SEPTEMBER 3, 1971

SUPPLEMENT NO. 3

TO

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

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1.0 INTRODUCTION

1.1 General

The Safety Evaluation by the Division of Reactor Licensing dated November 16, 1970, included a description of the Indian Point Unit No. 2 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 Supplements 12 and 13 of the Indian Point Unit No. 2 operating license application. These analyses were performed using computer codes developed by the Westinghouse Electric Corporation for analysis of large PWR systems 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 19, 1971 and published in the Federal Register on June 29, 1971 (36 F. R. 12247). By letter dated July 7, 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 Indian Point Unit No. 2 ECCS in accordance with the Interim Policy Statement. The applicant provided a revised analysis of the Indian Point Unit No. 2 ECCS performance in a document titled "Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance", dated July 13, 1971. This analysis was supplemented by additional information in a document titled "Additional Information on Emergency Core Cooling Analysis" dated August 16, 1971. The analysis was performed using the Westinghouse Evaluation Model in conformance with the Interim Policy Statement, Appendix A, Part 3. This portion of the Interim Policy Statement is attached for reference as Appendix I to this report. The analysis was performed assuming the occurrence of a loss-of-coolant accident during operation at 102% of the requested license power level.

1.2 Recent Experimental Information

Small-scale experiments have been conducted by the Idaho Nuclear Corporation (INC)*, 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

* Renamed Aerojet Nuclear Corporation on July 1, 1971

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 INC 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 tests 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 Semiscale 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 are (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 resistances, (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 Indian Point Unit No. 2 emergency core cooling system (ECCS) consists of a high pressure injection system, an injection system employing accumulator tanks, a low presssure injection system with external (to the containment) recirculation capability, and a seperate internal (to the containment) recirculation system. 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 rated at 400 gpm at 1200 psi each and discharging to two separate headers. Two of the three pumps discharging through either a single header or both headers will provide the required minimum flow. The high pressure injection pumps are located in the primary auxiliary building adjacent to the containment. Initially a concentrated boric acid solution from the boric acid injection tank is provided to the suction side of the high pressure pumps, followed by borated water from the refueling water storage tank. Discharge from the pumps is routed through the two headers so as to inject into each of the four primary coolant loops. Each of four high pressure injection lines is provided with a check valve and motor operated stop valve to isolate the system from the primary system. Opening of the stop valve is actuated by the safety injection signal. The system is designed to withstand a single failure of an active component without a loss of function.

The four accumulator injection tanks are located in the containment in the annular space between the secondary shield wall and the containment wall. Each accumulator has a total volume of 1100 ft^3 with a minimum stored borated water volume of 700 ft^3 pressurized with nitrogen to 600 psi. Each accumulator is connected to a separate loop of the primary system by a line incorporating two check valves and a normally open, remote-operated valve in series. The accumulator will therefore inject water automatically whenever the pressure in the primary system is reduced below the minimum accumulator pressure of 600 psi.

The low pressure injection system includes two pumps each rated at 3000 gpm at 150 psi arranged to discharge into each of the primary coolant loops. During the initial phase of ECCS function, the pump suction is connected to the Refueling Water Storage Tank. When the water in this tank has been used, suction is transferred to the containment sump for recirculation of the sump water. The low pressure injection pumps are located well below grade level in the primary auxiliary building so as to provide for adequate suction head from the containment sump during recirculation. A separate recirculation system, provided within the containment, includes two pumps rated at 3000 gpm at 150 psi which take suction from a separate containment sump. This system can recirculate water to the reactor via either the low pressure injection header or the high pressure injection header. Taken together the recirculation systems are redundant so as to withstand a single failure of an active or passive component without loss of function at the required flow.

Actuation of the injection pumps and the valves that isolate them from the primary coolant system is initiated by the safety injection signal (SIS) that results from coincidence of two of these low pressurizer pressure signals with two of three low pressurizer level signals, or from two of three high containment pressure signals.

All of the ECCS subsystems can accomplish their functions when operating on emergency (onsite) power as well as offsite power. If one of the three diesel generators should fail to start, the minimum ECCS requirement of the accumulators (which require no power), plus one low pressure pump, and two high pressure injection pumps would be available for operation and capable of providing the required performance. The diesel loads and ECCS starting sequence are arranged so that the system will be pumping at the full rated flow within 34 seconds following the safety injection signal.

3.0 PERFORMANCE ANALYSIS OF EMERGENCY CORE COOLING SYSTEM

3.1 General

The AEC has developed a set of conservative assumptions and procedures to be used in conjunction with the Westinghouse developed codes to analyze the ECCS functions. These assumptions and procedures were used by Westinghouse in analyzing the function of the Indian Point Unit No. 2 ECCS. This is described in Appendix A, Part 3 of the Interim Policy Statement (Appendix I to this report).

The design of the Indian Point Unit No. 2 ECCS has not been changed as a result of our reevaluation of its functional performance. The reassessment of the functional performance of the ECCS, presented in this section, applies to an analysis of the plant assuming that the postulated loss of coolant accident occurs at a power level of 102% of 2758 MWt using the Westinghouse evaluation model described in Appendix A. Part 3, of the Commission's Interim Policy Statement, adopted June 19, 1971. The applicant submitted this reassessment in supplemental testimony dated July 13, 1971 and August 16, 1971.

In order to meet the criterion limiting the calculated peak clad temperature to less than 2300°F, the applicant will be required to reduce the allowable nuclear peaking factors from the values previously specified in the proposed Technical Specifications. The Technical Specifications will be modified to require the nuclear hot channel factors F_q^N and $F_{\Delta H}^N$ to be reduced from 3.12 to 2.90 and from 1.75 to 1.66, respectively.

In our review of the revised analysis, we first considered the events that occur during the blowdown period, defined as the time for occurrence of the postulated pipe break to the time that the primary system pressure is reduced to containment pressure, the end of blowdown.

The second phase of the accident, called the refill and reflood period, starts at the end of blowdown and stops when the temperature transient of the fuel cladding is satisfactorily controlled.

3.2 Analysis of the Blowdown Period

The applicant used the SATAN-V and LOCTA-R2 computer codes for the analysis of the blowdown phase of the transient. Using these codes, and the evaluation model specified in Appendix A, Part 3 of the Interim Policy Statement, the applicant provided the information we needed to complete our reevaluation of the ECCS performance in compliance with the Commission's Interim Policy Statement.

Changes to analysis assumptions from those previously used in the ECCS performance calculations for Indian Point Unit No. 2 include:

(1) A 5% reduction in the nuclear peaking factor.

(2) A change in the model for the resistance of the reactor upper core support plate.

(3) A 20% increase in the decay heat with a decrease in heat deposition in the hod rod from 97.4% at steady state to 95% for the loss-of-coolant accident.

(4) A 20% reduction in core flow when applied to hot channel calculations.

(5) The time to departure from nucleate boiling should be equal to 0.1 seconds.

(6) A revision to the transition boiling correlation.

The changes had offsetting effects on the calculated peak clad temperature at the end of blowdown. In Supplement 12 for Indian Point Unit 2 dated July 30, 1970, on Figure 4 of Appendix 14B, a peak clad temperature of about 1600°F is predicted at the end of blowdown (16.4 seconds after the initiation of the accident) compared with the new value of 1550°F shown in Figure 10 of the additional testimony for Indian Point Unit 2 concerning ECCS performance dated July 13, 1971.

For the blowdown portion of the accident, we have concluded that the applicant's analysis conforms to the analysis requirements 1-6 specified in the Commission's Interim Policy Statement, Appendix A, Part 3.

3.3 <u>Analysis of the Refill and Reflood Period</u>

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

(1) The vessel refill is provided initially by the accumulator 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 the Appendix A, Part 3 of the Interim Policy Statement that accumulator water injected 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. Refill takes approximately 15 seconds to accomplish for the larger breaks, thus water reaches the bottom of the core approximately 30 seconds after the occurrence of the break.

(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 initial rate of rise is approximately 6 inches per second and the initial height before appreciable steam is generated is about 20 inches.

(4) When the amount of steam generated becomes appreciable, the pressure drop that occurs as a result of the steam flow to the break governs the rate of steam flow. The steam flow path is assumed to be only through the broken loop until accumulator discharge in the intact loops is complete.

This assumption is made in recognition of the fact that the accumulator water could block or partially block the lines in the intact loops. (For the Indian Point Unit 2 plant, for the double-ended cold-leg break, this assumption results in a reduced steam flow rate for about 10 seconds after water reaches the bottom of the core.) The steam flow resistance limits the rate of liquid rise in the core, but the annulus continues to fill with water until the liquid level reaches the inlet nozzle. After this it flows to the containment by way of the broken inlet pipe path.

(5) When the accumulators have completed their discharge, the intact loops become additional vent paths for steam generated from reflood water. This results in a substantial increase in the steam flow rate and core heat transfer. The peak temperature reached in the transient for the limiting double-ended cold-leg break occurs about 80 seconds after the break.

On page 72 of its additonal testimony of July 13, 1971, the applicant states that there were no deviations from Part 3 of the Commission's Interim Policy Statement. Based on our review of the additional testimony as supplemented on August 16, 1970, 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 Indian Point Unit No. 2 at 102% of a nominal power level of 2758 MWt:

Cold-Leg Pipe Break Area	Peak Clad Temperatures (F°)
8.24 ft.^{2} (double-ended)	2300
6.6 ft.^{2}	2280
4.5 ft.^{2}	2160
3.0 ft.^{2}	1715
0.5 ft.^{2}	2185

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

4.0 Conclusions

Our acceptance criteria, as described in the Commissions's Interim Policy Statement are:

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

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

(3) The clad temperature transient should be 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 were stated on page 40 of our Safety Evaluation on Indian Point Unit No. 2.

The results of the applicant's analyses for a pre-accident power level of 2758 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 with the nuclear hot channel factors F_q^N and $F_{\Delta H}^N$ reduced from 3.12 to 2.90 and from 1.75 to 1.66, respectively.

On the basis of our evaluation, we have concluded that the conclusions set forth on Page 40 of our Safety Evaluation dated November 16, 1970, are applicable to the operation of the Indian Point Unit 2 plant at 2758 MWt provided that the limits on peaking factors in the Technical Specifications are reduced as indicated above.

Appendix I
or
Appendix A
Part 3
Westinghouse Evaluation Model

Analyses should be performed for the entire break spectrum, up to and including the double-ended severance of the largest pipe of the reactor coolant pressure boundary. The combination of systems used for analyses should be derived from a failure mode and effects analysis, using the single-failure criterion.

The analytical techniques to be used are described in the topical report, "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident" WCAP-7422-L January, 1970 (Proprietary) and a supplementary proprietary Westinghouse report, "Emergency Core Cooling Performance," received June 1, 1971, and in an appropriate nonproprietary report to be furnished by Westinghouse, with the following exceptions:

For breaks greater than 0.5 ft^2-

1. The break discharge coefficient, (C_D), used with the Moody discharge flow model should be equal to 1.0 for all break sizes.
2. The decay heat curve described in the proposed ANS Standard, with a 20% allowance for uncertainty, should be used. The fraction of decay heat generated in the hot rod may be considered to be 95% of this value.

3. For large breaks in the range 0.6 to 1.0 times the total area of the double-ended break of the largest cold-leg pipe, two break models should be used. The first model should be the double-ended severance ("Guillotine"), which assumes that there is break flow from both ends of the broken pipe, but no communication between the broken ends. The second model should assume discharge from a single node ("split").
4. The time after the break for the onset of departure nucleate boiling at the hot spot should be equal to 0.1 second.
5. For cold leg breaks, all of the water injected by the accumulators prior to end-of-blowdown shall be assumed to be lost. In this context the end-of-blowdown shall be specified as the time at which zero break flow is first computed. The containment back pressure assumed for the blowdown analysis should not be higher than the initial pre-break pressure plus 90% of the increase in pressure calculated for the accident under consideration.
6. The pump resistance, K, used for analysis should be fully justified. The effect of pump speed upon K should be considered. The more conservative of two assumptions (locked or running) should be used for the pump during the blowdown calculation.
7. A calculation for the reflooding heat transfer should be performed. The containment back pressure assumed for the analysis should not be higher than the initial pre-break pressure plus 80% of the increase in pressure calculated for the accident under consideration.

The following items should be constraints on the calculation:

a. No steam flow should be permitted in intact loops during the time period that accumulators are injecting.

b. Core exit quality should be calculated from entering mass flow rate and nominal FLECHT heat transfer.

c. Pump resistance should be calculated on the basis of a locked rotor.

d. The effects of the nitrogen gas in the accumulator, which is discharged following accumulator water discharge, should be taken into account in calculating steam flow as a function of time.

e. The pressure drop in the steam generator should be calculated with the existing fluid conditions and associated loss coefficients.

f. All effects of cold injection water, in either a hot or cold leg, on steam flow (and Δ P) should be included in the calculation.

g. The heat transfer coefficient during reflood should be derived from FLECHT data.

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