reactor coolant system thus causing its overpressurization. Since no single random failure can cause the opening of these valves and the consequent discharge of the accumulators, the staff position is acceptably met.

During startup the operator must manually initiate gag removal and then manually initiate the opening of each of the four upper head injection isolation valves when the reactor coolant system pressure rises above the safety injection system unblock pressure. This manual restoration procedure conflicts with part (1) of

Instrumentation and Control System Branch Technical Position 4 which requires automatic opening of these valves to restore the upper head injection to operable status. However, our review has shown that (a) since no single failure can prevent operation of the upper head injection, power need not be removed from the valves or their controls, (b) since power is not removed, position indication and out-of-position alarms remain energized and functional for each isolation valve as required by Instrumentation and Control Systems Branch Technical Positions 4 and 18, (c) the McGuire standard technical specifications require that surveillance of the valve position be performed once per twelve hours whenever the reactor coolant system is pressurized and (4) we have determined that these valves need not be opened immediately upon reaching the safety injection system unblock point during the startup. Based on our review and these considerations we conclude that operator action to open the upper head injection isolation valves during startup is acceptable. We consider this matter resolved.

7.9 Cable Separation and Identification Criteria

In the Safety Evaluation Report we stated that the staff will review the cable separation and identification criteria in conjunction with the fire protection review and also verify the adequacy and implementation of the separation criteria during the site visit. Our evaluation and conclusion are addressed in the fire protection safety evaluation report, Appendix B of this supplement. We consider this matter resolved.

8.0 ELECTRIC POWER

8.3.3 Electrical Penetrations

In the Safety Evaluation Report we stated that we were reviewing the design of the McGuire electrical penetrations. The applicant stated in the McGuire Final Safety Analysis Report that medium voltage (6.9 kilovolts) electrical penetrations for reactor coolant pump power use sealed bushing for conductor seals. The assemblies incorporate dual seals along the axis of each conductor. Low voltage power, control and instrumentation cable (600 volts or less) enter the containment vessel through penetration assemblies (connector type) which have been designed to provide two leak tight barriers in series with each conductor. All electrical penetrations have been designed to maintain integrity for design basis accident conditions including the effects of pressure, temperature, chemical, and radiation. Double barriers permit testing of each assembly as required to verify that containment integrity is maintained.

To ensure that the failure of a single overload protective device associated with power circuits will not allow a fault current which could cause a loss of mechanical integrity of the penetration, the applicant has provided two circuit overload protective devices in series for these circuits. The staff has reviewed the applicant's design of electrical penetration overload protective devices (circuit breakers) for its conformance to the recommendations of Regulatory Guide 1.63, "Electrical Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants." The technical specifications will include a requirement for a periodic integrated test of these circuit breakers along with their associated fault sensors/trip relays to ensure that the breakers retain their fault interrupting capability within the required time.

Horizontal separation between reactor building penetrations for redundant channels is accomplished by routing cables through penetrations at opposite ends of the penetration room, maintaining minimum horizontal separation of five feet between each penetration.

Penetrations associated with engineered safety features train A are located between elevation 756 feet and 761 feet. Penetrations associated with engineered safety features train B are located between elevation 748 feet and 750 feet. Both sets of penetrations are above the maximum post loss-of-coolant accident water level (738 feet -6 feet).

The staff verified the cable routing and implementation of separation criteria for the penetration during a site visit and concluded that both are acceptable.

The applicant has submitted the medium voltage power electric penetration and the instrumentation and control cable penetration design specifications, the qualification test program used to qualify these penetrations and the test results. The applicant has stated that the general guidelines of Institute of Electrical and Electronics Engineers Standard 317-1972 "Standard for Electrical Penetrations", and Institute of Electrical and Electronics Engineers Standard 323-1971 "Standards for Qualifying Class IE Equipment" were followed in the tests. Also, the margins recommended in Institute of Electrical and Electronics Engineers Standard 323-1974 were included in the test conditions. The applicant concluded from these tests that the penetration assemblies remain leak-tight and electrically functional when subjected to the loss-of-coolant accident and the main steam line break environmental conditions.

The staff agrees that the qualification test results verified that the electrical penetrations can perform their required functions before, during, and following the abnormal environmental conditions specified by the applicant. Further, the staff has evaluated the McGuire containment loss-of-coolant accident electrical penetration environment envelope and finds it to be acceptable. However, the evaluation of the maximum containment temperature following a postulated main steam line break accident inside containment has not been completed. Therefore, the acceptance of main steam line break environmental qualification envelope is subject to the final resolution of containment maximum temperature response.

With regard to electrical penetration seismic qualification test, Institute of Electrical and Electronics Engineers Standard 317-1972 and Institute of Electrical and Electronics Engineers Standard 344-1975 require that those tests be performed under simulated installed conditions. Since electrical loading of the penetration was omitted during the tests, we required the applicant to fully justify the exception by:

(1) Determining the temperature distribution under full load operating conditions of the electrical penetrations, and

(2) Performing an analysis to determine the structural integrity of the electrical penetration when subjected to the combined thermal (electrical) and seismic loadings.

The applicant submitted the results of these analyses in Amendment No. 53 to the Final Safety Analysis Report. These results showed that:

(1) Only a slight increase in temperature (35 degrees Fahrenheit above ambient was observed in the worst case tested).

(2) The calculated structural changes due to thermal expansion are minimal even when using the maximum allowed temperature rise of 88 degrees Fahrenheit above the ambient temperature of 70 degrees Fahrenheit.

(3) Seismic prototype testing was performed at an acceleration value 90 percent above specification requirements for the McGuire plant.

(4) Fault current testing imparted more than twice the forces on the penetration conductors than did the seismic test.

Based on these results, the applicant has concluded that the lack of rate continuous current passing through the penetration did not alter the seismic withstand capability of the penetration for the required response spectrum.

We have reviewed the information and concluded that the applicant's justification is acceptable.

The containment maximum temperature response to a main steam line break accident is currently being reviewed. We will address this aspect of the penetration qualification in a forthcoming supplement to this report. With this one exception, we conclude that the electrical penetration qualification program is acceptable.

9.0 AUXILIARY SYSTEMS

9.1.2 Spent Fuel Storage

In the Safety Evaluation Report we stated that the applicant had proposed the storage of Oconee spent fuel at the McGuire Station and that this matter was under staff review.

The McGuire Station systems or portions of systems needed to support the storage of Oconee spent fuel will be complete and operational prior to shipment of Oconee spent fuel to McGuire Unit 1. A total of 300 Oconee assemblies, in batches of 60 assemblies, will be allowed to decay for at least 120 days before shipment to McGuire. Sufficient space will exist in the McGuire Unit 1 storage pool for any emergency unloading of the McGuire Unit 1 core.

The Oconee fuel assemblies will be accommodated within the McGuire Unit 1 fuel storage rack by the placement of 5 1/2 inch spacers in those locations designated to hold Oconee fuel. The spacers are necessary to permit handling of the Oconee fuel assemblies with the Oconee fuel handling tool in the McGuire storage rack.

The Oconee spent fuel array in the storage rack will be such that when flooded with unborated water, K_{EFF} will not exceed 0.95. Each spacer has a 25-square inch opening to allow adequate coolant flow. We have performed an independent evaluation of McGuire's spent fuel cooling system capacity. The pool water temperature can be maintained below 150 degrees Fahrenheit with both cooling trains in operation taking into account the decay heat generated by the stored Oconee spent fuel in addition to one full core of McGuire fuel due to emergency loading.

We have evaluated the location and method of attaching the spacers to McGuire's spent fuel rack and found that they will not nullify the seismic Category I design of McGuire's fuel rack. The spacers are not rigidly attached to the fuel rack but rather simply rest on the support plate due to gravity and are contained within the rack due to the fuel assembly guides. On this basis, we conclude that they have only negligible structural effects on the rack and structural support system. The mass of the spacer is very small compared to the mass of fuel rack and fuel assemblies; therefore, the structural impact due to this increase in mass is considered to be negligible. The effects of sloshing of water is also negligible.

We have reviewed the possibility of consequences of storing Oconee fuel assemblies in locations which are designed for McGuire fuel or vice versa, and conclude that such an event is incredible as the fuel handling tool design will preclude inadvertent storage of the Oconee spent fuel assemblies in locations reserved for McGuire fuel or vice versa. In addition, the placement of the spacers will be administratively controlled such that one spacer will be installed when a fuel assembly from Oconee is received. The hoist to be used to handle Oconee spent fuel has a load capacity of 4000 pounds with an overload interlock set at 2900 pounds which is well below the maximum uplift capacity of the storage racks.

We have also evaluated the consequences of dropping an Oconee fuel assembly and agree with the applicant that the consequences of such an event is less severe than the dropping of a McGuire fuel assembly since the Oconee assembly will have less stored radioactive material because it has decayed for at least 120 days prior to shipment to McGuire facility. The McGuire fuel assembly drop accident has been previously evaluated and the consequences found to be acceptable. The fire suppression and detection system for the McGuire spent fuel storage facility will be fully operational prior to the shipnment of Oconee spent fuel to the McGuire facility for storage.

Based on our review, we conclude that the design of the spent fuel storage facility meets General Design Criterion 61 of Appendix A to 10 CFR Part 50, and the recommendations of Regulatory Guides 1.13, "Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification," including seismic design and missile protection guidelines, and is therefore acceptable. We further conclude that there is reasonable assurance that the McGuire spent fuel storage facility can serve as a storage facility for the Oconee spent fuel without undue risk to the health and safety of the public.

9.5.1 Fire Protection System

In the Safety Evaluation Report, we stated that our review of the McGuire fire protection program was in progress. The results of our review are presented in Appendix D of this supplement and are summarized in this section.

The applicant has committed to provide a completely independent safe shutdown system (SSS) following commercial operation of Unit 1 to assure the hot shutdown capability for the McGuire Nuclear Plant. Until the safe shutdown system is installed, the applicant will establish interim emergency shutdown procedures to bring the plant to safe hot standby condition in the event of a damaging fire in the cable spreading room, the main control room, or the battery room common area. These procedures will be reviewed by the Office of Inspection and Enforcement prior to issuance of an operating license. Also, improvements are under way or planned for the plant fire protection system. Some of these will be installed prior to initial fuel loading and others by commercial operation of Unit No. 1 to assure safe cold shutdown without reliance on the cable spreading room or the control room (see Table 9.5.1). We require that the final design of the safe shutdown system be submitted for our approval by March 1980 and that the system be fully operational three months after the first refueling of Unit No. 1 but not later than 24 months after initial fuel loading of Unit No. 1.

We find that the Fire Protection Program for the McGuire Nuclear Plant with the improvements already made and being made by the applicant is adequate for the present and, with the scheduled safe shutdown system, will meet the guidelines contained in Appendix A to Branch Technical Position 9.5.1 and meets General Design Criterion 3 and is therefore, acceptable.

The Office of Inspection and Enforcement will assure implementation of these improvements in accordance with the applicant's commitments as shown in Appendix D of this supplement and Table 9.5.1.

TABLE 9.5-1

FIRE PROTECTION IMPROVEMENT IMPLEMENTATION PROGRAM

	To Be Installed	
	Prior to Initial Fuel Loading	Prior to Commercial Operation
Water Suppression Systems		
Cable spreading room (Manual Fog System)	X	
Residual Heat Removal pump rooms (automatic)	X	
Corridors adjacent to residual heat removal rooms (automatic)		X
Motor driven auxiliary feedwater pump room (automatic)	X	
Centrifugal charging pump rooms (automatic)	X	
Nuclear service water pump rooms (automatic)	X	
Component cooling water pump rooms (automatic)	X	
Reactor coolant pump (remote manual)	X	
Containment annulus (remote manual)	X	
Pipe corridor at elevation 725 feet (automatic)	X	
Battery room open area - east and west ends (automatic)		X
Smoke Detectors		
Immediate area of small exhaust fan servicing control panels in the main control room	X	
Each battery cell room	X	
Steam driven auxiliary feed pump room	X	
Peripheral rooms of the main control room	X	
Fire Barriers and Fire Barrier Penetrations		
Fire proofed angle iron along junction of barrier and ceiling between cable rooms (along column 56) and auxiliary building	X	
One hour rated ceilings and fire doors and dampers in peripheral rooms of the main control room		X
Barriers having 1 1/2 hour fire rating between redundant cooling pumps	X	
One half hour rated fire barriers to protect overhead instrumentation and control cables for the auxiliary feed water pumps	X	
One half hour rated fire barriers on four sides of the remote shutdown panel	X	
Fire proofed supports for heating, ventilation and air conditioning ducts in safety related areas	X	
Fire doors and dampers in penetrations in room 807 and 820 at elevation 750 feet of the auxiliary building		X
Safe Shutdown System		
Final design submitted to the staff	March 1980	X
System fully operational	3 months after first refueling of Unit No. 1 but not later than 24 months after initial fuel loading of Unit No. 1.	

10.0 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

In the Safety Evaluation Report we stated our concern that rapid steam bubble collapse in the steam generator pre-heater could create forces that could cause unacceptable damage to the system. The applicant has committed to a test program, which will be performed as a portion of the Unit No. 1 preoperational test program, to show that unacceptable feedwater hammer damage would not result from normal and transient operation. These tests will use the standard plant operating procedures which could allow cold feedwater to enter the preheaters and possibly cause waterhammer in the system. The applicant has committed to provide the test procedures, including the necessary instrumentation and acceptance criteria for the test to the staff for approval prior to performing the preoperational test. We conclude that completion of these tests without unacceptable feedwater hammer damage will accomplish our test objective. We find this program to be acceptable and consider this matter resolved.

10.5 Auxiliary Feedwater System

Since the issuance of the Safety Evaluation Report the applicant modified the controls of the valves connecting the seismic Category I nuclear service water system and the suction of the auxiliary feedwater pumps so that the valves will automatically open in the event of a low auxiliary feedwater pump suction pressure signal. Previously, the valves were remote manually operated from the control room.

Diversity for the auxiliary feedwater pump suction is provided by use of several water sources and adequate valving for source change. All three auxiliary feedwater pumps are normally supplied from a common header which is normally connected to the upper surge tank, the auxiliary feedwater condensate storage tank, or the condenser hotwell. Each of these sources are provided with motor operated valves with remote manual operation from the control room. Upon receipt of a low sunction pressure signal, the auxiliary feedwater pump suction is realigned to the nuclear service water system. The two motor driven pumps are connected to one nuclear service water system train and the turbine driven pump is connected to the other nuclear service water system train. Each nuclear service water system source train is provided with a normally closed Class IE motor operated valve which opens automatically on low pump suction pressure. A cross-over is provided between the two nuclear service water system trains. The cross-over contains parallel motor operated valves which can be operated from the control room. These valves have Class IE operators such that the cross-over is functional with either of the diesels operational.

We have reviewed the design modifications and conclude that the modifications do not affect our previous acceptance basis of the system and are therefore acceptable.

13.0 CONDUCT OF OPERATIONS

13.2 Training Program

Since the issuance of the Safety Evaluation Report, the applicant submitted revisions to the Final Safety Evaluation Report regarding fire protection training. We have reviewed this information and conclude that the plans for fire protection training and retraining of the fire protection staff, the fire brigade personnel and other staff personnel and plans for periodic drills provide reasonable assurance that the plant staff will be adequately trained to cope with fire related emergencies. Construction workers and temporarily assigned personnel will be sufficiently indoctrinated so that they will contribute to the overall fire protection program. Based on our review we find that the proposed training meets our requirements and is acceptable. We consider this matter resolved.

13.3 Emergency Planning

Subsequent to the issuance of the Safety Evaluation Report, the staff performed an additional review of the McGuire Emergency Plan in conjunction with the Fire Protection Program review. In this regard, the applicant submitted revisions to the Final Safety Analysis Report in Amendment No. 49. We now find that the applicant's emergency plans include measures for coping with fire emergencies that conform with the applicable provisions of Regulatory Guide 1.101, "Emergency Planning For Nuclear Plants." In particular, satisfactory written agreements are in effect with the Gilead Volunteer Fire Department and the Charlotte City/Mecklenburg County Fire Departments which assure the availability of additional trained personnel and equipment for fire fighting support when called upon. In addition, the applicant has provided for annual training of these personnel to assure their necessary familiarity with the plant, access procedures, and radiation protection precautions, and for their participation in an annual drill or test excercise. Based on our review and the provisions of Regulatory Guide 1.101 we conclude that the applicant has provided an acceptable state of fire emergency preparedness. We consider this matter resolved.

13.7 Industrial Security

In the Safety Evaluation Report, we stated that the applicant had submitted a Modified Amended Security Plan which we were reviewing. As a result of our evaluation, we identified certain areas in which additional information and upgrading was required in order for the Modified Amended Security Plan to comply with the requirements of Section 73.55 of 10 CFR Part 73.

Subsequently, the applicant filed five revisions to the Modified Amended Security Plan to satisfy our requirements. We consider that the revised plan meets the requirements of 10 CFR Part 73 and is acceptable.

An ongoing review of the progress of the security plan implementation will be performed by the staff to assure conformance to the performance requirements of Section 73.55 of 10 CFR Part 73 by February 23, 1979 or before the issuance of an operating license whichever is later. We consider this matter resolved.

14.0 INITIAL TEST PROGRAM

In the Safety Evaluation Report we stated that the applicant had not proposed to conduct in-plant testing that would simulate recirculation from the containment sump with the low pressure coolant injection system as recommended by Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." Subsequently, the applicant has conducted an out-of-plant scale model test program to demonstrate, under various flow conditions, that vortexing will not occur in the containment sump and that there is no air entrainment which might degrade pump performance. Staff representatives witnessed some of these tests and have reviewed the results of the test program (see Section 6.3.3 of this supplement). We conclude that this test program, in conjunction with the in-plant tests of the system described in the Final Safety Analysis Report, is an acceptable alternative to testing in accordance with Regulatory Guide 1.79. We consider this matter resolved.

The applicant had not proposed to conduct in-plant tests to demonstrate the capacity of the atmospheric steam dump valves. The applicant has submitted manufacturer's test data which provides assurance that the capacity of the valves does not exceed the Final Safety Analysis Report Chapter 15 assumptions. Based on our review we find this acceptable and consider this matter resolved.

15.0 SAFETY ANALYSIS

15.4.1 Loss-of-Coolant Accident

Subsequent to the issuance of the Safety Evaluation Report the applicant in Amendment 56 to the Final Safety Analysis Report provided a reanalysis of the radiological consequences due to a postulated loss-of-coolant accident assuming that four percent (previously one percent) of the containment leakage bypasses the containment annulus and is released to the environment without treatment. The assumptions used in calculating the consequences of the loss-of-coolant accident are listed in Table 15.4-4 (Revised).

On the basis of our review, we conclude that potential doses for this postulated accident as shown in Table 15.4-6 (Revised) remain within the 10 CFR Part 100 guideline value and are satisfactory.

15.4.2 Fuel Handling Accident

Subsequent to the issuance of the Safety Evaluation Report, the applicant in Amendment 59 to the Final Safety Analysis Report provided a reanalysis to demonstrate the capability of the fuel handling building ventilation system to mitigate the consequences of a fuel handling accident. The design of the fuel handling building did not meet the negative 1/4 inch water gauge pressure requirement of Standard Review Plan Section 15.7.5 and thus a reanalysis by the applicant was required. The ventilation assumptions used in calculating the consequences of the fuel handling accident are listed in Table 15.4-7.

The ventilation system for the fuel handling building consists of a supply system with multiport ducts running at two levels along one side of the fuel pool, and an exhaust system taking air from a multi-port duct running at a low level along the opposite side of the fuel pool. Thus, there is a sweep of air across the pool, and radioactivity which is released from the pool passes into the exhaust system. A safety grade radiation detector is located in the exhaust duct near the pool, and will actuate dampers to cause the air to be diverted through high efficiency particulate air and charcoal filters should an accidental release of radioactivity occur. The dual exhaust fans will then draw the air through a charcoal filter bed of two inch thickness with a tested removal efficiency of 95 percent for elemental and organic forms of iodine. We have conservatively assumed the lesser efficiencies given in Table 15.4-5 and have performed an independent evaluation of the radiological consequences of a fuel handling accident in the fuel building. Using the assumptions of Regulatory Guide 1.25, "Assumptions Used for Evaluating the Consequences of a Fuel Handling Accident In the Fuel Handling and Storage Facility For Boiling And Pressurized Water Reactors," the doses would be 333 rems to the thyroid at the site boundary for an unfiltered release, and 50 rems to the thyroid for a filtered release. The latter assumes all the radioactivity released from the fuel pool passed through the filters.

In addition, an analysis was made of the effectiveness of the ventilation system in controlling the pathway of the accident effluents from the fuel pool, including the effects of the ambient wind on the building. In this analysis, an allowance was made for as much as 20 percent of the area of aluminum siding subjected to a negative pressure on the outside equal to the magnitude of 100 percent of the velocity pressure of the wind. It was assumed that brick and concrete areas had insignificant leak rates compared to the aluminum siding. This model has been confirmed by comparison to measurements which this applicant and others have performed on similar buildings. An allowance was made for leakage around the railroad cask car door. The analysis also considered the effects of various wind directions in an effort to establish the direction which was most likely to cause radioactivity from the fuel storage pool to exfiltrate from the building into the atmosphere without being filtered.

It was assumed that air which infiltrated into the building was mixed uniformly with the supplied air as it passed over the fuel pool. In this way, for wind conditions which caused air to exfiltrate through the sides of the building unfiltered, the division of the radioactivity released could be assumed proportional to the flow rates of the filtered and unfiltered exhaust streams. The assumptions are detailed in Table 15.4-7, and the doses are presented in Table 15.4-6 (Revised).

The doses are well within the guideline values of 10 CFR Part 100, and are acceptable. We conclude that the proposed ventilation system for the fuel handling building will provide suitable mitigation of the consequences of a postulated fuel handling accident in the building, and that the standard technical specification requirement of negative 1/4 inch water gauge pressure in the fuel building can be waived. The applicant will be required in its Technical Specifications to maintain a minimum differential between the supply and exhaust flow rates in the building of at least 8,000 cubic feet per minute, since this is the principal factor assuring the efficiency of the ventilation system.

On the basis of our review, we conclude that potential doses for this postulated accident as shown in Table 15.4-6 (Revised) remain within the 10 CFR Part 100 guideline values and are satisfactory.

TABLE 15.4-4 (Revised)

ASSUMPTIONS USED TO ESTIMATE RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED LOSS-OF-COOLANT ACCIDENT

Power level, megawatts thermal	3565
Operating time, years	3
Primary Containment Leak Rate,* percent per day	0.2 to 24 hours 0.1 greater than 24 hours
Fraction of Core Inventory Available for Leakage from Containment:	
Noble Gases	100 percent
Iodine	25 percent
Bypass Leakage Fractions, percent of Primary Containment Leak Rate	
0 - 80 seconds	100
80 seconds to 30 days	4
Iodine Form Fractions, percent	
Elemental	91
Particulate	5
Organic	4
Annulus Ventilation System Filter Efficiencies for Iodine Forms, percent	
Elemental	95
Particulate	95
Organic	95
Relative Concentrations, seconds per cubic meter	
0 - 2 hours at 700 meters	9.5×10^{-4}
0 - 8 hours at 8850 meters	2.6×10^{-5}
8 - 24 hours at 8850 meters	1.7×10^{-5}
24 - 96 hours at 8850 meters	6.5×10^{-6}
96 - 720 hours at 8850 meters	1.6×10^{-6}

*Exclusion area boundary distance = 700 meters

TABLE 15.4-5

ASSUMPTIONS FOR A POSTULATED FUEL HANDLING ACCIDENT

Power Level	3656 Megawatts-thermal
Power Peaking Factor	1.65
Operating Time	3 years
Number of Rods Failed	289
Number of Rods in Core	50,952
Fraction of Inventory in Gap:	
Noble Gases	10 percent
Iodines	10 percent
Effective Iodine Decontamination Factor in Pool	100
Filter Efficiencies:	
Elemental Iodine	90 percent
Organic Iodine	70 percent
Iodine Fractions Leaving Pool	
Elemental	75 percent
Organic	25 percent
Shutdown Time	72 hours
Q/X Relative Concentration Values	
0 - 2 hours at 700 meters	9.5×10^{-4} seconds per cubic meter
0 - 2 hours at 8850 meters	2.6×10^{-5} seconds per cubic meter

TABLE 15.4-6 (Revised)

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Two-Hour* Exclusion Boundary		Course of Accident Low Population Zone**	
	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss-of-Coolant	117	2.5	24	< 1.0
Fuel Handling	50	1.0	< 1.0	< 1.0
Rod Ejection	66	< 1.0	6	< 1.0

*Exclusion area boundary distance = 700 meters
** Low-population zone distance = 8850 meters

TABLE 15.4-7

VENTILATION ASSUMPTIONS FOR POSTULATED FUEL HANDLING ACCIDENT

Wind direction	Toward west side of building (contains railroad cask car door)
Windward pressure factors	90 percent of wind velocity pressure
Lee and other pressure factors	-70 percent of wind velocity pressure
Exhaust flow rate from pool area	31,000 cubic feet per minute
Exhaust - Supply differential flow rate for building	8,000 cubic feet per minute
Windward area of aluminum siding	2,667 square feet
Leeward area of aluminum siding	2,175 square feet
Other areas of aluminum siding	2,175 square feet
Equivalent areas connecting to auxiliary building	500 square feet
Area subjected to 100 percent negative velocity pressure	1,403 square feet (20 percent of total)
Building leakage coefficient	40 cubic feet per minute per square foot per inch water guage
Mixing of supply and infiltrated air	100 percent
Fraction of release filtered	31,000/(31,000 & exfiltration flow)
Atmospheric dispersion model	No change in stablility with increased wind speed Dose inversely proportional to wind speed, for a given release quantity

APPENDIX A

CONTINUATION OF THE CHRONOLOGY OF RADIOLOGICAL REVIEW OF WILLIAM B. MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

OPERATING LICENSE REVIEW

May 1, 1978	Letter from applicant transmitting proprietary description of the proposed standby shutdown system designed to augment security systems
May 8, 1978	Submittal of Amendment No. 52, consisting of responses to letters dated February 8 and April 10, 1978
May 10, 1978	Letter from applicant providing clarification of response to staff letter of March 6, 1978
May 12, 1978	Letter from applicant transmitting description of analysis used to confirm the safety and integrity of the steam generator and pressurizer supports
May 17, 1978	Letter to applicant requesting information concerning electrical penetration seismic qualification tests
May 24, 1978	Issuance of Supplement No. 1 to Safety Evaluation Report
May 30, 1978	Letter from applicant advising that fire protection information will be submitted by July 31, 1978
June 1, 1978	Letter from applicant transmitting information on security provisions and revised pages for the security plan
June 5, 1978	Letter to applicant transmitting request for additional information concerning loss-of-coolant analysis
June 7, 1978	Letter from applicant concerning design requirements for upper head injection system
June 12, 1978	Letter to applicant transmitting safeguard handbooks
June 12, 1978	Letter from applicant transmitting information concerning steam generator and pressurizer supports in response to letter dated April 12, 1978

June 13, 1978	Letter to applicant transmitting request for additional information on reactor vessel fracture toughness properties
June 14, 1978	Letter to applicant transmitting request for additional information for fire protection review
June 19, 1978	Letter from applicant transmitting figures inadvertently omitted from revision to security plan submitted June 1, 1978
June 28, 1978	Meeting with applicant to discuss adequacy of revised Modified Amended Security Plan
June 29, 1978	Letter to applicant advising of schedule for review of security plan
June 29, 1978	Letter from applicant requesting extension of construction completion dates
June 30, 1978	Submittal of Amendment No. 53, including "Assessment of Flow Characteristics Within a Reactor Recirculation Sump Using a Scale Model," May 1978 and "Augmented Inservice Inspection for Pipe Rupture Protection," June 30, 1978
July 18, 1978	Letter to applicant transmitting "Barrier Penetration Database," NUREG/CR-0181
July 24, 1978	Letter from applicant transmitting information concerning requests for relief from requirements of American Society of Mechanical Engineers Code Section XI
July 26, 1978	Letter from applicant transmitting Revision 2 of Modified Amended Security Plan and other proprietary information regarding security provisions
July 31, 1978	Submittal of Amendment No. 54 in response to letter dated May 2, 1978
August 1, 1978	Letter to applicant transmitting draft Appendix I Technical Specifications
August 1, 1978	Letter from applicant transmitting information for fire protection review in response to letter dated June 14, 1978

August 2, 1978	Letter to applicant transmitting "Nuclear Security Personnel for Power Plants, Content and Review Procedure for a Security Training and Qualification Program," NUREG-0219
August 2, 1978	Letter to applicant transmitting request for information regarding preoperational testing of upper internals
August 3, 1978	Letter to applicant transmitting "Manpower Requirements for Operating Reactors" regarding fire brigades
August 9, 1978	Letter to applicant transmitting schedule for fire protection review
August 11, 1978	Letter to applicant concerning standard format for meteorological data on magnetic tape
August 14-15, 1978	Meeting with applicant to discuss augmented inservice inspection for pipe rupture protection
August 15, 1978	Letter to applicant advising of pressurized water reactor steam generator workshop to be held September 7 and 8, 1978
August 17, 1978	Letter to applicant transmitting comments by Idaho National Engineering Laboratory on "Pressurizer Surge Line Intermediate Pipe Rupture Protection"
August 21, 1978	Submittal of Amendment No. 55, consisting of updated information on piping stress and break analysis inside containment
August 23, 1978	Letter from applicant transmitting Revision 3 to Modified Amended Security Plan
August 25, 1978	Letter from applicant transmitting data on preoperational testing of upper internals
August 28, 1978	Letter to applicant advising of regional meeting to be held to discuss upgraded guard qualification and training requirements

August 30, 1978	Letter to applicant advising of site visit to obtain information in connection with fuel load forecast panel review
September 1, 1978	Letter from applicant transmitting missing page from August 1, 1978 letter on fire protection
September 5, 1978	Letter from applicant providing additional information to support request for relief from American Society of Mechanical Engineers Section XI requirements
September 5, 1978	Letter from applicant requesting exemption from 10 CFR Part 21 for relays and other materials supplied by Cutler-Hammer, Inc.
September 6, 1978	Letter to applicant transmitting final staff positions and unresolved issues in fire protection review and suggesting meeting be held to discuss analysis
September 8, 1978	Letter from applicant transmitting Revision 4 of Modified Amended Security Plan
September 11, 1978	Letter to applicant advising that meeting to discuss upgraded guard qualification is now scheduled to be held October 13, 1978
September 22, 1978	Letter from applicant transmitting proprietary information, environmental test reports for hydrogen skimmer and containment air return fan motors, Test Report FF-14282, Technical Paper RA 4081, Test Report X-604 (balance-of-plant Class IE equipment qualification)
September 25, 1978	Letter from applicant providing chronology of submittals concerning preservice inspection plan
September 26, 1978	Meeting with applicant to review applicant's plans and schedules for completion of construction, to observe the actual status of plant construction, and to assess the scheduled fuel load date
September 27-28, 1978	Meeting with applicant to discuss fire protection review and the 1978 resolution of outstanding issues

October 2, 1978	Submittal of Amendment No. 56, including report "Augmented Inservice Inspection for Pipe Rupture Protection," SRG-78-01, Rev. 1
October 3, 1978	Letter to applicant concerning criteria for piping modelling technique structural overlapping
October 5, 1978	Letter from applicant regarding inservice inspection report submitted with Amendment No. 56 and commenting on recommendations of document transmitted August 17
October 5, 1978	Letter from applicant transmitting information regarding qualification of balance of plant Class IE equipment, transmitting "Tests of Raychem Thermofit Insulation Systems Under Simultaneous Exposure to Heat, Gamma Radiat, Steam and Chemical Spray While Electrically Energized"
October 6, 1978	Submittal of Amendment No. 57, which includes "Revision 2 to the Evaluation of the Effects of Postulated Pipe Failures Outside Containment for McGuire Nuclear Station, MDS/PDG-77-1"
October 16, 1978	Letter from applicant requesting limited exemption to allow use of activated charcoal presently on site
October 20, 1978	Letter from applicant transmitting Revision 5 to Modified Amended Security Plan
October 25, 1978	Meeting with applicant to discuss mass energy transfer model for Westinghouse ice condenser design
October 26, 1978	Letter from applicant requesting approval of standby shutdown system concept associated with security and fire protection
November 1, 1978	Letter to applicant granting temporary exemption from technical specifications to permit use of activated charcoal
November 2, 1978	Letter from applicant transmitting information on fire protection
November 6, 1978	Letter to applicant transmitting request for information regarding fuel building ventilation system
November 7, 1978	Letter from applicant providing additional information to support request for extension of construction completion dates

November 16, 1978	Letter to applicant transmitting Revision 1 of Draft Radiological Effluent Technical Specifications and "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," NUREG-0133
November 17, 1978	Submittal of Amendment No. 58, consisting of response to letter dated June 5, 1978
November 20, 1978	Letter from applicant transmitting (1) nonproprietary version of qualification documentation for the hydrogen skimmer and containment air return fans submitted September 22 and (2) a summary listing of Westinghouse supplied safety-related transmitters and resistance temperature detectors
November 22, 1978	Letter to applicant transmitting request for information and advising of upcoming meeting to discuss augmented inservice inspection for pipe rupture protection
November 24, 1978	Letter to applicant in response to September 5 letter which requested exemption from certain requirements of 10 CFR Part 21
November 29, 1978	Letter to applicant requesting information regarding safety of bolted connections in linear component supports
November 30, 1978	Meeting with applicant to discuss augmented inservice inspection program for pipe rupture protection
December 18, 1978	Letter from applicant transmitting draft radiological effluent technical specifications and providing information on status of evaluation of solidification system and Offsite Dose Calculation Manual
December 21, 1978	Submittal of Amendment No. 59,* consisting of information concerning fuel building ventilation system, part length control rods, main steam line break protection, load follow capability and other miscellaneous revisions
December 26, 1978	Issuance of Order extending construction completion dates to April 30, 1979 (Unit 1) and December 31, 1980 (Unit 2)
January 17, 1979	Letter to JOY Manufacturing Company advising that staff unable to conclude at this time that the submittal of September 22, 1978 contains proprietary material
January 18, 1979	Appeal meeting as requested by applicant to discuss staff position on proposed augmented inservice inspection

January 23, 1979	Letter from applicant advising of change in fuel loading and commercial operation dates
January 25, 1979	Letter from applicant transmitting Offsite Dose Calculation Manual (first draft)
January 31, 1979	Letter from applicant transmitting revised table which summarizes seismic qualification requirements of electrical equipment
January 31, 1979	Letter from applicant requesting approval of flexible hose in construction of certain piping
January 31, 1979	Letter from applicant transmitting revision to "McGuire Nuclear Station Fire Protection Review"
February 6, 1979	Letter to applicant transmitting request for additional information regarding spent fuel cask drop accident analysis
February 7, 1979	Letter to applicant concerning technical specifications for solidification of radioactive wastes
February 8, 1979	Letter to applicant transmitting list of outstanding review matters
February 9, 1979	Letter to applicant concerning augmented inservice inspection for pipe rupture protection
February 27, 1979	Letter to applicant concerning requirement for analysis of containment temperature and pressure response to postulated main steam line break accident

*Unreviewed in this supplement to the Safety Evaluation Report.

APPENDIX B

SAFETY EVALUATION REPORT
MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2
EXEMPTIONS FROM CERTAIN REQUIREMENTS OF
APPENDICES G AND H OF 10 CFR PART 50

I. INTRODUCTION

The Duke Power Company provided information in the FSAR defining their method of compliance with 10 CFR Part 50, Appendices G and H for the McGuire Nuclear Station, Unit Nos. 1 and 2. As a result of our review of this information, we have determined that exemptions to 10 CFR Part 50, Appendices G and H are required and have also determined that exemptions regarding this matter are justified. Our bases for this conclusion for both Unit Nos. 1 and 2 are discussed in the subsequent paragraphs of this report.

II. TECHNICAL EVALUATION CONSIDERATIONS

A. The objective of Appendix G is to specify minimum fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. Specimens of the material of fabrication are required to be tested and the data used to develop safe operating condition limits for the reactor pressure vessel.

The objective of Appendix H is to monitor the change in fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens placed in the vessel before operation and withdrawn periodically during operation and tested to obtain fracture toughness data. These data permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The bulk of the detailed procedures and practices to be followed are given by way of reference to the ASME Code and ASTM Standards.

B. In the following evaluation the staff considered each area of variance with the regulations of Appendices G and H and have assessed the importance of those variances on the fulfillment of the safety objective of the regulations, as well as the feasibility of requiring absolute compliance with the regulations.

C. The ferritic materials of the Unit No. 1 reactor coolant pressure boundary were specified to meet the fracture toughness requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 1971 Edition, including Addenda and applicable Code Cases through Summer 1971. The reactor vessel for Unit No. 1 was fabricated by Combustion Engineering. The Unit No. 1 reactor vessel was fabricated with SA 533 plate material in the beltline region and has both longitudinal and circumferential weldments in this region.

The ferritic materials of the Unit No. 2 reactor coolant pressure boundary were specified to meet the fracture toughness requirements of Section III of the ASME Code, 1971 Edition, including Addenda and applicable Code Cases through Winter 1971. The reactor vessel for Unit No. 2 was fabricated by the Rotterdam Dockyard Company. The Unit No. 2 reactor vessel was fabricated with SA 508 forging material in the beltline region and has only a circumferential weldment joining the ring forgings in this region.

Specific exemptions for each unit are discussed separately because the different materials of construction and manufacturing processes require the consideration of different issues.

III. EXEMPTIONS REQUIRED

We have reviewed the information submitted by the Duke Power Company related to their method of compliance with 10 CFR Part 50, Appendices G and H. Based on this information and our review of the design, geometry, and materials of construction of the components, the requirement to comply with certain provisions of 10 CFR Part 50, Appendices G and H, has been determined to result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Therefore, pursuant to 10 CFR Section 50.12 specific exemption for those requirements is justified as follows:

A. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"

Exemption Required: The Duke Power Company has addressed the areas in which the McGuire Nuclear Station, Unit Nos. 1 and 2, is in non-compliance with certain requirements of 10 CFR Part 50, Appendix G. Based on our evaluation of this information we have determined that an exemption is required to enable the substitution of an alternative method of compliance with the requirements of 10 CFR Part 50, Appendix G.

Reason for Request: Based on our evaluation of the information provided by the Duke Power Company, we have determined that the requirements of Appendix G of 10 CFR Part 50 have been met except for the following:

Item 1: For Unit No. 1, Section III.C of Appendix G is not complied with to the extent that the unirradiated impact tests did not include all the required tests from separate weldment specimens taken from excess material from each of the six beltline shell plates with the corresponding heat of filler material used in the fabrication of the vessel.

Item 2: For Unit No. 1, Section IV.A.4 of Appendix G is not complied with to the extent that the Charpy V-notch test for the reactor vessel bolting material was not conducted in terms of the lateral expansion of the specimens and the test was not conducted at the lower of the preload temperature or at the lowest service temperature.

Item 3: For Unit No. 2, Section IV.A.4 of Appendix G is not complied with to the extent that the Charpy V-notch test for the reactor vessel bolting was not conducted in terms of lateral expansion of the specimens and the test was not conducted at the lower of the preload temperature or at the lowest service temperature.

Bases and Conclusions:

Item 1: The Unit No. 1 reactor vessel material testing program was formulated in accordance with Paragraph NB-2300 of the 1971 Edition of Section III of the ASME Code including Addenda through Summer 1971, which did not require the inclusion of samples from the weldment or the Heat Affected Zone (HAZ) of the beltline weldments in the subject testing program.

Paragraph III.C of Appendix G requires that test specimens are to be taken from the weldment and the HAZ material in the vessel beltline. This would require a consideration of testing the weldment and HAZ specimens representative of each plate/weld combination in the vessel beltline. In addition, Paragraph III.C requires that the Charpy V-notch impact test shall be conducted at appropriate temperatures over a temperature range sufficient to define C_v test curves (including the upper-shelf) in terms of both fracture energy and lateral expansion of specimens. This was not done for the Unit No. 1 reactor vessel since Appendix G was not effective at the time of fabrication of the vessel.

The FSAR contains sufficient data to establish the initial fracture toughness properties of the beltline materials. Tables 5.2.4-5 through 5.2.4-10 of the FSAR contains Charpy impact tests of the beltline plates through a temperature range of -40°F through +160°F. Table Q121.16-2 contains data on Charpy impact tests performed at 10°F of all the weldment material. Tables 5.2.4-3 and 5.2.4-4 of the FSAR contains Charpy impact tests of the weldment and HAZ of the limiting material through a temperature range of -80°F through +160°F in terms of impact energy, lateral expansion and percent shear. In addition, the reactor vessel material surveillance program includes weldment and HAZ material using the actual weld wire and flux in combination with the plate in the beltline with the most limiting properties.

The bases for our conclusions are as follows:

(1) The test data on all the base metal and the weldment material is sufficient to establish the limiting base metal and weld wire and flux combination. Full Charpy curves on the base metal and the limiting weld wire and flux combination are presented to demonstrate an adequate upper shelf in terms of both absorbed energy and lateral expansion.

(2) The test data on the other weldment material performed at only 10°F is sufficient to demonstrate an adequate upper shelf in terms of absorbed energy. The absorbed impact energy and the lateral expansion of the specimen are two closely related measurements of fracture toughness. Consequently, we have determined that the measurement of the absorbed energy is sufficient to establish the minimum acceptable fracture toughness properties. Conducting the impact tests at 10°F is conservative because the absorbed energy normally increases with increasing temperature.

(3) Test data is not available for the HAZ except for the limiting material. However, our experience with similar materials indicates that the Charpy impact test of the weldment is normally limiting when compared with the HAZ.

(4) Since the limiting material is defined and contained in the material surveillance program, sufficient data will be available to ensure adequate safety margins during operation.

Based on our evaluation of the information in the FSAR, we conclude that, while the precise requirements of Appendix G have not been complied with, the safety objective of Appendix G has been met.

Item 2: The Unit No. 1 reactor vessel bolting material testing program was formulated in accordance with Paragraph NB-2300 of the 1971 Edition of Section III of the ASME Code including Addenda through Summer 1971, which did not require that the Charpy V-notch test be conducted in terms of the lateral expansion of the specimens. In addition, Paragraph NB-2330 requires that the impact test temperature shall be 60°F below the lowest service metal temperature of the component except that for vessels the impact test temperature shall also be 60°F below the hydrostatic or pneumatic test metal temperature. The ASME Code required meeting an acceptance standard of an average of 35 ft-lbs.

Section IV.A.4 of Appendix G requires that material for bolting with nominal diameters exceeding one inch shall meet the minimum requirement of 25 mils lateral expansion and 45 ft-lbs in terms of Charpy V-notch tests conducted at the preload temperature or at the lowest service temperature, whichever temperature is lower.

The testing procedure, acceptance standards and testing temperature were performed in accordance with the ASME Code since Appendix G was not effective at the time of fabrication of the vessel. Tests were performed at 10°F on specimens from each end of seven bars used to fabricate the bolts and seven tubes used to fabricate the nuts and washers. The three impact tests on each end of the bars and tubes required by the ASME Code showed impact energy values that ranged from a low of 44, 46, and 44 ft-lbs to a high of 52, 52, and 52 ft-lbs for the bars and from a low of 42, 46, and 46 ft-lbs to a high of 54, 56, and 54 for the tubes. Three bars and one tube tested at 10°F showed impact values that were less than 45 ft-lbs.

The absorbed impact energy and the lateral expansion of the specimen are two closely related measurements of fracture toughness. Consequently, we have determined that the measurement of the absorbed energy, in accordance with the ASME Code requirements, is sufficient to establish the minimum acceptable fracture toughness properties. Further conducting the impact tests at 10°F, instead of the higher preload or service temperature, required by Appendix G, is conservative because the absorbed energy normally increases with increasing temperature. While the precise requirements of Appendix G have not been complied with, sufficient information has been provided to demonstrate that the safety objective of Appendix G has been met by performing the ASME Code impact tests.

We have evaluated the data presented in the FSAR and based on the results of our evaluation we have determined that the objective of Appendix G, as cited above, has been met.

Item 3: The Unit No. 2 reactor vessel bolting material testing program was formulated in accordance with Paragraph NB-2300 of the 1971 Edition of Section III of the ASME Code including Addenda through Winter 1971. The impact testing of the bolting for Unit No. 2 was conducted in a similar manner as Unit No. 1, described in Item 2 above, since the applicable ASME Code requirements did not change between the Summer 1971 and Winter 1971 Addenda.

Tests were performed at 10°F on specimens from each end of the 12 bars used to fabricate the bolts. The impact tests required by the ASME Code showed impact energy ranging from a low of 41, 44, and 44 ft-lbs to a high of 58, 60.5, and 60.5 ft-lbs. Three of the 12 bars tested at 10°F showed impact tests that were less than 45 ft-lbs.

We have evaluated the data presented in the FSAR for Unit No. 2 in a manner similar to Item 2 above. While the precise requirements of Appendix G have not been complied with, sufficient information has been provided to demonstrate that the safety objective of Appendix G has been met by performing the ASME Code impact tests. Based on the results of our evaluation we have determined that the objective of Appendix G, as cited above, has been met.

B. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

Exemption Required: The Duke Power Company has addressed the areas in which the McGuire Nuclear Station, Unit Nos. 1 and 2, is in non-compliance with a requirement of 10 CFR Part 50, Appendix H. Based on our evaluation of this information we have determined that an exemption is required to enable the substitution of an alternative method of compliance with the requirements of 10 CFR Part 50, Appendix H.

Reason for Request: Based on our revaluation of the information provided by the Duke Power Company, we have determined that the requirements of Appendix H of 10 CFR Part 50 have been met except for the following item.

For both Unit Nos. 1 and 2, Section II.C.2. of Appendix H is not complied with to the extent that the calculated neutron flux lead factor for four of the six surveillance capsules is 3.6 instead of the maximum allowable factor of 3.0.

Bases and Conclusions: Paragraph II.C.2. of Appendix H to 10 CFR Part 50 states that surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface.

The reactor vessel surveillance program for both units uses six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and are positioned directly opposite the center portion of the core. Dosimeters permit the evaluation of the neutron flux experienced by the surveillance specimens and the vessel wall. The reason that the lead factor (the ratio of neutron flux at the capsule to that at the vessel inner wall) in four of the capsules is higher than required by Appendix H is that new methods for calculating fast neutron fluence were developed after the reactor vessel internals were designed.

Our evaluation has determined that a lead factor up to 3.6 instead of 3.0 has no safety significance. Sufficient data from surveillance programs from operating plants have been generated and adequate radiation damage estimating techniques (Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials") are available to provide information to compensate for relatively small inaccuracies that may result from lead factors higher than 3.0 and to ensure that adequate safety margins are maintained.

We have evaluated the information presented in the FSAR and determined that the safety objective of Appendix H has been met.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH 10 CFR Part 50, APPENDICES G AND H

Our technical evaluation has not identified any practical method by which the existing McGuire Nuclear Station reactor vessels can meet the specific requirements of 10 CFR Part 50, Appendices G and H. Requiring specific compliance with these Appendices would delay the startup of the plant due to the need to complete the following actions: (1) obtain, if possible, sufficient material from the actual Unit No. 1 beltline plates to fabricate weldment and heat affected zone specimens for the remaining plate/weld combinations, and test the weldment test specimens, (2) for both Unit Nos. 1 and 2 replace the bolts, nuts, and washers having material with measured absorbed energies below 45 ft-lbs with material having higher fracture toughness, (3) remove and relocate the four installed material surveillance capsules in Unit Nos. 1 and 2 with a lead factor higher than 3.0.

We believe the public interest is served by not imposing certain provisions of 10 CFR Part 50, Appendices G and H, that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

V. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to 10 CFR Section 50.12, a specific exemption as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

APPENDIX C

SAFETY EVALUATION REPORT
MCGUIRE NUCLEAR STATION, UNITS NOS. 1 AND 2
EXEMPTION FROM CERTAIN REQUIREMENTS OF
APPENDIX J OF 10 CFR PART 50

I. Introduction

The Duke Power Company provided information describing their method of compliance with 10 CFR Part 50, Appendix J, relating to leak testing of the containment airlocks after each opening. As a result of our review of this information, we have determined that an exemption to 10 CFR Part 50, Appendix J is required and have also determined that an exemption regarding this matter is justified. Our bases for this conclusion are discussed herein.

II. Technical Evaluation Considerations

Containment Airlocks

Paragraph III.D.2 of Appendix J to 10 CFR Part 50 requires in part that containment airlocks be tested at six-month intervals and after each opening. Paragraph III.B.2 of Appendix J to 10 CFR Part 50 requires that these tests be performed at a pressure not less than the calculated peak containment internal pressure related to the design basis accident (14.8 pounds per square inch). The airlock design of the McGuire Nuclear Station, Units 1 & 2, includes dual seals on the airlock doors with the capability to apply a pressure between the seals. This will permit door seal integrity to be demonstrated without pressurizing the total airlock.

Based on plant operating experience, the leakage testing of containment airlocks after each opening as required by Paragraph III.D.2 of Appendix J to 10 CFR Part 50, when frequent airlock usage is necessitated over a short period of time is, in our judgment, impractical and unnecessary to assure the maintenance of the leaktight integrity of the airlocks.
It is our judgment that verification of the leaktightness of the airlock by leak testing of the door seals within 72 hours after being opened at the calculated peak containment pressure corresponding to the design basis accident pressure (Pa, 14.8 pounds per square inch gauge) provides the required assurance that the leaktight integrity of the airlock is maintained. The McGuire technical specifications provide that the airlocks be tested at six month intervals at the test pressure of 14.8 pounds per square inch gauge as required by Paragraphs III.D.2 and III.B.2 of Appendix J to 10 CFR Part 50 and the airlock door seals be leak tested within 72 hours after being opened.

These former tests involve pressurization of the entire airlock instead of just the gap between the door seals.

We, therefore, conclude that the methods for leakage testing the containment airlocks provided by the McGuire technical specifications represent an acceptable alternative to those required by Appendix J to 10 CFR Part 50.

III. Public Interest Regarding Compliance With 10 CFR Part 50 Appendix J

To require specific conformance with the applicable requirements of Appendix J to 10 CFR Part 50 for the leakage testing of the primary containment airlocks would necessitate after each opening (1) the installation of retainer clips on the interior airlock door (since the interior airlock door is designed to resist internal containment pressure), (2) the pressurization of the entire air lock to a pressure of 14.8 pounds per square inch gauge, (3) the determination of the leakage rate of the airlock, (4) the depressurization of the airlock, and (5) the removal of the retainer clips from the interior airlock door. The applicant estimates, based on previous experience, that approximately 8 hours would be required to perform this operation as opposed to approximately 15 minutes at least once every 72 hours when the air locks are being used for multiple entries. It is our judgment that to require that the entire air lock be leakage tested at a press. of 14.8 pounds per square inch gauge after each opening when result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

We, therefore, conclude that the public interest is served by not imposing the applicable requirements of Appendix J to 10 CFR Part 50 since such an imposition would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

IV. Conclusions

Based on the foregoing, we have determined that, pursuant to Section 50.12 of 10 CFR Part 50, a specific exemption for a period of three years as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

APPENDIX D

MCGUIRE 1 SAFETY EVALUATION REPORT
FIRE PROTECTION REVIEW
UNIT NOS. 1 AND 2

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MCGUIRE UNIT NOS. 1 AND 2
FIRE PROTECTION SAFETY EVALUATION REPORT

I. INTRODUCTION

We have reivewed the McGuire fire protection program and fire hazards analysis submitted by the applicant. The submittal was in response to our request to evaluate the fire protection program against the guidelines of Appendix A BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." As part of the review, we visited the plant site to examine the relationship of safety related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems.

The overall objective of our review of the McGuire Nuclear Plant fire protection program was to ensure that in the event of a fire at the facility, the units would maintain the ability to safely shutdown and remain in a safe shutdown condition and to minimize the release of radioactivity to the environment. Our review included an evaluation of the automatic and manually operated water and gas fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls and fire brigade training, and plant fire protection technical specifications.

Since Unit 1 and Unit 2 are of the same design, except where noted, the comments made in this report apply to both units.

II. FIRE PROTECTION SYSTEMS DESCRIPTION

A. Water Suppression Systems

The fire water system is common to both units and consists of three full capacity 2500 gallons per minute motor driven pumps, two gallons per minute pressure maintenance pumps (jockey pump) with a 5000 gallon pressure surge tank, and a yard loop with sectionalizing post-indicator isolation valves.

Power to fire pump A is from Unit 2, 2TB switchgear; power to fire pump B is from Unit 1 1TD switchgear; and power to fire pump C is from the 44 kilovolt substation independent of the McGuire Station auxiliary power system.

The jockey pumps take suction from the condenser circulating water system and the fire pumps take suction from Lake Norman. All pumps are installed in accordance with applicable National Fire Protection Association (NFPA) guidelines. A redundant starting scheme is used for the three main fire pumps. In the event of a fire, a drop in line pressure actuates a set of staggered set point pressure switches so that if the first pump fails to start, the second and third pumps will sequentially start automatically. Separate alarms monitoring pump running, drive availability, or failure to start are provided in the control room for each pump.

The fire pumps are located in the seismic Category 1 intake structure and separated by three hour fire rated barriers from the other pumps in that structure. Portions of the fire suppression system piping in the vicinity of safety related equipment or used to protect such equipment is designed to meet seismic conditions.

The automatic sprinkler system and standpipe system are fed by a main yard loop serving both Unit 1 and Unit 2 with multiple connections to interior fire protection system headers, the auxiliary building, turbine building, service building and reactor building. Each sprinkler system and manual hose station has an independent connection to the fire protection feeder, therefore a single failure cannot impair both the primary and backup fire protection systems.

Post indicator valves are provided to isolate sections of the fire loop for maintenance or repair. Valves in the fire protection system which are not electrically supervised, with indication in the control room, will be locked in normal operating position and checked periodically.

The automatic sprinkler systems, e.g., wet sprinkler system, pre-action sprinkler systems, deluge and water spray systems, are designed to the requirements of NFPA Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed System."

Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety related area in the plant. These systems are consistent with the requirements of NFPA Standard No. 14, "Standpipe and Hose System for Sizing, Spacing, and Pipe Support Requirements."

Areas that have been equipped or will be equipped with water suppression systems are:

(a) Cable spreading room (Manual Fog System)

(b) RHR pump rooms and adjacent corridor area (automatic)

(c) Motor driven auxiliary feedwater pump room (automatic)

(d) Centrifugal charging pump rooms (automatic)

(e) Nuclear service water pump rooms (automatic)

(f) Component cooling water pumps rooms (automatic)

(g) Reactor coolant pump (remote manual)

(h) Containment Annulus (remote manual)

(i) Pipe corridor @ EL 725' (automatic)

(j) Battery room open area - east and west ends (automatic)

We have reviewed the design criteria and bases for the water suppression systems and conclude that these systems meet the guidelines of Appendix A to Branch Technical Position 9.5.1 and are in accord with the applicable portions of the National Fire Protection Association (NFPA), Codes, and are, therefore, acceptable.

B. Gas Suppression System

A Halon 1301 suppression system is installed in the following areas:

(a) Emergency diesel generator rooms;

(b) Steam driven auxiliary feedwater pump room.

The Halon 1301 system for the diesel generator rooms is automatically actuated by fixed temperature detectors. Actuation of this system provides alarms and annunciate in the main control room. The ventilating systems for each room is shutdown automatically in the event of actuation of the Halon system, thus, isolating the affected diesel room.

The Halon suppression systems are designed and installed according to NFPA Standard No. 12A, "Halogenated Fire Extinguishing Agent Systems - Halon 1301." We have reviewed the design criteria and basis for these fire suppression systems. We conclude that these systems satisfy the provisions of Appendix A to Branch Technical Position 9.5.1 and are, therefore, acceptable.

C. Fire Detection Systems

The fire detection system consists of the detectors, associated electrical circuitry, electrical power supplies, and the fire annunciator panel. The types of detectors used at the McGuire Nuclear Plant are ionization (products of combustion), and thermal (heat sensors). The system is continuously supervised with a NFPA-72D Class B supervised system. The central supervising station is provided with two sources of power; primary (ac)and secondary (dc-inverter power). A trouble alarm is initiated and annunciated in the control room in the event of any sensor or circuit failure.

Fire detection systems will give audible and visual alarm and annunciation in the control room. Local audible and/or visual alarms are also provided.

At our request, the licensee agreed to install additional smoke detectors in the immediate area of the small exhaust fan servicing the control panels in the main control room. Smoke detectors will also be installed in each battery cell room, the steam driven auxiliary feed pump room and the peripheral rooms of the main control room to provide early notification of a fire.

The fire detection systems have been installed or will be installed according to NFPA No. 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protection Signalling Systems."

We have reviewed the fire detection systems to ensure that fire detectors are located to provide detection and alarm of fires that could occur. We have also reviewed the fire detection system's design criteria and bases to ensure that it conforms to the applicable sections of NFPA No. 72D. We conclude that the design and the installation of the fire detection systems with the additional detectors to be installed, meet the guidelines of Appendix A to Branch Technical Position ASB 9.5-1, and are therefore, acceptable.

III. OTHER ITEMS RELATING TO THE STATION FIRE PROTECTION PROGRAM

A. Fire Barriers and Fire Barrier Penetrations

All floors, walls, and ceilings enclosing separate fire areas have a minimum of a 3-hour fire rating. At our request, the applicant has conducted a full scale fire test for the wall separating the cable rooms along column line 56 in the auxiliary building, El 750. As a result of the fire test the applicant has agreed to modify this barrier by installing a fire proofed angle iron along the junction of the barrier and the ceiling. The main control room area contains peripheral rooms which are located within the main control room 3-hour fire barrier. These peripheral rooms will be provided with detectors and alarms and one-hour rated ceilings and fire doors.

Barriers having a 1 1/2 hour fire rating will be provided between redundant component cooling pumps. The barriers will extend from floor to ceiling and 3 feet beyond each pump.

In the area where the motor driven auxiliary feedwater pumps and remote shutdown panel are located, the applicant has agreed to provide 1/2 hour fire barriers to protect the overhead instrumentation and control cables for the turbine driven pump which pass through this area. In addition the remote shutdown panel will be protected on four sides from an exposure fire with a 1/2 hour barrier.

The applicant has provided documentation to substantiate the fire rating of the 3-hour barriers, the penetration seals used in the penetrations for cable trays, the conduits, and the piping.

B. Fire Doors and Dampers

We have reviewed the placement of the fire doors to ensure that fire doors of proper fire rating have been provided.

All doors which separate safety related redundant divisions, including doors separating the turbine building from any safety related equipment room, will be locked and/or alarmed in the control room.

Penetrations through rated barriers are sealed to provide fire resistance equivalent to the barrier itself. Ventilation penetrations through barriers are protected by standard fire door dampers. The applicant has provided NRC with necessary information to demonstrate that the fire dampers can provide a fire rating of three hours.

The applicant has further agreed to provide fire proofed supports for those HVAC ducts in safety related areas to ensure the integrity of the barrier penetration between the duct work, including the damper, and the fire barrier. The fire proofed duct support will be located at a distance no greater than 5 feet from the barrier.

We conclude that the fire barriers, barrier penetrations, fire doors and dampers are provided in accordance with the guidelines of Appendix A to Branch Technical Position ASB 9.5-1 and are, therefore, acceptable.

C. Armored Electrical Cable

The power and control cable used in McGuire is insulated with ethylene propylene rubber (EPR) then encased in a steel interlocked armor jacket or a braided wire armor jacket. The cable outside the containment has an outer PVC jacket over the metal armored jacket. The cable inside containment has the outer PVC jacket removed.

The applicant has conducted tests which demonstrate that no fire propagation from cable to cable or tray to tray occurs as a result of an electrically initiated fire. In addition, the cable used at McGuire passes the current IEEE 383 Flame Test. However, because of the PVC on the outside of the armored cable, we have required appropriate fire protection measures in the cable spreading room, as discussed in Section V of this appendix.

IV. Safe Shutdown System

The applicant will install, at a later date, a completely independent Standby Shutdown System (SSS) which will be located in a separate structure remote from the existing plant facilities. The system will incorporate its own AC and DC power supply and instrumentation. The SSS will provide a means of bringing the unit to a safe hot shutdown condition independent of loss of redundant safety functions in such areas as the cable spreading room, control room or the battery room common area and maintain this condition until damage control measures can be instituted to bring the unit to a cold shutdown condition.

Since the SSS will be installed after initial fuel loading, we required and the applicant has agreed to establish and implement, by initial fuel loading, interim emergency procedures to assure safe plant hot shutdown in the event of a damaging fire in the cable spreading room or the control room or the battery room common area. In addition, repair and operating procedures for cold shutdown following a fire incident will be established prior to plant start-up with the materials required to make the necessary repairs on site. The applicant will be able to make repairs and achieve cold shutdown within 72 hours; however, the SSS is capable of extending this time significantly longer.

We have reviewed the design concept and criteria of the SSS and conclude that it will be capable of achieving a safe hot shutdown condition and will meet the guidelines of Appendix A to BTP ASB 9.5-1 and are therefore acceptable. We will review the final design of the SSS when it is available.

Until the committed SSS is installed and operational, we consider that the interim procedures described above (to be reviewed by the Office of Inspection and Enforcement) as well as the applicant's established administrative procedures for control of combustibles and ignition sources, trained fire brigade and the fire protection technical specifications are sufficient to provide adequate protection against a damaging fire in the above mentioned areas. These areas have automatic fire detection and manual fire fighting capability.

V. Fire Protection for Specific Areas

A. Cable Spreading Room

Each unit has a cable spreading room. The two rooms are separated by walls having a three hour fire rating. The floors and ceiling in each room are also designed to have a fire rating of three hours. At present there is no fixed automatic system installed. Primary fire protection is provided by portable fire extinguisher and hose stations. Smoke detectors are provided that will initiate a local alarm and audible and visual alarm in the control room.

All power control or instrumentation cable pass the current IEEE No. 383 Flame Test. All cable within each cable spreading room is encased in a galvanized steel, interlocked armor jacket and covered with a polyvinyl chloride (PVC) jacket. All cables in this room are routed in cable trays. Aisle separation and overhead clearance is provided for access for fire fighting operation; however we were concerned that the large quantity of cable with the PVC jacket consitituted a significant fire load and that an exposure fire could disable the redundant safety related cable tray of one unit. At our request, the applicant has agreed to provide a manually initiated fixed waterspray (fog/mist) system for each cable room. The system will provide a level of open spray heads at the ceiling and an additional level below the lowest cable trans throughout both rooms.

The applicant will provide an independent safe shutdown system (see Section IV). As discussed previously, since the SSS is proposed for installation at a later date, the applicant will establish and implement by initial fuel loading interim emergency procedures to assure safe plant cold shutdown in the event of a damaging fire in the cable spreading room, the control room, or the battery room common area.

B. Battery Room Areas (Fire Area 13)

The battery cell rooms for both units are located in the battery room common area and are individually separated by 3-hour fire walls, with a metal deck ceiling above each battery cell room; therefore it is unlikely that a single fire could affect all the battery rooms at the same time and prevent safe shutdown of the plant. However, there is a large concentration of overhead cable trays at the east and west ends of the battery room common area. The cable trays at each end belong to the same safety division of each unit. Because of the heavy concentration of fire load in this area we requested and the applicant agreed to upgrade the existing fire dampers in the ventilation system to 3 hours. The applicant also agreed to provide a sprinkler system to protect the cable trays from an exposure fire.

We have reviewed the applicant's fire hazards analysis and fire protection provided for the cable spreading room, control room and battery room common area and consider that appropriate fire protection and interim emergency shutdown procedures have been provided for the period prior to the time the SSS is operational. Further, we find that after the SSS is operational, the system will conform to the provisions of Appendix A to BTP 9.5-1 and is therefore, acceptable.

C. Fire Protection Inside Containment

The major fire hazard in the containment is the lubricating oil contained in the reactor coolant pumps. Each reactor coolant pump is provided with an oil collection system around the upper and lower oil pots to contain any oilleakage and direct it to piping which goes to a drain tank. In addition a closed head sprinkler system is provided for each pump. The control valve for this system is manually operable from the control room. Ionization and fixed temperature detectors around the pumps alarm and annunciate in the control room.

Instrumentation cables within containment are encased in galvanized steel interlocked armor without a PVC jacket so that propagation of an electrically initiated fire is precluded.

Two containment auxiliary carbon filter units are located in the lower containment compartment. Each unit is protected by a fixed manual water spray system. Hose stations are provided as secondary protection throughout containment.

The annulus which contains armored cable penetrations without PVC jacketing is protected by a fixed manual extinguishing system with detection by both ionization and rate of rise heat detectors. The area is not readily accessible during normal plant operation. When containment access is possible, the area hose station and portable extinguisher (located outside containment) may be used for manual fire fighting.

We have reviewed the applicant's Fire Hazards Analysis for areas inside containment and conclude that appropriate fire protection has been provided and is acceptable.

D. Residual Heat Removal Pump Rooms

There are no fire doors installed on the three-hour fire barriers of each Residual Heat Removal pump rooms. Access for manual fire fighting is very limited by two open spiral stairways from the level above. At our request, the applicant has agreed to extend the sprinkler system in each Residual Heat Removal pump room to cover the adjacent corridor area where an exposure fire may occur and threaten the Residual Heat Removal pumps. We have reviewed the applicant's Fire Hazards Analysis for the Residual Heat Removal pump rooms and conclude that appropriate fire protection has been provided and is acceptable.

E. Other Plant Areas

The applicant's Fire Hazards Analysis addresses other plant areas not specifically discussed in this report. The applicant has committed to install additional detectors, portable extinguishers, hose stations, and some additional emergency lighting as identified in the applicant's installation schedule. We find these areas with the commitment made by the applicant to be in accordance with the guidelines of Appendix A of BTP 9.5-1, and the applicable sections of the National Fire Protection Association Code and are therefore acceptable.

VI. ADMINISTRATIVE CONTROLS

The administrative controls for fire protection consists of the fire protection organization, the fire brigade training, the controls over combustibles and ignition sources, the efire plans and procedures for fighting fires.

The applicant has agreed to revise his administrative controls and training procedures to follow supplemental staff guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated 6/14/77, and implement them by initial fuel loading for the following activities;

(a) Fire Brigade Training;

(b) Control of Combustibles;

(c) Control of Ignition Sources; and

(d) Fire Fighting Procedures

(e) Quality Assurance

The plant fire brigade of at least five members is organized to provide immediate response to fires that may occur at the site. Spare air cylinders and recharge capability are provided to satisfy the guidelines of Appendix A to Branch Technical Position ASB 9.5-1.

The plant fire brigade will also be equipped with breathing appartus, portable communications equipment, portable lanterns, and other necessary fire fighting equipment.

The fire fighting brigade participates in periodic drills. Liaison between the plant fire briade and the local fire departments has been established. The local fire departments have been on plant tours and have also been involved in training sessions with the plant fire brigade.

We conclude that the fire brigade equipment and training conform to the recommendations of the National Fire Protection Association, Appendix A to Branch Technical Position 9.5-1 and supplemental staff guidelines and are, therefore, acceptable.

VIII. TECHNICAL SPECIFICATIONS

We have reviewed the proposed Technical Specifications for McGuire Units Nos. 1 and 2 and find that they are consistent with our Standard Technical Specifications for fire protection. Following the implementation of the modifications of fire protection systems and administrative controls resulting from this review, the Technical Specifications will be modified accordingly to incorporate the limiting conditions for operation and surveillance requirements to reflect these modifications.

VIII. CONCLUSIONS

During the course of our review we have reviewed the applicant's submittals and his responses to our requests for additional information. In addition, we have made a site visit to evaluate the fire hazards that exist in the McGuire Nuclear Plant and the design features and protection systems provided to minimize these hazards.

The applicant has proposed to make many modifications to improve the fire resistance capability for fire doors, dampers, fire barriers and barrier penetration seals.

The applicant has also proposed to install additional sprinkler systems for areas such as the cable spreading rooms, battery room area, residual heat removal pump room area, and various other areas. To ensure that fires can be detected rapidly and the plant operators informed promptly, additional detectors will be installed in various areas of the plant.

The applicant has committed to making all improvements prior to initial fuel loading of Unit 1 with the following exceptions which will be implemented prior to commercial operation of Unit 1:

1. Extention of the residual heat removal pump room sprinkler system to protect the corridor connecting the pump rooms.
2. Automatic sprinklers installed to protect the cable tray stacks at the east and west ends of the battery room from an exposure fire.

3. Fire doors and dampers installed in penetrations in room 807 and 820 on elevation 750 feet of the auxiliary building which are adjacent to safety related equipment area.
4. Fire doors, dampers and the 1 1/2 hour rated ceilings for the peripheral rooms within the control complex.

In addition the applicant has committed to provide a completely independent safe shutdown system following commercial operation of Unit #1 to assure the hot shutdown capability for McGuire Nuclear Plant. Until the SSS is installed, the applicant will establish interim emergency shutdown procedures to bring the plant to safe hot standby condition in the event of a damaging fire in the cable spreading room, the main control room, or the battery room common area. Also, repair and operating procedures will be established to bring the plant to safe cold shutdown within 72 hours.

We have reviewed the applicant's schedule and find it acceptable. In summary, the fire protection system modifications to be completed by commercial operation as well as the applicant's emergency shutdown procedures, the control of combustibles and ignition sources, the barriers between fire areas, the trained onsite fire brigade with the capability to extinguish fires manually provide adequate protection from the adverse effects of a fire during the interim period prior to installation and operation of the Safe Shutdown System (SSS).

We find that the Fire Protection Program for the McGuire Nuclear Plant with the improvements already made and those being made by the licensee is adequate for the present and, with the scheduled SSSS, will meet the guidelines contained in Appendix A to Branch Technical Position 9.5-1 and meets General Design Criterion 3 and is, therefore, acceptable.

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16. ABSTRACT (200 words or less)

Supplement No. 2 to the Safety Evaluation Report for Duke Power Company's application for licenses to operate the McGuire Nuclear Station (Docket Nos. 50-369 and 50-370), located in Mecklenburg County, North Carolina has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this Supplement is to update the Safety Evaluation Report by providing the staff's evaluation of additional information submitted by the applicant since the issuance of Supplement No. 1 to the Safety Evaluation Report. Subject to favorable resolution of the items discussed in the Safety Evaluation Report and Supplement Nos. 1 and 2, the staff concludes that the plant can be operated by the Duke Power Company without endangering the health and safety of the public.

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