SUPPLEMENT NO. 1

TO

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT 1

DOCKET NO. 50-269

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1.0 INTRODUCTION

1.1 General

The Safety Evaluation by the Division of Reactor Licensing dated December 29, 1970, included a description of the Oconee Nuclear Station Units 1, 2, and 3 emergency core cooling system (ECCS) and our evaluation of the performance analysis of this system for the spectrum of break sizes up to and including the double-ended severance of the largest pipe of the reactor coolant pressure boundary. This evaluation was based upon ECCS analyses performed by the applicant and reported in the Oconee Nuclear Station operating license application. These analyses were performed using computer codes developed by B&W for analysis of large PWR reactors having safety injection systems.

Subsequently, the Atomic Energy Commission has reevaluated the theoretical and experimental bases for predicting the performance of emergency core cooling systems, including new information obtained from industry and AEC research programs in this field. As a result of this reevaluation, the Commission has developed interim acceptance criteria for emergency core cooling systems for light-water power reactors. These criteria are described in an Interim Policy Statement issued on June 25, 1971, and published in the Federal

Register on June 29, 1971, (36 F.R. 12247). By letter dated July 9, 1971, the Division of Reactor Licensing informed the applicant of the additional information that would be required for our evaluation of the performance of the Oconee Unit No. 1 ECCS in accordance with the Interim Policy Statement. The applicant provided a revised analysis of the Oconee Nuclear Station Unit 1 performance in a report titled "Multinode Analysis of B&W's 2568-MWt Nuclear Plants During a Loss-of-Coolant Accident" dated October 1971. The applicant also provided a supplement to this report, identified as Supplement 10 to the Oconee FSAR and dated December 17, 1971, that discusses the analysis of ECCS using unpressurized fuel pins in Oconee Units 1 and 3. The analysis was performed using the B&W Evaluation Model in conformance with the Interim Policy Statement, Appendix A, Part 4. The analysis was performed assuming the occurrence of a loss-of-coolant accident during operation at 102% of the requested power level of 2568 MW thermal.

1.2 Recent Experimental Information

Small-scale experiments have been conducted by the Aerojet Nuclear Corporation (formerly Idaho Nuclear Corporation) under contract to the U. S. Atomic Energy Commission as part of the reactor safety research and development work being carried out at the National Reactor Testing Station, principally to assist in the development of analysis methods to be used in the design and execution of the LOFT Project. During the past several years tests under this program have been performed to investigate the phenomena of blowdown of heated high pressure water from:

(1) a simulated reactor vessel with and without internals,

(2) a simulated reactor primary system with a vessel and single operating loop,

(3) a single loop system with an electrically-heated simulated reactor core, and

(4) a single loop, electrically-heated core system with accumulator ECC injection.

The results of some of these tests (LOFT Semiscale series 845-851) conducted in late 1970 and early 1971 showed that the analytical technique (RELAP-3 code) used by ANC at that time for blowdown analysis did not accurately predict the phenomena that occurred during blowdown after the cold ECCS water was introduced. The analysis had assumed that uniform and instantaneous mixing of the cold injection water and the hot residual fluid took place in the appropriate zones of the Semiscale system. The test showed that mixing is incomplete. In addition, the analysis did not predict that the cold ECCS water would be ejected from the vessel after injection. This phenomenon was observed in several cold leg Semiscale tests; the performance of the ECCS was satisfactory for the hot leg tests.

Although the LOFT Semiscale tests in this series have provided information for evaluation of the adequacy of analytical models, the results of these tests cannot be applied directly to describe the performance of pressurized water reactors following a loss-of-coolant accident because the test loop used was not designed so as to properly scale parameters affecting system performance. These include (1) the elevation head of the inlet annulus water, (2) the ratio of steam bubble diameters to the width of the vessel inlet annulus, (3) multiple flow loops, (4) relative loop and core resistances, (5) containment back pressure, (6) surface to volume ratios, (7) pump flow resistance, (8) steam generator model, (9) core heat rate, and (10) core internals.

Although the results of the small LOFT Semiscale experiments would not be expected to describe the performance of large power reactors, we have taken into account the results of these tests in establishing the acceptability of PWR interim evaluation models listed in Appendix A of the Commission's policy statement by including the conservative assumption that all of the water injected by the accumulators during blowdown is lost. Another consideration that led to this conservative assumption was the inadequacy of the currently used calculational techniques to predict accumulator water behavior during blowdown. As further experimental information or improved calculational techniques become available, this conservative assumption will be reevaluated.

2.0 DESCRIPTION OF EMERGENCY CORE COOLING SYSTEM

The Oconee Unit 1 emergency core cooling system (ECCS) consists of a high pressure injection system, an injection system employing core flooding tanks, and a low pressure injection system with external (to the containment) recirculation capability. Various combinations of these systems are employed to assure core cooling for the complete range of break sizes.

The high pressure injection system includes three pumps, each capable of delivering 450 gpm at 585 psig reactor vessel pressure and discharges to the reactor coolant inlet lines. One pump will provide the required minimum flow. The high pressure injection pumps are located in the auxiliary building adjacent to the containment. A concentrated boric acid solution from the boric acid water storage tank is provided to the suction side of the high pressure pumps during ECCS operation.

During normal reactor operation, the high pressure injection system recirculates reactor coolant for purification and for supply of seal water to the reactor coolant circulation pumps. The high pressure injection system is initiated at a low reactor coolant system pressure of 1500 psig or a reactor building pressure of 4 psig. Automatic actuation switches the system from normal to emergency operating mode. One of the three high pressure pumps is normally in operation. The system is designed to withstand a single failure of an active component without a loss of function.

The two core flooding tanks are located in the containment outside of the secondary shield. Each accumulator has a total volume of 1410 ft^3 with a minimum stored borated water volume of 1040 ft^3 pressurized with nitrogen to 600 psig. Each accumulator is connected to a separate reactor vessel core flooding nozzle by a flooding line incorporating two check valves and a motor operated normally open stop valve adjacent to the tank. The core flooding tanks will therefore inject water automatically whenever the pressure in the primary system is reduced below the core flooding tank pressure of 600 psig.

The low pressure injection system includes two pumps plus a spare pump each capable of delivering 3000 gpm at 100 psig reactor vessel pressure arranged to deliver water to the reactor vessel through two separate injection lines. One low pressure injection pump is capable of removing the heat energy generated after a loss-of-coolant accident.

The low pressure injection system pumps take their suction from the borated water storage tank (initially) and the reactor building emergency sump. The recirculation system components are redundant so as to withstand a single failure of an active or passive component without loss of function at the required flow.

The low pressure injection system is actuated on a low reactor coolant system pressure of 500 psig or a high reactor building pressure of 4 psig.

All of the ECCS subsystems can accomplish their function when operating on emergency (onsite) power as well as offsite power. If there is a loss of normal power sources the engineered safeguards power line is connected to the Keowee hydro unit which will start up and accelerate to full speed in 23 seconds or less. The pumps and valves of the injection system will be energized at less than 100% voltage and frequency to achieve the design injection flow rate within 25 seconds.

3.0 PERFORMANCE ANALYSIS OF EMERGENCY CORE COOLING SYSTEM

3.1 General

We have developed a set of conservative assumptions and procedures to be used in conjunction with the Babcock and Wilcox developed codes to analyze the ECCS functions. The assumptions and procedures used by B&W in analyzing the performance of the Oconee Unit No. 1 ECCS are described in Appendix A, Part 4 of the Interim Policy Statement published in the Federal Register on December 18, 1971 (F.R. Vol. 36, No. 244). Report BAW-10034 "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident," October 1971, covers the performance of cores for which all fuel pins are pressurized. In addition, Supplement 10 of the FSAR presents the B&W LOCA analysis for cores having unpressurized pins as will be the case for Oconee Units 1 and 3. Unit 1 will have unpressurized and pressurized (a mixture) pins for the first two cycles and Unit 3 will have a mixture for the first cycle only. The analysis for the core with a mixture of pressurized and unpressurized pins resulted in a heat rate limit of 17.4 KW/ft. for the 102% power case to meet the 2300°F maximum cladding temperature criteria. The applicant submitted an analysis in Supplement 10 to support his claim that the 17.4 KW/ft. limitation would not result in power penalty and that there would be adequate margin below this limit through core life. For comparison, the analysis reported in BAW-10034 is based upon an 18.15 KW/ft peak linear heat rate for cores with pressurized pins only. The 8.55 ft^2 cold leg split is the limiting case accident with a peak temperature of 2284°F in the case of the mixed core and 2177°F in the case of the pressurized pin only case.

3.2 Analysis of the Blowdown Period

The applicant used the CRAFT and THETA 1-B computer codes for the analysis of the blowdown phase of the transient. Using these codes, and the evaluation model specified in Appendix A, Part 4, of the Interim Policy Statement, the applicant provided the reevaluation of the ECCS performance in compliance with the Commission's Interim Policy Statement.

For the blowdown portion of the accident, we have concluded that the applicant's analyses as reported in BAW-10034 and Supplement 10 of the FSAR, conform to the requirements specified in the Commission's Interim Policy Statement, Appendix A, Part 4.

3.3 Analysis of the Refill and Reflood Period

The applicant has considered the thermal behavior of the core during the refill and reflood portion of the loss-of-coolant accident, which is explained as follows:

(1) The vessel refill is provided initially by the core flooding tanks, and later by the pumping systems, and is assumed to start at the end of the blowdown period. The reactor vessel is assumed to be essentially dry at the end of the blowdown period, as a result of the conservative assumption in Appendix A, Part 4, of the Interim Policy Statement that water injected by the core flooding tanks prior to end-of-blowdown is ejected from the primary system.

(2) No heat transfer in the core is assumed until the level of water reaches the bottom of the core, at which time refill is considered complete and the core reflood starts.

The end of blowdown is 14.6 seconds after rupture for the 8.55 ft^2 cold leg double ended break and reflood (to the bottom of the core) is complete about 23 seconds after rupture. The end of blowdown is 18.7 seconds after rupture for the 8.55 ft^2 cold leg split and reflood is complete about 26 seconds after rupture.

(3) The reflood of the core is characterized initially by a rapid liquid level rise both in the core and in the vessel annulus until enough of the core is covered to generate substantial amounts of steam. The re-flood rate

increases and peaks in about 8.5 seconds after the end of blowdown at about 11 to 12 inches per second, then decreases rapidly leveling off at about 5.5 inches per second about 10 seconds after the end of blowdown. At 10 seconds after the end of blowdown, the water covers about 12 inches of the core for the case of a double ended cold leg break and 20 inches of the core for the case of a 8.55 ft^2 cold leg split.

(4) The amount of steam generated in the core together with the steam flow path resistance governs the rate of steam flow. The steam flow path is assumed to be only through the vent valves within the reactor vessel and no credit is taken for steam flow around the loop. The steam flow resistance also limits the rate of liquid rise in the core, but the annulus water level continues to increase until the liquid level reaches the inlet nozzle. Core flood tanks and low pressure injection system water is piped directly to the reactor vessel with no intervening reactor coolant system piping.

(5) The peak temperature reached in the transient for the limiting 8.55 ft^2 cold leg split occurs about 30 seconds after the break.

Based on our review of "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident" BAW-10034, October 1971, and Supplement 10 to the FSAR we have concluded that the applicant has evaluated the refill and reflood events in an acceptable manner.

3.4 Results

The applicant has calculated the following temperatures for Oconee Unit No. 1 at 102% of a nominal power level of 2568 MWt:

Cold Leg Pipe Breaks		Peak Clad Temperatures (°F)	
(Area)	(Type Break)	Pressurized Pins	Unpressurized Pins
8.55 ft^2	(Double Ended)	2052	2072
8.55 ft^2	(Split)	2177*	2284*
3.0 ft^2	(Split)	1652	1662
0.5 ft^2	(Split)	1614	1561
Hot leg			
14.1 ft^2	(Split)	1621	1605

*Limiting case.

The total core metal-water reaction is less than 1% for each of the assumed pipe breaks.

4.0 CONCLUSIONS

On the basis of our evaluation of the additional B&W analyses, described in 3.1 above, we conclude that our acceptance criteria, as described in the Commission's Interim Policy Statement have been met:

(1) The maximum calculated fuel element cladding temperature does not exceed 2300°F.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor.

(3) The calculated clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

These are the same acceptance criteria that we stated on pages 42 and 43 of our Safety Evaluation on Oconee Unit 1.

The results of the applicant's analyses for a loss-of-coolant accident initiated at a core power level of 2568 MWt show that the acceptance criteria are met on the basis of analyses performed in accordance with an acceptable evaluation model given in the Interim Policy Statement.

On the basis of our evaluation of the additional B&W analyses described in 3.1 above, we have determined that the conclusion that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident, as set forth on page 43 of our Safety Evaluation dated December 29, 1970, remains applicable for the Oconee Nuclear Station reactors fo core powers up to 2568 MWt.