123 Main Street
White Plains, New York 10601
914 681.6240

New York Power Authority

John C. Brons
Executive Vice President
Nuclear Generation

November 13, 1987
IPN-87-053

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Safety System Outage Modification Inspection
(Design) Report 87-013

Reference 1. Letter from Mr. Steven A. Varga to Mr. John C. Brons, dated September 8, 1987, entitled: Safety System Outage Modification Inspection (Design) 50-286/87-013

Reference 1 provided the results and conclusions of the design portion of the Safety System Outage Modification Inspection (SSOMI) of Indian Point 3 (IP-3). The inspection noted weaknesses in the following areas: design, design verification, design criteria, design change control, performance of safety evaluations (10 CFR 50.59), control of FSAR information, and the ability to retrieve design basis information. Enclosure 1 to this letter provides the Authority's response to these findings and identifies the corrective actions taken to date and those that are planned to be implemented. It is organized in the format which corresponds to the subsections presented in Reference 1.

Enclosure 1 also provides a description of the Authority's "Design Control And Configuration Management Program". This program is an ongoing program that was initiated prior to the SSOMI. The implementation of this program is voluntary and consistent with current Authority goals, and is not pursuant to regulatory initiatives (10 CFR 50.109(a)(4)(i)).

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons

John C. Brons
Executive Vice President
Nuclear Generation

IE01
1/1

cc: See next page

cc: Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Joseph D. Neighbors, Senior Project Manager
PWR Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
Mail Stop 428
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia

ENCLOSURE 1

NEW YORK POWER AUTHORITY

INDIAN POINT 3

NOVEMBER 12, 1987

RESPONSE TO SSOMI REPORT 50-286/87-13

PREFACE

This report is prepared in response to SSOMI Report 50-286/87-013 transmitted by NRC letter dated September 8, 1987. The report is organized in a format which corresponds to the subsections presented in the SSOMI report. Some of the NRC findings have been summarized.

SUMMARY

1.1 INTRODUCTION

This report is prepared in response to the NRC's Safety System Outage Modification Inspection (SSOMI) Report 50-286/87-013 concerning design activities for Indian Point 3. The NRC inspection was performed during the cycle 5/6 refueling outage and involved an examination of the detailed design and engineering required to support plant modifications planned during the outage. A significant NRC finding concerned improper consideration of NPSH requirements for flows during postulated pipe ruptures (MOD 86-03-096 SWS). Both the design verification process and nuclear safety evaluation failed to detect this deficiency. This finding prompted the Authority to defer installation of the replacement pumps to a subsequent refueling outage. A description of the design approach applied to the replacement pumps and the Authority's assessment of the causes for this finding are presented below.

The main objective of the service water system (SWS) upgrade (MOD 86-03-096 SWS) was to improve the overall reliability of the service water system. The pump replacement was intended to be like-in-kind (i.e., a different manufacturer, but with-equivalent performance characteristics). Although the pumps were purchased to the same design operating point as the original pumps, it was done without consideration of the maximum flow demand and NPSH requirements for postulated pipe ruptures. Since the original pumps were specified prior to the NRC pipe rupture concern (Ref. IP-3 SER), the application of the original specification requirements as a design basis for the replacement pumps failed to consider the maximum flow demand and corresponding NPSH requirements resulting from conservatively postulated pipe ruptures identified in the FSAR. It was subsequently determined that for postulated pipe ruptures analyzed in accordance with Standard Review Plans 3.6.1 and 3.6.2 and proper balancing, the replacement pumps did have sufficent capacity.

Potential pump runout and inadequate flow distribution due to other postulated failures were also identified as concerns by the SSOMI team. Two potential failures detected by the SSOMI team are the effect of valve positions due to loss of instrument air or an accident signal (see 2.1.1(4)) and the single failure of a pump to start on the non essential header during manual transfer to the recirculation mode following a postulated LOCA (see 2.3.2). As a result of these postulated failures, additional analysis was required. Although they appear to be valid issues, the Authority has no record of these scenarios being analyzed in the original plant design or reviewed in the IP-3 SER. Therefore, the failure to consider these scenarios for the replacement pumps should not be perceived as a weakness in configuration control. These scenarios have been considered in the reanalysis of the existing pumps, and will be considered for the replacement pumps.

The failure to recognize the pipe rupture event as a valid licensing commitment was partly caused by the apparent lack of documentation addressing survivability of the original pumps following certain pipe ruptures (i.e., per FSAR Table 9.6.2, the maximum calculated pump flow of 7827 gpm due to a 24 inch pipe break exceeded the pump capacity curve). It was speculated that the original pipe break evaluation considered cavitation not to be harmful to the pumps for short operational periods.

Therefore, the Authority believes that the basic deficiency involving the SWS pump replacement is an incomplete analysis due to an insufficient assessment of an FSAR commitment concerning conservatively postulated pipe ruptures as identified in Subsection 2.3.1 combined with inadequate consideration of single failures in the original design (see 2.1.1(4) and 2.3.2).

The corrective actions designed to improve the Authority's control of plant modifications are presented in the following subsections.

1.2 CORRECTIVE ACTIONS IN RESPONSE TO SIGNIFICANT WEAKNESS

1.2.1 Configuration Control

Finding

During this inspection, the team observed certain weaknesses in control of plant configuration. The team noted, for example, the use of uncontrolled design inputs in development of plant modifications. In one instance, the team noted that nominal flow rates specified in the FSAR were used to calculate the NPSH of the service water pumps. This practice is a problem for two reasons: (1) the FSAR generally lags the plant configuration by at least six months, therefore it should not be relied upon as a design input document; and; (2) to determine the required pump NPSH, the design engineer needs to know the maximum system flow corresponding to pump runout, rather than the nominal design flows presented in the FSAR.

Corrective Action

The Authority recognizes that improvements are needed in the area of configuration control to assure that the plant remains in conformance with its licensing basis over its lifetime. A description of the Authority's plans to improve the configuration control program is presented in Attachment 1 to this response, "Design Control and Configuration Management Program".

1.2.2 Design Interface Control

Finding

Weaknesses were also noted by the team in the area of design interface control. The procurement specification for the safety-related service water pumps did not include enough information regarding seismic design requirements. Therefore, the seismic qualification report prepared by the vendor considered only a one dimensional earthquake rather than the three dimensional earthquake as committed in the FSAR. In addition, the procurement specification did not specify the committed design code, appropriate allowable stress levels, and minimum modeling requirements necessary to perform a dynamic analysis. This lack of specificity in the procurement specification contributed, in part, to the pump vendor's failure to produce a seismic qualification report that demonstrated that the pumps were seismically qualified.

Corrective Action

The Authority believed that the information specified was sufficient to enable the manufacturer to seismically qualify the pumps. Refer to subsection 2.2 of this response for corrective actions to resolve specific NRC findings pertaining to design interface control. The Authority is strengthening the control of design interface by reviewing and revising appropriate procedures as part of the previously noted Design Control and Configuration Management Program.

1.2.3 Design Verification

Finding

In addition to the lack of specificity in the procurement specification, several design deficiencies and computational errors were found in the seismic qualification report that were not detected by NYPA's technical review, indicative of a weakness in design verification. For example, the team's review of the dynamic analysis revealed that the pump operating speed was between the third and fourth resonant frequencies. The procurement specification required that the pump fundamental frequency be greater than 110 percent of the operating speed. Therefore, during pump startup and shutdown the pump would pass through the fundamental frequency and pump resonance would not be precluded during pump operation. In addition, the licensee failed to verify that the replacement service water pumps would provide design flow to essential components assuming a concurrent LOCA and guillotine failure of a moderate energy line as currently committed to and analyzed in the FSAR. Furthermore during this inspection, the team found one scenario where this commitment could not be met. If the break is postulated in the essential service water header upstream of the pump discharge check valves, service water flow would be lost to 2 of the 3 emergency diesel generators. Two diesel generators are required in the event of a LOCA. This FSAR commitment is more conservative than the guidelines set forth in Standard Review Plan 3.6.1 for moderate energy line cracks. This matter should be resolved between NYPA and NRC as soon as possible, since it relates to the original plant design.

The team also found that the licensee had failed to verify that the worst case system alignment had been selected to determine service water pump NPSH. This worst case pump runout condition is likely to exist following a LOCA, during manual transfer to the recirculation mode, assuming the single failure as the inability to start a service water pump aligned to the non-essential header. In this scenario, only a single service water pump would be operating since the technical specifications permit one of the three non-essential pumps to be inoperable without entering a limiting condition of operation (LCO); i.e., only two of the three pumps are considered operable at the initiation of the accident. Therefore, in the initial stages of recirculation, prior to isolation of turbine building non-essential heat loads, a single service water pump is running against minimum system resistance and consequently, providing a high (runout) flow rate. In addition to the question raised about the operability of the single non-essential service water pump at prolonged runout conditions, the team questioned whether the component cooling water (CCW) heat exchanger, initially aligned to the non-essential header during recirculation receives its design flow rate. The CCW heat exchanger is the heat sink for the containment following a LOCA.

rm/ssomi

Although the licensee has now decided to defer the installation of the new service water pumps, the above systems operability and flow balancing questions are equally valid for the original pumps. The licensee has been requested by the NRC staff to demonstrate service water system operability prior to restart.

Corrective Action

The Authority maintains that pump resonance will be precluded during pump operation. The specification requirement for first critical speed to be 110% of operating speed was based on a rigid bearing support system. The seismic analysis performed by I-R concluded that the bearing support was in fact not rigid. For cases such as these the criteria for acceptability is that sufficient (10% minimum) separation exist between operating speed and resonance frequencies. This was carefully addressed at formal design review meetings between I-R and NYPA and evaluated to be acceptable. Therefore, pump resonance during operation would have been precluded.

The operability and flow balancing was successfully demonstrated for the existing pumps prior to startup by performing a flow test per ENG 281, Rev. 1 (see Section 1.4).

For corrective actions in response to specific NRC findings concerning design verification, see subsection 2.3 of this response.

CONCLUSIONS

The intent of the Authority's Design Control and Configuration Management Program is to strengthen the design process and design verification process. These programs will ensure that (1) adequate controls are in place to maintain plant configuration (2) the plant licensing basis is updated to reflect design changes as required by 10CFR50.71(e) and; (3) the Authority's staff and design agents are aware of the plant's licensing basis. The corrective actions in response to specific NRC findings concerning modifications and the original plant design identified in the SSOMI report 50-286/87-013 are presented in the following subsections.

1.4 CORRECTIVE ACTIONS PRIOR TO RESTART

As stated in SSOMI Report 50-286/87-013, the original pumps were reinstalled and the following corrective actions were completed prior to restart:

1. Procedure ENG-281, Rev. 1 was prepared to establish the throttled positions for flow control valves to assure correct flow distribution in the service water system.

2. A system flow test per ENG-281, Rev. 1 was performed on August 12, 1987 to benchmark the analytical model developed to represent the service water system.

3. The flow control valves were set to their throttled positions and modifications were made to maintain throttled positions. As a temporary modification, the air supply was disconnected from valves FCV-1176 and FCV-1176A. Valve 1176 was set in the closed position and FCV-1176A was set in the throttled position determined per ENG-281, Rev. 1.

4. A system flow distribution test was performed per ENG-281, Rev. 1 to simulate system alignment during the injection phase of LOCA with concurrent failure of the nonsafety-related air supply. This test established the throttled positions and confirmed that minimum required flow to safety related equipment was achieved.

5. Procedure FS-1.3, Rev. 1 "Transfer to Cold Leg Recirculation" was revised to require that turbine building heat loads be isolated prior to starting a nonessential service water pump.

6. Operating procedure SOP-RW-6, Fan Cooler Unit Flow, has been eliminated. The provisions of the procedure, including valve positions, have been established in procedure ENG-281.

7. Alarm response procedure ARP-5 was revised to correct the setpoint for the containment fan cooling water low flow alarm to be consistent with revision 5 to NSE-81-03-055 FCU.

2.1 GENERIC CONTROL OF DESIGN INPUT

Finding (1)

The FSAR was incorrectly used as a source of design input instead of appropriate design documents. An example of an external agent's calculation which applied FSAR data as design input is described in Item 2.1.1.

Response (1)

The Authority recognizes that the FSAR should only be used as a reference source subject to verification with appropriate design documents. Design procedure DAP 3.18, Rev. 6 Baseline Design Information, has been modified accordingly and a description of the Authority's plan to review and revise other appropriate procedures as necessary to improve the control of design input is presented in Attachment 1 to this response. The response to Item 2.1.1 (1) describes the corrective action used to resolve the example cited.

Finding (2)

Sources of design input have not been identified in a design calculation. The examples cited for this finding are identified in Items 2.1.1 and 2.1.2 which are concerned with calculations performed by external organizations.

Response (2)

The Authority's existing engineering department calculation control procedure requires that design inputs including references for equations and procedures be verified. Therefore, controls do exist for control of design input for documents originated by NYPA engineering department. This procedure will be included as part of the Design Control and Configuration Management Program.

Other design organizations are required to have similar control procedures. The Authority's existing procedure for review, comment and acceptance of foreign technical documents requires that documents classified for "Review and Acceptance" be reviewed for compliance with technical specifications and concurrence with the "Design Concept". This procedure will be included as part of the Design Control and Configuration Management Program.

A description of the Authority's plans to review and revise appropriate procedures to improve the control of design input is presented in Attachment 1 to this response. To ensure correct implementation of these procedures appropriate instructions will be provided to individuals who prepare and review design documents. The responses to Items 2.1.1 and 2.1.2 describe the corrective actions used to resolve the specific examples cited.

Finding (3)

Assumptions have not been identified or justified in a design calculation. The examples cited for this finding are identified in items 2.1.1 and 2.1.2 which are concerned with calculations performed by external organizations.

Response (3)

The Authority's existing engineering department calculation control procedure requires that design assumptions be adequately described and verified. Therefore, controls do exist for control of design input for documents originated by the NYPA engineering department. This procedure will be included as part of the Design Control and Configuration Management Program.

Other design organizations are required to have similar control procedures. The Authority's existing engineering department procedure for review, comment and acceptance of foreign technical documents requires that documents classified "For Review and Acceptance" be reviewed for compliance with technical specifications and concurrence with the "Design Concept". This procedure will be included as part of the Design Control and Configuration Management Program.

A description of the Authority's plans to review and revise appropriate procedures to improve the control of design input is presented in Attachment 1 to this response. To ensure correct implementation of these procedures appropriate instructions will be provided to individuals who prepare and review design documents. The responses to Items 2.1.1 and 2.1.2 describe the corrective actions used to resolve the specific examples cited.

Finding (4)

Preliminary proposal information was used in a design calculation. An example of this finding is cited in Item 2.1.1 (5) which concerns a calculation from an external design organization.

Response (4)

The response to Item 2.1.1 (5) describes the corrective action to resolve the specific example cited. The Authority does not believe that this isolated finding is indicative of a generic problem.

The Authority's existing engineering department calculation control procedure requires that calculations be identified as preliminary or final. Therefore, a calculation based on proposal information would normally be identified as preliminary. Similarly, other design organizations are required to follow the same approach. The review of selected documents prepared by external design organizations using the procedure for review, comment and acceptance of foreign technical documents is intended to confirm the correct implementation of procedures.

2.1.1 SWS Pump Performance Calculation 6604-266-2-SW-003

Finding (1)

The calculation for the replacement SWS pumps used required design flows identified in the FSAR rather than pump operating flows. Higher flows require more service water pump NPSH to prevent cavitation and potential pump degradation or failure.

Response (1)

The Authority concurs that actual system operating flows may be higher than design flows and will revise the referenced calculation accordingly.

It should be noted that the external agent's SWS Pump performance calculation 6604-266-2-SW-003 was not intended, as a system flow balance analysis. This calculation was prepared as a preliminary check of the selected design operating point.

Finding (2)

A flow rate of 1400 gpm to the FCU's was used in lieu of test values of 1450 gpm to 1500 gpm.

Response (2)

The deviation between the design flow rates and previously tested flow rates is small such that it has minimal impact on NPSH requirements. A flow rate of 1400 gpm corresponds to the minimum analyzed flow rate to each FCU to achieve the required heat removal capacity (Ref. NSE-81-03-055 FCU Rev. 5). The NPSH calculation used to support the replacement pumps will consider the maximum flow conditions based on a system flow balance analysis including bench mark performance testing.

Finding (3)

A flow rate of 1350 gpm to the diesel generators was used in lieu of maximum flow rate thru valves FCV-1176 and FCV-1176A of 1500 gpm.

Response (3)

A flow rate of 1350 gpm was used on the basis of normal flow thru the flow control valves. The Authority agrees that the analysis should have considered the wide open position of the valves resulting from a safety injection signal. The revised calculation for the replacement pumps will reflect the flow rate to the diesel generators on the basis of a system flow balance analysis including bench mark performance testing.

Finding (4)

The failure position of non-safety related valves was not considered in the calculation.

Response (4)

The Authority does not have evidence to indicate that loss of instrument air was postulated in the original design calculations. However, the revised analysis conservatively reflects the flow rates thru non-safety related valves with consideration to their failure position resulting from loss of nonsafety related instrument air. Performance testing has been used to benchmark the analytical model.

Finding (5)

Data for required NPSH was based on a proposal.

Response (5)

At the time the calculation was performed, only proposal data was available. This calculation should have been considered preliminary until the information could be verified with the purchased pump data. In the revised calculation, NPSH required will be compared with the manufacturer's guaranteed data.

2.1.2 Replacement of Inverters Calc. 6604-0221-3-BR-02 (MOD 85-03-058 EL)

Comment (1)

Misleading references and inappropriate assumptions were used in a battery voltage sensitivity calculation to support the replacement of original safety-related inverters 31 and 32.

During a previous outage, modification MOD 85-03-058EL replaced the original 7.5 kVA safety-related inverters 31 and 32 with larger 25 kVA inverters. Calculation 6604-0221-3-BR-02, 125 Volt DC Load Study, Rev. 0, 8/30/85, was prepared to support this modification. This calculation was initially presented by NYPA as being the calculation which determined the adequacy of the batteries to supply the 25 kVA inverters. From this perspective the team considered the battery to be inadequate. It was later learned that this calculation was not a battery sizing calculation, but its purpose was to determine the maximum output of 25 kVA inverters based on the current battery capacity. It was determined that the batteries could not sustain a 25 kVA inverter output and that the vital ac loads sustainable was approximately 9 kVA, i.e., the inverters were oversized. NYPA subsequently determined that the vital loads supplied by the 25 kVA inverter were less than the maximum sustainable battery load (9 kVA) and therefore, the battery capacity was adequate.

Response (1)

As stated in the above comment, during a previous outage, modification MOD 85-03-058EL replaced the original 7.5 kVA safety-related inverters 31 and 32 with larger 25 kVA inverters. Calculation 6604-0221-3-BR-02, 125 Volt DC Load Study, Rev. 0, 8/30/85, was prepared to support this modification. Its purpose was to determine the maximum output of 25 kVA inverters based on the current battery capacity. This was done since an oversized inverter was provided and the maximum inverter load was to be determined such that the battery would not be overloaded. The calculation was conservative in that it over estimated the load on the battery.

At the time the inspection team was reviewing modification MOD 85-03-058EL which replaced the original 7.5 kVA inverters, the team's request for calculations was assumed to be for the calculations relating to the modification under review and not some other modification completed several years ago.

NYPA did not present calculation 6604-0221-3-BR-02 as the calculation that determined the adequacy of the batteries that were procured and installed 3-4 years before 1985. This calculation was presented as the one that was associated with the modification under review. As stated on the cover page (Problem Statement) of the above calculation, its purpose was to "Perform a load study to determine the adequacy of the 125VDC Bus #31 and Bus #32 with respect to the ability of the (existing) batteries to supply the load and the ability of the battery chargers to recharge the batteries while carrying normal load." This calculation was actually a sensitivity study to determine the maximum load that the new (25kVA)* inverters could sustain without exceeding the battery capacity using conservative loads.

As indicated in the battery calculation, various inverter loads of 100 Amps, 150 Amps and 250 Amps were assumed. The purpose of this was to establish the maximum inverter load that would still maintain compliance with the battery specification/size and FSAR requirements, i.e., minimum cell voltage (1.81V) after discharge and recharging time of 15 hours in accordance with the FSAR.

* The new inverters were 25kVA due to the fact that lower capacity IEEE qualified inverters were not available at the time of this modification.

2.1.2 Comment (2)

6604-0221-3-BR-02 contained the following incorrect references and inappropriate assumptions.

The battery voltage sensitivity analysis contained implicit assumptions concerning battery age, minimum electrolyte temperature and maintenance margin which were not conservative with respect to the calculation's objective and inconsistent with surveillance procedures. First, the battery capacity was implicitly assumed to be at 100 percent of the manufacturer's rating, even though a capacity as low as 80 percent is acceptable per the refueling outage Battery Load Test procedure. Second, the minimum electrolyte temperature was implicitly assumed to be 77 degrees F even though monthly and quarterly battery surveillance procedures specify an alert point at 65 degrees F and define the battery inoperable at 60 degrees F. Third, the average specific gravity was implicitly assumed to be that of a fully charged battery at 1.215 even though the quarterly battery surveillance procedure specifies an alert point at 1.205 and defines the battery inoperable at less than 1.195 specific gravity.

Response (2)

With regard to the comment on inappropriate assumptions used in performing calculation 6604-0221-3-BR-02, it should be noted that this calculation was not intended to provide specific information on margins and battery derating. Such information was provided in the 1981 battery calculation which was performed in accordance with the applicable IEEE standards.

The subject calculation, however, will be revised to clarify the assumptions and references.

2.1.2 Comment (3)

The sources of bus data and load profile data were identified as references attached to the calculation. However, the data used in the calculation for the loads could not be deduced from the attached references.

Response (3)

As noted in the comment, the sources for the load data were referenced in the calculation. These references and other related information were included in the battery sizing file and were provided to the inspection team.

2.1.2 Comment (4)

The team concluded that, in spite of the misleading 1985 calculation, the existing batteries have sufficient capacity for the identified loads. However, NYPA should develop suitable controls to assure that additional loads added to the vital ac bus do not exceed the load capacity of the batteries.

rm/ssomi

Response (4)

As previously noted, the 1985 calculation was not used to determine the battery capacity. The battery capacity was established by the 1981 calculation. The Authority has measures to ensure that additional loads added to the inverter as well as to the dc panels will not exceed the capacity of the batteries. Such measures will also be formalized in procedures.

2.2 GENERIC DESIGN INTERFACE CONTROL

Finding

Weaknesses exist in the implementation of controls associated with external organizations. The specific examples cited in support of this finding are identified in Items 2.2.1 and 2.2.2.

Response

For the specific example cited, the Authority did, according to engineering, procedures conduct a review of the seismic qualification report submitted by an external design organization. However, as noted by the inspection team, the vendor's qualification report contained errors which were not detected by the Authority's review and therefore, reanalysis was initiated.

Appropriate design related licensing commitments will be included in procurement specifications where appropriate. This requirement is being incorporated into appropriate engineering procedures.

As previously indicated, the Authority is strengthening the control of the design interface control by reviewing and revising appropriate procedures as part of the Design Control and Configuration Management Program.

2.2.1 SWS Pump Replacement Seismic Qualification Report and Specification MDA-SWP-84-0148-A

Finding

Procurement specification MDA-SWP-84-0148-A was developed to specify the design, fabrication, testing, preparation, shipping and delivery requirements for seven service water pumps, six of which were planned to be installed as part of modification MOD 86-03-096 SWS. Several design related licensing commitments were not specified in the procurement specification and were apparently not transmitted to the pump vendor in a controlled manner. These commitments included (1) a reference to the design code of record; (2) the basic seismic analysis method to be followed including the three-dimensional earthquakes (OBE and SSE), percent critical damping, method of modal combination, and pump operability requirements; (3) appropriate allowable stress levels and; (4) minimum modeling requirements.

Response

The Authority offers the following comments in regard to the above finding:

- The procurement specification did reference the design code of record in section 2.0. Although Section 3.3 "Seismic Requirements" did not reiterate the subject reference code, the Authority does not consider this a deficiency.

- Although the Authority did not specifically state that the "Equivalent Static Load Method" be used, Section 3.3 of the procurement specification provided the required 3 directional equivalent static loadings. In providing this data, it was intended that an equivalent static method for seismic qualification be used. The specified seismic requirements for the replacement pumps were consistent with the seismic analysis for the original pumps and satisfied the licensing commitment for a static analysis. Subsequently, a dynamic analysis was used to reduce the conservative static seismic loads. This dynamic analysis method was considered an adequate alternative to the simplified static approach.

- Since Section 2.0 of the procurement specification defines design codes to be used, and these codes provide "Allowable Stress Levels", we do not consider this as a deficiency.

However, as a result of the inspection, these four design considerations pertaining to a more rigorous dynamic analysis were incorporated into the Ingersoll-Rand seismic pump analysis and report. Consequently, Ingersoll-Rand (I-R) Qualification Report No. TR-8605 "Structural Integrity of the 26 APK-1 Service Water Pump", Rev. 1, 6/19/87, contains the details of these design-related licensing commitments. This report was provided to the inspection team for review.

Comment

The capability of the support structure to withstand operating and seismic loads is unresolved.

Response

Subsequent to the May 1987 audit, the support structure was reevaluated based on the revised I-R seismic analysis report (Rev. 1). Maximum reactions due to pump operations and seismic loading were computed at the three support locations, namely, the intake structure deck (El. +15'-0), seismic restraint platform (El. +6-0) and suction head pin cup support (El. -10'2 1/2").

The intake structure deck, the pump sole plate and the four anchor bolts (connecting the pump sole plates to the concrete slab) were evaluated for the new reactions and all stresses are within the allowable limits established in the FSAR.

At the seismic restraint platform, the original analysis was performed using conservative boundary conditions which resulted in higher pump reactions than those obtained from I-R seismic analysis report (Rev. 1). Based on this, the present design is adequate.

At the suction head pin cup support, the structure was reanalysed using the revised results from the I-R seismic analysis report (Rev. 1) and stress for all structural framing elements are within the allowable limits.

Based on the above, it is concluded that the question of the capability of the support structure to withstand operating and seismic loads is resolved.

2.2.2 SWS Pump Replacement Seismic Qualification Report TR-8605, Rev. 1

Finding (1)

Seismic operability, as a minimum, is normally evaluated by comparing the available clearance between the rotating component and the pump casing to the lateral deflection of the rotating component. While I-R did discuss pump operation as it affects wear and as a consequence to service life, the net clearance between the rotating impeller and the pump casing was not evaluated.

Response (1)

The dynamic analysis of the replacement pumps is more rigorous than the static analysis performed on the original pumps. During the NRC audit, after Rev. 1 of the Ingersoll-Rand pump seismic qualification report was reviewed, NYPA notified I-R by a letter (IPO-87-36) that they will have to evaluate seismic operability of the pumps in greater detail. In particular, the Authority requested that I-R expand Rev. 1 of the seismic report to include seismic operability by comparing the available clearance between the rotating component and the pump casing to the lateral deflection of the rotating component. The details of this comparison will be incorporated in a future revision of the I-R seismic qualification report prior to installation of the new pumps.

Finding (2)

The NRC team reviewed the seismic analysis results to assess if they appeared to be reasonable and consistent with expected behavior. While the mathematical modeling and natural frequency responses appeared to be reasonable, the square root of the sum of the squares (SRSS) results were inconsistent with expected behavior. Upon further review by the team, it was noted that the element data (forces and stresses) appeared to be incorrectly summed. The strain in the two perpendicular springs that modeled the spiders varied by an unrealistically large amount when, in fact, the strain values should have been essentially the same.

Response (2)

Since this computer program has been verified and is used industry wide, NYPA would not normally review the analysis to this level of detail. On July 15, 1987, at NYPA's request, Swanson Analysis Systems, Inc. (SASI), owner of the verified ANSYS program, conducted an investigation on the subject of SRSS results for spring element STIF14 (see Attachment 4). SASI reported the following findings of the investigation to NYPA:

1. This is the first time this type of error has been reported to SASI.

2. The SRSS operation only works with the Level 1 and 2 data. The spring force is Level 1 data and it can be processed by SRSS. The stretch of the spring element, however, is Level 3 data which cannot be processed by SRSS.

3. Normally Level 3 data can be obtained with the WRITE operation which not only causes the latest Post 27 results to be copied to outfile, but also automatically calculates the correct Level 3 data from the current Level 2 data. In the case of spring element, there is no Level 2 data for this type of element, therefore no calculation was performed to get the appropriate Level 3 data. Subsequently, the incorrect Level 3 data, i.e., the stretch of the spring element, was printed in the Post 27 output.

4. The ANSYS program has the following error: The stretch of the spring element (STIF 14), which was not processed by either SRSS or WRITE operation, should not be allowed to be printed in Post 27. In the current 4.2 revision of ANSYS, there is no way to avoid this error.

5. SASI will correct this error (stretch of spring element) in the future revision of ANSYS.

In light of this finding (program error), NYPA has notified I-R on 7/27/87 to take the following actions:

a) Ignore the SRSS results on stretch of the spring element. Instead, perform a hand calculation of stretch = spring force/rate for all spring elements. The spring force values are obtained from the SRSS operation. (Note: The incorrect strain values of spring elements were not utilized by I-R in the previous reports).

b) Expand the current seismic report to include seismic operability by comparing the available clearance between the rotating component and the pump casing to the lateral deflection of the rotating component.

Finding (3)

An additional inconsistency noted was the magnitude of the vertical force located at the mounting plate which supports the entire column assembly, pump casing, and end bell. The value determined from the analysis was 20,471 pounds. The entire weight of the pump (including water) is approximately 12,200 pounds. Therefore, a net vertical acceleration of approximately 2.68 g was developed. However, since the pump is assumed to be rigid in the vertical direction and the applied acceleration is 0.4 g, the magnitude of the force is inconsistent with expected results.

Response (3)

The Authority does not agree that there is an inconsistency in the vertical force. Attachment 3 demonstrates that the computer model used by I-R is correct and the results are in reasonable agreement with the expected behavior.

2.3 GENERIC DESIGN VERIFICATION

Finding

A weakness was found in the implementation of the design verification process which suggests a need for greater attention to detail. Errors included: (1) Failure to meet licensing commitments; (2) Failure to ensure that an appropriate design method was used; (3) computational errors and; (4) failure to ensure that specified parts and equipment are suitable for the required service. Specific examples of this finding are presented in the following subsections.

Response

The Authority does have procedures which control design verification. As previously noted, the Authority is strengthening the control of design verification through the Design Control and Configuration Management Program. A description of the Authority's plans to improve design verification is presented in Attachment 1 to this response.

2.3.1 SWS Pump Performance Following Postulated Passive Failure

Finding

The replacement pumps have not been evaluated to FSAR commitments for postulated line breaks.

Response

The Authority concurs that the replacement pumps have not been evaluated for all of the breaks analyzed in the FSAR and therefore, installation of the replacement pumps has been deferred until a later outage.

Finding

The existing essential service water system does not comply with the FSAR for several postulated breaks.

Response

The original concern in the IP-3 SER was the adequacy of cooling of the diesel generators during and after a postulated 10" service water line break in the diesel generator cooling loop. This is considered as the licensing basis for long term passive failures during the recirculation phase for IP-3. The existing system satisfies this licensing basis and commitment.

The error in the original model is associated with two other breaks in the 24" header where back flow would be precluded because of the presence of check valves in the system. The existing system can handle all postulated breaks in the system and still meet its cooling requirements except for the two 24" breaks mentioned above.

As described in Attachment 2, the Authority considers the postulated circumferential breaks in the seismically supported service water system to be overly conservative. Applying current NRC criteria (Standard Review Plans 3.6.1 and 3.6.2) for postulating cracks in moderate energy piping to the service water system results in acceptable consequences.

Also, a postulated break in the nonseismically supported service water piping to the turbine generator lube oil coolers, as an initiating event, has been evaluated with acceptable consequences.

The Authority is in the process of preparing a submittal to the NRC in order to revise the current methodology for postulated SWS breaks. Once NRC approval is granted the FSAR will be revised accordingly.

2.3.2 SWS Pump Performance During Long Term Recirculation Combined With a Single Active Failure

Finding

The replacement safety related, non essential SW pumps have not been evaluated for potential run out conditions resulting from a postulated single active failure of one pump to start on the non essential header during manual transfer to the recirculation mode following a postulated LOCA.

Response

The Authority has revised procedure ES 1.3, Transfer to Cold Leg Recirculation, to isolate FCV-1111 and FCV-1112 prior to starting the first non-essential pump in the eight step sequence. This procedure change will preclude the identified challenge to the non-essential service water pumps.

Finding

The existing pumps have the same concern as identified above.

Response

As indicated above and in Subsection 1.4, Item 5, revised procedure ES 1.3 resolves this concern for the existing pumps.

2.3.3 Replacement of Inverters Calc. 6604-0221-3-BR-02, MOD 85-03-058 EL

Comment (1)

A verified calculation had errors which should have been detected during the checking and verification process (2.3.3). . . . This calculation (6604-0221-3-BR-02) had been checked, but it contained the following errors:

It appeared that the load profile developed in this calculation may have left out 3 out of 4 buses amounting to a potential error of about 25 percent of the load.

Response (1)

No error was involved in this area. The calculation correctly listed all loads based on conservative assumptions. The tabulation given on sheet 3 of 38 of the calculation 6604-0221-3-BR-02 provided the total load on the 125VDC Power Panel 31 and also listed the loads of the individual 125VDC Distribution Panels that are fed from the 125VDC Power Panel 31. Therefore, since Power Panel 31 feeds the Distribution Panels, the total load of the Power Panel includes the individual loads of the Distribution Panels as well as the inverter load.

The above explanation was provided to the inspection team and the Authority believes the misunderstanding was resolved during the inspection.

2.3.3 Comment (2)

To demonstrate that the battery chargers have sufficient capacity to carry dc loads and recharge the battery within 15 hours per FSAR commitment, a calculation [6604-0221-3-BR-02] was prepared. . . . However, the results for maximum output available in KVA of the new inverters is overstated because values used in calculating the output were not conservative (i.e., to determine inverter output at a reduced dc bus voltage and at an inverter efficiency corresponding to that output).

An equation presented as Note 1 in the Summary/Conclusion section of the calculation uses a value for available dc bus voltage of 125 volts when the voltage will be much lower (approximately 10 percent) due to the inverter efficiency corresponding to a fully loaded inverter. Since the inverter will not be fully loaded, the efficiency will drop by 10 percent to 15 percent corresponding to approximately its half loaded condition.

Response (2)

The equation referenced in the above comment is clarified in the note associated with the equation which indicates that it is based on ac supply being available. During the battery recharging period ac power is available to the charger. In this condition the charger output will be at least 125 volts.

As indicated in the above comment the inverter efficiency at reduced load is lower than that at 100% load.

The efficiency of the subject inverters at 100% load is 86% and at 50% load the efficiency is 84%. As a result, the assumption of 80% efficiency at reduced inverter load for the subject calculation is also conservative.

2.3.3. Comment (3)

Voltage assessments at intermediate steps on the load profile incorrectly used a value of available Ampere-hours positive plate based upon a permissible cell voltage of 1.75 volts. This cell voltage corresponds to the original 60 cell battery and not the existing 58 cell battery which has a permissible cell voltage of 1.81 volts. The final step in the load profile used a slightly different method which avoided this error.

Response (3)

The intent of the calculation was to establish the final cell voltage after discharge based on increasing inverter loads and to establish that the battery recharge time did not exceed the 15 hours specified in the FSAR. In intermediate steps of the calculation, 1.75 volts per cell was referenced instead of the permissible cell voltage of 1.81 volts. The 1.75 volts per cell was only a reference point to establish the fact that after the discharge period, the cell voltage will be in excess of 1.75 volts per cell.

The summary and conclusions of the calculation provide the tabulation of cell voltage and battery voltage. It should be noted that the cell voltage after discharge exceeded the permissible 1.81 volts per cell in all cases. Therefore, reference to 1.75 volts per cell as a reference point within the body of the calculation, although misleading, has no significance on the results of the calculation.

2.3.4 MOV Overload Heater Protection

Finding

It appears that inadequate motor operated valve electrical protection is a generic problem at Indian Point 3 and could result in undetected damage to safety related valve motors.

Response

MOV overload protection is currently being reviewed for all safety related valves. While the current size of the overload protection is, in some cases, greater than that typically specified by the manufacturer, appropriate size for MOV overload protection must consider factors such as degraded grid voltage and valve operator resistance. These factors demand additional operating current which must be accommodated by the overload protection. In any event, the function of the valve (ie, its operation) is primary and motor protection is considered secondary.

Overload protection will be revised, if warranted, based on the results of the reevaluation of all safety related valves.

?.3.5 Pipe Supports MOD 86-03-009 RCS

Finding

Calculation No. 840223-CA is a pipe support calculation for modification MOD 86-03-009 RCS. Several of the support base plates were evaluated consistent with the methodology prescribed in the licensee's commitment to IE Bulletin 79-02. However, the base plate evaluation for 8 supports (RC-R-343-4A-H, RC-R-343-4B-H, RC-R-343-105-H, RC-R-343-106-H, RC-R-342-106-H, RC-R-70-204-R, RC-R,-70-205-R and RC-R-70-206-R) were performed with a less conservative approach. If the methodology of IE Bulletin 79-02 is applied to the supports in question, then some of the anchor bolts are inadequate. Although the use of the alternate approach may be technically acceptable, approval for relaxation of a licensing commitment should have been obtained from NRC prior to performing the modification.

Response

During the NRC audit, the 8 supports mentioned above were recalculated using the approach outlined in the response to NRC I&E Bulletin 79-02 as described in a letter from P. J. Early (NYPA) to B. H. Grier, OIE, NRC, Region I, (IPN-79-1979), dated July 6, 1979. As a result of this recalculation, it was determined that all of the anchor bolts for the 8 support base plates were adequate as designed. Therefore, the subject pipe supports can be considered adequate as designed and in compliance with I&E Bulletin 79-02 requirements.

In addition, Attachment 5 explains the intent of the current base plate qualification approach.

Finding

The use of Code case N-411 was generically endorsed for plants conforming to the 1984 edition of the ASME Section III code. However, the code of record is ANSI B31.1, 1967 for the piping in question at Indian Point Unit 3. The piping was originally designed for 1/2 percent critical damping. Code case N-411 specifies a variable damping; varying from 5 percent damping for frequencies less than or equal to 10 Hertz, then linearly decreasing to 2 percent damping for 20 Hertz and 2 percent damping above 20 Hertz. The use of Code case N-411 results in substantially higher damping than the original plant licensing commitment and as a result the piping system is designed to withstand lower seismic accelerations. Prior permission for relaxation of this licensing commitment should have been obtained.

Response

Since Code Case N-411 had been generically endorsed for plants conforming to the 1984 edition of the ASME code, it was assumed that it could be applied to a power plant conforming to the 1967 edition of the ANSI B31.1 power piping code. However, during the IP-3 SSOMI audit, the NRC indicated that prior permission from the NRC was required for use of this code case for ANSI B31.1 power plants. As a result, NYPA reanalyzed the pressurizer piping using the original design commitment of 1/2 percent critical damping and redesigned the pressurizer system modification to incorporate the higher loadings, prior to installation.

In the future, NYPA will obtain prior approval from the NRC, as required, before using ASME code cases in any analysis or design work. This guidance will be provided as part of the Configuration Management Program.

2.3.6 Heat Tracing For MOD 86-03-096 SWS

Finding

The basis for the selection or adequacy of the freeze protection proposed on the new screen wash water lines could not be defined. Installation drawing 860726-FE-531, Rev. 2 contained conflicting information regarding the position of the heat tracing cable and failed to identify how much heat tracing tape was required per foot of pipe.

Response

Heat tracing is not safety related for this application. The heat tracing had been selected from drawings for another modification that had been previously performed in this same area of the intake structure and formal design calculations were not performed.

A calculation has been subsequently performed to demonstrate the acceptability of the heat tracing for this modification. In the future, calculations will be performed when needed for all freeze protection requirements.

2.4 GENERIC DESIGN DOCUMENT CONTROL

Finding

Document control procedures permit issuing drawings with missing information without a tracking mechanism to ensure that the work is completed correctly. An example of this finding was identified on MOD 86-03-018 SWS.

Response

The Authority does not agree with the finding. The design document controls imposed are adequate. Drawings may be issued for a modification with holds and/or missing information. The modification cannot be completed and closed out until all holds and missing information are resolved by change notices or drawing revisions. Therefore, an independent tracking mechanism is not considered to be necessary.

2.4 SERVICE WATER FLOW INDICATION MOD 86-03-018 SWS

Finding

A drawing was incorrectly identified as non nuclear safety related and detailed material requirements were missing.

Response

This drawing was part of a modification which involved both safety related and non safety related items. All mechanical, structural, and respective Bill of Material (BOM) drawings for this modification were designated as "Nuclear Safety Related". The electrical components, not directly interfacing with the SW system, are considered non safety related. Therefore, the electrical drawing and BOM associated with the non safety related components is appropriately designated as "Non Nuclear Safety Related".

As indicated in the SSOMI report, the lack of detailed material requirements did not result in an inadequate installation. Detailed material requirements in the electrical BOM are not necessary when the description in the BOM identifies the vendor catalog number. (ex. Thomas & Betts Cat. No. 5252 Flex Conn. or Buchanan Cat. No. NQB106 Terminal Block)

2.5 GENERIC DESIGN CHANGE CONTROL AND MODIFICATION CLOSEOUT

Finding

A weakness was found in the closeout of modifications in that not all affected documents, procedures or the controlled list were being revised. Examples of this weakness were identified in MODS 81-03-055 FCU and 80-3-055 RPI.

Response

To assure that all affected documents for a modification are identified and revised, the Authority is developing procedural changes to require early identification of affected documents and an improved tracking system.

2.5.1 FCU Cooling Coil Replacement Modification MOD 81-03-055 FCU

Finding

The Loop Instrument Calibration Document, F-1124 dated May 7, 1973 was not revised to reflect the reduced flow rate to the FCUs.

Response

The Loop Instrument Calibration Document F-1124 has been revised to reflect the reduced flow rate to the FCUs per MOD 81-03-055 FCU.

Finding

The alarm response procedure ARP-5, Rev. 7 dated November 7, 1985 for safeguards panel SBF-2 was not revised to reflect the reduced cooling water low flow set point.

Response

Procedure ARP-5, has been revised to reflect the reduced cooling water low flow set point per MOD 81-03-055 FCU.

Finding

Standard Operating Procedure SOP-RW-6 was incorrectly revised to reflect a reduced flow rate to the FCUs on the basis of an incomplete engineering evaluation.

Response

Operating procedure SOP-RW-6, Fan Cooler Unit Flow, has been eliminated. The provisions of this procedure, including future manipulations of the Fan Cooler Unit Service Water outlet valves, will be performed in accordance with a performance test which will ensure the proper Service Water flow balance between the Fan Cooler Units.

Finding

The FSAR was not revised to reflect the results of the nuclear safety evaluation. (i.e., heat removal capacity, flow rate)

Response

The failure to update the FSAR for the FCU modification was an inadvertent omission. An FSAR change request is being prepared to reflect revision 5 of NSE 81-03-055 FCU and will be included in the next annual update. Refer to Subsection 2.6.2 for a summary of this revised NSE.

As previously indicated, to assure that all affected documents for a modification, including the FSAR are identified and revised, the Authority is developing procedural changes to require early identification of affected documents and an improved tracking system.

2.5.2 RPIS Power Supply MOD 80-3-055 RPI

Finding

Following completion of modification 80-3-055 RPI, the dc single-line diagram 9321-F-30083-27 was not revised to reflect the removal of the 5 kw Rod Position Control Rack primary inverter.

Response

Drawing 9321-F30083 has been "red-lined" to show the removal of the Rod Position Inverter. The drawing is being updated to correct this error.

2.6 SAFETY EVALUATION AND REPORTABILITY ANALYSES

Finding

Weaknesses were found in implementation of safety evaluation requirements.

Response

The Authority does not believe that the example cited is indicitive of a generic weakness in the implementation of safety evaluation requirements. As stated in the SSOMI report, the Authority's requirements for performing safety evaluations and the specified technical content are adequate.

The following responses clarify the basis for the statements made in the safety evaluations reviewed by the SSOMI team.

2.6.1 SWS Upgrade NSE 86-03-096 SWS

Finding

The NSE concluded that the service water pump and its mounting were evaluated for integrity during a seismic event so that the structural integrity of the service water pumps and motors is maintained. However, the seismic qualification report used as the basis for the NSE conclusion contained significant errors and did not confirm that the service water pumps could withstand a seismic event and remain operable (see sub sections 2.2.1 and 2.2.2).

Response

The safety evaluation was based on an assumed satisfactory seismic report. An error in the report does not indicate a weakness in implementation of safety evaluations. Based on requests from NYPA, Ingersoll-Rand has revised and corrected their service water pump seismic analysis report, TR-8605. Revision 1 to this report was provided to the inspection team for review and comments were returned to Ingersoll-Rand. All comments will be resolved prior to installation of the replacement pumps. (Also, see NYPA response to Sections 2.2.1 and 2.2.2.)

2.6.2 Safety Evaluations

NSE 81-03-055 FCU

Finding

NSE did not address discrepancy in heat removal rates applied in the current analysis, FSAR and Technical Specification.

Response

Both the FSAR and Section 5 of the Technical Specifications indicate that the fan cooler units (FCUs) are designed to remove 76.32×10^6 BTU/HR/FCU. The FCU coils, which were replaced by MOD 81-03-055 also were designed to achieve the original design heat removal rate of 76.32×10^6 BTU/HR/FCU. The following design parameters apply to the original and replacement fan coils.

	Original FCU	Replacement Coils
SWS Flow	2000 GPM/FCU	1810 GPM/FCU
SWS Temp.	85°F	85°F
% Plugged Tubes	0%	10%
Tube Side Fouling Factor	.001	.00135
Heat Removal Rate	76.32×10^6 BTU/HR/FCU	76.32×10^6 BTU/HR/FCU

On the basis of a Westinghouse analysis prepared to evaluate design parameters based on operational conditions at a core power level of 3216 MWT, it is concluded that the FCUs with replacement coils are capable of limiting the peak containment pressure to 40.6 psig for the following service water system parameters:

SWS Flow	1400 GPM/FCU
SWS Temp.	85°F
% Plugged Tubes	1.1%
Tube Side Fouling Factor	0.0015

This conclusion is presented in revision 5 of NSE 81-03-055 FCU which resolves the stated descrepencies in heat removal rates.

Finding

A power level inconsistent with that stated in the FSAR was used in the NSE.

Response

The NSE was performed to demonstrate the acceptability of the new FCU coils. The licensed power level of 3025 MWt was considered in revision 4 of the NSE to show the amount of conservatism that was built into the original design of the FCUs in order to account for tube plugging, fouling factors and reduced service water flow. To demonstrate that a lower FCU heat removal rate was compensated for by the fact that the original analysis conservatively assumed a core power of 3216 MWT, the Authority had Westinghouse perform an evaluation using 3025 MWT. This evaluation was provided in Westinghouse's letter of August 7, 1981 to NYPA. This evaluation showed that based on a core power of 3025 MWT a heat removal rate of 49.0×10^6 BTU/HR/FCU was acceptable.

The acceptability of the 49×10^6 BTU/HR/FCU heat removal capability was established by taking credit for the fact that the original LOCA Mass and Energy (M&E) Release Analysis conservatively employed a core power level of 3216 MWt, which was 6% greater than the licensed core power of 3025 MWT. The approach was as follows:

1) The 6% additional core power resulted in a conservatively high M&E release. Re-analysis with the licensed core power would result in a significant reduction in the containment pressure response due to the correspondingly lower releases.

2) The reduction in net energy released to the containment, if the actual licensed core power was employed, was estimated by reducing by 6% the integrated energy release associated with the 3216 MWT FSAR calculation. This was done for the integrated energy release rate corresponding to the time to peak pressure for the worst case (DEPSG-Minimum SI). It was estimated that this would result in a decrease of 1.07×10^6 BTU.

3) The reduction in heat removal by the FCU's was estimated in a similar manner by assuming that the decrease in integrated heat removal (up to the time of peak pressure), was directly proportional to the degradation of heat removal at the design point (36% reduction). Applying this assumption resulted in an estimated 0.717×10^6 BTU decrease in net FCU heat removal.

4) The estimated 1.07×10^6 BTU reduction in energy released to the containment would more than compensate for the estimated 0.717×10^6 BTU decrease in FCU heat removal capability. It was, therefore, concluded that there was no safety concern.

Since the replacement fan coils are actually capable of a significantly higher heat removal capacity than 49×10^6 BTU/HR/FCU, revision 5, of the NSE clarified that the heat removal capacity was adequate without consideration of a reduced power level. (See the response to the above finding concerning descrepancy in heat removal rates).

2.6.3 SWS Flow Balance

Finding

A nuclear safety evaluation was not performed for changing the setpoint of the turbine generator lube oil temperature control valve from 115°F to 105°F.

Response

The Authority routinely performs safety evaluations for safety as well as non safety related modifications. Resetting the temperature control valve was a change in the operating point within vendor specified limits and it was not considered a modification. Credit for non safety related components for mitigating the consequences of an accident is not normally assumed. The SWS reanalysis and flow test (ENG-281, Rev. 1) have incorporated the fail open position of the temperature control valve and have demonstrated the acceptability of the flow distribution in the system under worst case conditions.

Finding

On the basis of the above finding, setpoints may not be controlled adequately.

Response

Setpoints are controlled adequately. The Authority does not consider the change in operating temperature of a nonsafety related valve, as a change in setpoint.

Finding

A weakness exists in determing when a safety evaluation is required.

Response

The Authority routinely performs safety evaluations for both safety related and nonsafety related changes, tests and experiments. Therefore, this is considered to be an isolated incident.

POST MODIFICATION TESTING
MOD 81-03-055 FCU - FCU COIL REPLACEMENT

Finding

Acceptance criterion for SWS flow rates to fan cooler units was incorrectly determined.

Response

The 1300 gpm acceptance criteria was based on a Westinghouse evaluation which determined that a heat removal rate of $49x10^6$ BTU/HR/FCU was required for a licensed power rating of 3025 MWt. The 1300 gpm was derived using an overly conservative fouling factor of .004 and 4% plugged tubes. On the basis of the current analysis and as documented in the nuclear safety evaluation, a minimum flow rate of 1400 gpm per FCU is required to maintain the peak containment pressure within 40.6 psig for a power rating of 3216 MWT and actual tube plugging. The minimum flow rate of 1400 gpm was confirmed by both previous and recent performance testing.

Finding

Margin of safety as described in the FSAR was inadvertently reduced due to a lower heat removal rate.

Response

A Westinghouse analysis, based on the core power level of 3216 MWT, determined that a reduction in the heat removal rate from 76.32 BTU/HR/FCU to $49x10^6$ BTU/HR/FCU would slightly increase the peak containment pressure from 40.6 psig to 43.48 psig (preliminary analysis was 43.34 psig). As stated in the SSOMI report, this peak pressure is well within the design pressure for the containment. Also, since the paramaters used to derive the heat removal rate of $49x10^6$ BTU/HR/FCU were unnecessarily conservative, the FCUs are capable of providing a much higher heat removal rate. As noted in the Subsection 2.6.2 response, by applying design paramaters which reflect operational data, the peak containment pressure of 40.6 psig is not exceeded. Therefore, the margin of safety has not been reduced.

2.8 INCORRECT FSAR ANALYSIS

Finding

The analytical model used in the FSAR SWS pipe break analysis does not reflect the installed plant condition. For example, the FSAR analysis takes credit for back flow through the diesel generator coolers even though installed check valves will prevent this back flow. In addition, the FSAR analysis used extrapolated pump performance characteristics for NPSHR which may not be conservative.

Response

The original concern in the IP-3 SER was the adequacy of cooling of the diesel generators during and after a postulated 10" service water line break in the diesel generator cooling loop. This is considered as the licensing basis for long term passive failures during the recirculation phase for IP-3. The existing system satisfies this licensing basis and commitment.

The error in the model is associated with two other breaks in the 24" header where break flow would be precluded because of the presence of check valves in the system. The existing system can handle all postulated breaks in the system and still meet its cooling requirements except for the two 24" breaks mentioned above.

As described in Attachment 2, the Authority considers the postulated circumferential breaks in the seismically supported service water system to be overly conservative. Applying current NRC criteria (Standard Review Plans 3.6.1 and 3.6.2) for postulating cracks in moderate energy piping to the service water system results in acceptable consequences.

Also, a postulated break in the nonseismically supported service water piping to the turbine generator lube oil coolers, as an initiating event, has been evaluated with acceptable consequences.

The Authority is in the process of preparing a submittal to the NRC in order to revise the current methodology for postulating SWS Breaks. Once NRC approval is granted the FSAR will be revised accordingly.

Excluding the postulated 24" diameter breaks in the service water system header, pump performance characteristics have not been extrapolated beyond the manufacturer's supplied performance curve corresponding to a maximum flow of 7000 gpm for other postulated breaks identified in the FSAR. Since the 24" breaks need not be postulated to meet the SER licensing basis, the use of extrapolated pump performance beyond 7000 gpm is not required. The Authority's agent has contacted the pump manufacturer who has reaffirmed the validity of extending the NPSHR curve beyond the 6500 gpm test point.

SSOMI RESPONSE

ATTACHMENTS

1. Description of NYPA's Design Control and Configuration Management Program

2. Evaluation of the SWS Pipe break Criteria and Analysis for the Indian Point 3 Nuclear Power Plant

3. Response to SSOMI Finding 2.2.2 Concerning SWS Pump Replacement Seismic Qualification Report TR-8605, Rev. 1 Magnitude of Vertical Force

4. NYPA Letter to Swanson Analysis Systems, Inc. dated July 17, 1987

5. Evaluation of the Current Authority's In-House Pipe Support Base Plate Design Procedure Using Concrete Expansion Anchor Bolts

rm/ssomi

ATTACHMENT 1 To
SSOMI RESPONSE

DESIGN CONTROL AND CONFIGURATION MANAGEMENT PROGRAM

Over the past year, the Authority has initiated action to unify design and modification control procedures between the corporate office and the nuclear plants. This effort has addressed root causes of problems and identified areas for improvements, particularly in the area of design change control.

This change control program will be a part of the corporate Design and Modification configuration management program and will define the activities of all Power Authority organizations performing design and modification work for the nuclear facilities. The program manual will define the responsibilities and interfaces of all departments involved in design and modification work.

The design control and configuration management program is delineated in a Nuclear Administrative Policy (2.11, "Design and Modification Control Program"). The program consists of four major areas of control: design bases, design standards, design control and modification control. For each of these areas, a separate manual with implementing procedures is being developed. The policy defines the responsibility of various Authority organizations. The program elements are shown in the accompanying figure.

The Authority is taking a multipath approach in developing the design control and configuration management program. The intent of this approach is to identify and improve change control procedures in the short term while also working on issues which require more time to implement. The implementation of these procedures will provide immediate benefits and provide input to the overall program.

The progress in some of the major areas is provided below.

A. The Nuclear Administration Policy, 2.11 has been approved by the President.

B. The identification of those organizations responsible for design and the scope of their responsibility is contained in the Nuclear Administrative Policy. Based on this document, a more detailed implementing procedure has been prepared and approved. This procedure clearly delineates the interfaces between the Nuclear Generation, Engineering and Project Management Departments.

C. Selective procedures that control modifications to the plants are being prepared and/or revised. An initial list of procedures has been prepared and work is proceeding on a priority basis. The new procedures will incorporate consistent controls to be applied to all modifications at both plants and the headquarters office. These controls include appropriate measures to ensure that all applicable design inputs are accurately identified and factored into the design of the modifications and that adequate instructions are developed. The status of some of these procedures is as follows:

1. Conceptual Design Package - Approved

2. Initiation of Modifications - Approved

3. Project Management and Interface Controls for Nuclear Power Plants - Approved

4. Nuclear Safety and Environmental Evaluation - Submitted for Approval

5. Modification Package Preparation and Closeout - Under Review

6. Material and Component Substitution - Under Review

7. Selected procedures that control the design process are being revised. Two procedures (Preparation of Calculations and Approval of Vendor Documents) that will be implemented company-wide as part of the Design Control Manual are currently in use by the Engineering Department.

D. A pilot program has been initiated with United Engineers & Contractors, Inc. for the preparation of design basis documents. An initial meeting with Westinghouse has been held to discuss the content and format of design basis documents. These documents, which will be prepared for selected systems, include a detailed description of the design parameters developed in the original plant design and which are important in the preparation of design changes. A related task will consist of the identification and collection of design documents such as calculations, specifications, etc. Progress in this area is as follows:

1. System design basis documents for Indian Point 3 are in preparation for the the Main Steam and Auxiliary Feedwater systems.

2. The controls of plant drawings is being centralized. Drawing transfer from the IP3 site to headquarters is essentially complete (95%). Procedures for the control of these drawings have been implemented.

The procedures in the Design Control and Configuration Management Program are utilizing INPO guidance and industry experience. SSOMI and SSFI concerns identified as a result of inspections at IP-3 and other utilities are being addressed.

It is expected that design basis documents for two systems and the turnover of associated documentation will be completed by the summer of 1988.

DESIGN CONTROL AND CONFIGURATION MANAGEMENT

- DESIGN AND MODIFICATION CONTROL PROGRAM (Nuclear Administrative Policy)
  - DESIGN BASES MANUAL (Nuclear Generation)
  - DESIGN STANDARDS MANUAL (Design & Analysis)
  - DESIGN CONTROL MANUAL (Design & Analysis)
  - MODIFICATION CONTROL PROCEDURES MANUAL (Nuclear Generation)
  - GROUP SPECIFIC SUPPLEMENTARY INSTRUCTIONS

ATTACHMENT 2 TO SSOMI RESPONSE

EVALUATION OF THE SWS PIPE BREAK CRITERIA
AND ANALYSIS FOR THE
INDIAN POINT 3 NUCLEAR POWER PLANT

TABLE OF CONTENTS

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 guillotine or slot 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 at locations other than the 10-inch Diesel Generator supply line in the Service Water System (SWS) is overly conservative and 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. All the break locations were characterized as both guillotine and 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. PIPEFLOW 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 the results of the pipe break analysis (with or without the check valves noted previously) demonstrate that for a 10-inch diesel generator supply line guillotine or slot 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 (CDC) 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".

Another SECY-77-439 report statement 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 necesary 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. <u>Probabilistic Risk Assessment (PBA)</u>

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}. 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 rate 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 loss of instrument air:

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. A detailed discussion of the analysis and its results are presented in Appendix A. This appendix discusses other failure modes of the SWS other than the one discussed in this report.

V. CONCLUSIONS

a. The analysis presented in Section 9.6.1 of the IP3 FSAR is not valid for a full guillotine break of the 24" essential header during the post-LOCA recirculation phase. NYPA has concluded that such a break is not a credible event.

b. NYPA has concluded that the original pipe break analysis(2) can be used to predict that the IP3 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 or a slot 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 Section 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 IP3 SWS pipe failure analysis.

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

e. The IP3 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/ANS58.9-1931).

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. 2", Docket No. 50-186, 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.

ATTACHMENT 3
SSOMI RESPONSE

RESPONSE TO ITEM 2.2.2
(MAGNITUDE OF VERTICAL FORCE)

The vertical force of 20,471 lbs was the maximum force value in the member (El. 14) just below the mounting plate at the worst loading combination. As shown in Rev. 1 of the report by I-R, dated 6/19/87 (see Appendix A), the value of 20,471 lbs. can be broken down as follows:

Dead Weight	=	4,179 lbs.
Nozzle Load	=	1,200 lbs.
Pressure Thrust	=	13,655 lbs.
Seismic Load	=	1,437 lbs.
Maximum Force at El 14	=	20,471 lbs.

The contribution from seismic load was only 1,437 lbs. not 2.68 g (or 32,696 lbs.) as indicated by the NRC. NYPA has conducted the following independent checks to make sure the computer model used by I-R was correct and the results were consistent with the expected behavior:

1. Reduced mass distribution check
   The reduced mass distributions were found to be identical in three orthogonal directions, i.e., MASS (X) = 22.93, MASS (Y) = 22.93, MASS (Z) = 22.94 lb-sec^2/in. These reduced masses were only slightly less than the total mass of the whole pump (total mass of the whole pump including water = 9,450 lbs. = 24.456 lb-sec/in).

   These results indicate that the selection of dynamic degrees of freedom was appropriate and the dynamic motion of the pump could be adequately characterized by the model.

2. Symmetry check

   a. The frequency results show the mode shapes always come in pairs in the two horizontal directions.

      For example:

      f_1 = 12.05, f_3 = 18.52, f_5 = 25.99 Hz in X-Dir.

      f_2 = 12.59, f_4 = 21.57, f_6 = 26.49 Hz in Z-Dir.

b. The horizontal reactions at the suction head pin support due to seismic loading are almost identical.

Fx = 1411 lb. (SRSS result)

Fz = 1409 lb. (SRSS result)

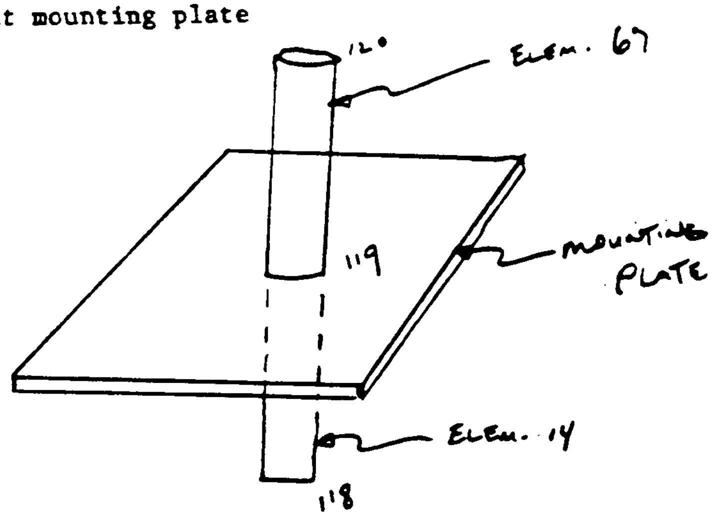
c. For the seismic event, the forces in the two perpendicular springs that modeled the spiders are almost identical. The spring forces in elements 51 and 52 (connected nodes 4 and 104), for example, are :

Fx = 203 lb. (SRSS result)

Fz = 204 lb. (SRSS result)

The computer model, with the exception of discharge nozzle located in X(E-W) direction, generally can be considered as a symmetric model. The results as shown above, appeared to be reasonable and consistent with the expected behavior.

3. Vertical force at mounting plate



a. Dead weight result

The external load should be balanced by the element (internal) force, i.e.,

$F_{ext} = F_{int}$.

Where F_{ext} = dead load carried through node 119

= total pump wt. - mounting plate - flanges

= (24.456-2.064-0.293)g = 8,539 lbs.

F_{int} = F_{14} (Tension in El. 14) + F_{67})

(compression in El. 67)

= 4,183 + 4,356 = 8,539 lbs.

As shown above, $F_{ext} = F_{int}$

b. Seismic result (as discussed in the NRC's report)

Assuming the pump is rigid in the vertical direction, the vertical acceleration, y, at the mounting plate shall fall somewhere between the upper and lower limits, i.e.,

$$\ddot{y}_{max} \geq \ddot{y}_{PL} \geq \ddot{y}_{zpa}$$

where:

$\ddot{y}_{max}$ = The maximum peak acceleration of the floor response spectrum (Upper Limit)

= 0.4g

$\ddot{y}_{zpa}$ = The Zero Period Acceleration of the floor response spectrum (Lower Limit)

= 0.14g

$\ddot{y}_{PL}$ = Acceleration of mounting plate = $\frac{F_{Pl}}{M_P}$

$$= \frac{(\text{SRSS result of } F_{14} + (\text{SRSS result of } F_{67})}{\text{mass of the pump}}$$

$$= \frac{1437 + 689}{24.456}$$

= 0.23g

Therefore, 0.4g   0.23g   0.14g

It can be seen that a net vertical acceleration of 0.23g was developed due to seismic loading (not 2.68g as mentioned in the NRC's report).

In summary, the independent checks conducted by NYPA on reduced mass distributions, symmetric behavior, dead weight distribution, and vertical acceleration of rigid body motion have demonstrated that the computer model used by I-R is correct and the results are in reasonable agreement with the expected behavior.

rm/ssomi

APPENDIX A
to ATTACHMENT 3
of SSOMI RESPONSE 2.2.2

ANSYS Output - Summary of Loads and Stresses

		DW	NL	Press	Seismic	Σ
Anchor	Fx	—	931	—	8160	3981
Bolts	Fy	—	1300	—	7126	3324
Node#315	Fz	—	660	—	8035	3685
El 446.267	Mx	—	17373	—	182428	159711
	My	—	40900	—	1533	42323
	Mz	—	21505	—	180087	171692
Col.	Fx	—	889	—	675	1384
Bolts	Fy	3718[1]	1200	13655[3]	1613	19907
El. 12	Fz	—	632	—	644	1181
Node#116	Mx	—	18573	—	18910	38443
	My	—	40900	—	1533	42323
	Mz	—	26855	—	18554	43906
Motor	Fx	—	—	—	3463	3463
Bolts	Fy	—	—	—	440	440
El 68	Fz	—	—	—	3382	3352
Node#183	Mx	—	—	—	58525	58525
	My	—	—	—	0	0
	Mz	—	—	—	61428	61428
Col	Fx	—	824	—	593	1417
Bolts	Fy	4179[1]	1200	13655[3]	1437	20471
El 84	Fz	—	644	—	566	1210
Node#118	Mx	—	[illegible]	—	36360	43434
	My	—	40900	—	1533	42323
	Mz	—	21885	—	28447	49832

		DW	NL	Seismic	Σ
Column	SDIR	287	111	133	631
El # 14	SBND	6	752	1061	1819
Node 119	ST	0	559	21	580
	SSE	2	193	152	347
Motor	SDIR	-.332	—	51	
Support	SBZ	—	—	1934	
El 66	SBY	—	—	7167	
Node 130	FX	—	—	3508	
	FY	—	—	685	
	MY	—	—	0	

See next page for notes

ANSYS OUTPUT - NOTES

1\. DW

	DIR	BEND	ST	SP	SSF
EL # 12	344	8	0	0	2

Node # 116

(Outer Column)

$$344 \text{ PSI} \times \frac{\pi}{4} (14^2 - 13.5^2) = 3715 \text{ lbs.}$$

2\. DW

	DIR	BEND	ST	SP	SSF
EL # 14	387	5	0	0	2

Node # 119

(Outer Column)

$$387 \text{ PSI} \times \frac{\pi}{4} (14^2 - 13.5^2) = 4179 \text{ lbs.}$$

3\. PRESSURE THRUST AT COLUMN

$$F \text{ Press} = P \times A$$

$$= 95.4 \text{ PSI} \times \frac{\pi}{4} (13.50)^2$$

$$= 13655 \text{ lbs.}$$

Attachment 4

TO SSOMI RESPONSE

New York Power Authority

July 17, 1987

Mr. Mike Wheeler
Swanson Analysis Systems, Inc.
P.O. Box 65
Houston, PA 15342

Ref: Error in STIF 14 of ANSYS

Dear Mike:

This is a record of our two (2) telephone conversations on July 15, 1987 concerning the SRSS combination of the response spectrum analysis results in ANSYS Post 27 solution.

My question to you was:

> In the Post 27 output of the SRSS operation, the stretch of the spring element (STIF 14) appears to be in error. All results (such as node force, element force, and spring rate) are consistent and checked, with the exception of the stretch of spring element. NYPA wanted to know if this is a program error or an error caused by the user's incorrect input.

Your initial response was:

> This is the first time this type of error has been reported to Swanson. Swanson will conduct an investigation to see if this is indeed an ANSYS error.

After two hours of investigation on a similar model, you indicated the following:

(1) The stretch of the spring element is Level 3 data which cannot be processed by SRSS operation (SRSS only works with Levels 1 and 2 data), and

(2) Level 3 data can be obtained with the WRITE operation which not only causes the latest Post 27 results to be copied to outfile, but also automatically calculates the correct Level 3 data from the current Level 2 data. In the case of spring element, there is no Level 2 data for this type of element, therefore no calculation was performed to get the appropriate Level 3 data. Subsequently, the incorrect Level 3 data, i.e., the stretch of the spring element, was printed in the Post 27 output.

With the combination of these two (2) items, it is clear that the ANSYS program has the following error:

> The stretch of the spring element (STIF 14), which was not processed by either SRSS or WRITE operation, should not be allowed to be printed in Post 27.

You said that the only way to correct this error is to change the ANSYS program so that the stretch of the spring element is consistent with the spring rate and the spring force in the SRSS output. You further stated that you will call Mr. Joel Blackman of WESTEK (NRC Consultant), if requested by NYPA, to clarify this problem.

It is my understanding that this error can be eliminated by implementing any of the following items in the future revision of ANSYS:

(1) Change the stretch of the spring element from Level 3 to Level 2 data, so that this data can be processed by SRSS, or

(2) Do a Stretch = Force/Rate calculation in the WRITE operation, so that the correct stretch is obtained and printed, or

(3) Delete the printing of Level 3 data for STIF 14 in Post 27, so that the results will not be misused.

Thank you for your effort in resolving this problem.

Sincerely,

Wensen Chen

Wensen Chen
Civil/Structural Design & Analysis
New York Power Authority
(914) 681-6957

WC:mjd

cc: J. Brunetti
N. Coleman
N. Mathur
J. Bencivenga
L. Garofolo

& SRSS Result

```
NODE FORCE   11.3844        11.3844

ELEMENT   43   STIF 14  ITYPE=  5  MAT=  5  NODES=   15  115
STRETCH OR TWIST= -0.210055E-06  RATE=  0.776000E+07
FORC OR TORQ=          3.58169
NODE FORCE   3.58169        3.58169
```

& SRSS Result (Typical,

```
ELEMENT   44   STIF 14  ITYPE=  5  MAT=  5  NODES=   17  117
STRETCH OR TWIST=  0.648285E-06  RATE=  0.776000E+07
FORC OR TORQ=          12.4788
NODE FORCE   12.4788        12.4788

ELEMENT   45   STIF 14  ITYPE=  5  MAT=  5  NODES=    6  106
STRETCH OR TWIST= -0.203944E-16  RATE=  0.776000E+07
FORC OR TORQ=          17.1905
NODE FORCE   17.1905        17.1905

ELEMENT   46   STIF 14  ITYPE=  5  MAT=  5  NODES=    8  108
STRETCH OR TWIST= -0.220806E-16  RATE=  0.776000E+07
FORC OR TORQ=          10.6237
NODE FORCE   10.6237        10.6237

ELEMENT   47   STIF 14  ITYPE=  5  MAT=  5  NODES=   10  110
STRETCH OR TWIST= -0.141332E-15  RATE=  0.776000E+07
FORC OR TORQ=          32.9838
NODE FORCE   32.9838        32.9838

ELEMENT   48   STIF 14  ITYPE=  5  MAT=  5  NODES=   12  112
STRETCH OR TWIST= -0.460370E-16  RATE=  0.776000E+07
FORC OR TORQ=          11.3224
NODE FORCE   11.3224        11.3224

ELEMENT   49   STIF 14  ITYPE=  5  MAT=  5  NODES=   15  115
STRETCH OR TWIST=  0.373512E-16  RATE=  0.776000E+07
FORC OR TORQ=          4.68808
NODE FORCE   4.68808        4.68808

ELEMENT   50   STIF 14  ITYPE=  5  MAT=  5  NODES=   17  117
STRETCH OR TWIST= -0.115395E-15  RATE=  0.776000E+07
FORC OR TORQ=          11.8534
NODE FORCE   11.8534        11.8534

ELEMENT   51   STIF 14  ITYPE=  4  MAT=  5  NODES=    4  104
STRETCH OR TWIST= -0.121479E-03  RATE=   42500.0
FORC OR TORQ=          202.749
NODE FORCE   202.749        202.749
```

X-Direction } Near the Impeller

```
ELEMENT   52   STIF 14  ITYPE=  5  MAT=  5  NODES=    4  104
STRETCH OR TWIST=  0.205935E-13  RATE=   42500.0
FORC OR TORQ=          204.072
NODE FORCE   204.072        204.072
```

Z-Direction

```
ELEMENT   53   STIF 14  ITYPE=  4  MAT=  5  NODES=   99  100
STRETCH OR TWIST= -0.342703E-03  RATE=   24000.0
FORC OR TORQ=          1410.52
NODE FORCE   1410.52        1410.52

ELEMENT   54   STIF 14  ITYPE=  5  MAT=  5  NODES=   99  100
STRETCH OR TWIST=  0.569050E-13  RATE=   27300.0
FORC OR TORQ=          1408.76
NODE FORCE   1408.76        1408.76
```

EVALUATION OF THE CURRENT AUTHORITY'S IN-HOUSE PIPE SUPPORT BASE PLATE DESIGN PROCEDURE USING CONCRETE EXPANSION ANCHOR BOLTS

An evaluation was performed to demonstrate the adequacy of the current in-house methodology for designing pipe support base plates using concrete expansion anchor bolts.

Currently, the Authority uses the design approach outlined in the July 6, 1979 NRC submittal (IPN-79-45) in response to I.E. Bulletin 79-02 with the following three changes:
(Attachment 1 of the IPN-79-45 submittal is attached for reference) Appendix A

(1) Prying factor (α) is equal to 1.0 for $a + b \leq 6$. The prying factor differs from the submittal in that it does not increase the design tension load on the concrete expansion anchor bolt for base plates with $2t < a + b \leq 6t$.

(2) The shear/tension interaction equation uses a power factor of 5/3 vs. 1.0. The interaction equation differs from the submittal in that it increases the allowable shear/tension interaction loading on the concrete expansion anchor bolts to conform with test results.

(3) The moment arm (h_2) is equal to (d) for $2t < a + b \leq 6t$. The moment arm differs from the submittal in that it decreases the effective moment arm from (d + 2t) to (d), thus increasing the calculated design bolt tension proportionally.

A detailed evaluation of these changes follows:

Prying Factor - NRC IE Bulletin 79-02, Rev. 1, dated June 21, 1979, Page 2, item 1, states in part that:

> "In lieu of supporting analysis justifying the assumption of rigidity, the base plates should be considered flexible if the unstiffened distance between the member welded to the plate and the edge of the base plate is greater than twice the thickness of the plate. It is recognized that this criterion is conservative. Less conservative acceptance criteria must be justified and the justification submitted as part of the response to the bulletin. If the baseplate is determined to be flexible, then recalculate the bolt loads using an appropriate analysis."

The current Authority design method limits the base plate (a + b)/t ratio to equal or less than 6, thus contributing to the limitation of the prying action. The Authority's analyses (Reference 3) show that for these base plates, the design bolt loads always exceed those obtained using a qualified finite element (F.E.) program (Ref. 4) which fully accounts for the prying effect.

In addition, the impact of prying action on the anchor bolts was determined not to be essential for the following reasons:

- When the anchorage system capacity is governed by the concrete shear cone, the prying action would result in the application of an external compressive load in the cone and would not, therefore, affect the anchorage capacity.

- When the bolt pullout determines the anchorage capacity, the additional load carried by the bolt due to the prying action will be self-limiting since the bolt stiffness decreases with increasing load. At higher loads the extension will be such that the corners of the base plate will lift off and the prying action will be relieved.

Shear/tension interaction tests performed by various organizations (e.g. EPRI, Teledyne, References 1 and 2) have shown that a linear interaction shear/tension equation is conservative for the Hilti Kwik bolts that the Authority specifies. Figure 1 makes a comparison between the interaction equation with power factor of 5/3 vs. the Teledyne test results. The 5/3 power is a more realistic design equation and still is bounded by the test results. Also, additional conservatism is introduced by ignoring the interface friction, due to the applied moment, between base plate and concrete.

It should be noted that NYPA uses the highest of the FSAR "faulted" or "upset" loads in evaluating the safety factor of 4 for the bolts. Hence, the allowable loads shown in Figure 1 are determined to be highly conservative.

The IP-3 original installation procedure (IPN-79-45) for concrete expansion anchor bolts assured that the bolts had been set properly and will achieve the manufacturer's recommended capacity.

The moment arm currently used for calculating the bolt tension was reduced by 2t to add more conservatism, since the design bolt tension is inversely proportional to the moment arm. It should be noted that the current design bolt tension due to applied moment is far higher than that calculated by assuming a rigid plate.

In conclusion, the current design procedure maintains a safety factor (i.e. ratio of bolt ultimate capacity to the highest of the FSAR "faulted" or "upset" loads) of 4 or higher. The methodology is consistent with accepted industry practice and does not degrade the original design basis of the plant.

References

1. EPRI Report No. NP-5228, Volumes 1 and 2
"Seismic Verification of Nuclear Plant Equipment Anchors"
May, 1987

2. Teledyne Report No. TR-3501-2
"Summary Report Generic Response to USNRC I & E Bulletin
No. 79-02, Base Plate/Concrete Expansion Anchor Bolts"
August 10, 1979

3. NYPA Calculation No. C/S D&A:87-0994-01
August 3, 1987

4. ANSYS (Rev. 4.2A) Swanson Analysis Inc. with Baseplate Investigation Processors (BIP) Boeing Corporation Services Rev. 1.2.0 October 18, 1984

Appendix A to SSOMT Response

ATTACHMENT 1

(From IPN-79-45 Submittal)

Anchor Bolt Load Calculation

A. Flexibility

A base plate shall be assumed rigid if the unstiffened distance between the member welded to the plate and the edge of the plate is less than or equal to twice the thickness of the plate.

B. Anchor Bolt/Concrete Edge Distance

The base plate and supporting structure designs shall be reviewed to verify that the allowable minimum edge distance is maintained. Allowable loads shall be reduced according to the manufacturer's specifications when minimum edge distances are not met.

C. Anchor Bolt Spacing

Anchor bolt spacing must be maintained for full anchor strength as for the anchor to edge spacing in B. above.

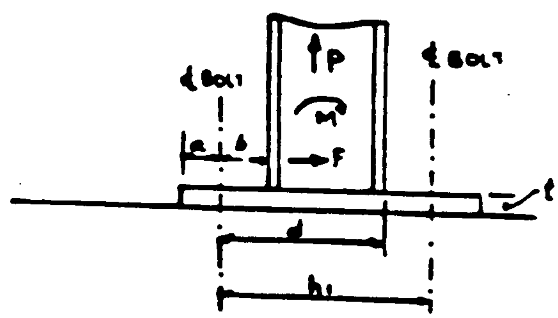
D. Anchor Bolt Load Calculation

The method of anchor bolt load calculation applied to a typical pipe support base configuration is given below:

Rigid when: $a + b \leq 2t$
Flexible when: $a + b > 2t$

$$T = \alpha_i \left(\frac{M}{N_1 h_i} + \frac{P}{N_2} \right)$$

$$V = \frac{F}{N_2}$$



Where: T,V = Anchor design tension and shear loads
M,F,P = Moment, shear and axial force acting on the connection
N_1 = Number of anchor bolts in tension
N_2 = Total number of anchor bolts
i = Flexibility index
i = 1 when rigid
i = 2 when flexible
αi = Prying action factor for given plate flexibility
hi = Moment arm

h_1 = Centerline distance between bolts

h_2 = d + 2t (not to exceed h_1)

Where the connection is subject to biaxial loading, the aforementioned approach must be repeated for the other principal plane and the absolute sum of the bolt reactions combined.

E. Anchor Bolt Allowables

The design tension load for each anchor shall be less than or equal to the Maximum Allowable Design Load (MADL). The MADL is defined as follows:

$$MADL = \frac{Fu}{SF}$$

Where: Fu = ultimate static capacity based on manufacturer's published data.

SF = Safety Factor

SF = 4 for wedge and sleeve anchors

SF = 5 for shell anchors

When both shear and tension act on an anchor, a straight line shear-tension interaction must be assumed as follows:

$$\frac{T}{Ta} + \frac{V}{Va} < 1.0$$

Where: T = Design tension force

Ta = MADL in tension

V = Design shear force

Va = MADL in shear

Tension and Shear Interaction.

Hilti Kwik Bolts.