POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

PRELIMINARY

Prepared for the Power Authority of the State of New York

by

EBASCO SERVICES, INCORPORATED

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INDIAN POINT NO. 3 NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

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OWNER: POWER AUTHORITY OF THE STATE OF NEW YORK

OPERATING COMPANY: POWER AUTHORITY OF THE STATE OF NEW YORK

PROJECT: INDIAN POINT No. 3 NUCLEAR POWER PLANT

UNIT NO.: 3 NOMINAL MW: 965

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CHAPTER 1

1.0 INTRODUCTION

This report documents the first step of a multiphase effort intended to address the concerns of systems interactions. The identification, evaluation, and correction or modification of adverse systems interactions, if any, will enhance the level of protection afforded to the health and safety of the public from the continued operation of the Indian Point No. 3 Nuclear Power Plant.

This report was prepared by Ebasco Services Incorporated in conjunction with personnel from the Power Authority of the State of New York and consists of 2 volumes. Described in this report are the methods used to identify and evaluate systems interactions in general (Volume I) and, in particular, the application of these methods to the Auxiliary Feedwater System (Volume II).

Volume I contains 8 chapters. Subsequent to these few introductory paragraphs, Chapter II presents pertinent background information which will be beneficial in understanding the philosophy of the systems interaction concern as it has evolved over the years. The objectives and scope of this study are presented in Chapter 3. The project organization structure is presented in Chapter 4. Chapter 5 presents the methodology employed in conducting this study. Chapter 6 provides the evaluation criteria by which the application of the review methodology will be judged. The Quality Assurance program utilized during this study is presented in Chapter 7. Chapter 8 is a listing of Reference documents used to develop this study. The results of the application of both the methodology and evaluation criteria to the Auxiliary Feedwater System are presented in Volume II, Appendix A. Subsequent system evaluations will be presented in Appendices B, C etc as necessary.

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CHAPTER 2

2.0 BACKGROUND

From an historical point of view it is noted that the Nuclear Regulatory Commission's (formerly AEC) General Design Criteria (GDC) for nuclear power plants and the Indian Point No. 3 Nuclear Power Plant design were developed concurrently during the late 1960's and early 1970's. The GDC are now incorporated in the NRC's regulations as Appendix A to 10CFR Part 50.

While Criteria 2, 3 and 4 of the GDC require that structures, systems and components important to safety be able to accommodate natural phenomena such as earthquakes, the effects of fires, and other environmental effects without loss of capability to perform their intended safety functions, the systems interaction issue was not specifically raised as a potential concern until the Advisory Committee on Reactor Safeguards (ACRS) formally raised the question in 1974.

In 1977 systems interaction formally appeared as NRC Generic Task Action Plan A-17. The first phase of this NRC plan has just recently been completed with the publication of the Sandia Report "Final Report - PHASE I Systems Interaction Methodology Applications Program". TMI-2 events have to a large extent been factored into this sytsems interaction plan. Additional detail on the regulatory developments on systems interaction are found in:

a) Generic Task Action Plan A-17 (NUREG 0606 Rev. 2) Systems Interaction In Nuclear Power Plants

2.0 BACKGROUND (Cont'd)

b) NUREG 0510
Identification Of Unresolved Safety Issues Relating To Nuclear Power Plants

c) NRC Information Notice 79-22
Potential Interactions Between Non-Safety Related Control Systems And Safety Systems

d) NUREG 0585
TMI-2 Lessons Learned Task Force Final Report
Recommendation 9 - Review Of Safety Classifications And Qualifications

e) NUREG 0660
Action Plans For Implementing The Recommendations Of The President's Commission And Other Studies Of TMI-2 Accident

TASK II.C.I - Systems Engineering, Reliability Engineering And Risk Assessment

The NRC has recently distributed three reports prepared by independent laboratories which address the different methodologies being utilized by various utility groups, consultants, etc. They are, NUREG/CR-1859, UCRL-53016, Systems Interaction: "State-of-the-Art Review and Methods Evaluation", prepared by Lawrence Livermore Laboratory for NRC-ONRR, November, 1980; NUREG/CR-1901, BNL-NUREG-51333, "Review and Evaluation of System Interactions Methods", prepared by Brookhaven National Laboratory for NRC-ONRR, January 1981; NUREG/CR-BMI-2055, R-2 "Report on Review of Systems Interaction Methodologies", prepared by Battelle Columbus Laboratories for NRC-ONRR, January 1981.

2.0 BACKGROUND (Cont'd)

Discussions between the industry via the Atomic Industrial Forum (AIF) and NRC are anticipated this spring or early summer at which time the NRC's plan for future work, taking into acccount the conclusions of the reports noted above, is scheduled for completion. At this time there is no universally accepted methodology for conducting systems interaction studies.

In order to derive a working definition of systems interaction, it is necessary to consider a number of associated concepts. In the design of a nuclear power plant provisions are made to make the release of radioactivity to the environment an extremely unlikely event by providing independent ways in which a safety function can be performed. These provisions are expressed in terms of redundancy and diversity so that multiple independent system failures would be necessary to have a safety function failure. Systems which support safety functions may be designed to interact with each other. These interactions are intentional. An "interaction" of concern results when the conditions in one system affect (degrade) the ability of another system to perform it's safety function. Therefore, system interactions are those events that affect the safety of the plant by one system acting upon one or more other systems in a manner not intended by design.

It is important to recognize that the systems interaction process is an attempt to reevaluate in a systematic fashion those potential events whose direct effect or natural cascading features could reduce plant safety margin. The criteria employed are considered new only to the extent that effects of nonsafety systems on safety systems are considered in a more thorough fashion. Currently, neither the NRC nor any industry body (such as AIF or ANS) have published an accepted methodology for performing systems interaction.

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CHAPTER 3

3.0 OBJECTIVES AND SCOPE

3.1 OBJECTIVES

The objectives of this study are (1) to develop the methodology and evaluation criteria to be used to identify and assess potential systems interactions and (2) to apply these criteria to a systems interaction review of the IP-3 Auxiliary Feedwater System. This methodology and evaluation criteria can then be utilized to evaluate other systems that may be identified for review.

For the "connected system" portion of the study, Ebasco utilized the work done by PL&G in the area of fault tree analyses, developed safe shutdown logic diagrams where necessary, and performed a Failure Mode and Effects Analysis (FMEA) to identify potential adverse interactions. For the "nonconnected system" portion of the study, Ebasco investigated the possibility of adverse interactions transported to the AFS via spatial or physical proximity considerations during design basis events such as earthquake, tornado, fire, high energy pipe rupture, internal or external flooding and internally or externally generated missiles. These latter events were investigated for interactions via the plant walkthrough and by a review of reports previously prepared on these subjects.

3.2 SCOPE

Although Chapter 5, Methodology, will supply many of the details, the general scope of work for the Auxiliary Feedwater System (AFS) study will include the investigation of systems physically connected to the AFS (Refer to Section 5.3) via the use of Failure Modes and Effects Analyses (FMEA), and systems not directly connected to the AFS (Refer to Section 5.4) via the in-situ plant walkdown technique.

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CHAPTER 4

4.0 STUDY TEAM ORGANIZATION

4.1 PASNY TEAM ORGANIZATION

The Authority retained Ebasco Services Incorporated to assist in the study of systems interaction for Indian Point No. 3 Nuclear Power Plant.

The Nuclear Technical Support Division (NTS) of the Nuclear Generation Department (NG) has the primary responsibility for accomplishing the Systems Interaction Study and for controlling and monitoring the activities of Ebasco Services.

The Authority's IP-3 Project Engineer appointed a Task Force Leader from his staff with the concurrance of the Manager, NTS, and the Senior Vice President, NG. For the Systems Interaction Study the Authority's IP-3 Project Mechanical Engineer was the Task Force Leader.

The Task Force Leader is responsible for monitoring and controlling day to day Ebasco activities and to ensure a sound multi-disciplinary review of work done by Ebasco. This was accomplished by choosing the following Authority personnel to be part of the review team:

Nuclear Operations Engineer-Nuclear Operations Division, NTS, NG, NYO.
Nuclear Licensing Engineer - Nuclear Licensing Division, NG, NYO.
IP-3 Project I&C/Electrical Engineer - NTS, NG, NYO
IP-3 Project Nuclear Engineer - NTS, NG, NYO.
IP-3 Project Mechanical Engineer - NTS, NG, NYO.
Nuclear Operations Senior BOP Engineer - Nuclear Operations Division, NTS, NG, NYO.
IP-3 Project Civil/Structural Engineer - NTS, NG, NYO.

4.0 STUDY TEAM ORGANIZATION (Cont'd)

4.1 PASNY TEAM ORGANIZATION (Cont'd)

Senior Structural Engineer - Design & Analysis Division, Engineering Dept,
NYO.
Senior Nuclear Engineer - Design & Analysis Division, Engineering Dept, NYO.
Site Engineer - Technical Services Department, IP-3 Site
QA Engineer - QA Dept, NYO.

The plant walk through team was comprised of the Task Force Leader, Site Engineer, Nuclear Operations Engineer and the Ebasco Systems Interaction team personnel.

Figure 4-1 indicates the structure of the Authority's reporting relationships among the Systems Interaction Task Force.

4.2 EBASCO TEAM ORGANIZATION

Within the Ebasco Organization, Systems Interaction Study for the Indian Point No. 3 Nuclear Generating Unit No. 3, is administered by the Mechanical-Nuclear Engineering Department under the direction of a Project Manager. Personnel from various Ebasco Engineering and Design disciplines are assigned to the project and take functional directions from the Project Engineer.

Figure 4-1 also indicates the reporting relationships among Ebasco Engineering and Design personnel who fulfill the key roles in the Systems Interaction Study.

The responsibilities within the Ebasco Study Team are outlined as follows:

4.0 STUDY TEAM ORGANIZATION (Cont'd)

4.2 EBASCO TEAM ORGANIZATION (Cont'd)

Project Manager

The role of the Ebasco Project Manager is to provide central leadership, planning, scheduling, budgeting and coordination of all services supplied by Ebasco to the Authority in addition to developing and administering controls to achieve schedule and budget compliance.

Systems Interaction Project Engineer

The role of the Project Engineer who reports to the Project Manager, is to provide advice, guidance, and support to the Project Team in performance of their function, manage the overall engineering effort, and integrate the multiple engineering activities.

His responsibilities include the following:

a. Writing the System Interaction Study description.

b. Coordinating the efforts of other Ebasco engineering and design disciplines who are preparing the study, preparing implementing procedures, determining study inspection and evaluation criteria, and reviewing resolutions proposed by the Interaction Team.

c. Providing functional and technical direction to the Interaction Team.

d. Reviewing and approving the resolutions proposed by the Interaction Team.

4.0 STUDY TEAM ORGANIZATION (Cont'd)

4.2 EBASCO TEAM ORGANIZATION (Cont'd)

Systems Interaction Project Engineer (Cont'd)

e. Preparing interim reports and the final program report.

f. Communicating the activities of the Interaction Team and the results of the program to the Project Manager.

The Project Engineer will use in-house engineering and design disciplines to recommend technical decisions, provide administrative assistance, recommend resolutions, and provide analysis as needed. All engineering and design disciplines will report to the Project Engineer.

Project Quality Assurance Engineer (PQAE)

The PQAE is responsible for the implementation of the Quality Assurance Program for the System Interaction Study. He reports directly to the Chief Quality Assurance Engineer and has the authority and responsibility to identify quality related problems, to intiate or recommend solutions to control nonconformances until properly dispositioned and to verify implementation of approved dispositions.

Interaction Team

The interaction team members are required to have considerable experience in their area of assignment. They have been involved with previous systems interactions studies on other nuclear projects.

4.0 STUDY TEAM ORGANIZATION (Cont'd)

4.2 EBASCO TEAM ORGANIZATION (Cont'd)

Interaction Team (Cont'd)

a. The Interaction Team comprises the following discipline Lead Engineers and their staffs:

(1) Mechanical-Nuclear Engineering
(2) Instrumentation and Control Engineering
(3) Electrical Engineering
(4) Civil/Structural Engineering
(5) Heating, Ventilating, and Air Conditioning Engineering
(6) Licensing

b. The discipline Lead Engineers are selected from the staff of the engineering and design departments and are under the technical direction of the Discipline Chief Engineer.

FIGURE 4–1

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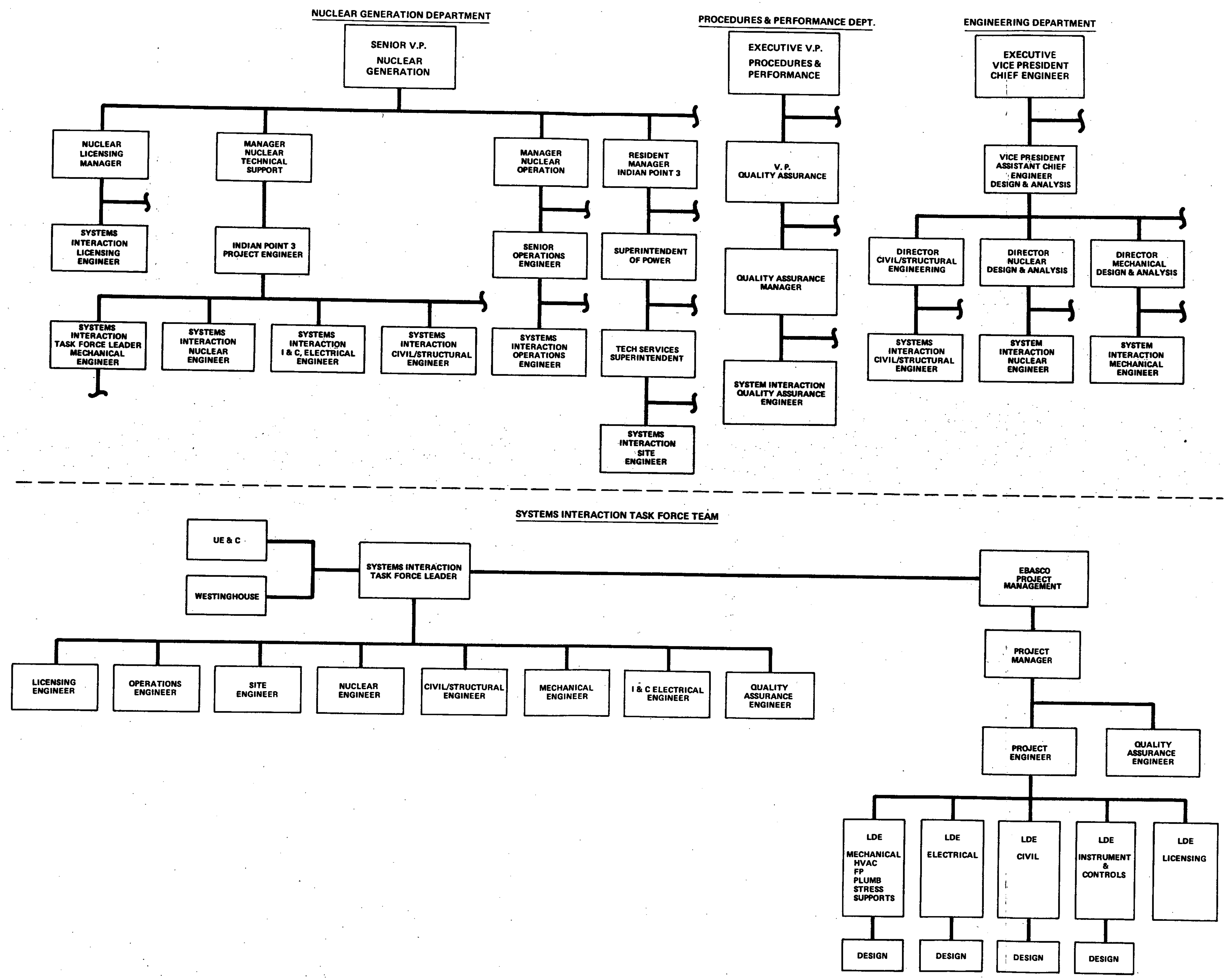

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CHAPTER 5

5.0 METHODOLOGY

5.1 PURPOSE

This section describes the methodology and documentation process to be used for performing the systems interaction study for the Indian Point No. 3 Nuclear Power Plant.

Through an evaluation of methodology techniques prescribed by state-of-the art reviews, it was concluded that any one method can not perform an adequate review for determining adverse systems interactions. However, all of the methods evaluated included a process of "sifting-out" adverse systems interactions by 1) selecting specific systems for detailed evaluation, 2) the identification of dependencies or commonolites, and 3) evaluation of the systems interactions through the determination of their relative importance to safety. It is this three-step process which provided the foundation for performing a systems interaction study for the Indian Point No. 3 Nuclear Power Plant.

5.2 INITIAL ACTIVITIES

The initial task of this study was to determine if an adverse systems interaction could occur and if so, whether or not a significant impact on the degradation of the reactor core and the release of unacceptable levels of radioactivity to the site environs could result. Those conditions considered to be adverse were those which have a significant potential for leading to core damage are, failure to achieve or maintain reactor subcriticality, failure to remove decay heat, failure of the reactor coolant system pressure boundary and containment integrity. Table 5-1 describes the safety functions associated with reactor subcriticality, decay heat removal, reactor coolant pressure boundary and containment integrity and the corresponding systems and major components for the Indian Point No. 3 Nuclear Power Plant.

5.0 METHODOLOGY (Cont'd)

5.2 INITIAL ACTIVITIES (Cont'd)

Of the systems described, the Auxiliary Feedwater System is required for decay heat removal and therefore was chosen as a test system to apply the investigative methodology and evaluation criteria. Refer to Appendix A for the results on the Auxiliary Feedwater System.

5.3 INTERCONNECTED SYSTEMS INTERACTIONS

5.3.1 Identification of Interconnected Systems

Interconnected systems are defined as those mechanical and electrical complexes which are process coupled to one another physically via piping, instrumentation tubing or electrical wiring. Included in this definition is HVAC equipment, which although not physically connected, may be necessary to support the continuous safe operation of interconnected systems, e.g., an air handling unit which has been specifically designed to cool an essential safety related pump/motor set.

The first step in "sifting-out" adverse systems interactions, the selection of specific systems for evaluation, is accomplished by developing functional shutdown logic diagrams that describe the general functions necessary to prevent core damage and the release of radioactivity to the environs. The logic diagrams are based upon system descriptions, instrumentation and control logic diagrams and electrical schematic, block and wiring diagrams. In addition important information about system interfaces is obtained at the site by inspecting physical facilities and by meeting with plant personnel familiar with the design, operation and maintenance of the system.

5.3 INTERCONNECTED SYSTEMS INTERACTIONS (Cont'd)

The second step, the identification of dependencies or commonalities is accomplished by further developing the functional system shutdown logics into subsystems more commonly referred to as auxiliary diagrams. The auxiliary diagrams provide the link between the functional system and support systems necessary to achieve a safety function.

The third step, evaluation of the systems interaction through the determination of their relative importance to safety is accomplished by using deterministic logic, eg, failure modes and effects analyses or the equivalent.

5.3.2 Plant Operating Mode

The systems interaction study for interconnected systems is conducted for Condition I, II, III and IV events as described by the N-18 Committee of the American Nuclear Society (ANS/N-18).

5.3.3 Failure Criterion

Postulated system interactions induced by random failures of safety related components will be considered acceptable if it does not compromise the functional capability of the system to perform its intended safety function. Additional active failures need not be assumed in the redundant safety train(s) of the interconnected system. The basis for this is that the random component failure is itself the assumed active failure.

5.0 METHODOLOGY (Cont'd)

5.3 INTERCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.3.4 Analysis Techniques For Interconnected Systems

5.3.4.1 General

For interconnected mechanical or electrical complexes (process connected systems), logic models (fault trees) will form the basis for the systems interaction analysis. The purpose of the fault trees is to model the combinations of components which, if assumed to fail, would result in loss of any of the four basic functions, i.e. reactor-subcriticality, decay heat removal reactor coolant pressure boundary, or containment integrity and by assumption result in possible core damage. The fault trees are the vehicles for the identification and evaluation of systems interactions which could significantly compromise the safety of the Indian Point No. 3 Nuclear Power Plant.

Fault trees that have been prepared by Pickard, Lowe and Garrick, "Zion/Indian Point, Probabilistic Risk Assesment," Draft, for the Indian Point No. 3 Nuclear Power Plant were considered in this study. In those instances where the level of detail is insufficient to complete a comprehensive review of random component failures within an interconnected system, further investigation in that area is identified. In some cases although the level of detail is not down to the component, it is possible to arrive at a conclusion based upon sound engineering judgement and logic. A failure mode and effects analysis (FMEA) is then performed for all components within the system or subsystem.

5.3.4.2 Fault Trees

Fault trees are used to determine the likelihood of failure of the various systems identified in the event tree accident paths. A fault tree starts with the definition of an undesired event, such as the failure of a system to operate, and then determines, using engineering and mathematical logic, the ways in which the system can fail.

5.3.4.3 Shutdown Logic Diagrams

Having described the safety functions and the specific plant safety systems the next step is to identify the required responses, or safety actions that must be accomplished in order to achieve a safety function. The sensed variables are identified that cause or require the system response. In cases where the system does not automatically respond, the operator action required to initiate the safety system (e.g., starting a pump locally from the control room) is identified. As the safety systems and their action are identified, they are arranged in functional order forming success paths, or operation sequences, leading to the required safety function. The arrangement of success paths becomes the Shutdown Logic Diagram (SLD) for the event.

To depict the level of redundancy in the plant design on the SLD, a sufficient number of independent parallel paths is developed for each safety function such that no single component failure can prevent the achievement of the required safety function.

5.3.4.3.1 Safety System Auxiliary Diagram

After completion of the SLD for a postulated event, each safety system displayed on the SLD is analyzed to determine the specific support requirements necessary to produce its safety action. Examples of these support requirements are electric power, component cooling, or instrument air supply. The analyst refers to the SLD to determine every sequence in which a safety system is required, thereby ensuring all support requirements are identified. After identification of the support requirements, the plant systems that provide these support requirements are identified. These systems are the Auxiliary Safety Systems. A Safety System Auxiliary Diagram is then prepared on which the prime safety system and its auxiliary safety systems are displayed.

5.0 METHODOLOGY (Cont'd)

5.3 INTERCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.3.4.3.1 Safety System Auxiliary Diagram (Cont'd)

In developing the Safety System Auxiliary Diagram the analyst ensures that each support requirement is functionally redundant by developing design information about the plant sufficient to positively identify the auxiliaries essential to the required response of the safety system, and by identifying plant design changes so that the auxiliary safety systems can support their safety system with the needed level of redundancy.

To complete the Safety System Auxiliary Diagram the analyst must review the Shutdown Logic Diagrams for all the postulated events to identify all safety sequences in which the subject auxiliary safety system appears.

5.3.4.3.2 Auxiliary Safety System Commonality Diagram

After completion of the Shutdown Logic Diagrams for each postulated event and the Safety System Auxiliary Diagrams, the Auxiliary Safety System Commonality Diagram (ASSCD) for each Auxiliary Safety System is developed. This diagram indicates all the safety systems that a given Auxiliary Safety System supports. The ASSCD is developed mainly as an information diagram, rather than a primary design review diagram. The ASSCD allows evaluation of the overall plant response to the operations of each Auxiliary Safety System.

5.3.4.4 Failure Modes and Effects Analysis Description

The failure mode and effects analysis provides for the evaluation of partial success or failure, off-normal operation, inadvertent operation, and time dependence. From these FMEA'S, performed with the fault tree as a backdrop, interconnected system interactions can be identified and evaluated.

5.0 METHODOLOGY (Cont'd)

5.3 INTERCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.3.4.5 Criteria for Selecting Random Equipment/Component Failures

5.3.4.5.1 The Scope of Failures To Be Excluded are :

- Operator Induced Failures, and
- Equipment Unavailability Due To Testing Or Maintenance, and
- Sabotage

The subject of operator induced failures in system interaction studies have been excluded from this criteria. The subject of the operator's influence on plant safety is not, however, being neglected since much of the available industry resources have been directed at improving operator training, developing advanced simulators, improving the human-machine interface through additional instrumentation and control room reevaluation, and the development of improved operational procedures.

Equipment unavailability due to testing or maintenance is also excluded from the scope of this study since generally technical specifications limit the time safety related equipment may be removed from service while the plant is in the operating mode.

5.3.4.5.2 The scope of Random Failures does include the consideration of:

Random failures caused by adverse interactions of interconnected systems and components that result as a direct consequence of off-normal events or actions for which the effected equipment has been prescribed to operate. The off-normal events or actions which will be considered in this study are:

5.0 METHODOLOGY (Cont'd)

5.3 INTERCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.3.4.5.2 The scope of Random Failures does include the consideration of: (Cont'd)

- Loss of Power (both motive and control power of the electrical, hydraulic and pneumatic type)
- Chemistry (Fluid Purity)
- Cooling (including HVAC equipment)
- Lubrication
- Operating Vibratory Motion.

5.4 NONCONNECTED SYSTEMS INTERACTIONS

5.4.1 Identification of Nonconnected Systems

Nonconnected systems are defined as all safety and nonsafety mechanical, electrical and civil systems which are associated with the physical arrangement or spatial coupling of each other.

The identification of nonconnected spatially coupled systems is based upon the review of plant general arrangements. The plant general arrangement and its association with spatially coupled systems is determined by performing a systematic plant "walkdown" of the areas comprising the system function described in Table 5-1.

5.4.2 Plant Operating Mode

The systems interactions study for nonconnected or spatially coupled systems is conducted for those Design Basis Events described in section 5.4.3 for the corresponding plant operating mode.

5.0 **METHODOLOGY** (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.3 Failure Criterion

When considering systems interactions of nonconnected systems for the design basis events described herein, the structures, systems and components important to safety (defined as those required to bring the plant to, and maintain it at cold shutdown from normal operation or a transient condition and/or those required to mitigate accident conditions) shall not be prevented from carrying out their required safety functions because of physical, mechanical, fluid or electrical interactions caused by the event induced failure of equipment not qualified/designed to withstand event consequences. The structures, systems and components important to safety shall not lose their redundancy required to compensate for single failures, because of event induced interactions.

In this report "an event" will include the following:

1) Earthquake: up to and including the safe shutdown earthquake.

2) Pipe failure: pipe whip, jet impingement, jet reaction, severe environment - (temperature, pressure, humidity)

3) Physical Impact: from missiles generated internally and externally.

4) Flooding from internal failures (pipe and tank failure) or external effects due to rain, snow etc.

5) Tornado Depressurization

6) Fire

5.0 METHODOLOGY (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.4 Analysis Techniques

For nonconnected systems interactions the first step is to classify nonconnected spatially coupled systems, components and structures as either a "source" or a "target."

Target Definition:

Equipment which requires protection from potential event induced interactions are designated as targets. All structures, systems and components important to safety are considered targets. In addition, specific portions of the fire protection systems are also designated as targets in accordance with a November 13, 1978 NRC letter from Philip A. Crane Jr. to John F. Stoltz, chief of Light Water Reactor Branch No. 1.

5.4.4.1 Seismically Induced Systems Interactions

Source Definition:

For seismically induced events, the sources of detrimental interactions are any non seismically supported or qualified structures, systems or components which, by their proximity and/or connection to targets, may interact through physical, mechanical, or electrical means to compromise the integrity or operability of the target.

5.4.4.1.1 Interaction Walkdown

A plant walkdown will be performed by an interdisciplinary team of experienced engineers. During the inspection, all possible interactions will be postulated for source equipment that might affect the target system to be

5.0 METHODOLOGY (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.4.1.1 Interaction Walkdown (Cont'd)

protected. Consideration will be given to local equipment arrangement and geometry, and the possible results of these failures. The interaction team, after identifying all possible interactions between source and target equipment, will utilize the established criteria in chapter 6 to determine if these interactions are credible. Once the field system evaluation has been completed the following information will be documented.

a. Location of the potential interaction
b. Components and systems involved in the potential interaction are identified on an interaction matrix form and documented on the interaction documentation forms.
c. The specific criteria used for the evaluation (which includes the type of interaction) is documented on the documentation forms.
d. A photographic record of each identified interaction is made. The photograph is cross referenced with the interaction matrix form and the interaction documentation form. A small arrow indicates the general location of a target(s). A key plan is made indicating the general location of the interaction.

e. Recommendation of the interaction team. This may take the form of one of the following:

(1) Finding whether or not an interaction occurs
(2) Determine that, if interaction does occur, no safety function is impaired.
(3) Recommendation that a physical modification be designed and installed.
(4) Recommendation for further evaluation.

5.0 METHODOLOGY (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.4.1.1 Interaction Walkdown (Cont'd)

The Interaction Team will consider failures to non-essential systems (e.g., loss of electricity and pressure) which may have an effect on the operation of target equipment.

During the plant walkdown, each component on the target matrix listing to be evaluated for interactions will be inspected by the Interaction Team. Each unit of source equipment in the vicinity of the item will be considered to fail by any or all of the specific mechanisms listed in the criteria (Chapter 6). These Mechanisms will be considered to act singly and in combination. When failure has been postulated, it will be possible during the inspection or, afterwards by offsite analyses, to determine interactions with the target equipment. All such interactions will be listed and evaluated using the established criteria as described in Section 5.4.3.

The plant walkdown by the interdisciplinary team will consider the effects of intercompartmental interactions. All possible intercompartmental interactions will be identified and relevent data such as location will be documented. The walkdown team will physically inspect all adjacent compartments that may have interaction effects. Items such as fire, flooding, electrical, pressure, and dynamic effects will be considered. Further interaction effects that may be determined from evaluation of the data base information may require a second intercompartmental walkdown.

5.0 METHODOLOGY (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.4.1.2 Interaction Criteria

An interaction is identified whenever the event induced behavior of a source could lead to detrimental effects on a nearby target. Pairings of targets and sources are based on physical proximity or direct system connection. Then an assessment is made of the possible event induced behavior of the sources. An interaction is not identified by the field walkdown team if it can be established by inspection that no credible failure mode can be induced in the sources by events of credible severity, which would violate the acceptance criterion

In general, event induced interactions identified will be in one or more of the following categories:

a. Contact between a source and a target that would compromise operability of the target.

b. Fluid leakage from one or more sources that would degrade the environment of the target component.

c. Contact between a missile generated by a source and a primary target that would compromise the pressure boundary of a secondary target component.

d. Contact between a missile generated by a source and a primary target that would compromise operability of a secondary target component.

e. Failure of non-safety related electrical equipment that would compromise the operability or integrity of target equipment.

f. Obvious secondary effects or cascading influences (mechanical, electrical or fluid) caused by any of the above interactions.

5.0 METHODOLOGY (Cont'd)

5.4 NONCONNECTED SYSTEMS INTERACTIONS (Cont'd)

5.4.4.2 Pipe Failure Induced Systems Interactions

The methodology employed for determining the effects of pipe failure induced systems interactions is consistent with the guidelines provided in NRC Standard Review Plans 3.6.1 and 3.6.2 and Regulatory Guide 1.4.6.

5.4.4.3 Missile Induced Systems Interactions

The methodology employed for dtermining the effects of internally and externally generated missile induced systems interactions is consistent with the guidelines provided by NRC Standard Review Plans 3.5.1, 3.5.2 and 3.5.3.

5.4.4.4 Flooding Induced Systems Interactions

The methodology employed for determining the effects of flooding induced systems interactions is consistent with the guidelines provided by NRC Standard Review Plans 3.4.1 and 3.4.2.

5.4.4.5 Fire Induced Systems Interactions

The methodology employed for determining the effects of fire induced systems interactions is consistent with the guidelines provided in NRC Standard Review Plan 9.5.1 and companion Branch Technical Position APCSB 9.5-1.

5.4.4.6 Severe Environment Induced Systems Interactions

The methodology employed for determining the effects of severe environment induced systems interactions is consistent with the guidelines provided by NRC Standard Review Plans 3.3, 3.4, 3.5, 3.6 and 3.11.

TABLE 5-1

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

SYSTEM INTERACTION STUDY

SAFETY FUNCTIONS ASSOCIATED WITH

REACTOR SUBCRITICALITY, DECAY HEAT REMOVAL, REACTOR COOLANT PRESSURE BOUNDARY, CONTAINMENT INTEGRITY

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
Subcriticality	Trip Reactivity Control	Rapid insertion of negative reactivity into the core to produce subcriticality immediately following an event.	Rod Control System
		Insertion of negative reactivity into the core sufficient to compensate for cooldown of the reactor coolant system.	Safety Injection System Chemical & Volume Control System 1) Refueling Water Storage Tank (RWST) 2) Boron Injection Tank 3) Hi-Head Safety Injection Pumps 4) Boric Acid Storage Tank

TABLE 5-1 (Cont'd)

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
Subcriticality	Long Term Reactivity Control	Establishment of a sufficient boron concentration in the core such that the reactor is maintained subcritical following the event.	Safety Injection System Chemical & Volume Control System 1) Boric Acid Transfer Pumps, Tanks and Charging Pumps 2) RWST
Decay Heat Removal	Emergency Core Cooling - Injection Phase	Injection of coolant to the reactor core immediately following an accident and prior to the time that manual action can be taken.	Safety Injection System, Chemical & Volume Control System, Residual Heat Removal System
Decay Heat Removal	Emergency Core Cooling - Recirculation Phase	Provision of coolant to the reactor core some time after the accident has occurred and at a time when manual action can be taken and in such a way that the core coolant is recirculated back into the primary system after it leaks out.	Safety Injection System, Residual Heat Removal System 1) Hi-Head Injection Pumps 2) RHR Pumps

TABLE 5-1 (Cont'd)

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
	Reactor Decay Heat Removal	Cooling of the core by other than injection of coolant directly to the core.	Residual Heat Removal System Auxiliary Feedwater System
Reactor Coolant Pressure Boundary	Pressure Control - Primary System	Maintenance of primary system pressure within allowable pressure limits and ensuring that the primary system steam bubble remains in the pressurizer.	Reactor Coolant System, Chemical & Volume Control System, Auxiliary Feedwater System, Main Steam System 1) PRZ Heaters & Spray 2) Auxiliary Spray 3) Charging Pump 4) AFS Pumps 5) MS Code Safety Valves 6) MS Atmospheric Steam Dump Valves

TABLE 5-1 (Cont'd)

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
Containment Integrity	Pressure Control - Containment	Maintenance of containment pressure within allowabal pressure limits when containment is required.	Safety Injection System, Containment Spray System 1) Fan Cooler Units
Containment Integrity	Combustible Gas Control	Conditioning of post-accident atmosphere or treatment of accident-generated flammables to prevent formation of flammable or explosive mixtures.	Hydrogen Recombiners, Charcoal Filters Fan Cooler Units
Containment Integrity	Radioactive Material Treatment	Mechanical or chemical treatment of radioactive materials to reduce the quantity that escape or are discharged to the environment.	Weld Channel, Isolation Valve, Seal Water & Gas Post-Accident Filtration System
Containment Integrity	Establish Containment	Trapping of radioactivity inside the containment to prevent escape to the environment.	Weld Channel, Isolation Valves, Seal Water and Gas Containment Isolation System
Containment Integrity	Primary System Isolation	Isolation of all or part of the primary system to prevent coolant loss or radioactivity discharge.	Phase A Containment Isolation System

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
Containment Integrity	Secondary System Isolation (blowdown)	Isolation of all or part of the secondary system to prevent or reduce the discharge of secondary system coolant into the containment, so that containment temperature and pressure are maintained within allowable limits.	Main Steam System Main Feedwater System Steam Generator Blowdown System 1) MSIV's 2) Feedwater Isolation 3) Blowdown Isolation (Operator Action)
Containment Integrity	Secondary System Isolation (heat sink)	Isolation of all or part of the secondary system to prevent or reduce the discharge of secondary coolant, so that at least the minimum number of steam generators can function as a heat sink for primary system energy removal.	Main Steam System Main Feedwater System Steam Generator Blowdown System 1) MSIV's 2) Feedwater Isolation 3) Blowdown Isolation (Operator Action)

TABLE 5-1 (Cont'd)

"FUNDAMENTAL SAFETY FUNCTION"	GENERAL SAFETY FUNCTION	FUNCTIONAL DESCRIPTION	APPLICABLE SYSTEM
Containment Integrity	Secondary System Isolation (radioactivity)	Isolation of all or part of the secondary system to prevent the discharge of radioactive materials to the environment.	Main Steam System Blowdown System
Decay Heat Removal	Steam Generator Inventory Control	Maintenance of a proper level in at least the minimum number of steam generators for use as a primary system heat sink and to preclude from injecting cold feedwater into a dry and hot steam generator.	Auxiliary Feedwater System
	Control Station Habitability	Conditioning of the post-event control station (Control room and other locations where manual actions are essential) atmosphere to ensure habitability and control of personnel radiation exposure.	Control Room Ventilation System Emergency locker equipment

INDIAN POINT No.3 NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

CHAPTER 6

6.0 EVALUATION CRITERIA

The evaluation of event induced systems interactions and their effects on plant safety rests heavily on experienced engineering judgement. Reliance is placed on assigned engineering and design personnel in various relevant disciplines applying their knowledge and experience in evaluating the problems.

6.1 INTERCONNECTED SYSTEMS

The evaluation of interconnected system interactions and their effects on plant safety will be based upon satisfying the failure criterion presented in Section 5.3.3 using the techniques of failure mode and effects analysis. As described in Section 5.3.3, postulated system interactions induced by random failures of safety related components will be considered acceptable if it does not compromise the functional capability of the system to perform it's intended safety function.

6.2 NONCONNECTED SYSTEMS

6.2.1 Evaluation of Sources

Potential sources are evaluated as part of the program to determine if events can credibly lead to detrimental interaction with targets.

a. Events will not lead to interaction because of defensible qualification of the sources by analysis, test, or experience with the same or similar items.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.1 Evaluation of Sources (Cont'd)

b. Events may lead to damage or failure of the sources, but the credible failure modes are no threat to the safety function of the target.

c. Events may lead to a credible failure mode of the source which has the potential to cause adverse interaction.

6.2.1.1 The following criteria provide minimum guidance for evaluation of sources for seismically induced events:

a) Structural Source Evaluation

All structural sources are evaluated by the single failure criterion:

Any non safety related structural element determined to be a potential source will be assumed to fail, unless seismic qualification by analysis, test or comparison to similar previously qualified elements has been performed to ensure integrity.

b) Mechanical Source Evaluation

The following is a set of failure modes for mechanical equipment which must be considered when evaluating potential sources in these categories.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

b) Mechanical Source Evaluation (Cont'd)

In addition to the specific failures below, complete loss of power for all source equipment and control power has been postulated. Relative motion between the source and target are considered during the walkdown examination.

- Overturning of tanks, pumps, filters or other unsupported equipment where the center of gravity location as measured from the base is longer than one-half the base width in all directions. Each direction will be evaluated independently. A horizontal acceleration equivalent to at least that value associated with the plant SSE, would be required to overturn an unsupported component whose height is less than 1/2 base width from the base. Overturning is not considered where the distance from the base to the center of gravity is small. Further conservatism is obtained because mechanical equipment is held down by bolting, brackets, etc. However, if any component structure or system experiences a horizontal acceleration of greater than the SSE, it will be evaluated on a case by case basis.

- All non-seismically qualified valves, pumps, tanks and vessels are assumed to fail in the "worst credible mode" possible. (E.g., partial failure of valves and operation of pumps below design flow rate have to be considered).

  The "worst credible mode" will be based on sound engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation

Several categories of failure type must be considered with regard to seismic effects on electrical sources (equipment and cabling). They are discussed below:

c.1 Electrical Equipment

c.1.1 Overturning of cabinets, transformers, switchgear or other unsupported equipment where the center of gravity location as measured from the base is longer than one-half the base width in all directions. Each direction will be evaluated independently.

The same considerations discussed in regard to overturning of mechanical equipment apply to electrical equipment, i.e., overturning is assumed only for cases where the distance to the center of gravity is significant compared to the base width.

c.1.2 All nonseismically qualified electrical equipment (except cable trays will be assumed to fail in the worst credible mode possible. The "worst mode failure" will be based on sound engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1 Evaluation of Sources (Cont'd)

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation (Cont'd)

c.1.3 All nonseismically supported electrical equipment (except raceways) will be assumed to be a source of the "worst possible" physical and electrical interaction.

c.2 Cable Trays

c.2.1 Seismically Supported Cable Trays

Cable trays that are determined to be seismically supported/restrained are assumed to remain physically intact in the event of an SSE (i.e., they do not become a source) and also that they will develop no electrical faults as built.

c.2.2 Non-Seismically Supported Cable Trays

A non-seismic cable tray in the vicinity of essential safety related equipment is to be a potential source and assumed to collapse. Also cables contained within the tray are assumed to develop electrical faults. The "vicinity" is defined by the criteria assumed and illustrated in Figure 6-1 & 6-2.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

c) Electrical Source Evaluation (Cont'd)

c.3 Conduits

Non-seismically supported/restrained conduits are assumed to be the source of mechanical and electrical interactions in an SSE.

d) HVAC Source Evaluation

d.1 Non-seismically supported ductwork that run directly over essential safety related targets will be considered a source of potential interaction. The interaction boundary envelope is illustrated in Figure 6-3.

d.2 While considering systems interaction of HVAC systems, the effects of ductwork crimping, adverse operation (or non-operation) of non-safety related fans that might spread combustible or toxic fumes through the ductwork has to be considered.

d.3 Failure of in-line HVAC equipment will follow the source evaluation criteria for Mechanical equipment. Support failure resulting in tipping, falling, sliding or overturning may occur. Overturning will be assumed possible when the distance as measured from the base to the center of gravity is more than one-half the width of the base. Each direction will be evaluated independently.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

e) Piping System Source Evaluation

High energy pipe rupture, jet impingement, flooding and internal missile analyses are not included in this seismically induced interaction assessment except in the cases where these effects are seismically induced.

All piping and asscoiated components identified as an essential safety related component fall under the category of targets. Also they are assumed to be seismically supported or restrained and hence will not become seismically induced souces.

Non-seismically designed piping will be considered as a source in the following context:

Physical Impact: All non-seismically designed/supported piping running in the vicinity of targets could fall or physically impact the target within the pipe's volume of influence. The volume of influence is defined as five (5) pipe diameters or five (5) feet whichever is greater, laterally from the pipe center line. The pipe is assumed to fail anywhere along the piping run, during or post SSE. This criteria is illustrated in Figure 6-4.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.1 (Cont'd)

e) Piping System Source Evaluation (Cont'd)

Flooding: A non-seismic piping run in the vicinity of target equipment will be assumed to have a circumferential or longitudinal rupture during or post SSE that could flood the room (attention must be paid to the instrumentation cabinets, motors, etc. in the room), flood any cable tray runs immediately above or below the piping run.

Environmental: Piping failures or a resulting chain interaction could cause unacceptable environmental conditions enveloping a target equipment, (e.g., auxiliary steam line failures could result in a steam environment with elevated temperatures and high humidity). Specific targets could either cease functioning or malfunction in this environment.

f) Instrumentation and Control, Source Evaluation

All instrumentation that is not seismically qualified will be assumed to malfunction in the "worst credible mode". Instrumentation that is not seismically mounted will be assumed to fail structurally and could becomes missile. The "worst credible mode" will be based on engineering judgement.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.2 The following criteria provide minimum guidance for evaluation of sources for pipe failure induced events

The criteria provided by the NRC Standard Review Plans 3.6.1 and 3.6.2 with companion Branch Technical Positions BTP APCSB 3-1 and MEB 3-1 were used to evaluate systems interactions associated with pipe failure induced events Table 6-1 summarizes the acceptance criteria for external and internal challenging events relative to the system, component or structure being evaluated.

6.2.1.3 The following criteria provide minimum guidance for evaluation of sources for missile (internally and externally) generated induced events.

The criteria provided by the NRC Standard Review Plans 3.5.1, 3.5.2 and 3.5.3 were used to evaluate systems interactions associated with the effects of internally and externally generated missile systems interactions. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system component or structure being evaluated.

6.2.1.4 The following criteria provide minimum guidance for evaluation of sources associated with flooding induced events.

The criteria provided by the NRC Standard Reveiw Plans 3.4.1 and 3.4.2 were used to evaluate adverse systems interactions associated with the effects of flooding. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system, component or structure being evaluated.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS

6.2.1.5 The following criteria provided minimum guidance for evaluation of sources resulting from the effects of fire induced events.

The criteria provided by the NRC Standard Review Plan 9.5.1 with companion Branch Technical Position BTP APCSB 9.5-1 were used to evaluate adverse systems interactions associated with the effects of fire. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system component or structure being evlauated.

6.2.1.6 The following criteria provide minimum guidance for evaluation of sources resulting from the effects of severe environment.

The criteria provided by the NRC Standard Review Plans 3.3, 3.4, 3.5, 3.6 and 3.11 were used to evaluate systems interactions resulting from severe environmental conditions. In addition the guidance provided by IE Bulleting 79-OIB was used to the degree practicable for this evaluation. Table 6-1 summarizes the acceptance criteria for challenging events relative to the system, components or structure being evaluated.

6.2.2 Modification Criteria

Modifications may be required to resolve identified event induced adverse systems interactions. These modifications may be any of the following:

a. Modification of the source to eliminate the adverse behavior by bracing, supporting, or reinforcing the source component.

b. Shielding or relocation of the target to preclude the physical interaction.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2 Modification Criteria (Cont'd)

c. Modification of the target to permit retention of the required safety function in spite of the interaction.

d. Alteration of system design to provide alternate means of accomplishing the safety function.

The criteria for structural or mechanical modifications are the same as documented for safety related structures and equipment.

For relocation or modification of non-safety related equipment, the criterion for acceptability is that the modified configuration, when re-evaluated for interactions using the evaluation criteria previously stated, is found to have resolved the original interaction and not created any new interactions.

6.2.2.1 Interaction Effects Evaluation Criteria

Once an interaction is identified as sufficiently credible to require more evaluation than can be done from inspection, it must be resolved in an acceptable manner and the resolution documented. Interactions considered are direct physical interactions such as target impact from a falling or moving source. Typical interactions are listed below.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Mechanical:

- impact from vibrating bodies
- impact from falling bodies
- pipe whip
- missiles

Electrical:

- unwanted open circuit (loss of control power)
- unwanted closed circuit
- unwanted energization

Pneumatic:

- loss of pressure (loss of control)
- unwanted pressurization
- jet impingement
- hostile gas

Hydraulic:

- loss of pressure
  - (a) loss of control
  - (b) loss of lubrication
- unwanted pressurization
- jet impingement
- flooding
- hostile fluids

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Environmental:

- elevated temperatures
- humidity
- radiation

Interactions are evaluated for their impact on the required safety functions and redundancy of identified targets. The results of the evaluation will then determine the method of resolution. In order of preference, the following are categories of acceptable methods of resolution of identified interactions.

a. Target Operability Evaluation:

The first approach to resolution is to show that the target's safety function is not impaired. This may be accomplished by studying the means by which impairment occurs and the possible extent of the impairment. For example, a pneumatically operated valve may be required to close during shutdown, but falling equipment could sever the air line so air supply to the operator is lost. If the valve is a "fail open" type, then shutdown capability is compromised, but if the valve is a "failed closed" type, then shutdown capability is not compromised even though the air supply is lost. In this example it is also necessary to consider consequences of crimping the air line, as well as the effect of a lost air line.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

a. Target Operability Evaluation: (Cont'd)

This example is typical of the reasoning process that is necessary in the evaluation of each interaction. A substantial degree of engineering judgement is, of necessity, expected to be used. Decisions based on judgement, along with the rationale, are documented.

b. Source Behavior Evaluation:

The second approach to resolution is to perform a more careful evaluation of the source behavior resulting from an event. If tests, analysis, or applicable experience can be developed to demonstrate that the item in question is qualified to withstand the postulated event, the interaction can be declared resolved on the basis that it will not credibly occur. Identification and resolution of indirect or chain-reaction source events shall use individual source failure criteria for each component source.

c. Modification:

If resolution is not possible by analysis or by test, the Interaction Team will recommend that physical modifications be made to prevent detrimental interaction. The range of possible modifications includes guard structures, protective covers, and restraining structures. The criterion is to prevent impairment of function.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

d. Change of Procedures:

The last method of resolution is by reordering the operating procedures or defining alternate means of providing the required safety functions. The Interaction Team will not specify procedural changes to resolve an adverser systems interaction, other than to present generic options.

The evaluation and resolution methods are discussed below in more detail.

Evaluation of Direct Interaction Effects

Where evaluation is directed to showing that the safety function of a target is not impaired by an identified direct interaction, the following guidance has been established. For cases not covered, criteria are developed and documented to provide an analagous level of rigor to the guidance herein provided.

a. Dynamic effects of breaks in piping are evaluated using the criteria in Section 6.2.1.2. For example one criterion to be used is that no damage will result if the target pipe size is at least equal to the size of the source pipe and the wall thickness of the target pipe is at least equal to that of the source pipe.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Direct Interaction Effects (Cont'd)

b. Direct impact of missiles or falling objects on structures and components are evaluated when necessary using the criteria of Sections 6.2.1.3. Care must be taken to consider such appurtenances as instruments, power connections, cooling and lubrication connections.

c. Direct impact of missiles or falling objects on HVAC ducts have to be evaluated on a case by case basis.

d. Flooding effects of broken or leaking pipes are evaluated using the criteria of Section 6.2.1.4.

e. The effects of fire are evaluated using the criteria of Section 6.2.1.5.

f. Environmental effects of broken or leaking piping, tanks, etc. are evaluated by comparison of the estimated environment with the target's qualification profile. Helpful criteria and data are contained in Section 6.2.1.6.

Evaluation of Secondary Effects or Cascading Influences

Two types of secondary effects on cascading influences are considered; chain-reaction failures and degraded operation.

6.0 EVALUATION CRITERIA (Cont'd)

6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Secondary Effects or Cascading Influences

For the chain-reaction events, the criteria for evaluation are the same as for the direct interactions and are successively applied to each member of the chain. It must be remembered that each step in chain scenarios has an associated probability less than one and that judgement must be applied to consider only credible scenarios.

In order for the plant to safely shut down, it is necessary for the required safe shutdown valves and drive elements to operate in the required manner, or fail in the required position. For this to occur, their control systems must remain intact after the interaction event, or else be damaged only in such a way to fail in the design failure mode. For example, if an air operated valve is required to fail in a certain mode, the design is such it will go to that failure mode on loss of air. If, however, the air line between the control device and the valve were to be impacted during a seismic event, the line might be pinched. This could prevent the venting of air and thereby prevent the valve from failing in its proper mode.

In electrically operated devices, a non-qualified component could impact the signal cable and cause damage which would adversely affect proper operation.

6.0 EVALUATION CRITERIA (Cont'd)

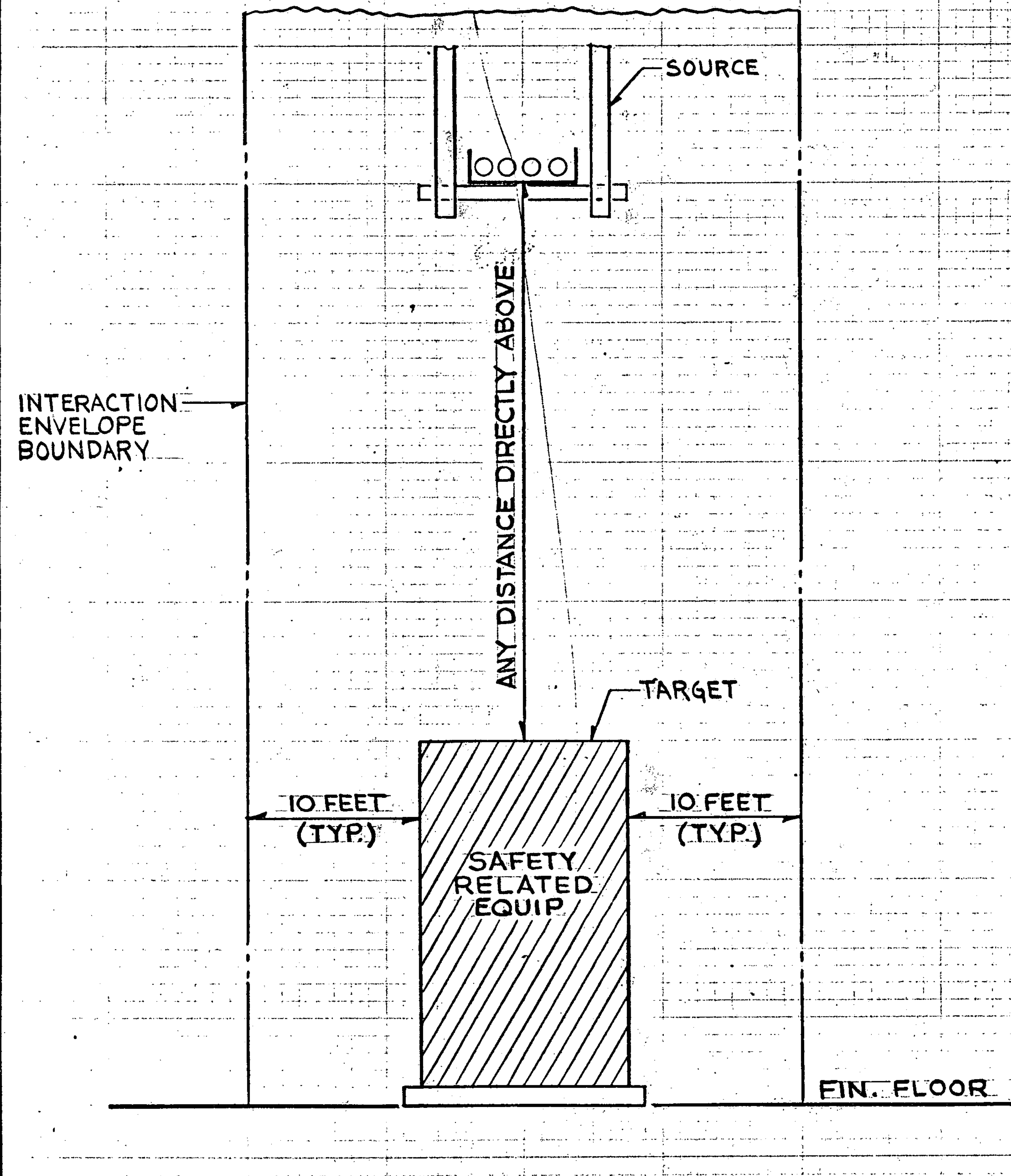
6.2 NONCONNECTED SYSTEMS (Cont'd)

6.2.2.1 Interaction Effects Evaluation Criteria (Cont'd)

Evaluation of Secondary Effects or Cascading Influences (Cont'd)

The walkdown will identify process tubing, instrumentation, electrical cables and cable trays requiring protection from unacceptable interactions.

When questionable secondary interactions are identified which are not readily evaluated to be acceptable, the resolution then becomes one of modification such as redesign or replacement of the source equipment or the rerouting or upgrading of control and electrical wiring and/or process and air tubing.



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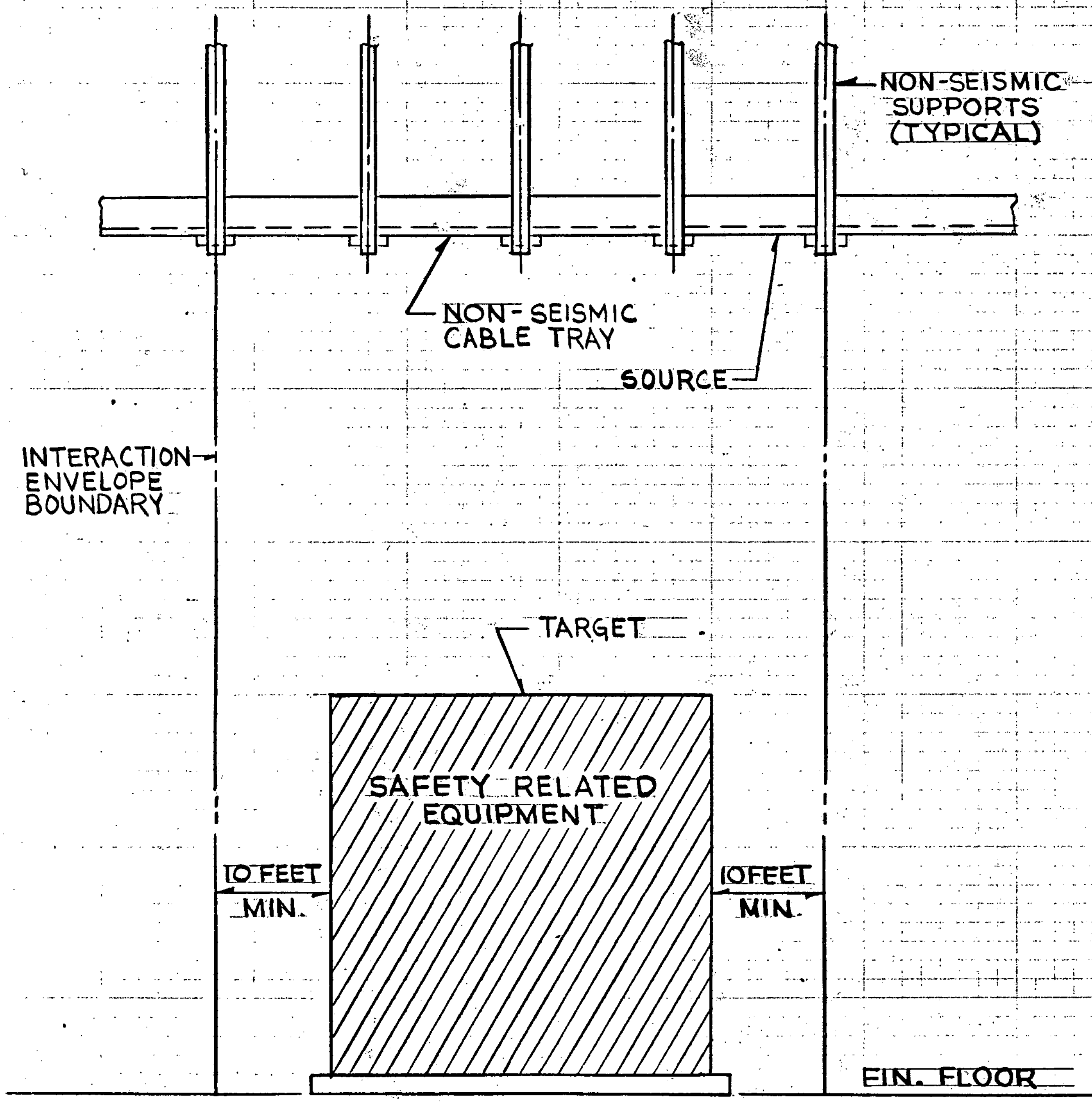
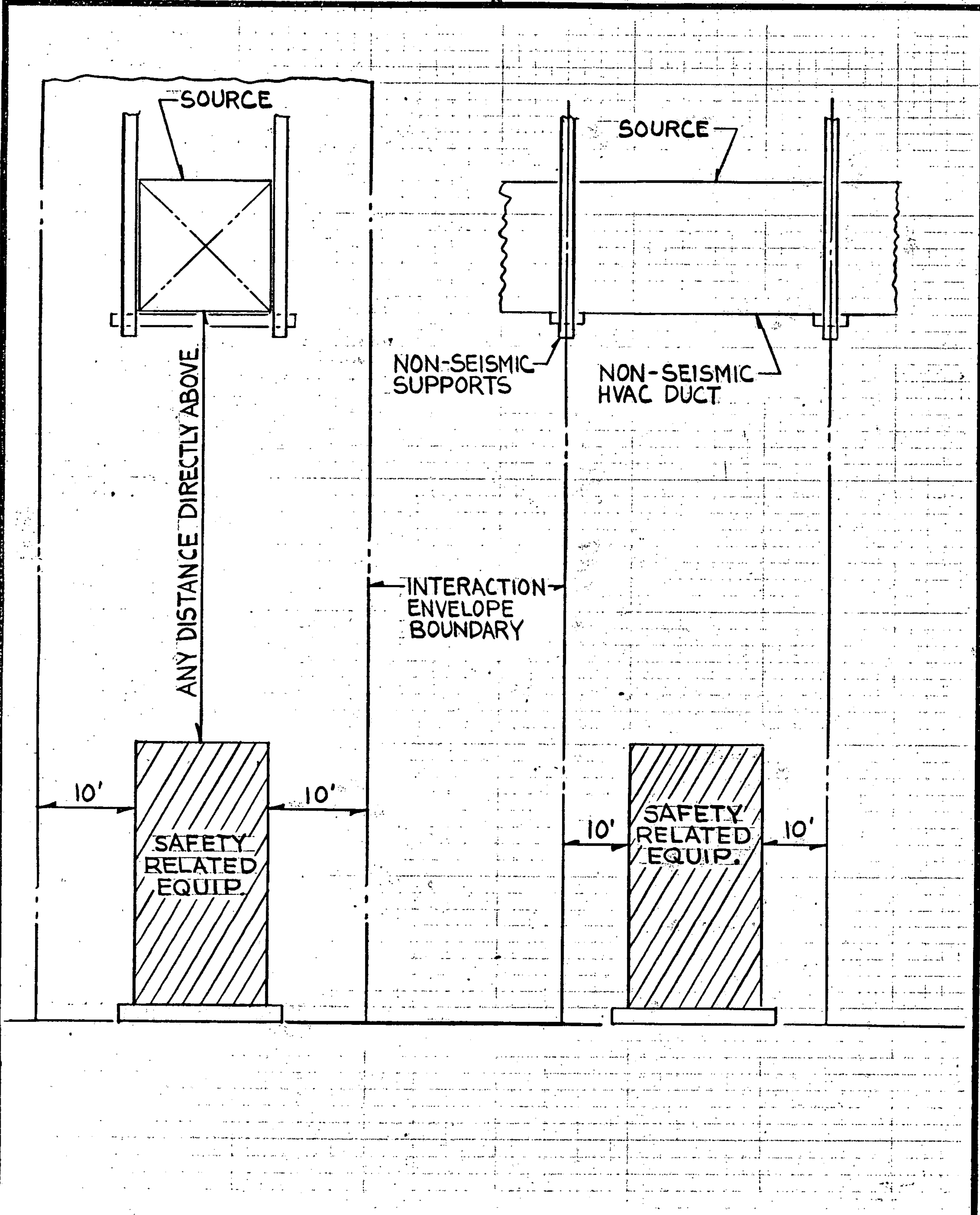
SCALE NONE

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT No. 3
NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY
INTERACTION CRITERIA
ELECTRICAL CABLE TRAYS

FIGURE 6-1



EBASCO SERVICES INCORPORATED	POWER AUTHORITY OF THE STATE OF NEW YORK INDIAN POINT No. 3 NUCLEAR POWER PLANT	FIGURE
DIV. MNE DR. JM DATE 3-20-81 CH. [illegible] SCALE NONE	SYSTEMS INTERACTION STUDY INTERACTION CRITERIA ELECTRICAL CABLE TRAYS	6-2



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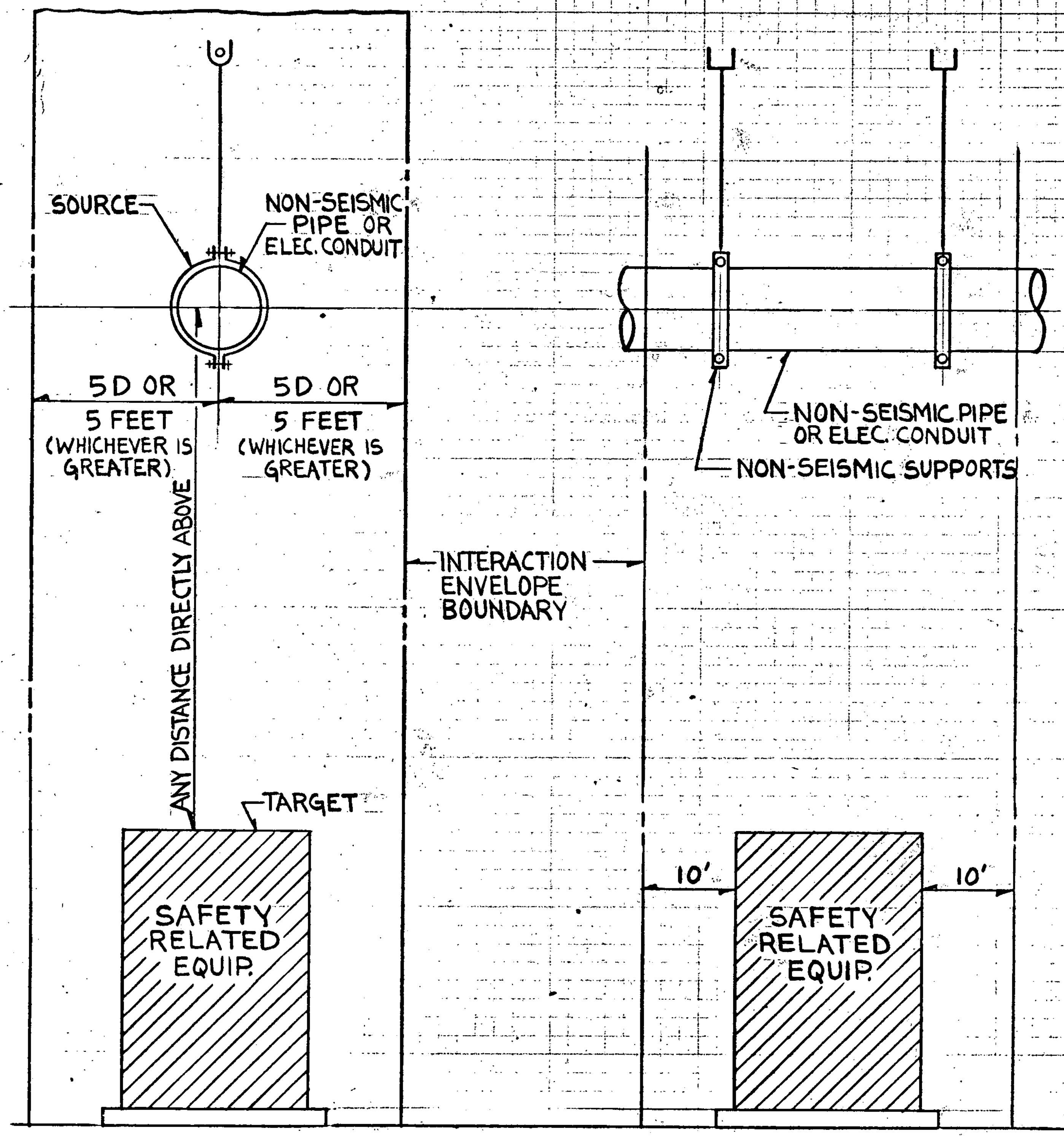
SYSTEMS INTERACTION STUDY
INTERACTION CRITERIA
HVAC DUCTWORK

FIGURE 6-3



EBASCO SERVICES INCORPORATED	POWER AUTHORITY OF THE STATE OF NEW YORK INDIAN POINT No. 3 NUCLEAR POWER PLANT SYSTEMS INTERACTION STUDY INTERACTION CRITERIA MECH PIPES OR ELECT CONDUITS	FIGURE 6-4
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INDIAN POINT NO. 3 NUCLEAR POWER PLANT
SYSTEMS INTERACTION STUDY

NON-CONNECTED SYSTEMS INTERACTION-EVALUATION CRITERIA
TABLE 6-1

Initiating Common Cause Event	Potential Follow-On Event	Acceptance Criteria Relative Location of Challenging Event Relative To System Component Being Evaluated: External	Internal	Corresponding Supporting Standard Review Plan/Regulatory Guide Criteria
A. Earthquake	a. Structural failure b. pipe failure c. flooding d. severe environment e. missile	See Chapter 6.0	See Chapter 6.0	10CFR part 50, Appendix A GDC 2 Regulatory Guide 1.29 "Seismic Design Classification" Standard Renew Plan 3.2.1, Seismic Classification
B. Pipe Failure	a. missile b. flooding c. severe environment d. structural	. structure, system, component capable of withstanding the resulting effects of pipe whip, jet impingement, flooding and severe environment	. Whip restraints . Barriers/Shields . Separation	Standard Review Plan 3.6.1/APCSB 3-1 Standard Review Plan 3.6.2/MEB 3-1 Regulatory Guide 1.46
C. Missile	a. pipe failure b. flooding c. fire d. severe environment e. structural failure	. structure, system, component capable of withstanding the resulting effects of pipe failure, flooding, fire, severe environment, impact	. Barriers/Shields . Separation .	Standard Review Plan 3.51 Standard Review Plan 3.52 Standard Review Plan 3.53
D. Flooding	a. severe environment b. structural failure	. structure/compartment designed to adequately prevent flooding entry	. Drainage system capable of handling maximum expected flood rate . Components capable of functioning in submerged (flooded) environment	Standard Review Plan 3.4.1 Standard Review Plan 3.4.2
E. Fire	a. severe environment	. Fire resistant construction . No communicating paths . Limited combustables	. Fire Detection and Suppression Systems . Limited Combustables	Standard Review Plant 9.5.1/ BTB APCSB 9.5-1
F. Severe Environment	a. flooding b. fire c. temperature d. humidity e. radiation f. wind g. missile h. depressurization	. structure/compartment capable of withstanding the resulting environmental condition . No communicating paths	. Equipment/component qualified to the resulting environmental condition . Compartment enviroment controlled	Standard Review Plan 3.3 Standard Review Plan 3.4 Standard Review Plan 3.5 Standard Review Plan 3.6 Standard Review Plan 3.11

INDIAN POINT No. 3 NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

CHAPTER 7

7.0 QUALITY ASSURANCE

In order to assure that the Systems Interaction Study project meets the requirements of the Quality Assurance Program, the Project Quality Assurance Engineer shall assign qualified internal auditors to review the inprocess activities of the SIS Project personnel. Results of these audits shall be distributed to the SIS Project Manager and the PQAE for information and corrective action if required.

Independent reviews will be performed by qualified personnel in accordance with Ebasco Procedure E-76 "Guidelines for Design Verification." Adherence to E-76 shall be verified by Quality Assurance Engineering. Records and data generated as a result of this study will also be reviewed by Quality Assurance to assure conformance to existing requirements

INDIAN POINT No. 3 NUCLEAR POWER PLANT

SYSTEMS INTERACTION STUDY

CHAPTER 8

8.0 References

1 - Zion/Indian Point Generating Unit No. 3, Probabilistic Risk Assessment, Pickard, Lowe and Garrick, Inc - DRAFT -

2 - Indian Point Station Unit No. 3 - System Description No. 21 Feed Water System, Revision 0, August 1975.

3 - Westinghouse Letter INT-80-51, dated August 21, 1980, Auxliliary Feedwater System, Flow Design Basis Responses

4 - Power Authority of the State of New York Indian Point No. 3 Nuclear Power Plant, SOP-FW-4, Rev 3 - Auxiliary Feedwater System Operation

5 - Line Schedule - Westinghouse Electric Corporation Nuclear Line Schedule Sheets 1-38, dated June 15, 1973.

6 - FSAR - Consolidated Edison Company of New York Indian Point Nuclear Generating Unit No. 3

7 - Review of the Indian Point Station Fire Protection Progran Volume I: Revision 1, April 1977.

8 - Safety Evaluation Report by the Director of Licensing U. S. Atomic Energy Commission in the Matter of Con Ed Company of New York, Inc. 1973.

9 - AFW Pump Building - AF Piping Sheets 1 & 2
UE&C DWG No. 9321-F-21263-13
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10 - Flow Diagram, Condensate & Boiler Feed Pumpt Suction, (W), UE & C DWG. No. 9321-F-20183-17 Rev. 17,

11 - Flow Diagram, Aux, Boiler Feed Pump Recirculation UE & C DWG. No. 6604-114-D-1, Rev. 0 May 24, 1979

12 - Flow Diagram - Boiler Feedwater UE & C DWG. No. 9321-F-20193-16, Rev. 16

CHAPTER 8 - REFERENCES, Cont'd

13 - Schematic and Wiring Diagram
Auxiliary BFP 31 & 33
Recirculation
Scale: None 6604-114-D-5
Final Issue Date: 8/23/79

14 - Instrument Block Diagram Int.
Indian Point Nuclear Power Plant
Unit No. 5
Reactor Protection & Control System
Westinghouse Electric Corporation Nuclear

Energy Systems Division
Pittsburgh, PA
Foxboro 9-35572 DWG NO. BD-11

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16 - Consolidated Edison Company
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18 - Consolidated Edison Company
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