ATTACHMENT I TO
IPN-88-040

EVALUATION OF THE SERVICE WATER SYSTEM (SWS) PIPE BREAK CRITERIA AND ANALYSIS FOR THE INDIAN POINT 3 NUCLEAR POWER PLANT

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I. SUMMARY

The objectives of this study are to review the analyses performed for the diesel generator cooling water loop to determine whether the results of those analyses satisfy the conditions stipulated in the Safety Evaluation Report for IP3[1] and to review current regulatory and industry guidance for postulating passive failures in moderate energy lines in order to formulate a position on postulating single failures for the Service Water System.

This report produced the following major results:

1. The open item in the SER Section 9.5.4, requiring adequate cooling water flows following passive failures in the Diesel Generator Cooling Water Loop, was satisfactorily resolved by the conclusions of the break study performed by Con Ed[2]. The requirement to postulate full size guillotine and slot ruptures in the Service Water System (SWS) is overly conservative and, except for the break in the 10-inch diesel generator cooling water supply header, could not be traced to any outstanding safety issue identified by the staff.

2. The IP3 SWS is capable of performing its intended safety function under active and passive failure conditions consistent with the design of the system within the context of the SER.

3. The crack locations and sizes postulated under the guidance of SRP Sections 3.6.1 and 3.6.2 are believed to be bounding in terms of the consideration of passive failures as addressed in SECY-77-439 and ANSI/ANS 58.9-1981, and should be applicable to the IP3 SWS pipe failure analysis.

II. CHRONOLOGY OF DEVELOPMENT

1. Safety Review

During the safety review of the Indian Point 3 Nuclear Power Plant (IP3), prior to issuance of the facility's operating license, the Atomic Energy Commission (AEC) expressed concern as to whether the IP3 emergency diesel generators would be adequately cooled in the event of a break in the Diesel Generator Cooling Water Loop.* At a meeting between the Consolidated Edison Company of New York, Inc. (Con Ed) and the regulatory staff on July 20, 1973, five break locations in the vicinity of the IP3 diesel generators were identified.

Subsequently, the Atomic Energy Commission's Safety Evaluation Report (SER)(1) in the matter of the application by Con Ed to operate the IP3 unit was issued on September 21, 1973. In this SER, the staff indicated that additional information involving a number of safety related issues was required from Con Ed to complete the staff's evaluation of the IP3 application.

Section 9.5.4 of the SER discusses the Diesel Generator Cooling Water System. In that section of the SER, the AEC staff postulated a break in the ten inch Service Water System (SWS) line supplying the three diesel generators which would result in inadequate cooling of the diesel generators and their eventual burnout. The break was to occur during the recirculation phase following a LOCA, (during the injection phase, only active component failures are addressed).

* The Diesel Generator Cooling Water Loop is part of the Service Water System.

Following publication of the SER, at a meeting held with the AEC on October 3, 1973, Con Ed was asked to consider pipe breaks anywhere in the SWS. These break locations were characterized as either guillotine or slot failures. In one case full circumferential failure occurred with free discharge from both ends of the broken pipe, in the other (slot break) case only partial losses of fluid were considered. The analysis of break locations other than at the ten-inch SWS line supplying the diesel generators was not incorporated into the SER as a condition for issuance of the operating license for IP3.

2. Pipe Break Study

The break analysis(2) utilized design parameters as input to the program. The program used in the analysis was named PIPEFLO, which has been used to analyze two and three dimensional fluid piping networks. PIPEFLO used the Newton-Raphson method of solving a system of non-linear equations.

As reported in Supplement No. 1 to the SER, dated February 21, 1975(9), Con Ed, on the basis of the results of the break analysis, proposed an alternative method of coping with postulated Service Water System line breaks. The method, which is described in the updated FSAR(7), splits the essential and non-essential recirculation loads between the designated nuclear and conventional service water headers.

The results of the break analysis for the SWS alignment in the recirculation phase proposed by Con Ed demonstrated the capability of the system to survive various breaks and still perform its intended safety function. It should be noted that the conclusions of the pipe break analysis(2) are valid for all of the breaks postulated in that study except for breaks which involve complete severance of the 24" essential header, during post-LOCA recirculation, upstream of the header check valves.

Following verification of flows during functional testing in April 1975, this issue was concluded within the Safety Evaluation Report, as discussed in Supplement No. 2 of the SER, dated December 12, 1975[10]. This supplement states: "We conclude that the diesel generator cooling water supply from the existing service water system can accommodate the passive failure postulated in the Safety Evaluation Report and, therefore, is acceptable". Note again that the passive failure postulated in the SER is a break in the 10-inch line supplying the diesel generators.

3. Adequacy of the Pipe Break Model

During an NRC review of a proposed modification to the IP3 Service Water System in May 1987, a discrepancy was noted between the network utilized for the break study for the SWS (Figure 1) presented in Section 9.6.1 of the Updated IP3 Final Safety Analysis Report (FSAR)[7], and the actual system configuration.

The network used for the break analysis[2] did not account for two check valves, one on each main SWS header, which will prevent backflow through two diesel generators under certain break conditions, hence challenging their operability. The IP3 facility requires that two of three diesel generators be operable in any combination to satisfy minimum safeguards requirements.

This error, however, only affects the guillotine break postulated to occur upstream of the check valves, specifically the break of the 24" essential header.

Hence, the results of the pipe break analysis are still applicable to the 10-inch line break identified by the NRC as the unresolved safety issue in their SER.

The method proposed by Con Ed to cope with breaks in the SWS piping specifically fulfills the requirements of the SER, since evaluation of the results of the pipe break analysis (with or without the check valves noted previously) demonstrates that for a 10-inch diesel generator supply line pipe break, adequate cooling is maintained to the diesel generators and the intended safety function of the SWS is satisfied. This is consistent with the conclusions in Supplement 2 of the SER.

III. REGULATORY AND TECHNICAL CONSIDERATIONS

NYPA believes that the outstanding safety issue stated in the SER was properly addressed by the break analyis performed and by the injection to recirculation switchover procedure.

Several issues are discussed below which are relevant to the evaluation of pipe breaks in the SWS for the IP3 facility and which have been reviewed by NYPA.

1. General Design Criteria

The General Design Criteria (GDC) which formed the basis for the IP3 design were published by the Commission on July 11, 1967 and were subsequently made part of 10CFR50. Of these original GDC, only Criterion 41 appears to apply to the SWS. This criterion requires that:

> Engineered Safety Features ... shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

Criterion 41 (1967) did not require consideration of passive failures for engineered safety features and, of course, no coincident Loss of Offsite Power (LOOP) following a Loss of Coolant Accident (LOCA).

However, in 1971 (prior to issuance of the SER and during the safety review by the staff of the IP3 facility) the Commission issued new GDC in Appendix A to 10CFR50. Criterion 44 was specifically applicable to the IP3 SWS. This criterion states that:

> A system to transfer heat from structures, systems and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions.

> Suitable redundancy in components and features and suitable interconnection, leak detection and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Within GDC 44 (1971), the single failure criterion is not specifically defined to encompass active and/or passive failures. A footnote 2 to Appendix A to 10CFR50 does however indicate that: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development".

As further clarification of the single failure criterion for the SWS, a review of NUREG-0800, Standard Review Plan, Section 9.2.1, Station Service Water System, does not require consideration of passive failures of the SWS under design basis accident conditions. However, the singular wording of footnote 2 to 10CFR50, Appendix A appears to indicate an element of judgement on the part of the staff when considering passive failure in fluid systems.

As noted in Section II of this report, the postulated break in the 10" cooling water line to the diesel generators during the recirculation phase following a LOCA forms the design basis for IP3 and the SWS is capable of accommodating such a break while still fulfilling its intended safety function. But NYPA does not believe that the size of break postulated in the break analysis(2) is representative of the type of break to be expected for SWS piping.

2. SECY-77-439[4]

As further clarification for defining the types of passive failures to be considered for fluid systems in nuclear power plants, in a memo from the staff to the Commissioners (SECY-77-439), NRC has concluded that:

> "... on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the single failure criterion to assure safety of a nuclear power plant."

Elsewhere, the SECY-77-439 report asserts that:

> "In the study of passive failures, it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks. No other passive failures are required to be assumed."

The SECY-77-439 report continues:

> "... an example of the application of a passive failure requirement is the approach to long-term recovery subsequent to a loss-of-coolant accident. Applicants are required to consider degradation of a pump or valve seal and resulting leakages in addition to initiating failure (LOCA)."

3. Formulation of Passive Failure Criteria

A review of NRC regulations relative to passive failures indicates that whereas consideration of passive failures is required for high energy systems (SRP Section 6.3, Emergency Core Cooling System), the passive failure criteria is more relaxed for moderate energy lines (in particular for the Service Water System, refer to SRP Section 9.2.1). Furthermore, although limited size breaks in moderate energy lines have been required, they have been taken as initiating events and not coincident with LOOP and LOCA. The intent has been to eliminate or reduce the risk of affecting the operation of a system important to safety as a result of breaks in other moderate energy systems nearby.

However, if piping failures in a moderate energy fluid system, such as IP3's Service Water System piping are to be evaluated, questions arise as to available guidance regarding the location and size of the postulated failure.

Enveloping passive failures in fluid systems are those which result in the loss of structural integrity of the system; i.e., a pipe break of undefined size. A review of industry standards for piping has shown that in determining the criteria for postulating passive failures in fluid systems, it is important to distinguish pipe failures as initiating events from long term passive failures subsequent to the initiating event. A crack in a moderate energy line which is evaluated according to criteria in SRPs 3.6.1 and 3.6.2 is considered as an initiating event. To satisfy General Design Criterion 44, current industry standards ANS51.7, and ANSI/ANS58.9(5)(6) require the consideration of a long term passive failure during post-LOCA recirculation in addition to the initiating event (in this case a LOCA). However, when supported by an analysis, the long term passive failure is limited to the "maximum flow through packing or mechanical seal rather than based on complete severance of the piping". (Ref. ANS 51.7-1976 and SECY-77-439)(4)(6). Further, no passive failures need be postulated in the short-term (up to 24 hours after the initiating event).

Again, the NRC does provide guidance for the evaluation of pipe breaks to support their review of a licensee's conformance with General Design Criteria 44 in NUREG-0800, Standard Review Plan (SRP) Sections 3.6.1 and 3.6.2(3). These sections address the review of postulated ruptures of piping systems and the evaluation of the impact of the dynamic effects associated with postulated rupture on structures, system and components important to safety.

It should be re-emphasized that the review under SRP Sections 3.6.1 and 3.6.2 does not deal with individual system design requirements necessary to ensure that the system performs as intended, but rather considers the protection necessary to assure the operation of such systems in the event of nearby piping failures. In addition, the criteria for evaluating postulated breaks in piping considers breaks only as single initating events occurring during normal plant conditions and not as passive failures postulated during the recirculation phase of plant cooldown following a LOCA.

These conditions notwithstanding, the criteria which have been developed for determination of pipe rupture locations and sizes are based on the governing conditions of stress and fatigue.

The point in a given piping system where a rupture would most likely occur would be associated with points of high relative stress and high relative fatigue. These points can be predicted for any piping system for various operating conditions and design loadings; therefore, the criteria for selecting break sizes and locations are intended to provide the maximum practical protection by postulating breaks at those locations with the greatest potential for failure under loading conditions associated with specific seismic events and plant operational conditions. These same criteria are thus assumed to be applicable for the consideration of passive failures in piping during the recirculation phase of plant cooldown following LOCA.

Since the SRP Section 3.6.1 and 3.6.2 criteria primarily are concerned with the protection of essential plant features from the dynamic effects associated with postulated pipe ruptures, only those portions of the SRP Criteria dealing with the size and location of postulated ruptures can be considered appropriate for use in this review of passive failures in the IP3 SWS piping.

The IP3 SWS is considered a moderate energy fluid system. The definition of a moderate energy fluid system adopted by NRC is presented in SRP 3.6.1 as a system that experiences an operating temperature of 200°F or less and a maximum operating pressure of 275 psig.

The break type postulated in the SRP on the basis of stress and fatigue for all seismically analyzed moderate energy systems is a leakage crack which is described as a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. The leakage crack is considered applicable to all moderate energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

For the IP3 SWS, which is comprised mainly of cement lined carbon steel pipe, the break width should be based upon the thickness of the carbon steel pipe only, since the cement lining does not contribute to pressure retaining capacity of the pipe, but is specified only for its corrosion-resisting properties.

In summary, to postulate passive breaks in the Service Water System during the recirculation phase of plant cooldown, the following methodology should be employed: for seismically designed portions of the service water leakage cracks (1/2 pipe diameter x 1/2 pipe wall thickness) should be postulated to occur at any point on the pipe. This crack size is taken to envelope and bound other passive failures to be taken into consideration.

4. Probabilistic Risk Assessment (PRA)

To support the use of limited size breaks in the analysis of passive failures for the IP3 SWS, the likelihood of catastrophic pipe failures has been reviewed.

The use of PRAs and limited PRAs has been utilized by NRC and utilities as a state-of-the-art tool in predicting the consequences of specific events on nuclear power plant safety.

As shown in the Indian Point Probabilisitic Safety Study (IPPSS)(11) Table 1.6.2.3.8-4, failure data show that the mean value for the probability of failure of a single pipe section for the SWS is of the order of 8.6×10^{-10}/hr.The pipe failure rate in any of 10 critical sections of SWS pipe identified in the IPPSS is 8.6×10^{-9}/hr. Piping failures during plant operations are assumed to be promptly detectable and result in either orderly plant shutdowns or header realignment for repair. Only pipe failures which occur after the start of the initiating event are addressed. The time period of interest is assumed to be 24 hours and so the anticipated failure value for SWS piping during that period is 2.1×10^{-7}. The IPPSS also reports a mean failure value of 1.36×10^{-3} for the SWS pump to start on demand and a mean failure value of 4.68×10^{-5} per hour for the pump to continue to run (1.12×10^{-3} for a 24-hour period). It is thus more likely that three pumps fail to start simultaneously or fail to run from common failure than the occurrence of a pipe break. If common mode failures are discounted, the probability of pipe failure during the critical 24-hour period is one order of magnitude less than the probability of two pumps failing to start on demand, and one order of magnitude less than the probability of two pumps failing to continue to run for that same period.

In addition, an attempt was made to calculate an approximate value of the probability of core damage, utilizing some of the values in the IPPSS for the accident scenario postulated in this evaluation of SWS piping failures(8). The conclusions are that the probability of core damage for the sequence of events postulated has a very low frequency of occurrence and may be considered as an incredible event.

5. Safety Evaluation

Based on the arguments presented in this report with regard to the use of moderate energy piping failure criteria as delineated in SRP Sections 3.6.1 and 3.6.2, NYPA feels that such criteria is applicable and bounding in the evaluation of passive failures in the IP3 SWS piping.

NYPA has concluded that the margins of safety have not been reduced. This conclusion is based on the review of current NRC and industry standards and the Probabilisitic Risk Assessment. The PRA underscores the fact that the probability of failure of the service water piping during the critical 24 hour period after a LOCA is so low that it does not constitute a credible event.

IV. RESULTS OF ANALYSIS

Utilizing the line break criteria as identified in Section III, and developed from SRP 3.6.1 and 3.6.2, the following passive failures were analyzed for the SWS piping during the recirculation phase following a LOCA with control valves TCV-1104 and TCV-1105 in the fully open position:

A.	24" essential header crack	Low River Water Level
B.	20" essential header crack	Low River Water Level
C.	20" non-essential header crack	Low River Water Level
D.	18" essential header crack	Low River Water Level
E.	10" essential header crack	Low River Water Level
F.	10" non-essential header crack	Low River Water Level

The flow distributions calculated for these cracks are within the capability of the SWS pumps.

V. CONCLUSIONS

a. The analysis presented in Section 9.6.1 of the IP-3 FSAR is not valid for a full guillotine break of the 24" essential header during the post-LOCA recirculation phase. However, the analysis of such a break was not included in the NRC's SER as a condition for the issuance of the operating license for IP-3. Additionally, NYPA has concluded that a full guillotine break of the 24" essential header is not a credible event.

b. NYPA has concluded that the original pipe break analysis (2) can be used to predict that the IP-3 SWS will satisfy the cooling water flow requirements of the diesel generator during the recirculation phase following a LOCA even after a full circumferential break (2) of a 10-inch supply line to the diesel generators.

c. NYPA has also concluded that the crack locations and sizes which were postulated under the guidance of SRP Sections 3.6.1 and 3.6.2 would be bounding in terms of the consideration of passive failures as addressed in SECY-77-439 and ANSI/ANS58.9-1981, and are thus applicable to the IP-3 SWS pipe failure analysis.

d. The IP-3 SWS is capable of performing its intended safety function under active and passive failure conditions consistent with the design of the system.

e. The IP-3 FSAR will be revised to reflect the new break criteria and analyses as discussed above.

VI. REFERENCES

1. "Safety Evaluation of the Indian Point Nuclear Generating Unit No. 3", Docket No. 50-286, U.S. Atomic Energy Commission Directorate of Licensing. September 21, 1973.

2. United Engineers & Constructors Inc., "Service Water System Pipe Break Analysis", for the Westinghouse Electric Corp./WEDCO Corp., Indian Point Generating Station Unit No. 3, Consolidated Edison Company of New York, Inc. August 1975.

3. NUREG-0800, "Standard Review Plan", as revised.

4. SECY-77-439, A memo to the Commissioners from Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation, re: Single Failure Criterion.

5. ANSI/ANS58.9-1981, Single Failure Criteria for Light Water Reactor Safety Related Fluid Systems. February 1981.

6. ANSI 31.7 (ANSI N658-1976), Single Failure Criteria for PWR Fluid Systems. June 21, 1976 (replaced by ANSI/ANS58.9-1981).

7. Updated Final Safety Analysis Report for the Indian Point 3 Nuclear Power Plant, as revised.

8. Memorandum (SEAP-MDA-SSI-54-87); SS Iyer to N. Mathur; on "IP3NPP Service Water System". July 31, 1987.

9. "Safety Evaluation of the Indian Point Nuclear Generating Unit No. 3, Supplement No. 1", Docket No. 50-286, U.S. Atomic Energy Commission, Directorate of Licensing. February 21, 1975.

10. "Safety Evaluation of the Indian Point Nuclear Generating Unit No. 3, Supplement No. 2", Docket No. 50-286, U.S. Atomic Energy Commission, Directorate of Licensing. December 12, 1975.

11. Indian Point Probabilistic Safety Study, Amendment 2, December, 1983.