File this instruction sheet in the front of Volume 1 as a record of changes.

The following information and check list are furnished as a guide for the insertion of new sheets for Amendment 22 into the Preliminary Safety Analysis Report for the Skagit/Hanford Nuclear Project. This material is denoted by use of the amendment date in the upper right hand corner of the page.

New sheets should be inserted as listed below:

Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)
CHAPTER 1	
Appendix 1B	Appendix 1B (reuse tabs)
CHAPTER 3	
3.8 - 8e/8f	3.8 - 8e/8f
CHAPTER 13	
Figure 13.1-3	Figure 13.1-3
Figure 13.1-4	Figure 13.1-4
CHAPTER 17	
17.1-i/ii	17.1-i/ii
17.1-1-3/4 through 17.1-9/10	17.1-3/4 through 17.1-9/10
17.1-13/14 through 17.1-19/20	17.1-13/14 through 17.1-19/20
17.1-25/26 through 17.1-37/38	17.1-25/26 through 17.1-37/38
17.1-43/44	17.1-43/44
Figure 17.1-2	Figure 17.1-2
17.2-1/2 through 17.2-5/blank	17.2-1/2 through 17.2-5/6
Appendix 17A	---------

APPENDIX 1B

COMMENTS RELATED TO REVIEW OF THE INCIDENT AT THREE MILE ISLAND UNIT TWO

APPENDIX 1B

The following pages identify the Applicant's commitments regarding the design, construction, and operation of the S/HNP in response to the review of the incident at Three Mile Island Unit 2.

Commitments in this Appendix supersede any conflicting statements elsewhere in the PSAR where such conflicting statements were made earlier than the date of the current revision of this appendix.

The following text consists of responses to NUREG-0718, Rev. 1, entitled "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", dated June, 1981. These responses meet the requirements of the proposed amendment to 10 CFR 50, entitled "Licensing Requirements for Pending Construction Permits and Manufacturing License Applications", dated July 10, 1981, as sent to all parties to pending construction permit proceedings by Generic Letter No. 81-26.

APPENDIX 1B

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APPENDIX 1B

TABLES

APPENDIX 1B

FIGURES

ITEM I.A.4.2 LONG-TERM TRAINING SIMULATOR UPGRADE

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

The use of a simulator will be an integral part of the S/HNP training program. Table I.A.4.2-1 provides an estimate of the manpower schedule to support operator training and assignment. The S/HNP license candidate training program will be typical of that defined in Appendix A of ANS 3.1-1978, "Standard for Selection and Training of Nuclear Power Plant Personnel." The S/HNP licensed operator training program will also meet the requirements of 10 CFR 55, "Operators Licenses." 21

The simulator used in the S/HNP operator training program, will meet RG 1.149, 4/81, "Nuclear Power Plant Simulator for Use in Operator Training," which meets the requirements of NUREG-0660 Item II.K.3.54. These requirements will be accomplished in a timely manner to support startup and operation.

The Black Fox Simulator represents a close approximation to the S/HNP control room, the principal differences consisting of S/HNP's Westinghouse main turbine generator and turbine driven feedpumps as compared to the General Electric supplied equipment used at the Black Fox Station. There are also anticipated to be some control layout differences associated with the long term response balance of plant bench board, reactor core cooling bench board and auxiliary panels. In the event that S/HNP elects to utilize the Black Fox Simulator for operator training, a detailed study of the differences that exist will be conducted. The study will show how the Black Fox Simulator can be successfully used to simulate the Westinghouse-supplied equipment. This study will also provide the base

for a supplementary training program utilizing the S/HNP full scale control room mockup to focus the knowledge gained during Black Fox Simulator training on the specific layout of the S/HNP control room. (The mockup is discussed under Item I.D.1, Control Room Design Review.)

The technique of (a) utilizing comparable equipment for dynamic training followed by (b) control location training on a static mockup and then (c) a period of actual operation under close supervision is a recognized training methodology. This is particularly true for the aircraft industry, where pilot training has demonstrated the validity of this methodology and provided a basis for the initial acceptance of simulator training for reactor operators. If selected, the integrated training program utilizing the Black Fox Simulator and the results of the study verifying the S/HNP-Black Fox Simulator similarity will be presented to the NRC for review prior to operator training. 21

The S/HNP will construct a Plant-unique simulator if either:

a. A Plant-unique simulator is justified on the basis of commercial considerations; or

b. The Black Fox Simulator is not found acceptable.

The Plant-unique simulator will meet NRC requirements for the similarity that must exist between the simulator and its reference plant.

Simulator option selection, the necessary NRC notification of the alternate selected, and the submittal for approval of the training program will be undertaken at the appropriate time in the S/HNP construction schedule. There are no concerns as to the technical details or feasibility of either of the operator training approaches described.

ITEM I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall submit a description of their administrative procedures for evaluating operating, design and construction experience and describe how they will assure that applicable important industry experiences originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant. Applicants shall submit a general discussion of how the requirements will be met. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experiences; (3) Identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; 21
(4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license."

RESPONSE

As the Applicant, Puget has the primary responsibility for assuring that applicable operating, design and construction experience is factored into the S/HNP. The Northwest Energy Services Company (NESCO) has been assigned to manage the design, procurement, fabrication and construction of the S/HNP with oversight by Puget.

NESCO, Bechtel Power Corporation and General Electric Company will each have administrative procedures for the evaluation of operating, design and construction experience. The procedures of each company complement and

overlap each other to assure that applicable industry experience is incorporated into the S/HNP. The following is a description of those procedures:

1. Organizational Responsibilities

Within NESCO, Nuclear Licensing & Safety (NL&S) is responsible for reviewing and categorizing the information received from outside the Project and identifying those experiences which may be of interest to S/HNP. NL&S is also responsible for categorizing these experiences such that operating, design and construction experience are directed to the respective organizations within NESCO and Puget for their review and use.

Design and construction experiences from within the Project are directed to the S/HNP Project Manager, who is responsible for reviewing, screening and directing the information to the S/HNP engineering organization, or Construction Manager, as appropriate, for action.

Bechtel and GE are responsible to NESCO for implementing feedback programs for design, operating, and construction experience within their respective organizations as described in the following sections. 21

2. Administrative and Technical Review Steps; and

3. Recipients of Information

a. General

Puget has contracted for design and construction of S/HNP with Bechtel and GE. As part of its responsibilities the General Electric Company has, within its Nuclear Services Department, established and maintained a formal service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance informations and recommendations. In addition, GE routinely reviews other available industry experience for applicability to the equipment and services it supplies for the S/HNP. Similarly Bechtel reviews available industry experience for applicability to the design, construction and other activities it provides for the S/HNP. In addition, NESCO is responsible to advise Bechtel and GE of operating design and construction

experience data uniquely available to NESCO such as from utility owners groups.

b. NESCO

NESCO functions within the program for review of operating, design and construction experience to: (a) review and approve Bechtel's and GE's programs, (b) audit and monitor Bechtel and GE implementation of their programs (c) furnish data uniquely available to NESCO or unlikely to be available to Bechtel and GE, and (d) provide direction to Bechtel for incorporating and implementing design and construction experience into the S/HNP design.

Operating, design, and construction experience information from external sources enters the NESCO program from two general categories: (1) Regulatory Agencies and (2) Industry Sources. Examples of documents reviewed are as follows:

(1) Regulatory Agency Information 21

- (a) License Event Reports
- (b) Regulatory Guides
- (c) Regulations (10 CFR and 49 CFR)
- (d) IE Bulletins, Circulars, Orders and Notices
- (e) NUREGs
- (f) Standard Review Plans (including Branch Technical Positions)

(2) Industry Sources

- (a) Topical Reports from GE and the Nuclear Safety Analysis Center
- (b) IEEE, ANS, ANSI, and ASME Codes and Standards
- (c) License Event Reports
- (d) NSAC/INPO Significant Events Evaluation Information Network Owners Group Activities
- (e) Owners Group Activities

As external information enters the NESCO system, it is directed to NL&S. There it is categorized, screened for applicability, and documented. The review at this stage is two-fold in purpose: (a) to reduce the quantity of information received to manageable amounts by culling out information

clearly not relevant to S/HNP, and (b) to broadly categorize the information into operations, design, or construction categories. NL&S transmits the information to either the Puget S/HNP operations organization, in the case of operational information, or the NESCO S/HNP Engineering Group, in the case of design information or the S/HNP Construction Manager, in the case of construction information, along with a specified time by which disposition of the items must be fed back to NL&S.

NESCO will provide continuous assessment of the efficacy of the experience feedback programs at NESCO, Bechtel and GE by using a commitment tracking system which will provide feedback to NL&S as to the ultimate resolution of the information that NL&S sends out to the Project. The same commitment tracking will be utilized for NESCO internally generated experience information.

Design Experience

For design experience, the S/HNP Engineering Group has the primary responsibility for resolving concerns once the information is received from NL&S. Engineering will review the information and direct it to the appropriate discipline leader for a determination of the necessary action. The engineer may consult with either or both Bechtel and GE to evaluate the concern. From this point, normal design control processes are used. 21

Construction Experience

For construction experience, the S/HNP Construction Manager will have the primary responsibility for resolving concerns once the information is received from NL&S. He may use assistance from S/HNP Engineering and either or both Bechtel and GE, as appropriate. Construction concerns that affect Plant design will be resolved in accordance with the Project's normal design process.

Operating Experience

Information on operating experience will be received from NL&S by Puget. The Plant Superintendent will perform a more detailed review of the information. This review will determine if the information should be factored into operations

planning activities or if it is of sufficient concern to pursue with Bechtel and GE. If warranted by the nature of the item, the Plant Superintendent will consult with the S/HNP Engineering Group, obtaining assistance as necessary, and recommend a course of action.

In some cases, the appropriate action will be decided without involving Bechtel or GE, particularly if it is in Plant maintenance or operations. The operational or maintenance concern may then be resolved as part of the normal process of operator training or procedures development.

c. Bechtel

Operations and Design Experience

Both on and off Project personnel have the responsibility for identifying and resolving design and operations feedback concerns. Sources utilized for feedback include:

(1) NRC Inspection and Enforcement Bulletins, Circulars and Notices 21

(2) Licensee Event Reports

(3) INPO/NSAC Significant Operating Experience Reports and Significant Event Reports

(4) Various internal Bechtel sources

Bechtel receives these documents through direct distribution, the Bechtel Licensing Information System, the Atomic Industrial Forum, etc. The focal point for this information is the Nuclear Discipline Licensing Group which reviews and distributes the information to the nuclear Project Engineers and Project Licensing Engineers within Bechtel.

The design discipline groups are responsible to determine the applicability of the concern to the S/HNP and for written disposition.

Significant experience feedback, if applicable, is also incorporated into generic engineering documents such as design standards, guides and specifications. These generic engineering

documents are utilized in developing Project-specific documents.

The Bechtel organization also reviews information transmitted to them by: GE on experience feedback from operating GE plants; NESCO; and other sources of unique experience.

Items applicable to the S/HNP will be resolved in the design or, if significant enough to warrant a NESCO decision on the resolution, submitted to NESCO for review and approval. Such submittals may be in the form of design documents submitted for NESCO review, studies, or correspondence. NESCO then reviews the Bechtel recommendation for disposition as described earlier.

Construction Experience

The Bechtel Construction Engineering Staff obtains construction experience data through reports from the field, review of I&E Bulletins, Circulars and Information Notices, and review of construction practices at the various sites. The significant experience data obtained from these sources is communicated to the Site to alert construction personnel to potential problems that may be encountered during the construction phase. 21

In addition, Project-level construction reviews are held to discuss and avoid problems that may have arisen during construction or as a result of feedback. Problem resolutions are incorporated in the various construction-related manuals and instructions.

d. General Electric

(1) Formal Advisory Service

The GE-Nuclear Services Department maintains a formal service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance information and recommendations. This system, implemented by the Service Information Letter (SIL), is designed to collect, process, and disseminate information pertinent to:

(a) unique operating conditions and experiences

(b) improved methods, techniques and procedures for operating and maintaining BWR plant equipment

(c) plant performance improvement and equipment upgrading

(d) safety, licensing and other regulatory matters

The major sources of information, including data, drawings, equipment, catalog/part numbers, problem definition, technical work recommendations, and other technical material required to prepare SILs include:

(a) Application Information Documents (AIDs)

(b) Field Engineering Memos (FEMs)

(c) Product Experience Reports (PERs) 21

(d) Safety and Licensing Reports

(e) Reports and Instructions prepared by GE engineering organizations

(f) GE and Vendor Equipment Instruction Manuals

(g) Equipment Failure and Reliability Reports

(h) BWR Plant Owner-Operator(s) and utility management suggestions

(i) Startup and Preoperational Test Reports

Occasionally, a need may arise to transmit to the utility owners with operating BWRs an urgent announcement of a potential operational hazard or other information which could seriously impact plant operations. In general, such announcements will consist of a brief but adequate explanation of the situation with advice or precautionary measures to be observed.

Prior to release from GE-Nuclear Services Department, SILs will undergo formal review by the responsible design engineer, other cognizant engineers, and GE management representing various disciplines including engineering, startup testing, licensing, and services.

(2) NRC Information

Information received from the Nuclear Regulatory Commission falls into the following categories:

(a) I&E Bulletins, Circulars and Information Notices

(b) NUREGs, Regulatory Guides and SRPs

I&E documents are received by one individual within the GE licensing department, who reviews and routes it to the proper unit within the department. In turn, that particular unit will review and communicate with each project to which that information may be applicable. 21

NUREGs, Regulatory Guides and SRPs are received directly from the NRC distribution list by the following organizations within the GE licensing department:

(a) Standardization
(b) Operating Reactor Services
(c) BWR Project Licensing
(d) BWR System Licensing
(e) Washington Liaison Office

Each organization reviews the documents received and cross communicates within to disseminate the data to the proper individual within the licensing organization. At that time all Project Managers are made aware of the information if the particular project is affected.

(3) Field Information

Within the General Electric Nuclear Division, all systems are assigned a Lead System Engineer with the prime responsibility for

that particular system. If at any time, a problem is encountered in the field by the A/E or GE field representatives, GE personnel will write a Field Deviation Disposition Report (FDDR) describing in detail the problems encountered. At the same time, that report may suggest a solution which is transmitted back to the GE Lead System Engineer in San Jose. That particular Lead Engineer will review the FDDR for its application. If it is a generic problem, an Engineering Change Authorization (ECA) will be written for review and approval. If the ECA is approved, then an Engineering Change Notice (ECN) will be issued to all projects to correct the problem. If the Lead System Engineer finds that the problem is only applicable to a certain project, the same procedure described above will take place but only the specific project management will be notified. Applicable ECNs & ECAs are transmitted to the S/HNP at NESCO.

4. Avoidance of Extraneous and Unimportant Information; and 21

5. Avoidance of Conflicting or Contradictory Information

Within NESCO, NL&S will assure the avoidance of extraneous and unimportant information through its normal screening process.

Within NESCO, NL&S will assure that potentially conflicting or contradictory information is identified and transmitted to the appropriate organization for resolution.

Within Bechtel the Nuclear Licensing Group, through its normal screening process, will assure the avoidance of extraneous and unimportant information.

6. Practical Interim Audits

NESCO will assure compliance with these requirements by monitoring and periodic audits of NESCO, Bechtel, and GE implementation of their programs. NESCO audits the implementation of experience feedback as part of their auditing of quality-related design and construction activities at NESCO and at GE and Bechtel.

ITEM I.C.9 LONG-TERM PROGRAM PLAN FOR UPGRADING PROCEDURES

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe their program plan, which is to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Puget will establish a program for the development of Plant operating procedures during the construction period. The plan for this program will be developed within two years
following receipt of a Construction Permit. This program 21
plan will be developed and implemented by the Plant Superintendent who will also be responsible for training of operators.

All Plant procedures described in Section 13.5 of the PSAR will be reviewed and approved by the Plant Operations Review Committee (PORC) or subcommittees of PORC. These subcommittees will be structured to ensure that the Plant Superintendent and Technical Supervisor will review all procedures with additional reviewers depending on the type of procedure. For example, the Operations Supervisor would be required to review and approve procedures involving operations, surveillance testing, etc. Specific requirements governing the review and approval of Plant procedures will be included in the S/HNP FSAR and Technical Specifications.

Development of the program for preparation of operating procedures will include, but not be limited to, consideration of the following:

1. Incorporation of the pertinent results from the human
   factors review of the control room described in Item 22
   I.D.1.

2. The results of the probabilistic risk studies, described in the response to Item II.B.8(1). 22

3. Results from applicable portions of generic efforts on procedures, such as those being sponsored by the BWR Owners Group and currently underway, efforts by INPO, or other applicable industry activities that may become available. Emergency procedure improvements will follow closely the efforts of the BWR Owners Group Emergency Procedures Guidelines.

4. Evolving NRC requirements, such as the requirements in NUREG-0737, Item I.C.1 currently being applied to operating procedures for operating plants and applicants for operating licenses.

5. Scheduling procedures development to support operator training, including the training of operators during preoperational testing of completed systems, with 21 plant-specific procedures. (The use of procedures during the preoperational testing program is discussed further in S/HNP PSAR Section 14.1.3.4).

6. Development of suitable analytical bases for procedures. The emergency procedures for training will be documented with references that identify the analytical or technical bases that demonstrate conformance to the BWR plant safety requirements.

7. Results of the on-going operating experience evaluation program described in the response to Item I.C.5.

8. Crisis management techniques will be applied to emergency procedures governing the interaction of control room activities with those in the Operations 22 Support Center, the Technical Support Center and the NRC Operations Center. Control room occupancy and shift turnover procedures will include consideration of emergency and crisis conditions.

ITEM I.D.1 CONTROL ROOM DESIGN REVIEWS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that comply with human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs is unacceptable. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs reflecting human factor principles prior to issuance of a CP or ML and shall supply design information for review prior to committing to fabrication or revision of fabricated control room panels and layouts."

RESPONSE 21

The response to this question is arranged in the order suggested in NUREG-0718, Rev. 1, requirement. The sections are:

a. General Discussion of Approach to Control Room Design.

b. Design Concept is Technically Feasible, Within the State of the Art and Will be Properly Implemented.

c. Commitment to Control Room Designs Complying with Human Factors Principles.

d. Summary of Control Room Design: this section provides general design information. 22

1. General Discussion of Approach to Control Room Design

a. Human Factor Principles Design Concept

The S/HNP was initiated with a strong commitment to incorporating human factor principles into the control room design. A further desire to utilize state-of-the-art concepts led to the selection of the General Electric Nuclenet 1000 Advanced Control Room Option. The S/HNP 22

Control Room is an advanced design control room utilizing CRTs and computers, and was designed after a full system analysis consistent with the philosophy of Appendix B of NUREG-0659.

The systems/operations analysis process by which the generic Nuclenet 1000 design was developed included an analysis of all functions necessary to safely operate the plant, an allocation of functions between operator and machine and a qualitative verification of the functional allocation. Detailed information on GE's methodology in developing the Nuclenet 1000 Advanced Control Room closely following the recommendations of NUREG-0659, Appendix B, has been presented to the NRC in the Allens Creek Nuclear Generating Station PSAR, Docket No. 50-466, Amendment 59, Item I.D.1, Part 1, and is included herein by reference. 22

In summary, the underlying human factor concepts implemented in the S/HNP Nuclenet 1000 Control Room are:

(1) Provide a more efficient, coordinated control of the BWR than that attained with a conventional control room. 21

(2) Integrate planned operations functions into a single operator's station for steam supply and power conversion systems.

(3) Optimize the quantity of data and number of display devices the operator must continuously survey, analyze, and comprehend with the goal of improving response time and reducing operator errors.

(4) Optimize, centralize and integrate the control devices and the number which the operator must routinely manipulate.

(5) Incorporate efficient hardware and software display techniques to present timely, useful information which is meaningful to the operator.

(6) Design and arrange the control room to permit one operator to perform planned operations and emergency shutdowns.

NUREG-0659 (March, 1981), "Staff Supplement to the Draft Report on Human Engineering Guide to ControlRoom Evaluation," provides a listing of ten major topics of human engineering to be considered in control room design. These ten topics are, upon examination, a restructuring of the six human factors concepts listed above which were applied to the S/HNP control room from its first beginnings in 1976. Table I.D.1-1 shows the principal relationships between these six human factor concepts, as applied by S/HNP, and the ten human engineering topics of NUREG-0659.

In addition to the principal relation-ships shown in Table I.D.1-1, there is an overall human factors influence in each of the areas. As a result, the S/HNP Nuclenet 1000 Advanced Control Room design incorporates the preferred human factor features.

b. Supporting Design Bases and Criteria 21

The principal human engineering bases and criteria established to guide the design of the S/HNP Nuclenet 1000 control room may be generally stated as follows:

(1) Insofar as possible, the controls, information displays and alarm displays shall be separated and grouped to enhance the operator interface involved.

(2) All controls and displays which form the operator interface with a given power system or subsystem shall be logically grouped.

(3) Control functions shall be arranged in such a manner as to present to the operator either an order of sequence activity or represent a two-dimensional layout of the system similar to the appropriate drawing.

(4) The reactor core cooling and BOP control boards shall provide for optimum convenience of a standing, mobile operator with control functions oriented for comfort of manual activities and accurate interpretation of displays.

(5) Control room panels shall be designed and arranged to ensure operability and

maintainability based on anthropometric considerations.

(6) Provision shall be made within the control room area for writing surface and storage space for reference material.

(7) The control room panel layout should be such that a single operator can survey and perform related control functions with a minimum walking motion.

These principal criteria conform to the guidance provided by NUREG-0659.

2. <u>Design Concept is Technically Feasible, within the State of the Art, and Will Be Properly Implemented</u>

a. <u>Technically Feasible</u>

The Nuclenet 1000 control room concept was 21
reviewed by the NRC through the following documents.

(1) GE Topical Report NEDO-10939, "Design Criteria and Technical Description of Plant Operator Interface of the Nuclenet 1000 Control Complex"

(2) GESSAR-251 Nuclear Steam Supply System Standard Design Docket No. STN 50-531

(3) Skagit Nuclear Power Project PSAR

The NRC staff's concurrence with the technical feasibility of the Nuclenet control room has been documented in the Safety Evaluation Report NUREG-0151. The technical feasibility has also been confirmed by the successful startup of the Black Fox Simulator and the progress and reviews conducted of other projects utilizing the Nuclenet control room concept.

b. <u>State of the Art</u>

As previously described, the General Electric Nuclenet 1000 Advanced Control Room represents the most advanced control room design available for the BWR6. The extensive use of computer generated displays and the priority given to human factor

engineering assures the S/HNP of an advanced state-of-the-art control room.

c. <u>Proper Implementation of the Nuclenet Control Room</u>

In order to assure the proper implementation of the Nuclenet Advanced Control Room design, the S/HNP initiated four major activities concurrent with the design effort. First, a Puget, Bechtel, and GE Control Room Design Task Force was formed. Second, a Control Room Design Review Team was formed. Third, a full scale control room mockup of the operator interface panels was constructed. Fourth, active membership in appropriate BWR owners groups was maintained, and significant industry meetings and conferences were attended.

The Control Room Design Task Force was charged with the responsibility to assure that the GE Nuclenet 1000 Advanced Control Room design was properly integrated into the Skagit control room design. Particular emphasis was placed on the preservation of all human factor design considerations. When the design work was initiated, the Control Room Design Review Team was formed. 21

The Control Room Design Review Team recognized the iterative nature of the task and the importance of an integrated functional review. The first action was the review of Plant system drawings, such as piping and instrument drawings, with the respective panel drawings so that the controls and instrumentation shown reflected the actual system design. After the necessary corrections were made, the panel insert drawings were redrawn to actual full scale with all labels included to give a true perspective. With this set of corrected drawings serving as a data base, a detailed system analysis was initiated to ensure that there were both sufficient and efficient controls and instrumentation.

This detailed review process began in October, 1976. The Skagit Control Room Review Team was directed to: (1) achieve the optimum panel arrangement of instruments and controls; (2) achieve an integrated review of the various systems interfacing with the control room operator; (3) effectively utilize the multi-discipline experience of the review team; and, (4) verify that the functional and operational

sequences that must be performed by an operator could proceed in an efficient and logical fashion.

As the review progressed, the need for a control room mockup consisting of panels forming the "horseshoe" (P680, P601, P877, P863, P800, P870, P682, P678 and P681) to achieve the review team goals was realized. In mid-1977 the decision to construct the mockup was made. The control room panels were built to full scale and coated with a metallic-based paint which provided a surface closely resembling both the color and texture of the surface of the actual panels to be used in the Skagit Plant. The full scale control and instrument layout drawings were trimmed and affixed in the appropriate locations on the control room panels. Fabrication was completed in the fall of 1977 and the control room was set up in Puget's Offices with the correct panel arrangement completing the establishment of the full scale control room mockup.

The control room mockup became the primary design review tool assuring that all aspects of the man-machine interface were included in the review process. Subsequent working sessions of the combined review team were held within the control room confines in Puget's Offices, to reinforce this focus. The review process continued to a close out meeting in January, 1980, when engineering was deferred. During the course of the review the organizations active in the design work also participated in the review effort. This close interaction assured that the reviewers clearly understood the functionality and limitations involved, and, the designers were fully aware of the reviewers' concerns. Since the control and instrument layout on the panels was provided by full scale drawings, it was possible to use the panels as a form of active three dimensional checklist for the reviewers. On occasions when the reviewers determined that arrangements were not yet in a logical sequence or mimics were misleading or incorrect, system and panel designers were able to make preliminary hand markups. Operational analysis was used to review both the assignment of the various system controls and instruments to a particular front or back row panel, and the arrangement of the system devices on the panel selected. Where the sequence or operational functions between systems were found 21

to be inefficient or potentially leading to operator error, the arrangement could be readily restructured. These changes and other Review Team comments were then investigated by the appropriate design group and responses prepared for the subsequent review sessions. When the review of a particular section was complete, the in-place markups and rearrangements became the source for the drawing revision. The revised drawings were placed on the control room mockup allowing the control room design review to proceed with all previous work fully integrated.

The Skagit Control Room Review Team member disciplines included Nuclear Engineering, Human Factors Engineering, Systems Analysis, Operations Research, Architectural Engineering, and Senior Reactor Operations; this membership is consistent with that recommended for a review team by NUREG-0659. The review team followed a review process based on system analysis and human factor considerations. The individuals participating in the control room design reviews participated in 21
owners group and industry activities, assuring that the broad experience base existing and developing in the industry was factored into the development of the Skagit control room design.

This review, utilizing the full scale control room mockup, has finalized the control panel arrangement and system to panel assignment provided below.

Control room instrument and control panel insert details are not yet finalized. The Skagit Control Room design review process described above will resume when the S/HNP design effort is activated and will be utilized to assure the proper implementation of the Skagit Nuclenet Advanced Control Room design at this final level of detail.

3. Commitment to Control Room Designs Complying with Human Factors Principles

The S/HNP commitment to incorporate human factors engineering into the design of the control room is described above. The S/HNP believes this commitment 22
and the review work performed satisfies the requirements of NUREG-0718, Rev. 1, and the guidelines presented in NUREG-0659. The design information presented below summarizes the status of the control

room design. The remaining design work, which includes the associated human factors review, focuses on the detailed layout of the controls and instruments in the inserts associated with the control room operator interface panels. This human factors review will be done in accordance with the guidelines and requirements applicable at the time the detailed final design process is resumed. (It is anticipated that these review guidelines will be issued as NUREG-0700.) This design effort will resume after the construction permit is issued, and the requisite review of the control room panel inserts will be completed prior to committing to fabrication of the inserts. 22

4. Summary of Control Room Design

a. The general control room arrangement utilizes the preferred horseshoe pattern as shown in Figure I.D.1-1. The Nuclenet console is at the central location with the Reactor Core Cooling Benchboard on the left and the Balance of Plant Long Term Response Benchboard on the right. The Auxiliary Long Term Response Panel and HVAC Control Panel are located at the back of the control area. Figure III.A.1.2-1 in the response to Item III.A.1.2 shows the orientation of the control room with the related support areas. 21

b. Panel configuration is generic to the BWR6 Nuclenet Control Room and meets the human factor anthropometric needs. The particular panel designs are as follows:

(1) Nuclear Control Console P680, Views A & B (Figure I.D.1-1).

This console is designed for operation by a seated operator. Consequently it incorporates a low profile with controls placed comfortably within the operators reach on the lower portion and indications positioned in the arc best used for visual scanning.

System Assignments:

Nuclenet Control Console P680
- Reactor Water Cleanup
- Condensate Pumping
- Feedwater Pumping
- Level Control
- Reactor Recirculation
- Rod Control and Information

Neutron Monitoring
Reactor Protection
Containment and Reactor Vessel Manual Isolation Initiation
Steam Bypass and Pressure Regulation
Turbine Controls
Generator and Main Transformer
Performance Monitoring System
Display Control System

(2) Reactor Core Cooling Benchboard P601, Diesel Generator Control Board P877, Balance of Plant Benchboard P870, View C (Figure I.D.1-1).

These benchboards are intended for use by a standing, mobile operator and consequently use a tall profile. The uppermost portion is sloped forward for ease in viewing and the vertical portion is located within the comfortable viewing arc, and easy control reach. The sloped lower portion provides comfort in control manipulation as well as ease in viewing from a standing position. 21

System Assignments:

Reactor Core Cooling Benchboard P601
High Pressure Core Spray
Low Pressure Core Spray
Residual Heat Removal
Reactor Core Isolation Cooling
Main Steam Pressure Relief
Main Steam Isolation
Main Steam Drain
Control Rod Drive Hydraulics
Standby Liquid Control System
Containment and Reactor Isolation
RHR Service Water
Drywell and Containment Sumps
Reactor Head Vent

Diesel Generator Control Board P877
Standby Power System

Balance of Plant Benchboard P870
Auxiliary Electrical System
Turbine and Generator Auxiliaries
Condenser Air Removal
Steam Seal System
Condensate System
Reactor Feedpump Auxiliaries

Moisture Separate Reheater
Extraction Steam
Feedwater Heater Drain Pump System
Feedwater Heaters

(3) Other Control Room Panels

The other panels that are utilized in the control room area are the Standby Information Panel P678, Post Accident Monitoring Panels P681 and P682, Auxiliary Long Term Response Panel P800, HVAC Control Panel P863 and the back row panels. These panels are of a vertical face configuration. This is the optimum configuration for the utilization of these panels providing the maximum flexibility for functional grouping of instruments and controls.

System Assignments:

Standby Information Panel P678

Post Accident Monitoring P681, P682

Auxiliary Long Term Response P800 21

Service Water System
Turbine Building Closed Cooling Water
Reactor Component Cooling Water
Condensate Transfer System Containment Isolation
Instrument Air Containment Isolation
Service Air Containment Isolation
Fire Protection Containment Isolation
Demineralized Water Containment Isolation
Circulating Water System
Cooling Tower System
Raw Water System

HVAC Control Panel P863

Auxiliary Building HVAC
Pump Room and Steam Tunnel HVAC
Containment Building HVAC
Drywell HVAC
Fuel Building HVAC
Diesel Generator Rooms HVAC
Standby Service Water Pump Rooms HVAC
Control Room HVAC
Standby Gas Treatment System

Back Row Panels

Assignments are shown on Figure I.D.1-1. 21

c. The detailed layout of the individual controls and instruments is not yet final. The design finalization and human factors review will be performed as described above. The final insert design consisting of insert location and layout will be provided for NRC review prior to insert fabrication. 22

In addition, any changes to the control room panels and arrangement described herein will be provided to the NRC for review prior to committing to fabrication or revision of fabricated panels or arrangements.

ITEM I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for a plant safety parameter display console. The console shall display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE 21

The S/HNP will review the available NRC approved Safety Parameter Display Systems (SPDS) applicable to the BWR6 design after issuance of the S/HNP Construction Permit and will select a system, utilizing the guidelines of NUREG-0696 (February, 1981) at the final design stage. The SPDS selected will be a computer-based system of high quality and reliability and will be capable of displaying the full range of important Plant parameters. The SPDS will be capable of functioning properly in the environments that are present during transient and accident conditions. Human factor engineering will be incorporated into the SPDS design to enhance the ability of control room personnel in evaluating the safety status of the Plant. Displays will be as simple as possible and will incorporate human factor considerations into grouping of parameters, patterns and coding techniques to assist the operator in detection of unsafe operating conditions. Time-rate-of-change for selected parameters will be provided through trending or derivation in a manner designed to both optimize operator-process communication and allow flexibility in the selection of variables for display. The system will also indicate when Plant parameters are approaching or exceeding process limits.

The SPDS will contain the minimum parameter set from which Plant safety status can be determined quickly and accurately at a single display location. This set of parameters will incorporate those determined by the BWR Owners Group when approved by the NRC. The Plant functions presented will include, but not be limited to: reactivity control; reactor core cooling and heat removal from the primary system; reactor coolant system integrity; radioactivity control and containment integrity.

The SPDS will be designed in accordance with NUREG-0696 (February, 1981) and will be located within the control room area based on an analysis of the operator's needs and a functional analysis of the use of the SPDS. This analysis will be an integral part of the final selection of the SPDS design. SPDS displays will also be provided in the Technical Support Center (TSC) and the Emergency Operations Facility (EOF). 21

The S/HNP has no concerns regarding the technical feasibility of meeting the requirements of NUREG-0696, and there are no concerns as to the ability to implement the SPDS design prior to OL issuance.

ITEM I.D.3 SAFETY SYSTEM STATUS MONITORING

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe how their design conforms to Regulatory Guide 1.47, 'Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems'. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be 21
implemented properly prior to the issuance of operating licenses."

RESPONSE

The S/HNP design includes automatic indication of the bypassed and inoperable status of safety systems in conformance with Regulatory Guide 1.47 as indicated in PSAR Appendix 3A. The basic design is described in GESSAR Sections 7.2.2.2.2.2, 7.3.2.1.2, and 7.3.2.2.3.7, which address GE-designed safety system status monitoring in accordance with Regulatory Guide 1.47. Bypassed and inoperable status indication for non-GE balance of plant safety systems will use the same design as described for GE safety systems.

Systems covered by Regulatory Guide 1.47 are shown on PSAR Figure 7.1-2. To the extent practical, inputs to the Safety System Status Monitoring System will be direct measurements of the desired variables.

ITEM I.F.1 EXPAND QA LIST

NUREG 0718, REV. 1, REQUIREMENT

"Prior to issuance of the construction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR 50, and shall provide a commitment to apply the revised QA program to all such items and activities."

Response

The QA list includes all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR 50 as described in the S/HNP PSAR in Appendix 3A and Section 3.1.

Bechtel Project Engineering is responsible for preparation and maintenance of the Q-List Each revision of the Q-List contains the issue date, approval date, and authorized signature. The Q-List and revisions thereto require the approval of the Bechtel Project Engineer and Chief Nuclear Engineer. For items that fall within the GE Scope of Work, input to the Q-List will be implemented based on GE recommendations. All changes to the Q-List are reviewed and approved by the S/HNP Principal Engineer. 21

The test, assuming single failure in the installed safety system in accordance with the single failure criterion, for the safety-grade functions, i.e., those portions of structures, systems or components whose failure could reduce equipment performance to an unacceptable safety level is as follows:

a. Will the failure or off-normal operation of the non-safety system or component degrade the capability of installed safety systems such that those safety systems cannot mitigate accident consequences and assure adequate safety.

b. Will the effects of failure or off-normal operation of the non-safety system or component exceed the capability of installed safety systems to mitigate accident consequences and assure adequate safety, if installed safety systems are operated properly so that full credit can be taken for their functioning to design capability throughout the accident sequence.

c. Is the non-safety system or component that may be called upon actually required to mitigate accident consequences and assure adequate safety, if installed safety systems are operated properly so that full credit can be taken for their functioning to design capability throughout the accident sequence.

If the answer to any of these questions in all of its aspects is affirmative, then:

a. the system or component in question would be upgraded to safety-grade, or

b. the design of the facility and/or the capability of the existing systems would be improved such that the answer is negative to all three questions.

The existing S/HNP Q-List includes those safety-grade items meeting the above criteria. The Q-List may be expanded as a result of ongoing activities related to the TMI-2 event. Any such items (e.g., hydrogen control, additional post accident monitoring systems) will be added to the list at the appropriate time using existing procedures. 21

To add further verification to the Q-List, it will be examined using systems analysis techniques. The systems or components identified by the systems analysis as safety-related system requirements will be checked against the existing Q-List and modifications to the Q-List will be made as appropriate. The systems analysis study is described below.

The systems analysis is performed to provide a systematic classification of components by examining Plant events by frequency of occurrence, radiological impacts, and allowable limits of the safety criteria.

The systems analysis is constructed by first defining categories of Plant operation and potential events in each Plant operating category. The events are ordered by frequency of occurrence and unacceptable safety criteria are established according to the expected frequency of occurrence.

For planned (normal) operation, the unacceptable results criteria are:

a. Release of radioactive material to the environs that exceeds the limits of either 10 CFP 20 or 10 CFR 50.

b. Fuel failure to such an extent that if the freed fission products were released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR 20 would be exceeded.

c. Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.

d. Existence of a Plant condition not considered by Plant safety analysis.

For anticipated (expected) operational transients, moderate probability of occurrence - once per day to once in 20 years - the unacceptable results criteria are:

a. Release of radioactive material to the environs that exceeds the limits of 10 CFR 20.

b. Any fuel failure calculated as a direct result of the transient analyses. 21

c. Nuclear system stresses exceeding those allowed for transients by applicable industry codes.

d. Containment stresses exceeding those allowed for transients by applicable industry codes when containment is required.

For abnormal (unexpected) operational transients, less than one event in 20 years to one in 100 years, the unacceptable results criteria are:

a. Radioactive material release exceeding the guidelines values of a small fraction of 10 CFR 100.

b. Failure of the fuel barrier as a result of exceeding mechanical or thermal limits (Failure means gross core-wide fuel cladding perforations).

c. Nuclear system stresses exceeding those allowed for transients by applicable industry codes.

d. Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

For design basis (postulated) accidents, low probability events - once in 100 years to once in 10,000 years - the unacceptable results criteria are:

a. Radioactive material release exceeding the guidelines values of 10 CFR 100.

b. Failure of the fuel barrier as a result of exceeding mechanical or thermal limits. Failure includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalphy (control rod drop accident).

c. Nuclear system stresses exceeding those allowed for accidents by applicable industry codes.

d. Containment stresses exceeding those allowed for accidents by applicable industry codes when containment is required.

e. Plant main control room personnel overexposure to radiation.

Nuclear safety operational requirements are diagrammed for each event to obtain minimum acceptable results and identify those systems required to function. The systems required to function become, by definition, systems important to safety. By inspection of Protection Sequence Diagrams (described below) those systems required to function will be determined and the requirements for satisfaction of single failure criteria observed. 21

Four operating states are identified in order to establish initial conditions of each protection system sequence analysis. The four states are: (a) reactor shutdown, vessel head off (b) reactor not shutdown, vessel head off (c) reactor shutdown, vessel head on, and (d) reactor not shutdown and vessel head is on. For each state, required safety actions are defined to assure adequate control. For example, in state (d) the required safety actions are as follows:

Radioactive material release control
Core coolant flow rate control
Core power level control
Core neutron flux distribution control
Reactor vessel water level control
Reactor vessel pressure control
Nuclear system temperature control
Nuclear system water quality control
Nuclear system leakage control
Core reactivity control

Control rod worth control
Containment and Reactor/Auxiliary Building pressure and temperature control
Stored fuel shielding, cooling, and reactivity control

Planned operations for each operating state are identified and safety action sequences are diagrammed to demonstrate system requirements. The six planned operations are: refueling, achieving criticality, reactor heat up, power operation, achieving reactor shutdown, and reactor cooldown. In addition to planned operation, anticipated operational transients, design basis accidents, and special events are defined for each operating state and planned operating condition.

For each event, protection sequences are diagrammed to show acceptable success paths including consideration of single active component failure and single operator error conditions. From these diagrams, safety-related system requirements are determined. 21

Those additional systems or components identified as safety-related from the results of the systems analysis diagrams, will be added to the Q-List as described above. Then, for any such additions, the QA Program will be applied to all subsequent system design, procurement, construction and operation activities.

ITEM I.F.2 DEVELOP MORE DETAILED QA CRITERIA

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item, including the establishment of a quality assurance (QA) program based on consideration of: (a) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (b) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (c) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (d) establishing criteria for determining QA programmatic requirements; (e) establishing qualification requirements for QA and QC personnel; (f) sizing the QA staff commensurate with its duties and responsibilities; (g) establishing procedures for maintenance of "as-built" documentation; and (h) providing a QA role in design and analysis activities. 21

Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters."

NRC ACCEPTANCE GUIDANCE

Establish a quality assurance (QA) program based on consideration of ensuring independence of the organization performing checking functions from the organization responsible for performing the functions.

The QA program includes:

2A1 (1B2) Verification of conformance to established requirements is accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified. Rationale and justification must be provided if performed by other than the QA organization.

2A2 (10B1) The QA organizational responsibilities for inspection are described. Individuals performing inspections report to the QA organization.

2A3 (7A2) Verification of suppliers' activities during fabrication, inspection, testing and shipment of materials, equipment, and components is planned and performed with QA organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

2A4 (7B1) Receiving inspection is performed by the QA organization to assure: 21

a. The material, components, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified inspection, test and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the S/HNP prior to installation or use.

2A5 (8B3) Correct identification of material, parts, and components is verified and documented by the QA organization prior to release for fabrication, assembling, shipping, and installation.

2A6 (9B2) Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. The QA organization verifies the recorded evidence and documents the result.

2A7 (10C3) (11C1) Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA

organization as a minimum evaluates, verifies, and documents completeness of this activity.

2A8 (16.3) Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

RESPONSE (Puget, NESCO, and Bechtel)

For the S/HNP, Bechtel has been designated responsible for supplier control in accordance with Section 7, pages 33-37 of the Bechtel Topical Report BQ-TOP-1, Rev. 1A. The Bechtel QA program requires that inspection personnel are independent of the individual or group performing the activity being inspected. Bechtel Procurement Supplier Quality, Site Construction Quality Control, and Quality Assurance comprise the quality group, which is delegated the authority and responsibility for inspection and verification functions. 21

Where reference is made to the appropriate QA organization in the following responses, depending on the organization and the function to be performed, this may be either the quality assurance or the quality control group of the involved organization. In either case, this group is independent of the group performing the work.

2A1 Verification of conformance to established requirements at all levels is accomplished by individuals or groups within the appropriate QA organization who do not have direct responsibility for performing the work being verified.

2A2 Bechtel Site Construction Quality Control is responsible for surveillance inspection of the Site contractors' activities.

2A3 Verification of suppliers' activities during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed by Bechtel Procurement Supplier Quality in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the

extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

2A4 Receiving inspection is performed by Bechtel Site Construction Quality Control to assure:

a. The material, components, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Materials, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified inspection, test and other records, (such as certificates of conformance attesting that the materials, components, and equipment conform to specified requirements) are available at the S/HNP prior to installation or use. 21

2A5 Correct identification of materials, parts, and components is verified and documented by Bechtel Site Construction Quality Control prior to release for fabrication, assembling, shipping, and installation.

2A6 Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. Bechtel Site Construction Quality Control verifies the recorded evidence and documents the result.

2A7 Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. Bechtel Site Construction Quality Control as a minimum evaluates, verifies, and documents completeness of this activity.

2A8 Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

NRC ACCEPTANCE GUIDANCE

Performing quality assurance/quality control functions at construction sites to the maximum feasible extent;

The QA program provides provisions to assure that:

2B1 (1C3) The person at the construction site responsible for directing and managing the site QA program is identified by position. He reports to the off-site QA organization and has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.

2B2 (1B6) Designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments). 21

RESPONSE (Puget, NESCO, and Bechtel)

2B1 The person at the construction site responsible for directing and managing the Site QA program is NESCO's Site QA Manager. He reports to the off-Site QA organization and has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the Plant Site is being effectively implemented.

2B2 Site QA personnel are involved in Plant activities important to safety and are kept abreast of work schedule and construction activities by periodically attending construction status meetings. In addition, Site QA personnel actively participate in day-to-day planning, scheduling and construction status meetings to review problem areas and evaluate these areas to determine if they are chronic and/or are developing a trend. Problem areas are also evaluated to determine the extent corrective action is taken and its effectiveness. 22

Site QA personnel ensure that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments. 22

NRC ACCEPTANCE GUIDANCE

Including QA personnel in the documented review of and concurrence in quality-related procedures associated with design, construction, and installation.

The QA program includes:

2C1 (2B1a) Provisions are established to assure the quality-affecting procedures required to implement the QA program are consistent with QA program commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official.

2C2 (2B1b) The QA organization reviews and documents concurrence with these quality-related procedures. 21

2C3 (4A1) Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents should require contractors and subcontractors to provide an acceptable quality assurance program. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents is performed by QA personnel.

2C4 (6A2) Procedures for the review, approval, and issuance of documents and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization reviews and documents concurrence with these documents with regard to QA-related aspects.

2C5 (10C1) Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA

aspects and other technical organizations, as appropriate:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of item 2A2 (10B1).

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements. 21

2C6 (11B1) Test procedures or instructions provide for the following as reviewed and concurred with by the QA organization for QA aspects and by other technical organizations for technical aspects:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

2C7 (12.3) Procedures are established and described for calibration (technique and frequency), maintenance, and control of the measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components. The review and documented concurrence of these procedures is described and the organization responsible for these functions is identified.

2C8 (13.2) Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity. The QA organization reviews and documents concurrence of these procedures. 21

2C9 (14.1) (14.4) Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components and throughout fabrication, installation, and test. The QA organization reviews and documents concurrence with these procedures.

2C10 (14.2) (14.4) Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps. The QA organization reviews and documents concurrence with these procedures.

2C11 (14.3) (14.4) Procedures are established and described to control altering the sequence of required tests, inspections, and other operations important to safety. Such action should be subject to the same controls as the original review and approval. The QA organization reviews and documents concurrence with these procedures.

2C12 (15.1) Procedures are established and described for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components and as applicable to services (including

computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformances, including disposition and closeout.

2C13 (15.2) QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconforming items and involvement of the QA organization in documenting concurrence to the disposition, satisfactory completion of the disposition, and corrective action.

2C14 (16.1) Procedures are established and described indicating an effective corrective action program has been established. The QA organization reviews and documents concurrence with the procedures.

RESPONSE (Puget, NESCO, and Bechtel)

2C1 2C2 The Puget S/HNP QA Program is described by the Quality Assurance Manual for the S/HNP. A statement of Policy by the President and Chief Executive Officer of Puget states that Puget is dedicated to the construction and operation of a dependable, safe nuclear facility and recognizes that a strong QA Program is a prerequisite to the achievement of this objective. This Statement of Policy and the Introduction makes the Quality Assurance Manual mandatory for all activities that affect quality of those structures, systems, and components that are on the Q-list. Procedures are reviewed by appropriate QA personnel during preparation for inspections, surveillance, implementation reviews and audits to ensure consistency with QA program commitments. Additionally, these procedures are reviewed and concurred with by the appropriate QA organization prior to issuance. 21

2C3 Procedures are established for the review and documented concurrence of procurement documents by QA personnel to determine that (1) quality requirements are correctly stated, inspectable, and controllable; (2) there is adequate acceptance and rejection criteria and; (3) that procurement documents have been prepared, reviewed, and approved in accordance with QA Program requirements. To the extent necessary, procurement

documents will require contractors and subcontractors to provide an acceptable QA Program.

2C4 Procedures for the review, approval and issuance of documents (including procedures, instruction, specifications, and construction drawings) and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. These documents are reviewed and concurred with by the appropriate QA organizations for QA related aspects.

2C5 Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the appropriate Bechtel QA organization for QA aspects and other technical organizations as appropriate:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection. 21

c. Identification of the individuals or groups responsible for performing the inspection operation.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements.

2C6 Test procedures or instructions are reviewed and concurred with by the appropriate Bechtel QA organization for QA aspects and by other technical organizations for technical aspects and provide for the following:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment, and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

2C7 Procedures are established for calibration (technique and frequency), maintenance, and control of the measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components. The review and documented concurrence of these procedures are described and the Bechtel organization responsible for these functions is identified. Requirements for such description and identification are included in procurement documents, as appropriate for contractors and suppliers. 21

2C8 Procedures are established to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity. The appropriate QA organization reviews and documents concurrence of these procedures.

2C9 Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components throughout fabrication, installation, and test. The appropriate QA organization reviews and documents concurrence with these procedures.

2C10 Procedures are established to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps. The appropriate QA organization reviews and documents concurrence with these procedures.

2C11 Procedures are established to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. The appropriate QA organization reviews and documents concurrence with these procedures. 21

2C12 Procedures are established for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components, and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformance, including disposition and closeout.

2C13 Puget requires that its contractors have procedures and a system for controlling nonconforming items. The QA organization is involved in documenting concurrence to the disposition, satisfactory completion of the disposition and corrective action.

NESCO receives, for information, nonconformance reports as soon as they are issued and supplier deviation reports that are dispositioned "repair or use as is." Nonconformance reports will be reviewed and analyzed by NESCO's QA organization and the results routinely reported to Puget's QA Manager. 22

Puget and NESCO QA personnel review nonconformance reports to assure:

a. Disposition has been satisfactorily completed and closed out.

b. Cause of the nonconformance has been determined and, when applicable, proper effective action has been taken to preclude repetition.

c. Chronic problems and trends have been reviewed and analyzed and forwarded to management.

d. Nonconformance reports pertaining to conditions outlined in the requirements of 10 CFR 50.55 (e) have been properly processed.

The following examples include situations which may warrant a corrective action request:

a. Repeated failure to follow approved procedures after previous violations have been reported.

b. Nonconformances which, due to their repetition or impact (potential or actual) upon quality, should be brought to management's attention for special action. 22

c. Repeated failure to implement action to correct deficiencies discovered in audits by the commitment date if the lack of such action may contribute to a failure of the quality program.

d. Repeated disregard for documentation requirements.

NESCO's Site QA discipline engineers review and analyze nonconformance reports and corrective action requests relating to their discipline and follow the action taken to assure that it is effective. The analysis will consist of evaluating chronic problems and trends to aid in determining adequate QA/QC staffing and in determining where QA/QC efforts should be concentrated in regard to witness and hold points, scheduling of surveillance and monitoring activities, training, etc.

For Puget and NESCO issued nonconformance reports, QA and Nuclear Projects approval/disapproval is required for dispositioning. QA documents satisfactory completion of the dispositioning and corrective action.

2C14 Procedures are established indicating an effective corrective action program has been established for contractors and suppliers performing safety-related activities. The appropriate QA organization 21

reviews and documents concurrence with the procedures.

NRC ACCEPTANCE GUIDANCE

Establishing criteria for determining QA programmatic requirements.

The QA program provides provisions to assure that:

2D1 (2B3) The QA organization and the necessary technical organizations participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspections, tests, special processes, records, audits and others described in 10 CFR 50, Appendix B.

2D2 (7B4) For commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verificiation requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser. 21

2D3 (10A) The scope of the inspection program is described that indicates an effective inspection program has been established. Program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed. The QA organization participates in the above functions.

2D4 (10C2) Procedures are established and described with the involvement of the QA organization to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

2D5 (11A1) The description of the scope of the test control program indicates an effective test program has been established for tests including proof tests prior to installation and preoperational tests. Program procedures provide criteria for determining the accuracy requirements of test equipment and

criteria for determining when a test is required or how and when testing activities are performed.

2D6 (1...) Audit data are analyzed by the QA organization and the resulting reports indicating any quality problems and the effectiveness of the QA program, including the need for reaudit of deficient areas, are reported to management for review and assessment.

RESPONSE (Puget, NESCO, and Bechtel)

2D1 Puget and NESCO's QA organization and the necessary technical organizations are participating early in the QA definition stage to identify the extent QA controls are to be applied to specific structures, systems, and components and will continue to do so. 22

Bechtel Engineering considers the importance of design features and characteristics when defining technical, inspection, and test requirements in the technical specifications. Bechtel Engineering, with QA participation, utilizes a unique ordering approach when specifying the QA criteria for procurements and contracts. Bechtel's Quality Control and Procurement Supplier Quality Representatives consider the specification requirements when preparing inspection instructions. The "graded approach" has been utilized by Bechtel for applying QA criteria to non-Q-listed items when formalized QA programs are required. 21

For items determined to be important to safety where specific QA controls cannot be imposed in a practical manner, an evaluation will be made to determine special quality verification requirements to be applied during installation or testing to provide the necessary assurance that the item(s) meet project requirements.

2D2 For commercial "off-the-shelf" items where specific QA controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item.

2D3 An effective inspection program is being established. Inspection program procedures will provide criteria for determining the accuracy requirements of inspection equipment and criteria

for determining when inspections are required or define how and when inspections are performed. The appropriate QA organization participates in the above functions.

2D4 Procedures are established and described with the involvement of the appropriate QA organization to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

2D5 A test control program will be established to include proof tests prior to installation and preoperational tests. Procedures will provide criteria for determining accuracy requirement of test equipment and criteria for determining when a test is required and how and when testing activities are performed.

2D6 Audits are conducted and the results analyzed by the appropriate QA organization. Audit reports indicate any quality problems and the effectiveness of the audited QA Program. Reaudits of deficient areas are conducted as necessary to assure implementation of corrective action and recurrence control. Audit results are reported to management for review and assessment. 21

NRC ACCEPTANCE GUIDANCE

Establishing qualification requirements for QA and QC personnel.

The QA program provides provisions to assure that:

2E1 (2D) Indoctrination, training, and qualification programs are established such that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs, documentation includes the objective, content

of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, re-examining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Rev. 1. 21

2E2 (10B2) A qualification program for inspectors (including NDT personnel) is established under direction of the QA organization and documented, and the qualifications and certification of inspectors are kept current.

RESPONSE (Puget, NESCO, and Bechtel)

2E1 The training qualification and certification programs are established so that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (1) the specific functions personnel are qualified to perform, and (2) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, re-examining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory positon in Regulatory Guide 1.58, Rev 1.

2E2 A qualification program for inspectors (including NDT personnel) is established under direction of the QA organization and documented, and the qualifications and certifications of inspectors are kept current. 21

NRC ACCEPTANCE GUIDANCE

Sizing the QA staff commensurate with its duties and responsibilities.

The QA program provides provisions to assure that:

2F1 (1A5) Organization charts identify the "onsite" and "offsite" organizational elements which function under the cognizance of the QA program (such as design engineering, procurement, manufacturing, construction, inspection, test, instrumentation and control, nuclear engineering, etc.), the lines of responsibility, and a description of the criteria for determining the size of the QA organization including the inspection staff.

2F2 (-) The QA organization is involved in establishing long-range projected work schedules and staffing of QA and QC personnel and evaluates these periodically (i.e., monthly) to assure they are valid or, if necessary, modify staffing level.

The QA program provides provisions to assure that:

2G1 (6A1) The scope of the document control program is described, and the types of controlled documents are identified. As a minimum, controlled documents include: As-built documents.

2G2 (6C1) Procedures are established and described to provide for the preparation of "as-built" drawings and related documentation in a timely manner to accurately reflect the actual plant design.

RESPONSE (Puget, NESCO, and Bechtel)

2G1 2G2 Puget's PSAR, Section 17.1.6, describes the scope of the document control program and includes "as-built" drawings in the document control system. Project procedures will be established to provide for the preparation of "as-built" drawings and related documentation in a timely manner to accurately reflect the actual Plant design. "As-built" drawings is an item on the checklist prior to fuel loading. 22

NRC ACCEPTANCE GUIDANCE

Providing a QA role in design and analysis activities.

The QA program provides provisions to assure that: 21

2H1 (3E1) Procedures are established and described requiring a documented check to verify the dimensional accuracy and completeness of design drawings and specifications.

2H2 (3E2) Procedures are established and described requiring that design drawings and specifications be reviewed by the QA organization or other individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necesary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

RESPONSE (Puget, NESCO and Bechtel)

2H1 Procedures are established and require documented checks to ensure the dimensional accuracy (including tolerance for accept/reject criteria and

RESPONSE (Puget, NESCO, and Bechtel)

2F1 PSAR Chapter 17 describes the Project QA organizations of Puget, NESCO and Bechtel and provides organizational charts which indicate on-Site and off-Site personnel. PSAR Section 13.0 shows Project personnel on-Site and off-Site from other organizations. Criteria for determining the QA/QC staffing needs include (1) work load and schedule and (2) personnel efficiency. 21

The methods used to establish the work load and schedule include: work packaging; development of specific work plans (including inspections); identification of areas of inspection (QA/QC) expertise needed; and participation in day-to-day staff, planning and schedule meetings.

The factors considered in determining personnel efficiency include experience and training; quality of work coming to the inspector; quantity of work coming to the inspector (work flow); procedure effectiveness; and productivity of personnel.

The criteria for determining if QA/QC staff is adequate include the degree of: uninspected work; quality problems detected after inspection; inspector preparation time; close-out of nonconformances and audits; follow-up of corrective actions; and complaints. 2

2F2 Long range matching of QA/QC resources with work load will be accomplished by QA review of projected work forces of the utility and its contractors. This review will permit recruiting and training activities to be carried out in such a manner as to provide trained QA personnel necessary to assure the quality of work.

QA/QC personnel will reevaluate staffing levels periodically (i.e., monthly) to assure they are adequate and modify as necessary.

Effectiveness of the staffing program will be assured by QA participation in the work planning, surveillance and audit and by the authority to stop work when it appears staffing is inadequate.

NRC ACCEPTANCE GUIDANCE

Establishment of procedures for maintenance of "as-built" documentation. 21

inspectability) and the completeness of the drawings and specifications. QC inspections of quality-related activities will be conducted using procedures or inspection checklists developed from the engineering specifications and drawings for the system, component, or structure.

2H2 Procedures are established to require that design drawings and specifications be reviewed by individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with procedures and that documents contain the necessary QA requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results. 21

ITEM II.B.1 REACTOR COOLANT SYSTEM VENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall modify their plant designs as necessary to provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting these requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

Venting capability of the S/HNP reactor vessel is addressed in two parts (refer to Figure 5.1-3a of the 251 NSSS GESSAR which is incorporated by reference into the S/HNP PSAR):

1. Up to the main steam line nozzles: the presence of noncondensible gases in the vessel below the main steam line nozzles could interfere with continued core cooling, so the capability for venting this region is essential. Venting can be accomplished by opening any one of the 22 safety relief valves (SRVs) on the main steam lines (which may be open already depending on the mode of core shutdown cooling in use). These valves and their actuators are safety grade, seismically and environmentally qualified for accident conditions, and are powered from the Class 1E electrical system and operable from the control room. Eight of the valves have a safety-related air supply, thus providing redundant venting capability.

   In addition, this region of the vessel can be vented through the RCIC steam supply line which connects to main steam line A, without opening the SRVs. This path is through the RCIC steam turbine exhaust, which discharges to the suppression pool.

2. Above the main steam line nozzles: the presence of noncondensible gases in the vessel above the main steam line nozzles will not interfere with continued core cooling, and as such, venting this region of the vessel is not considered to be a safety concern. Even so, there are two means of venting this space:

   a. Normally open 2" reactor head vent line and valve B21-F005, which discharges to main steam line A (which can be vented to the suppression pool/containment via any one of four SRVs).

   b. Normally closed 2" reactor head vent line and series valves B21-F001 and B21-F002, which discharge to the drywell equipment drain sump. 21

   These valves are safety grade and their actuators are Class 1E, seismically and environmentally qualified, but are not powered from the Class 1E electrical system. They are operable from the main control room.

Consideration has also been given to the potential for the accumulation of noncondensible gases interfering with the operation of the ECCS. In the post-LOCA condition, it is possible for noncondensible gases to come out of solution while operating the RHR system. It is expected that these gases would be swept through the RHR system, but some gases could potentially accumulate in the upper portions of the RHR heat exchanger during the steam condensing mode of RCIC operation should substantial amounts of noncondensibles be generated. The upper portion of the RHR heat exchangers are provided with separate 2" vent lines to the suppression pool for the removal of such noncondensible gases. The isolation valves on these lines are Class 1E, and are operable from the main control room. 22

All of the above venting paths lead to the containment via the suppression pool, which is the basis for hydrogen mixing analyses. The control of large amounts of hydrogen in containment is discussed in the response to Item II.B.8.3. 21

Procedures for the use of all of the above vent lines will be summarized in the FSAR. 22

The above supplements the PSAR information on capability for RCS venting and is consistent with preliminary design information normally required at the CP stage of review. There is no new, novel design, and there are no concerns regarding technical feasibility, state of the art or ability to implement the intended RCS venting design. 21

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844* source term radioactive material and (2) implement plant design modifications necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

*TID 14844, U.S. Atomic Energy Commission, 1962. 21

RESPONSE

1. Purpose

A post-accident radiation and shielding design review of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive material is being performed to respond to NUREG-0718. The review is scheduled for completion prior to issuance of a Construction Permit. Its purpose is (a) to ensure that vital areas in which personnel will be present during post-accident operations will be accessible; (b) to determine the accessibility of areas where it may be beneficial (although not essential) to have access to support post-accident operations; and (c) to verify the adequacy of protection provided for safety-related equipment.

Should the shielding design review so indicate, design modifications will be implemented as the detailed design progresses to permit adequate post-accident access or to protect safety equipment from the radiation environment. Any required design and/or procedural changes will be made to maintain personnel

exposures in vital areas within 10 CFR 50, Appendix A, GDC-19, specified design bases.

2. Post-Accident Source Terms and Systems Containing Sources

TID 14844 source terms are used in this shielding review. Table II.B.2-1 provides information on source activities, dilution volumes and systems containing the source terms.

Tabulations of the initial inventory of radioisotopes
for these sources are given in Tables II.B.2-2, 22
II.B.2-3 and II.B.2-3a.

For the calculation of the post-accident radiation source terms, the following assumptions are employed:

a. No credit will be taken for radioactive decay prior to transport of the source terms to the systems under consideration. 21

b. A detailed mechanistic approach to develop radiation source terms has not been used.

Systems listed in NUREG-0737, Item II.B.2, Section (2), which are not considered as sources are listed as follows with justification:

a. Hydrogen Recombiner System: S/HNP utilizes thermal recombiners which are completely internal to the containment.

b. Gaseous and Liquid Radwaste Systems: The radwaste systems are isolated from the containment and other systems which may contain primary coolant after an accident.

c. Chemical and Volume Control System: This is a PWR system.

3. Vital and Potential Post-Accident Support Areas

Tables II.B.2-4 and II.B.2-5 list the post-accident vital areas and potential post-accident support areas, and their anticipated occupancy requirements. These areas are defined as follows:

Post-Accident Vital Areas: Those areas in which personnel will be present during post-accident operations to perform monitoring and control functions.

Potential Post-Accident Support Areas: Those areas other than vital areas in which it is beneficial, although not essential, to have access to support post accident operations.

4. Personnel Radiation Exposure Guidelines

The general design basis for personnel radiation exposure guidelines is 10 CFR 50, Appendix A, GDC 19. The maximum allowable radiation dose to personnel shall not exceed 5 rem to the whole body or its equivalent to any part of the body, for the duration of the accident.

Doses received by personnel in areas of continuous occupancy will be determined using the control room occupancy factors contained in SRP 6.4 as discussed in NUREG-0737.

Doses received by personnel in an infrequent occupancy area will be determined by taking into account the frequency and duration of the activities anticipated for that area, and will be consistent with GDC 19 limits. 22

Potential support areas will be reviewed to determine under what circumstances they will be accessible, consistent with GDC 19 limits.

Average area dose rates will be used to determine personnel doses, although local hot spots may exist. 21 The dose rates will include contributions from containment shine and equipment shine from all significant sources.

5. Protection of Safety Related Equipment

A preliminary analysis for equipment qualification will be performed using the source terms specified in NUREG-0737 to establish the integrated dose, including 22 post accident operation, under which safety-related mechanical and electrical equipment located inside and outside containment are required to function. The results of this analysis will be used in the design and specification of this equipment. A final analysis will be performed and the results reported in Section 3.11 of the FSAR. Design modifications will be implemented where necessary to assure that the safety-related equipment will function when exposed to the radiation fields resulting from systems involved in the mitigation of an accident.

6. Options for Solving Potential Problems

The study will verify the adequacy of the existing design and indicate where changes will need to be made. If changes are required to meet acceptable operator and/or equipment dose levels in certain locations, the following options are available:

a. Move the offending radiation source to a less sensitive location.

b. Move the target equipment or operator control/work station to a location with an acceptable radiation field.

c. Place additional shielding around the offending radiation source.

d. Place local shielding around the target equipment or operator control/work station. 21

e. Purchase equipment designed to withstand the newly specified radiation environment.

In selecting the option to be used, emphasis will be placed on minimizing building structural modifications, since the buildings potentially affected are mostly designed and are early in the construction sequence.

If problems are encountered as a result of the shielding analysis, they are expected to be of a physical or design detail nature rather than questions of technical feasibility or state of the art.

ITEM II.B.3 POST-ACCIDENT SAMPLING

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844* source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

*TID 14844, U.S. Atomic Energy Commission, 1962.

RESPONSE

The capability for post-accident sampling of the reactor coolant and the drywell and containment atmosphere will be provided by the Post-Accident Sampling System (PASS). The PASS and the on-Site analysis capability will meet the requirements of NUREG-0737, II.B.3. Details are as follows:

1. Sample Collection and Transport

   a. Liquid: The capability to collect liquid samples from the Reactor Coolant System (RCS) and suppression pool will be provided. Sample location for the Reactor Coolant System is shown on Figure II.B.3-1. The length of the sample lines will be as short as possible to minimize plateout and the volume of liquid involved. Sample collection will not require an isolated

auxiliary system to be placed in operation. The sampling operation under post-accident conditions, utilizing TID 14844 source terms, will result in a personnel dose of less than the 5 rem whole body and 75 rem extremities criterion of GDC 19 to the workers involved in the sample collection and transport operation. A shielded transport cask will be provided and the access pathways analyzed to minimize the dose to workers while transporting samples. The sample station will have provisions for purging of all sample equipment and components. The sample return line will be to the suppression pool. Provisions will be incorporated to minimize radioactive release and the spread of contamination from the sample station.

b. Gaseous: The capability to collect containment atmosphere samples will be provided through the containment/drywell H_2 sampling system described in Appendix 1A.4. The capability to collect samples representing the secondary containment atmosphere will also be provided. The same dose-to-workers criterion for liquid sampling will be met for the gaseous sampling operation. The sample return line will be to the containment atmosphere. 21

2. Sample Analysis

For those samples not analyzed by in-line analysis equipment, the analysis of the post-accident samples collected will be performed in the counting room and laboratory. These facilities are located in the Service Building and will be designed and shielded such that the required analyses can be performed without interference from external radiation sources. Radiological analyses for certain radionuclides that are indicators of core damage (e.g., noble gases, iodines and cesium and nonvolatile isotopes) will be performed. The chemical sample analysis stations are equipped with fume hoods which are exhausted through HEPA filters to the environment through the Service Building vent. Doses to workers involved in sample analysis will not exceed those specified for the sample collection and transportation operation. Sample lines and panels will be provided with adequate shielding in those areas where access is required.

Time for the sample collection and analysis will not exceed the following:

- radiological: three hours

- boron: three hours, if boron injection was initiated
- chlorides: twenty-four hours
- total dissolved gas or hydrogen: three hours
- dissolved oxygen: verification that dissolved oxygen is less than 0.1 ppm if chloride concentration exceeds 0.15 ppm.

Accuracy, range and sensitivity will be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the Reactor Coolant System. 21

There are no questions regarding technical feasibility of state of the art regarding the post-accident sampling capability, nor are there any concerns as to the ability to implement the design prior to OL issuance.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC 21 within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.

It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim."

RESPONSE

1. Risk Study

A Plant/Site-specific probabilistic risk assessment will be performed.

The initiating events to be considered will include those indicated in Table II.B.8(1)-1, together with 22 the accidents and transients identified in the S/HNP PSAR and those applicable accidents in WASH-1400. These will be screened to identify the basic set of initiating events requiring operation of the key safety systems for core protection and release mitigation.

The PRA program will focus on core and containment cooling systems in performing event tree/fault tree analyses, and will include environmental effects, system interactions, human error and performance data, interdependence of support systems, and system

unavailabilities in the event tree/fault tree analysis.

The PRA study will identify common-mode failure mechanisms, and the sequences and system/component failures which are the dominant contributors to core damage risk. The data base used in the system fault trees will include methodologies to adjust failure data for varying testing and surveillance strategies. Human errors will be considered in the development of the data base. Furthermore, uncertainty analyses will be performed to determine propagation of component failure data, including error ranges, through the fault trees. Sensitivity analyses will be performed by varying the failure rates of key basis events which contribute to dominant event sequences in order to determine the effect on system failure rates and overall results.

The methodology to be used will be similar to that employed in WASH-1400 updated to be consistent with the IEEE/ANS, and IREP efforts to establish a standard methodology. A component failure data base for use in system fault tree analysis will be developed from recognized reference sources including WASH-1400 and IEEE-500. In addition, prototype specific failure data will be requested from vendors of selected components being supplied to S/HNP. 22

An additional decay heat removal system with its functional requirements and criteria derived from the PRA study will also be considered.

The final report will follow the outline presented in Table II.B.8(1)-2.

2. Performance of Study

Puget/NESCO is responsible for performance of the PRA and will ensure that the study will be performed by engineers who are highly qualified and experienced in risk assessment methodology. Puget/NESCO engineers will be actively involved in this study and will provide direction in the development of the program. Prior to decisions relating to identified design improvements, Puget/NESCO will appoint a third party to conduct a peer review on the risk study. Puget/NESCO retains ultimate authority and responsibility of implementing design improvements as a result of this study.

3. Application of Results to Final Design

Acceptance criteria for system probabilistic risk analyses will be established during the initial phase of the program. These acceptance criteria will include both quantitative and qualitative considerations of potential design changes on plant cost, schedule, and availability.

The results of the probabilistic risk analyses will be evaluated using the acceptance criteria to determine design or other changes. The results of the study will be used to improve reliability of component selection, specification, and testing, and to improve system interaction. Furthermore, the results of the study will be used to identify improvements to be considered for maintenance, procedures, operator training, and operating feedback and to identify those areas where additional quality assurance activities would improve reliability of core and containment cooling systems.

4. Schedule

The program will commence at issuance of Construction Permit. The initial phase of the program is expected to take approximately 15 months and will consist of a preliminary PRA of the present S/HNP design. The final study, including radionuclide release quantification, will be completed within two years of CP issuance. 22

Two-thirds of S/HNP engineering design has been completed. However, a subsequent setback in overall engineering completion to less than 50% has resulted from Project relocation to the Hanford Reservation. Such relocation has had negligible impact on the design of NSSS/ECCS systems and the control room; and the design of these systems remains more than two-thirds complete. In addition, most of the NSSS/ECCS components have already been fabricated and delivered into storage.

5. Acceptance Criteria

There are currently no established regulatory requirements or acceptance criteria for judging the acceptability of the PRA. Thus, the need for implementing changes in design or operating, testing, or maintenance procedures to achieve improvements will be based on judgmental acceptance criteria which are not directly related to licensing requirements.

6. Radioactive Release

The total radioactive release to the environment will be estimated for the various release groups to serve 22 as a basis for assessing the effect of improvements to the reliability of the core and containment cooling systems.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(2) include provisions in the containment design for one or more dedicated penetrations, equivalent in size to a single 3-foot diameter opening. This shall be done in order not to preclude the installation of systems to prevent containment failure, such as filtered vented containment systems." 21

RESPONSE

The S/HNP containment design will provide a single dedicated 3-foot diameter penetration in order not to preclude the installation of systems to prevent containment failure. This dedicated penetration will be located at approximately 491' elevation in the southwest quadrant of the containment building and will be capped and seal welded. Space inside the containment is dedicated for the containment penetration assembly, and a future inboard isolation valve, if required (space provision only). This penetration will meet all the requirements of existing penetrations indicated in the S/HNP PSAR Section 3.8.1.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(3) provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction."

RESPONSE 21

The S/HNP will provide a hydrogen control system capable of handling hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction. The hydrogen control system will consist of igniters distributed throughout the drywell, the containment, and any local area which has the potential of pocketing hydrogen. These igniters will burn hydrogen as it is generated and will reasonably assure that uniformly-distributed hydrogen concentrations will not exceed 10% during and following an accident.

22

The following criteria will be used to design the hydrogen igniter system:

a. Burning of the hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction such that:

   (1) Uniformly-distributed hydrogen concentrations will not exceed 10% during and following the accident.

   (2) Local pocketing of hydrogen in the drywell, the containment, or local areas will be prevented. 21

b. The system will be single active failure proof.

c. Operation of the hydrogen ignition system will not adversely affect the safe shutdown of the Plant.

d. The system will be protected from tornado and external missile hazards.

e. The system will not compromise the containment design.

f. The equipment inside the drywell and the containment necessary for achieving and maintaining safe shutdown of the Plant will be designed to perform its intended function during 21 and after being exposed to the environmental conditions created by activating the hydrogen igniter system or will be protected from these environmental conditions.

22

ITEM II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicant shall:

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service 21
Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions."

RESPONSE

1. INTRODUCTION

The following sections describe the two year program
to be pursued by the S/HNP. The program, which is 22
intended to be conducted post-Construction Permit, will be split into a first phase (six months) and a final phase (two year) effort.

Currently, the containment integrity has been evaluated against an internal pressure of 45 psig as determined by the NRC to be the minimum pressure to be

considered. Work performed by others suggests that 45 psig is adequate to assure containment integrity in the event of a 100% active fuel clad metal-water reaction using a distributed ignition system (DIS). The above mentioned program, together with appropriate hydrogen control measures, will be pursued to confirm the adequacy of the S/HNP containment design and the control system measures to be implemented.

As stated in the response to II.B.8(3), the S/HNP has chosen a distributed ignition system (DIS) for controlling the release of hydrogen from a 100% active fuel clad metal-water reaction. The selection of the DIS was based upon a review of the existing industry work, in particular the Sequoyah docket (50-327/328) and the Grand Gulf docket (50-416/417). This industry work considered various hydrogen control options and narrowed these to the two most viable alternatives, DIS and post-accident inerting. Studies by the lead plants identified the DIS as the most viable method and this selection was consistent with the conclusions reached by the S/HNP engineering and operating staff. 22

It was recognized that uncertainties still exist in providing an adequate basis for defining a DIS for hydrogen control for a Mark III containment. The S/HNP, therefore, decided to monitor and participate in industry-wide efforts in two ways: first, as a funding member of the BWR/6 Hydrogen Control Owners Group (HCOG) and second, by monitoring other activities including submittals on individual plant dockets, research by national laboratories (e.g., Sandia, Livermore), EPRI and the just-beginning IDCOR program of which the S/HNP is an active member. This latter effort includes reviewing already completed testing programs (at TVA's Singleton Laboratory, hydrogen burn testing at Fenwal and Lawrence Livermore Laboratory's igniter tests) and monitoring on-going and future testing (Acurex, Factory Mutual, HEDL, etc.).

Of all these activities, the greatest emphasis has been placed on the two most directly applicable to the S/HNP design: following the submittals made by Grand Gulf and participating with the HCOG. Grand Gulf, as the lead plant BWR/6, has been leading the application of refined analytical capability through the CLASIX-3 code to evaluate the release, distribution and ignition of hydrogen in a Mark III containment. This work is of particular interest to S/HNP because the Grand Gulf containment is very similar to the S/HNP containment, as will be described in greater detail

below. The HCOG represents a pooling of utility resources and manpower to efficiently address, on a generic basis, the key issues related to hydrogen control in a Mark III containment. Its primary focus has been to define the areas requiring study and quantification and to establish programs to address these areas. The HCOG will fund and oversee these programs. The S/HNP will use the results of the HCOG programs as the basis for certain elements of its own Plant-unique program described below.

The HCOG and other industry activities may identify additional practicable alternative hydrogen control systems. The S/HNP will actively monitor this work and will include an evaluation of such alternatives in the two-year program described below.

2. COMPARISON OF S/HNP WITH GRAND GULF

To assure that the pressures and temperatures resulting from the combustion of hydrogen released from a 100% active fuel clad metal-water reaction are within the S/HNP containment structural capability, the S/HNP has been compared with Grand Gulf. Grand 22
Gulf was selected because: (a) both have the same size reactor pressure vessel (see Table II.B.8(4)-1), (b) both are Mark III containments, (c) both are reinforced concrete-steel lined containments, (d) both containments were designed by the same A/E, (e) both containments are of similar size and arrangement (see Table II.B.8(4)-2), and (f) Grand Gulf is presently the Mark III lead plant in hydrogen control.

The hydrogen control parameters for Grand Gulf were identified by Mississippi Power & Light Company (MP&L) in its June 19, 1981, (AECM-81/221) letter to Mr. R. L. Tedesco of the NRC entitled "Description of Hydrogen Control Measures." A comparison of Grand Gulf and S/HNP is listed in Table II.B.8(4)-2. It should be noted that the Grand Gulf analysis is for a 75% active fuel clad metal-water reaction whereas the S/HNP will be for a 100% active fuel clad metal-water reaction. However, the amount of oxygen available for combustion with the hydrgen is limited by the air inside the containment and drywell. This amount of air can supply only enough oxygen to burn with approximately the amount of hydrogen associated with a 75% active fuel clad metal-water reaction.

Therefore, the additional hydrogen released from a 100% reaction will be incapable of burning in an oxygen depleted environment. The amount of total

hydrogen assumed burned will be approximately the same for both Grand Gulf and S/HNP.

By comparing the parameters of Table II.B.8(4)-2, it has been concluded that the results of a hydrogen burn analysis for Grand Gulf and S/HNP will be similar. It should be noted that S/HNP is slightly larger in volume than Grand Gulf and therefore contains slightly more oxygen. However, the S/HNP containment has more volume for the pressure to expand into. Also, S/HNP has larger passive heat sinks and a greater volume in the suppression and upper pools. It should also be noted that although there is some slight difference in the flow rates in the containment spray systems, the initiation logics are similar, and no significant impact on the analysis is expected. The "mixing" systems for the post-LOCA combustible gas control systems are slightly different. In both plants, the mixing systems are manually initiated. However, in Grand Gulf air is taken from the drywell volume and exhausted to the containment by bubbling it through the suppression pool, while the S/HNP design exhausts drywell air directly to the containment atmosphere above the wetwell region. This difference will be investigated to determine any significant impact on the overall hydrogen burn analysis. Appropriate modifications to the mixing system will be made if necessary in order to make the two systems essentially identical for purposes of hydrogen burn analysis results, including, if necessary, the bubbling of the drywell atmosphere through suppression pool. 22

In AECM-81/221, MP&L identified its worst case sequence as a delayed containment burn initiated by a small break LOCA in the drywell resulting in 42 psig pressure to the containment. Because of the similarities of the two containment designs, the 42 psig result is believed to be applicable to the S/HNP containment design. MP&L is continuing its analysis and is presently investigating the sensitivity of various parameters. The results of these additional analyses will also be factored into the S/HNP analysis.

3. PRELIMINARY STRUCTURAL ANALYSIS AT 45 PSIG

Subsubarticle CC-3720, Factored Load Category, considering pressure of 45 psig and dead load alone is interpreted as follows: 21

All load factors shall be taken as 1.0 and thermal effects shall be considered. The resulting load combination is,

$$D + P_{mw} + T_{mw}$$

Where D = Dead Load

P_{mw} = 45 psig resulting from a degraded core accident that releases hydrogen generated from 100% fuel clad metal water reaction accompanied by hydrogen burning

T_{mw} = thermal effects and loads resulting from a degraded core accident

General yielding of cross sections shall not be
permitted. However, local yielding will be permitted
as long as serviceability and containment integrity
are maintained. The liner plate will be considered
effective as a load carrying member. Liner plate 22
strains shall not exceed those of Table CC-3720-1 for
factored load combinations. These changes are
incorporated in revisions to Section 3.8.1.4.b, Loads
and Loading Combinations and Table 3.8-1 of the PSAR. 21

Using these criteria, a preliminary analysis of several sections of the containment concrete cylinder has been performed using loads generated from the original design in combination with the 45 psig metal-water reaction internal pressure and associated thermal effects.

This analysis was conducted using a computer program to predict stresses and strains in a reinforced concrete element with multidirectional reinforcing, subjected to applied loads and thermal gradients. The program takes into account concrete cracking and stress redistribution due to concrete cracking.

Using this method, the containment general cylindrical shell, away from major discontinuities, as presently designed and reinforced, was verified to be adequate to withstand the 45 psig internal pressure and accompanying thermal effects.

The investigation of the containment cylindrical shell near or adjacent to major discontinuities (e.g., junction of cylindrical shell to base mat and at locations of large diameter openings such as the equipment hatch) is not complete and continued investigation will be performed to verify the adequacy of the

design. This analysis is expected to confirm the adequacy of the current design, but if that does not result, modification will be made to assure the design does meet the criteria.

The reinforced concrete dome design is not complete. A review of the preliminary design indicates the amount of reinforcing to be inadequate for a 45 psi metal-water reaction accident pressure. However, the review also indicates that the dome can be designed to the 45 psi criteria by increasing the amount of the dome reinforcing. The dome will be analyzed and designed for the degraded core accident criteria prior to construction. 21

The reinforced concrete base mat was not investigated, because the relocation to the Hanford Reservation will necessitate a complete redesign. The base mat will be analyzed and designed for the degraded core accident criteria prior to construction.

Containment locks and hatches are designed by vendors. The requirements of the ASME Code, Div. 1, Subsubarticle NE-3220, Service Level C limits will be specified to the vendors and reviewed before incorporation into the design. The containment penetration sleeves and flued heads will also be designed and verified to the requirements of Service Level C. These requirements are indicated in PSAR Paragraph 3.8.2.6.

4. <u>POST CP PROGRAM - FIRST PHASE</u>

In order to better quantify the expected containment requirements during an accident that releases hydrogen generated from a 100% active fuel clad metal-water reaction accompanied by hydrogen burning, plant-specific studies will be conducted for the S/HNP containment configuration. These studies will provide the expected pressure response and confirm the adequacy of an internal pressure capability of 45 psig. In the event that the pressure response is significantly above 45 psig, modifications to the hydrogen control system or containment structual design will be provided as described below. The following paragraphs discuss the first phase post-construction permit program. A report will be provided to the NRC describing the results of this first phase program within 6 months after receipt of the Construction Permit. 22

a. S/HNP Unique Analysis

The HCOG prgram includes generic evaluations of areas such as plant event combinations leading to hydrogen generation, igniter characteristics, and hydrogen mixing in containment. The S/HNP will generate a specific plant-unique analysis of the limiting hydrogen burn case for the S/HNP configuration. The burn scenarios will be based upon relative probability of hydrogen generation. This anlaysis will include the dynamic pressure and temperature time histories suitable for establishing structural design and analysis criteria.

b. Current Issues

A number of specific issues will be addressed in this first phase in order to provide a realistic analysis of containment response to hydrogen combustion. Some of these issues are:

Hydrogen generation rate 22
Igniter performance
Spray effectiveness
Hydrogen mixing in containment
Specific Plant events leading to potential hydrogen generation
Release points for hydrogen into containment
Hydrogen burn flame speeds
Single failure assumptions
Potential for and consequences of local detonations

To the extent that these issues require specific considerations on the S/HNP following receipt of the Construction Permit, they will be addressed uniquely. In addition, through participation in the HCOG and from monitoring other programs, the S/HNP may become aware of additional issues that have to be addressed specifically.

c. Sensitivity Studies

To the extent that the above issues are not resolved or that new issues appear, sensitivity studies will be performed for the S/HNP containment. Currently, it is expected that sensitivity studies would be performed to determine the effect on pressure response of

varying hydrogen generation rates, hydrogen mixing in containment, and hydrogen release locations.

d. Alternative Systems

Alternative methods for accommodating hydrogen releases from a 100% active fuel clad metal-water reaction will be studied and will be evaluated as they mature. Maturity of candidate methods is expected to be developed on predecessor BWR/6 project(s), with assistance from the HCOG, EPRI, IDCOR and other sources. State-of-the-art for all candidate methods for hydrogen control will be assessed for S/HNP adoption immediately after receipt of a Construction Permit. Results of that assessment will be reported within six months after receipt of a Construction Permit.

e. Commitment to Meet Maximum Pressure

Examination of the results developed to date indicate a high level of confidence that the peak response pressure for S/HNP will not exceed 45 psig. In the event that Plant-unique calculations show pressures beyond 45 psig, a number of possibilities will be examined: (1) Further improvements in the DIS will be evaluated such as revised location of igniters, modified containment spray conditions, or modified containment atmosphere mixing capability; (2) Containment strengthening will be considered; and (3) Work on alternative systems will be examined for potential adoption. 22

If containment design to a higher pressure than 45 psig is required, stress in the reinforcing steel and strain in the liner plate will be recalculated and compared with the limits set forth in section 3 above. If these limits are exceeded, design modification will be initiated to the reinforced concrete containment design. Locks, hatches, and penetrations will also be designed to the higher pressure.

Preliminary analyses performed to date at 45 psig indicate that the liner strain is well within the allowable limit. This suggests that modest increases above 45 psig might be accommodated without reaching allowable liner strain limits.

5. POST CP PROGRAM - FINAL PHASE

Two years after issuance of the Construction Permit, the S/HNP-unique programs described in section 4 will be completed. A reexamination of the applicable hydrogen burning event, including a reevaluation of the various accident scenarios considered to be appropriate, will be completed. Sensitivity studies of hydrogen burn assumptions, including examination of any new variables determined to be significant during development of the DIS, will be completed.

Igniter performance and endurance characteristics will be evaluated for the S/HNP containment based on available data.

The report will include updated response of the containment structure and essential equipment to local detonations and to the environmental conditions resulting from the combustion of hydrogen in the S/HNP containment. 22

a. Evaluation of Alternative Systems

As stated above in Section 4 for the first phase of this program, awareness will be maintained of development of alternative hydrogen control systems. An assessment of the potential applicability of these alternates to the S/HNP will be completed and justification for the selected means of hydrogen control will be provided.

b. Secondary Fires

Potential for secondary burning, which could be initiated by hydrogen burn, will be examined. Candidates for any such secondary burning will be determined by examination of combustible materials under hydrogen burn conditions. Either the consequences of any secondary burning (including combustion products) will be found tolerable, or the presence of combustible material in the containment building will be limited by design or by administrative controls for the operating plant. The secondary burn analysis will be described and any necessary limits or combustibles within containment will be identified.

c. <u>Structural Response</u>

The dynamic effects of the pressure-time histories 22
from the hydrogen burn analysis will be determined. Impact on the final design of the containment structure, including locks, hatches, and penetrations will be described.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACTIVITIES

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(4) provide preliminary design informaton at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(b) The containment and associated systems will provide reasonable assurance that uniformly-distributed hydrogen concentrations do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion." 21

RESPONSE

The igniter system will be designed and the igniters will be strategically located such that the hydrogen generated from a 100% active fuel clad metal-water reaction will be ignited in a manner that provides reasonable assurance that uniformly distributed hydrogen concentration will not exceed 10% during and following an accident.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(c) The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of 21
hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features."

RESPONSE

The relatively open configuration of the Mark III containment incorporated into the S/HNP design generally serves to preclude pocketing of hydrogen. Igniters will be strategically placed throughout the drywell, the containment, and local areas which have the potential for pocketing hydrogen. These igniters will provide reasonable assurance that uniformly-distributed hydrogen concentration does not exceed 10% during and following an accident in any area which has the potential for pocketing.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(4) provide preliminary design information consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(d) If the option chosen for hydrogen control is post-accident inerting: (a) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loading specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Service Load Category), (b) demonstrate that a pressure test, which is required, of the containments at 1.10 and 1.15 times for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting can be safely conducted, (c) demonstrate that inadvertent full inerting of the containment can be safely accommodated during plant operation." 21

RESPONSE

Inerting, as a hydrogen control measure, is not proposed for the S/HNP containmnt design; therefore this item is not applicable.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall:

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: 21

   (e) If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system."

RESPONSE

The equipment required to achieve and maintain safe shutdown of the Plant and to maintain containment integrity will be designed and qualified to perform its intended function during and after being exposed to the environmental conditions created by activation of the DIS. The qualification program will be developed and submitted for NRC approval within two years after receipt of a Construction Permit. 22

The expected extreme containment environmental conditions are determined as part of the 2-year program described in the response to II.B.8(4)(a). This will include enveloping pressure and temperature curves for all parts of the containment where the vital equipment is located.

The location of equipment required to achieve and maintain safe shutdown of the Plant and to maintain containment integrity, and any necessary equipment protection, will be described in the FSAR.

ITEM II.D.1 TESTING REQUIREMENTS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants and their agents shall provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

Performance testing of BWR safety/relief valves (SRVs) will be done beyond the current qualification requirements. This testing will be sponsored by the utilities of the BWR Owners Group (BWROG) in response to NUREG-0578, Requirement 2.1.2.

In July, 1979, the NRC issued its TMI Short-term Lessons Learned Report (NUREG-0578). In this report, the NRC required that testing be conducted "to qualify the Reactor Coolant System relief and safety valves under expected operating conditions for design basis transients and accidents." The "expected operating conditions" were to be determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The discussion

accompanying the requirement gave primary emphasis to two-phase and liquid flow conditions.

Reference 1 presents an evaluation of those Regulatory Guide 1.70, Revision 2, events which have the potential of producing liquid or two-phase flow discharge from the SRVs. This report is applicable to most BWRs. It is specifically applicable to all plants that have level 8 trips (high water level) on high pressure inventory maintenance systems (e.g., Feedwater, High Pressure Core Spray and Reactor Core Isolation Cooling). The S/HNP is included within this group of plants.

The conclusion reached, after a detailed review of all identified events (see Table 2-1 of Reference 1) is that a test which simulates the alternate shutdown cooling mode should be performed. This event is an anticipated operating condition which has been considered in the design analysis of plants. The BWROG has committed to perform
liquid flow safety valve tests for the conditions which can 22
occur for this mode of operation (Reference 2). All other
events which were identified are either of sufficiently low
probability or low consequence such that no additional
testing is warranted. 21

A description is given in Reference 3 for those tests which will be run on typical SRVs for BWR2 through BWR6 plants to demonstrate ability to perform satisfactorily under the condition in which low pressure (i.e., up to 250 ± 20 psig) water passes through the valve instead of saturated steam. This corresponds to conditions expected during the Alternate Shutdown Cooling Mode, i.e., the mode in which low pressure pumps are injecting cold water into the reactor vessel and this water is vented through the SRVs back to the suppression pool.

The low pressure water test will serve the following two purposes:

a. To demonstrate the capability of each type of SRV to operate satisfactorily under the bounding cases of release of low pressure water with resultant, typical BWR pipe loads on the SRV.

b. To measure the SRV discharge line (SRVDL) loads during water discharge through SRVs.

Six different SRVs will be tested in the relief mode (normal operating mode for low pressure). The specimens to be tested consist of 6 x 8 (inlet diameter x outlet diameter in inches) pilot-operated Electromatic Relief Valve, 6 x 10 two and three-stage pilot-operated Target Rock SRVs,

6 x 10 and 8 x 10 Crosby direct-acting SRVs and 8 x 10 Dikkers direct-acting SRV.

The S/HNP will use the Crosby 8 x 10 direct-acting SRVs and is thus covered by the testing program. In addition, the data gathered on discharge piping response will be considered in the design of the S/HNP SRV discharge piping, which will be designed for the same solid water flow conditions for which the valves are being tested. 22

REFERENCES

1. "Event Evaluation for BWR Safety Relief Valve Testing Required by NUREG-0578, 2.1.2" enclosed with letter from D. B. Waters (BWROG) to R. H. Vollmer (NRC) dated September 17, 1980, and titled "NUREG Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves." 21

2. Letter, T. D. Keenan (BWROG) to D. G. Eisenhut (NRC) dated December 14, 1979, and titled "BWR Owners Group Implementation of NUREG-0578 Requirement 2.1.2."

3. "NUREG-0578 BWR Safety/Relief Valve Test Description" enclosed with letter from D. B. Waters (BWROG) to R. H. Vollmer (NRC) dated September 17, 1980, and titled "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves."

ITEM II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to issuance of operating licenses."

RESPONSE

Safety relief valve (SRV) position indication will be determined by pressure measurement in the discharge pipe. This has been verified by the BWR Owners Group (BWROG) to be an adequate indication of SRV position indication by studying data from operating plants, which was submitted to the NRC by letter, T. Keenan (BWROG) to D. G. Eisenhut (NRC) dated October 17, 1979. 21

The actual pressure setpoint to be used for the S/HNP will be determined from a combination of analysis and field test data and will be submitted with the FSAR. Indication in the main control room will be on two light matrices, one for each division of position measurement, on the Reactor Core Cooling Systems Benchboard, P601 (see Figure I.D.1-1 for location), above the manual control switches for the relief valves. The indication will be redundant, safety grade, seismically and environmentally qualified, and powered from a Class 1E power source. An alarm indicating that an SRV is open will be provided but will not be safety grade.

There are no questions regarding technical feasibility or state-of-the-art capability of this SRV position indication design, nor is there any concern that it cannot be implemented prior to OL issuance.

ITEM II.E.4.1 DEDICATED PENETRATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessary to include redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. Applicants 21 shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly."

RESPONSE

The S/HNP design utilizes thermal recombiners which are internal to containment and have no associated mechanical penetrations. As such, this item is not applicable.

ITEM II.E.4.2 ISOLATION DEPENDABILITY

NUREG 0718, REV. 1, REQUIREMENT

"Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, and describe the basis for selection of each essential system. All nonessential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981.

For post-accident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive diverse isolation signals. 21

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The

pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II.3(f) during operational conditions 1, 2, 3 and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit state of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

The Containment Isolation System (CIS) is discussed in Section 6.2.4. The criteria for the design of the CIS Control System are listed in Section 7.1.2.1.2. The following are the responses to NUREG-0718, Item II.E.4.2:

1. Compliance with SRP 6.2.4, Rev. 1

   The design of the S/HNP Containment Isolation System will meet the recommendations of Standard Review Plan Section 6.2.4, Rev. 1. The details of how these requirements will be met will be described in the S/HNP FSAR. The present Containment Isolation System has been reviewed and accepted by the NRC (Reference: Skagit Nuclear Power Project SER, NUREG-0309, Section 6.2.15).

2. Identification of Essential and Nonessential Systems

PSAR Table 6.2-11 lists the systems which penetrate the containment. These systems will be categorized as essential, intermediate or nonessential in the FSAR. The following definitions will be applied in categorizing the systems.

Essential 21

Essential systems are those critical to the immediate mitigation of the consequences of a LOCA. Essential systems are not automatically isolated by accident signals. 22

Intermediate

Intermediate systems are those which could be useful (although not critical) in mitigating an accident which results in containment isolation. Intermediate systems are automatically isolated by accident signals. If automatically isolated, the operator may choose selectively to reopen the valves as they are needed, while the accident signal is still present. This permits the operator to use all available systems to cope with an accident, while still maintaining the effectiveness of the containment.

In summary, the isolation provisions for intermediate systems have the same essential features as nonessential systems (double barrier isolation, automatic isolation on diverse accident signals). The main difference is that non-essential systems can be manually re-opened by the operator while the accident signal is still present. 22

Nonessential

Nonessential systems are those which are not required or used in the mitigation of an accident which results in containment isolation. All nonessential systems are automatically isolated by the Containment Isolation Actuation Signal and cannot be reopened by the operator while the accident signal is still present. 21

3. Isolation of Nonessential Systems

As required for post-accident situations, each nonessential penetration (except instrument lines)

will have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56 and 57 as clarified by Standard Review Plan, Section 6.2.4. Isolation will be performed automatically with no credit being taken for operator action. All manual valves will be sealed closed so as to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration will receive independent isolation signals, derived from diverse parameters.

4. Reopening of Isolation Valves on Isolation Signal Resetting

The design of the controls for automatic containment isolation will be such that the resetting of the isolation signals will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves will require deliberate operator action on a valve by valve basis. Administrative provisions to close all isolation valves manually before resetting the isolation signals will not be utilized. Ganged reopening of containment isolation valves will not be utilized.

5. Containment Pressure Isolation Setpoint 21

The drywell high pressure set point to initiate containment isolation is 2 psig. This will allow 1 psi for operational pressure variations and 1 psi for instrument error to minimize the potential for spurious containment isolation.

6. High Radiation Isolation of Path Lines

All systems that provide a path from the containment atmosphere to the environs (e.g., the containment purge and vent systems) will close on a safety-grade high radiation signal.

Radiation monitors are located in the containment purge lines such that containment atmosphere releases through the purge line prior to isolation from the radiation signal will not result in doses in excess of 10 CFR 100 guidelines for a spectrum of accidents (see PSAR Section 9.4.5). An analysis has been performed to determine the amount of radioactivity released from the containment as a result of Low Purge System operation and post-LOCA isolation valve closure. The results of this analysis are presented in PSAR Section 15.1.39.5.2.2.6 which indicate that the radiological consequences are well below the guidelines set forth in 10CFR 100.

7. Containment Purge Isolation Valves

The containment purge and vent isolation valves will satisfy the operability criteria of CSB 6-4. See Item II.E.4.4 for details.

8. Level of Information

The above supplements the PSAR information on the CIS and is consistent with preliminary design information normally required at the CP stage of review. There is no new or novel design and there are no concerns regarding technical feasibility, state of the art or ability to implement the intended CIS design. 21

ITEM II.E.4.4 PURGING

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall (1) provide a capability for containment purging/venting designed to minimize purging time, consistent with ALARA principles for occupational exposure, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability, (4) adopt procedures and restrictions consistent with the revised requirements, and (5) provide and demonstrate high assurance that the purge system will be reliably isolated under accident conditions.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

The general safety concern over containment purging stems from the presumption that the purge line provides a path for accident releases prior to isolation, and further, that the dynamic effects of the accident may interfere with effective isolation of the purge line.

These presumptions are not directly applicable to the Mark III containment design. The Reactor Coolant System piping is enclosed in the drywell which communicates with the containment only through the suppression pool. Releases from the primary system are subjected to the quenching and scrubbing action of the suppression pool before entering the containment, so the purge system does not provide a path for primary system releases in the same sense as other containment designs. Even so, special care is being taken in the purge system design, specifically for valve operability assurance (Items 2 and 3 below).

The specific points of NUREG-0718 are addressed below:

1. Purging Consistent with ALARA

The basis for the purge system design is justified in the response to NRC Question 042.63 and was found acceptable in Section 6.2.15 of the Skagit SER (NUREG-0309).

The present design provides for continuous purging of the containment during power operation at 6000 cfm through an 18" line to reduce airborne radionuclide concentrations to a level which permits continuous access. This is in keeping with occupational ALARA considerations, because extensive containment access for routine maintenance is required.

2. Performance of Purge and Vent Valves Against Accident Pressure

The performance of the purge isolation valves has been evaluated and meets the requirements of BTP CSB 6-4, Section B, for isolation and dependability under accident pressures as presented in response to NRC Question 042.63. 21

The purge and vent containment isolation valves are not expected to have to close against the containment design pressure even assuming that a DBA LOCA occurs. The purge and vent lines begin to isolate when drywell pressure reaches 2 psig (almost instantaneously, as shown on Figure 6.2-2). Containment pressure is virtually unaffected for the first several seconds of the accident, and does not rise to near the design pressure for many hours. Regardless, the purge and vent containment isolation valves will be designed to close against the containment design pressure of 15 psig.

3. Interim NRC Guidance on Valve Operability

The purge and vent containment isolation valves will meet the applicable portions of the Setember 27, 1979, interim NRC guidance, "Guidelines for Demonstration of Operability of Purge and Vent Valves." 22

4. Procedures and Restrictions Consistent with Revised Requirements

There are no additional procedures or restrictions on containment purge deemed necessary.

5. Assurance of Purge and Vent Isolation Reliability

The inherent design of the Mark III containment and the added conservatism in isolation valve design and testing give a high level of assurance that the purge and vent lines are reliable to isolate under accident conditions. 21

ITEM II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall provide instrumentation to measure, record and readout in the control room: (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident release points. The requirements for the specific monitors are listed in NUREG-0737. Applicants shall also provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential, accident release points, and for onsite capability to analyze and measure these samples. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the 21
construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

The additional accident monitoring instrumentation required by NUREG-0737, Item II.F.1, Subparts (1) and (3)-(6) will be provided as discussed below:

Subpart (1)

Noble gas effluent radiological monitors at all potential accident release points:

Release Point	Range	
A. Multipurpose vent	10^{-6} to 10^{3} microcuries/cc	22
B. Multipurpose vent which includes drywell or SGTS purge.	10^{-6} to 10^{4} microcuries/cc	

These monitors meet the design criteria specified in NUREG-0737, Table II.F.1-1. 21

Subpart (3)

Containment high-range radiation monitor:

High range (1-10^7 R/hr gamma only) radiation monitoring instrumentation will be provided in the containment and in the drywell. These monitors will meet the design specifications in NUREG-0737, Item II.F.1, Attachment 3, and the requirements of Regulatory Guide 1.97, Rev. 2 (see Item II.F.3). 21 22

Subpart (4)

Containment pressure monitor:

Containment pressure instrumentation with a wide range of -5 psig to at least 45 psig (three times design pressure) will be provided. The upper range may be higher depending on the exact range of the transmitter to be purchased. These instruments are in addition to the normal range containment pressure monitors and meet the requirements of both Regulatory Guide 1.97, Rev. 2, (see Item II.F.3) and NUREG-0737.

Subpart (5)

Containment water level monitor: 21

Suppression pool water level instrumentation, covering the range from the bottom of the ECCS suction line to 5' above the normal suppression pool level will be provided. These instruments are in addition to the normal range suppression pool water level monitors and meet the requirements of both Regulatory Guide 1.97, Rev. 2, (see Item II.F.3) and NUREG 0737.

Subpart (6)

Containment hydrogen concentration monitor:

Containment hydrogen monitoring instrumentation covering the range of 0-30% will be provided which includes the 0 to 10% range requirement of NUREG-0737 and meets the requirements of both Regulatory Guide 1.97, Rev. 2, (see Item II.F.3) and NUREG-0737.

The instrumentation in (1) and (3)-(6) will be redundant, safety grade, seismically and environmentally qualified for accident conditions including the span of its own measured parameter range, and powered from the on-Site electrical system. This instrumentation is known to be commercially available (with the exception of Item (3), as discussed in

the next paragraph), and space has been allocated for transmitter locations in the Plant. The display location in the main control room may be in dedicated post-accident panels or adjacent to or integrated with the existing normal range instrumentation display.

The above supplements the PSAR information on additional accident monitoring instrumentation and is consistent with preliminary design information normally required at the CP stage of review. There are no concerns regarding technical feasibility, state of the art or ability to implement this instrumentation design, with one exception. A fully environmentally qualified high range containment radiation monitor has not yet been found. However, this is not viewed as critical or even significant at this time.

The requirement of Subpart (2), Sampling of Plant Effluents, is not monitoring instrumentation per se, but is rather a sample collection and analysis capability. This will be provided in the manner specified in NUREG-0737, as described below:

Sample collection: The release points with high range noble gas effluent monitors will also have particulate and iodine sampling capability. Iodine samples will be taken with a charcoal or silver-zeolite cartridge and particulate samples with a filter. The post-accident iodine and particulate samples are extracted from the release point via the same sample line as the monitoring line. 21

Sample transport: The sample cartridges will be placed in a portable shielded cask and taken to the counting room.

Sample analysis: Capability for the analysis of sample cartridges will be provided. Design of the counting facility will consider the design basis sample.

The precise location of the sample collection station will be selected upon completion of the post-accident shielding study (Item II.B.2), and the location will assure that a worker involved in the sample collection and transport operation will not receive an exposure greater than 5 rem to the whole body and 75 rem to the extremities.

ITEM II.F.2 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants shall provide instruments that provide in the
control room an unambiguous indication of inadequate core
cooling, such as primary coolant saturation meters in
PWR's, and a suitable combination of signals from indica-
tors of coolant level in the reactor vessel and incore
thermocouples in PWR's and BWR's. 21

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Puget concurs with the techniques described in the BWR Owners Group (BWROG) Emergency Procedure Guidelines submitted to the NRC by letter, D. B. Waters (BWROG) to D. G. Eisenhut (NRC), dated January 31, 1981, to recognize and recover from conditions leading to inadequate core cooling. The guidelines appropriate to the S/HNP BWR6 will be utilized in developing the S/HNP emergency operating procedures.

The existing instrumentation used to measure the reactor
vessel water level includes: the four narrow range, the
three wide range. the two fuel zone, the upset range and 22
the shutdown range instruments shown in Figure 2.3.2.2-1E
of NEDO 24708A. Design descriptions and criteria are given
in Section 7.6.1.2.3.1.2 of the 251 GESSAR which is
incorporated into the S/HNP PSAR.

Water level instrumentation used to detect the approach to inadequate core cooling as required by Regulatory Guide 1.97, Rev. 2 will be provided in the final design and described in the FSAR.

The S/HNP agrees with the stated purpose "to provide in the control room an unambiguous indication of inadequate core cooling". At the present time there is no qualified design for BWR core thermocouples. The S/HNP will install in-core thermocouples which meet the requirements of NUREG-0737 and RG 1.97 Rev 2 if the instrumentation is available at the time of final design commitment. However, S/HNP retains the option of presenting for NRC approval alternate improved instrument design to detect the approach to inadequate core cooling. 22

ITEM II.F.3 INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases and Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

The S/HNP will meet Regulatory Guide 1.97, Rev. 2, with the clarifications of items in Table 1 of the Regulatory Guide noted below.

The S/HNP, recognizing that some generic issues are not resolved, has no concerns regarding technical feasibility, state of the art, or ability to implement the post accident monitoring instrumentation prior to OL issuance. The generic issues of instrument range and qualification applicable to the BWR6 are being worked on by the BWR owners and GE and are being reviewed by appropriate NRC staff. The S/HNP design will incorporate the results of these efforts.

A preliminary list of Type A variables is provided in Table II.F.3-1. The Type A variables list is subject to change as the final design and planned manually controlled actions evolve. The final choice of Type A variables and the instrumentation for those variables will satisfy the provisions of Regulatory Guide 1.97, Rev. 2. 22

Clarifications

1. The following items are not applicable to the S/HNP BWR6 and Mark III containment design:

   a. Suppression chamber spray flow (Note: Containment spray is part of the RHR System which will incorporate flow instrumentation meeting the requirements of Regulatory Guide 1.97, Rev. 2). 22

   b. Drywell spray flow.

   c. Isolation Condenser System shell side water level.

   d. Isolation Condenser System valve position.

   e. Containment and drywell oxygen concentration (for inerted containment plants).

2. Clarification of Terminology

   a. Reg. Guide 1.97, Rev. 2, lists a Drywell Sump Level under Type B instruments which does not 21 appear to have a parallel in the S/HNP Mark III containment design. Reg. Guide 1.97, Rev. 2, lists "Drywell Drain Sumps Level (identified and unidentified leakage)" under Type C which corresponds to the S/HNP drywell equipment drain sump and drywell floor drain sump respectively. Consequently S/HNP will treat these requirements as requiring the drywell equipment and floor drain sumps to have full range level indication and be included under Type C category 1.

   b. Reg. Guide 1.97, Rev. 2, Type D HPCI Flow, is not applicable to the BWR6. The BWR6 high pressure injection system is the High Pressure Core Spray (HPCS) System. The S/HNP design will incorporate HPCS flow instrumentation meeting the requirements 22 of RG 1.97, Rev. 2.

   c. Core Spray System Flow has been changed to Low Pressure Core Spray (LPCS) Flow for the BWR6. This is essentially the same parameter and the S/HNP design will incorporate flow instrumentation meeting the requirements of Reg. Guide 1.97, 22 Rev. 2.

d. Standby Liquid Control System (SLCS) Flow Type D

Due to the relatively low flow rate involved and
the wide pressure range from atmospheric to 21
maximum reactor vessel pressure involved, a
credible direct flow measurement is not possible.
A derived flow measurement is possible using the 22
rate of change in the level of the standby liquid
control tank. This is the technique that is used
to demonstrate that the SLCS meets its design
criteria during preoperational testing. The same
technique using the test tank is employed for
periodic system tests. The S/HNP will use a
derived flow signal to satisfy this requirement.
The instrumentation used to display the derived
flow signal will have a range of 0 to 110% of
design flow. The instrumentation will meet the 22
design criteria for Category 2 instrumentation in
Regulatory Guide 1.97, Rev. 2.

e. High Radioactivity Liquid Tank Level Type D

The S/HNP BWR6 design uses sumps as the primary
collection point. The primary containment sumps 22
incorporate an automatic leak detection system
which provides the operator with an alarm when the
detected leakage rate exceeds a predetermined
value. This system consequently automatically
monitors sump operation for the operator. In
addition to the leak rate detector, the sumps have
high-high level alarms and the key drywell 21
equipment and floor drain sumps incorporate full
range level indication. This integrated system
satisfies the requirement stated in Reg. Guide
1.97, Rev. 2.

f. Area Radiation and Exposure Rate Monitoring

S/HNP will develop a plan for the selection and
location of radiation monitors in containment
penetration areas and in areas where access to
service safety equipment is required. This plan
will be developed in conformance with the 22
provisions of Regulatory Guide 1.97, Rev. 2,
identifying expections, and provide justification
for any exceptions noted. This plan will be
submitted to the NRC prior to the procurement of
any of these monitors.

g. Airborne Radioactive Materials Released from the Plant During and Following an Accident

The S/HNP design utilizes two common vents for identified discharges, the main Plant vent and the Fuel Building vent. The main Plant vent discharge includes the drywell and containment purge. The 21
Fuel Building vent discharge exhausts the Standby Gas Treatment System. The required range for the monitors on these common plant vents is therefore 10^{-6} microcuries/cc to 10^{4} microcuries/cc. Under 22
the current design, system status inputs to the effluent monitoring computer and values determined during preoperational testing are used to derive the necessary flow rates. A study will be undertaken to determine the feasibility of installing flowmeters in the current design and alternative designs which could accommodate flowmeters. The study will be completed within two years of construction permit issuance and the results will be provided to NRC. The method selected for accident airborne radioactive materials release monitoring will comply with Reg. Guide 1.97, Rev. 2, or justification acceptable to NRC for an exception to Reg. Guide 1.97 requirements will be provided. Final design information will be presented in the FSAR. 22

h. BWR Core Thermocouples, Type B and Type C

The S/HNP agrees with the stated purpose "to provide in the control room an unambiguous indication of inadequate core cooling." At the present time, there is no qualified design for BWR core thermocouples. The S/HNP will install in-core thermocouples which meet the requirements of NUREG-0737 and Reg. Guide 1.97, Rev. 2, if the instrumentation is available at the time of final design commitment. However, S/HNP retains the option of presenting for NRC approval alternate improved instrument design to detect the approach to inadequate core cooling.

ITEM II.J.3.1 ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include: (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect/engineer and nuclear steam supply vendor, (4) proposed procedures for handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 21
shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly prior to issuance of the construction permits or manufacturing license."

INTRODUCTION

Section 13.1.1 has been revised to describe the present Puget corporate organizational structure including the role of the Northwest Energy Services Company (NESCO) in engineering and construction management for the Skagit/Hanford Nuclear Project (S/HNP). The following supplements material in Section 13.1.1 and describes Puget's program for management of design and construction activities and for transition to operation.

1. Organizational and Management Structure

Puget has overall responsibility for the design, procurement, fabrication, construction, preoperational testing, operation and quality assurance (QA) activities for the S/HNP. NESCO is responsible to manage the design,

procurement, fabrication and construction of S/HNP with
oversight by Puget.

NESCO is a management and engineering services company
established in 1980 by the four investor-owned Pacific
Northwest utilities. The co-owners are Pacific Power &
Light Company, Portland General Electric Company, Puget
Sound Power & Light Company and The Washington Water Power
Company. NESCO was formed to provide increased emphasis on
project management, engineering and construction management
services for major electrical generating projects of the 21
owner utilities. This organization performs a strong
technical integrating function between Puget and the
Principal Contractors.

PSAR Section 13.1.1 describes Puget's and NESCO's organi-
zational and management structures for the S/HNP. Figure
13.1-1 shows the overall S/HNP organization and indicates
the interface relationship between Puget, NESCO, Bechtel,
GE and Westinghouse. Figure 13.1-2 shows Puget's organiza-
tion for management of design and construction of the
S/HNP. Figure II.J.3.1-1 shows the NESCO organizational
and management structure. The NESCO organization is a
matrix structure consisting of thermal project and 22
technical support organizations. Figure 13.1-3 shows
NESCO's organization for management of design and
construction of the S/HNP. Figure 13.1-4 shows Puget's
interface relationship with NESCO.

2. Technical Resources Directed by the Applicant

a. Staffing Levels

During S/HNP construction, Puget and NESCO will
maintain an in-house staff of engineers and
managers to oversee the design, procurement, 21
fabrication and construction management activities
and verify conformance with applicable
regulations, codes and design criteria. In
specific cases where Puget/NESCO's in-house staff
is not sufficient to meet S/HNP responsibilities,
temporary technical support is assigned from
Puget's in-house line organizations or outside
consultants contracted to work under the direction
of NESCO personnel. To support the construction
of the S/HNP, Puget and NESCO have scheduled
staffing levels as shown in Table II.J.3.1-1.
Puget and NESCO systematically develop manpower
plans annually based on projected work
requirements developed by cognizant managers. 22
These manpower plans are reviewed quarterly and

updated periodically as required by actual workload. 22

b. Level of Education and Experience

Puget and NESCO have and will continue to retain a highly trained and capable technical staff to meet the responsibilities of managing the design and construction of S/HNP. Tables 13.1-1 and 13.1-1a list respectively Puget's and NESCO's current S/HNP technical resources and experience. In addition to these project resources, there is a wide range of technical expertise within the Puget and NESCO corporate organizations covering all major engineering disciplines plus some of the more highly-specialized fields (e.g., instrumentation and control engineering, metallurgy and materials, chemistry, health physics.) Also available are program supervisory expertise in geology/seismology, meteorology, fisheries and aquatics and general environmental and construction impact disciplines. If a technical issue arises that is outside the scope of the Puget and NESCO's technical staff's engineering capabilities, services of outside experts may be utilized to assist in resolving the issue. 21 22

c. Training and Experience Feedback

In addition to hiring experienced individuals, Puget and NESCO have an active technical training program. All professionals have the opportunity and are expected to attend outside developmental courses, seminars and workshops as a means of staff development. They also remain cognizant of current industry concerns and activities. Examples of these activities are: 21

(1) Edison Electric Institute Nuclear Operations Subcommittee

(2) Edison Electric Institute Health Physics Task Force

(3) BWR Owners Group

(4) NSAC/INPO Significant Event Evaluation Information Network (SEE-IN) Program

(5) Edison Electric Institute Quality Assurance Task Force

In addition, Puget and NESCO also use temporary assignments of personnel to other utilities and organizations as a means of staff development.

It should be noted that the above participation significantly broadens the experience of Puget and NESCO personnel assigned to the S/HNP. Additional information on the feedback of design, construction and operating experience is provided in the response to Item I.C.5.

3. Details of the Interaction of Design and Construction Activities

a. General

The following supplements the material in PSAR Section 13.1 on the interaction of design and construction activities by Puget and NESCO and the Principal Contractors, Bechtel Power Corporation (Bechtel) and General Electric Corporation (GE), for the nuclear steam supply system, and Westinghouse for the turbine-generator. Establishment of the division of responsibility and the means of assuring close integration of the work is established in contractual documents, Project (inter-company) procedures, and the GE Customer Interface Data Document. 21

Puget has the overall responsibility for the design, construction and operation of the S/HNP in accordance with NRC regulatory requirements, including the quality assurance requirements of 10 CFR 50, Appendix B. NESCO has the responsibility for providing management oversight of Principal Contractor activities, approving basic design criteria and releasing design documents. Puget retains stop work authority over design and construction activities.

Bechtel is responsible for project engineering management, planning, cost control, engineering, procurement, construction management, contract administration, quality control, quality assurance, and balance of plant preoperational testing. Bechtel is also responsible for design interface control among Bechtel, GE and Westinghouse and between Bechtel and its contractors. Bechtel will perform its services in accordance with applicable Federal, State and local codes and regulations including the quality assurance requirements of 10 CFR 50, Appendix B. NESCO

monitors and evaluates Bechtel's performance of these responsibilities by requiring Bechtel to obtain NESCO's approval of the basic design criteria and NESCO's release of selected design documents prior to purchase or construction and NESCO's acceptance upon completion of construction.

GE is responsible for design and fabrication of the Nuclear Steam Supply System (NSSS), including preparation of design documents and procurement of related hardware. Bechtel reviews these documents to provide interface coordination between the NSSS and balance of plant. NESCO also reviews the GE design and interfaces with BOP systems. Otherwise GE has authority to determine the NSSS design subject to NESCO QA surveillance. GE prepares: interface criteria; safety analyses; other design information; test procedure guidelines; and technical support for NSSS installation. GE is accountable to NESCO to perform its services and provide NSSS designs and equipment in accordance with applicable Federal, State and local codes and regulations, including the quality assurance requirements of 10 CFR 50, Appendix B. 21

Westinghouse is responsible for design and fabrication of the turbine-generator, which does not include activities subject to the quality assurance requirements of 10 CFR 50, Appendix B.

b. Oversight of Design

Puget has the overall responsibility for design of the S/HNP. NESCO is responsible for the management of design activities. NESCO's organization is described in PSAR Section 13.1.1.2.2 and is shown in PSAR Figure 13.1-3.

The NESCO S/HNP staff consists of a Project Manager and technical managers whose function is to manage the design and procurement of the S/HNP. The NESCO S/HNP Project Manager reports to NESCO's Vice President, Nuclear Projects, and is accountable to him for the cost, schedule and quality of S/HNP. The NESCO S/HNP staff manages the contracts of Bechtel, GE, Westinghouse, and outside consultants. All technical direction from NESCO to the Principal Contractors is provided through the NESCO S/HNP technical staff.

In addition to the specific NESCO control aspects over design and procurement activities, NESCO monitors the quality, cost and timeliness of other activities performed by the Principal Contractors. Management oversight of contractor design activities is facilitated by the issuance of several status and performance reports which are directed to various levels of management. Also, copies of correspondence among contractors are sent to NESCO for information.

c. <u>Oversight of Construction</u>

The NESCO internal organization described in Section 13.1.1.2.2 includes the NESCO S/HNP Project Manager, Site Construction Manager and resident engineering staff for management oversight of contractor construction activities on the S/HNP. The NESCO S/HNP organization, which has responsibility for construction management oversight, is described in PSAR Section 13.1.1.2.2.3. Reporting to the S/HNP Project Manager is the Site Construction Manager as shown in PSAR Figure 13.1-3. The Site Construction Manager and his staff are responsible for construction overview of contractor performance. The contractors and subcontractors under Bechtel construction management are responsible for 21 construction activities that conform to design quality requirements. The NESCO Site Construction Manager and his staff monitor construction activities; approve schedules, field procurements, selected invoices, and other financial controls; monitor compliance with permit and license requirements; monitor procedure compliance; monitor coordination of Bechtel field engineering with Bechtel home engineering staff; and coordinate the Bechtel Field Construction Manager's turnover of Plant systems to Puget.

Quality assurance responsibilities are described in Section 17 of the PSAR. NESCO QA provides construction oversight through the NESCO Site QA group which is responsible for monitoring the QA aspects of Site construction, including review of contractor Site procedures; audits and surveillance of construction; identification of quality problems and monitoring of their resolution; and acceptance reviews of components, constructed structures, and completed systems. The Site QA group interacts with Principal

Contractor Site organizations through Bechtel and with the NESCO home office QA organization.

NESCO will assure that approved procedures exist for construction management and control prior to the start of each safety-related construction activity. These procedures will reflect the organization and conform to applicable regulatory requirements, contractual arrangements, and the NESCO Quality Assurance Manual. Procedures will exist for each organizational element involved in safety-related construction oversight activities.

4. Transition to Operation

NESCO has a single Vice President responsible for management of nuclear design and construction. This Vice President functions under the oversight of Puget Vice President, Generation Resources, during design and construction. Puget's Vice President, Generation Resources, has overall responsibility for the S/HNP design, construction, fuel, QA and operation. Puget's staff, functioning under the direction of the Vice President, Generation Resources, will oversee the S/HNP design and construction. This will greatly facilitate the transition from construction to operation. Since the NESCO resident engineering staff will be physically located at the Site during construction and startup, the individuals will have excellent familiarity with the equipment. These individuals will be a basic resource for actual transfer to the operations or engineering support groups. The NESCO technical staff responsible for review and approval of plant design will also be available, as a technically cognizant expert resource, during S/HNP startup and operation. 21

Once the S/HNP becomes operational, Puget will provide the required technical support necessary to assure safe and reliable Plant operation. This support will be consistent with the guidelines suggested in NUREG-0731. Prior to start of S/HNP operation, Puget will consider various organizational alternatives to ensure that: (1) unambiguous management control and effective lines of authority and communication are maintained among the organizational units involved in the management, technical support and operation of the S/HNP; and (2) any potential conflict with the application of resources to non-nuclear functions of the utility are minimized.

Puget intends to employ the operating staff with ample lead time for them to learn the S/HNP design and operation as discussed in PSAR Section 13.2. Furthermore, because NESCO

is a subsidary of Puget, it is Puget's personnel policy to open new technical staff positions to internal staff and to encourage transfers from NESCO to Puget and vice versa. Thus, engineering and management personnel involved in the S/HNP design and construction phases will be encouraged to transfer to Puget S/HNP operating staff positions as they become available, facilitating the transfer of experience to S/HNP operation.

GE, the NSSS supplier, will provide instruction manuals for various pieces of NSSS equipment. These manuals will include operation and maintenance instructions which will be used as references during formation of the S/HNP startup, maintenance, and operation procedures. Puget may request additional procedure guidance from GE during all phases of Plant construction or operation. This will help ensure that Plant operations reflect the engineering expertise in Plant design.

Operating and maintenance procedures will be written by the Plant staff with assistance, as necessary, from the startup organization, NESCO, Principal Contractors and consultants with BWR experience. During this period, the operating staff will have the opportunity to interface directly with 21
personnel in the design organizations. NESCO, Bechtel and GE will provide inputs to the procedures and will review completed draft procedures as appropriate to ensure that design information is accurately reflected. These procedures will be developed on a schedule which will permit their use for operator training and the startup test program.

The operating staff will be directly involved in the preoperational and startup test programs. The startup organization will be under Puget's direction and will be an integrated group including NESCO, Bechtel, GE and Westinghouse personnel. The integrated nature of this group should facilitate communications between these organizations and thus enhance the transfer of design and equipment performance information to the Plant staff.

The trial use of Plant procedures during the test program is described in Section 14.1.3.4. This process should provide further assurance that design information and base line data are incorporated into the Plant operating procedures.

5. Management Oversight

Puget and NESCO corporate functions, responsibilities and authorities are summarized in PSAR Section 13.1.1.2. Puget, under a joint ownership agreement with other

utilities, has sole responsibility and is fully authorized
to act for the owner utilities with respect to design,
construction and operation of the S/HNP.

Puget exercises top level management oversight by assigning
the responsibility for design, construction and operation
of the S/HNP to the Vice President, Generation Resources.
The Vice President, Generation Resources, regularly:
reviews status and progress information; is informed of
significant project decisions, issues, problems and Project
plans for resolution of issues and problems. Weekly
meetings are held by the Vice President, Generation
Resources, to discuss Project status and problems. The
Vice President, Generation Resources, holds monthly Project 22
status meetings with the owner utilities.

NESCO exercises top level management oversight by assigning
the responsibility for design and construction of the S/HNP
to the NESCO Vice President, Nuclear Projects. The Vice 21
President, Nuclear Projects, regularly reviews status and
progress information; is informed of significant project
decisions, issues, problems and project plans for
resolution of issues and problems. Weekly meetings are 22
held by the Vice President, Nuclear Projects, to discuss
Project design and construction status and problems.

The NESCO S/HNP Project Manager provides weekly reports to 22
Puget's Vice President, Generation Resources, and the NESCO
Vice President, Nuclear Projects, and to the S/HNP owner
utilities. These reports identify progress, current
difficulties and planned activities over the next reporting
period. These reports ensure that top-level management is
aware of the S/HNP activities.

Puget's Vice President, Generation Resources, and NESCO's
Vice President, Nuclear Projects, are in frequent communi-
cation and hold regular Executive Review Meetings. NESCO's
Vice President, Nuclear Projects, is in frequent
communication and holds regular meetings (approximately
monthly) with executives of Bechtel and GE thus enabling 22
Bechtel and GE management to be informed regularly of the
Project status, management and technical issues, and plans
for the future.

Additional management oversight is provided by the NESCO
Board of Directors. This board consists of experienced
executives from each of the owner utilities and the 22
President of NESCO. This board meets as required, usually
quarterly, to review NESCO activities.

ITEM II.K.1.22 DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM IS NOT OPERABLE

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. A general explanation of how this requirement will be met is required prior to issuance of the construction permits. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly."

RESPONSE

Operating procedures for the S/HNP have not been written, but the design is such that no manual actions are required initially to mitigate the consequences of a loss of feedwater, although the operator may take anticipatory actions before automatic actions. These manual actions will be specified in the operating procedures, and will be summarized in the FSAR. The following is a discussion of the response of the BWR6 to loss of feedwater transients to demonstrate that no manual operator action is required 21
immediately.

The BWR6 NSSS is designed with self-actuating systems to assure core cooling. An isolation event can be totally accommodated initially by automatic operation of engineered safety feature systems and the Reactor Core Isolation Cooling (RCIC) System which are redundant and diverse. These systems restore and maintain system parameters. During the long term, however, there is adequate time for the operator to take appropriate action. The operator need monitor and control only reactor vessel pressure and level. Furthermore, the operator has multiple parameters available to provide information on system conditions.

All the loss-of-feedwater flow cases result in a proportional reduction of vessel inventory causing the vessel water level to drop. Corrective action normally begins as soon as low feedwater flow is sensed (any one or all pumps) and low level alarm (L4) is reached. At this time, a reduction of the core recirculation flow is initiated to reduce power and thereby reduce the rate of level decrease. The first automatic protective action is the low level (L3) scram trip actuation. The reactor protection system responds within one second after this trip to scram the reactor. The low level (L3) scram trip function meets the single failure criterion.

For the loss-of-feedwater (LOF) and LOF + stuck open relief valve (SORV) cases, main steam line isolation occurs from low steam line pressure.

For the LOF + no HPCS/RCIC case main steam line isolation occurs from low water level (L1) signal. The main steam line isolation signal also initiates a main steam line isolation valve position scram trip as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time due to the L3 scram.

Loss-of-Feedwater

Vessel water level continues to drop reaching the L2 trip at about 20 seconds. At this time, the recirculation system is completely tripped, and High Pressure Core Spray (HPCS) and RCIC systems operation is initiated. After the initiation delay, HPCS and RCIC inject into the vessel causing the vessel water level to reach its minimum value about 6.5 feet above the top of active fuel (TAF). In addition, operation of both HPCS and RCIC will cause the vessel to depressurize which causes a low pressure isolation to occur (assuming the reactor has remained in the RUN mode). After the HPCS and RCIC have tripped on high vessel water level or are regulated by operator action, the vessel will repressurize to the setpoint of the 21
lowest set relief valve which will open to limit the pressure rise.

Loss-of-Feedwater with Stuck-Open Relief Valve

Vessel water level continues to drop reaching the L2 trip at about 20 seconds. At this time, the recirculation system is completely tripped and HPCS and RCIC operation is initiated. After the initiation delay, HPCS and RCIC inject into the vessel causing the vessel water level to reach its minimum value about 6.5 feet above the TAF. In addition, operation of both HPCS and RCIC will cause the vessel to depressurize which causes a low pressure isolation to occur (assuming the reactor has remained in the RUN mode). After the HPCS and RCIC have tripped on high vessel water level or are regulated by operator action, the vessel will repressurize to the setpoint of the lowest set relief valve which will open to limit the pressure rise. It is assumed, in this case, that the relief valve fails to close when the reactor pressure drops below the relief valve reset point, thus remaining stuck open. The SORV causes the reactor to depressurize to the point where the shutdown cooling system can be put into operation.

Loss-of-Feedwater with no HPCS/RCIC

Vessel water level continues to drop, reaching the L2 trip at about 20 seconds. At this time, the recirculation system is completely tripped. With the failure of HPCS and RCIC, the vessel water level continues to drop and the level outside the core shroud reaches the low level (L1) trip. At this time, the main steam line isolation valves will close. The operator can maintain adequate core cooling by manual actuation of the relief valves or Automatic Depressurization System to lower reactor pressure and allow use of the low pressure Emergency Core Cooling System in time to prevent core uncovery. In this case, it was assumed that the operator performed the manual operation at the low level (L1) trip point. 21

ITEM II.K.2.16 IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

NUREG 0718, REV 1, REQUIREMENT

"Applicants shall perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs."

NUREG 0626 ITEM B.4

"The licensees should determine by analysis or experiment, on a plant specific basis, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current power for at least two hours. Adequacy of the seal design should be demonstrated." 21

RESPONSE

1. Nature of Study

Puget is a member of the BWR Owners Group (BWROG) addressing this item.

The concern relates to the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. Adequacy of the seal design should be demonstrated.

The recirculation pump design incorporates a dual mechanical shaft seal assembly to control leakage around the rotating shaft of the recirculation pump. Each assembly consists of two seals built into a cartridge that can be replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for full pump design pressure and can adequately limit leakage in the event that the other seal should fail.

Even though General Electric uses two different recirculation pump configurations, the seal designs are essentially the same. Both designs use hydro-

statically balanced mechanical shaft seals. The subsequent discussion is applicable to both pump designs.

The recirculation pump seals require forced cooling due to the temperature of the primary reactor water and due to the friction heat generated in the sealing surfaces. For all BWR6 reactors, two systems accomplish this forced cooling: (a) the Equipment Protection Closed Cooling Water System (for the S/HNP, this system is delineated as the Reactor Component Cooling Water System), and (b) the Seal Purge System. Cooling water, provided by the Equipment Protection Closed Cooling Water (EPCCW) System, flows through a heat exchanger around the seal assembly. This EPCCW flow cools primary reactor water which flows to the lower seal cavity thereby maintaining the seals at the correct operating temperature. The seal purge system injects clean, cool water from the control rod drive system into the lower seal cavity. This seal purge flow also provides an efficient cooling function for the seals. 21

The Seal Cooling System described above has been examined to determine the consequences of a total loss of cooling on the effectiveness of recirculation pump shaft sealing.

2. Conduct of Study

Under normal conditions, with the primary reactor system at or near rated temperature and pressure with the recirculation pumps either operating or secured, both EPCCS and Seal Purge System are operating. These two systems maintain the seal temperatures at approximately 120°F.

Recirculation pump vendor test data have shown that the pump seals may begin to deteriorate when seal temperatures exceed 250°F. If an event occurs where both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow are totally lost, the recirculation pump seals will heat up. Vendor test data, taken while operating at approximately 530°F/1040 psia, indicate that the seals will heat up, reaching 250°F approximately 7 minutes after the total loss of cooling.

Similar test data indicate that if either one of the seal cooling systems is operating, the seal temperatures remain well below 250°F and no seal deterioration should occur.

If both closed cooling water and seal purge are totally lost, and if the seals heat up to exceed 250°F, seal deterioration may occur, resulting in primary coolant leakage to the drywell. In order to evaluate the fluid loss through a degraded seal, an analysis was performed using the RELAP-4 computer program (see Reference 1).

This analysis modelled the fluid leakage path as a series of fluid volumes with interconnecting junctions, each having appropriate initial conditions. Also, the model assumed gross degradation of the mechanical seals. Gross failure of these seals 21
encompasses warpage, fractures and grooving of the seal faces due to excessive thermal gradients and dirt.

The results of this leakage analysis show that, even with gross degradation of the seals, the leakage would be less than 70 gallons per minute. This amount of leakage is within normal reactor fluctuations, and the normal vessel water level control systems will easily compensate for it. Also, 70 gpm is much less than the bounding values of loss-of-coolant accident analyses, hence there are no adverse effects on LOCA analyses.

3. Completion Date

Puget is closely monitoring the activity of the BWROG, including the study already completed (Reference 2) which is described above. Puget will continue to review this matter and will provide any further 22
information needed to resolve this item to the NRC within 2 years. The resolution of this item will be factored into the final design of the S/HNP and will be submitted in the S/HNP FSAR.

REFERENCES

1. NEDO-24083, "Recirculation Pump Shaft Seal Leakage Analysis," November, 1978 (Licensing Topical Report). 21

2. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated May 22, 1981, and titled "BWR Owners Group Evaluation of NUREG-0737 Requirements II.K.3.25." 22

ITEM II.K.3.13 SEPARATION OF HPCS AND RCIC SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the requirements set forth in Item A.1 of NUREG-0626 as they apply to HPCS and RCIC systems, and perform an evaluation of the safety effectiveness of providing for separation of high pressure core spray system (HPCS) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initates at a higher water level than the HPCS system, and of providing that both systems restart on low water level. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs."

NUREG 0626 ITEM A.1

"Currently, the Reactor Core Isolation Cooling (RCIC) system and the high pressure coolant injection (HPCI) system both initiate on the same low water level signal and 21
both isolate on the same high water level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the RCIC system initiation logic should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed by GE to evaluate these changes. The analyses should be submitted to staff and changes should be implemented if justified by the analyses."

RESPONSE

1. Nature of Study

Puget is a member of the BWR Owners Group (BWROG) study that addresses NUREG-0626, Item A.1.

The concerns cover two aspects of the HPCS and RCIC systems. The first concern is with the initiation levels of these two systems and requests analysis to determine if benefit could be obtained from allowing

the RCIC System to initiate from a higher water level than the HPCS. The second concern is with automatic restart of the RCIC System, and requests analysis to determine if benefit could be gained by introducing this feature.

As previously confirmed in discussions with the NRC, the fundamental issue of the separation of initiation setpoints (water level) is the potential benefit of reducing the number of thermal cycles on the reactor vessel and internals resulting from HPCS operation. It is noted that the S/HNP employs HPCS which does not inject via the feedwater nozzle, consequently the fatigue usage on this component is reduced. Thus the study of this issue, which was based mainly on the BWR4 HPCI arrangement, is conservative for the S/HNP.

Analysis was also made to evaluate the proposed logic change for the RCIC System which permits this system to restart automatically following isolation from high water level. This evaluation considered the logic changes involved, effect on system availability, impact on design reliability and the operator/ equipment interface. 21

2. Conduct of Study

a. Setpoint Separation

The analyses conducted are for typical BWR3 and BWR4 designs where the HPCI and RCIC systems inject via the feedwater spargers. Later plant designs (BWR5 and BWR6) have a separate injection location for HPCS and are less limiting in comparison to the typical BWR3 and BWR4 configuration. Differences in the thermal fatigue analyses are identified where appropriate.

The discussion of the study addresses the potential for reducing the thermal cycles due to HPCI and RCIC initiation. The transients considered are those cited in PSAR Chapter 15. Two classes of transients can cause RCIC and HPCI initiation:

(1) Initiation of HPCI and RCIC on low water level after feedwater is tripped on high reactor water level. For these transients, the inventory is slowly lost due to decay heat steam generation.

(2) Initiation of HPCI and RCIC following a sudden loss of feedwater. For these transients,

inventory loss is rapid with HPCI and RCIC initiation occurring approximately 20 seconds after event initiation.

The details of this study are provided in Reference 1.

b. Automatic Restart of RCIC System

NUREG-0626, Item A.1, requires evaluation of changes to the RCIC System to allow automatic restart following a trip of the system at high reactor vessel water level. The evaluation of this change showed that it would contribute to improved system reliability and that it could be accomplished without adverse effect on system function and Plant safety. The recommended change would be to relocate the existing high level trip from the RCIC turbine trip valve to the steam supply valve. Once the level reaches a predetermined high level the steam supply valve would be closed. One additional relay in the logic circuitry would be required to accomplish the new function. Closure of the steam supply puts the system in a partial standby configuration because of the existing interlocks associated with closure of this valve. Very little modification to the logic circuitry is required to automate realignment of the system in preparation for low water level initiation. This approach was one of several options considered. 21

The details of this study are provided in Reference 2.

3. Completion Date

Completed.

4. Program for Implementation of Results

a. Separation of HPCS and RCIC Setpoints

The results of the analyses for this issue indicate that no significant reduction in thermal cycles can be achieved by separation of these setpoints. It is therefore proposed that the current design values be retained.

b. Automatic Restart of RCIC System

The results of the analyses for this issue indicate that the proposed logic change would contribute to improved system reliability, be of assistance to the Plant operator, and generally enhance safety. This change can be incorporated into the S/HNP design and will be, after NRC approval of the BWROG study. This change will be described in the FSAR.

REFERENCES 21

1. Letter from R. H. Buchholz (GE) to D. G. Eisenhut (NRC) dated October 1, 1980, and titled "NUREG-0660 Requirement II.K.3.13."

2. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated December 29, 1980, and titled "BWR Owners Group Evaluation of NUREG-0737 Requirements."

ITEM II.K.3.16 REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES - FEASIBILITY STUDY AND SYSTEM MODIFICATION

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the requirements set forth in Item A.4 of NUREG-0626, and perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs."

NUREG 0626 ITEM A.4

"The record of relief valve failures to close for all BWRs in the past three years of plant operation is approximately 30 in 73 reactor years (0.41 failures/reactor year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break LOCA. 21
The high failure rate is the result of a high relief valve challenge rate and a relatively high failure rate per challenge (0.16 failures/challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater,
(2) Revised relief valve actuation setpoints,
(3) Increased emergency core cooling (ECC) flow,
(4) Lower operating pressures,
(5) Earlier initiation of ECC systems,
(6) Heat removal through emergency condensers,
(7) Offset valve setpoints to open fewer valves per challenge,
(8) Installation of additional relief valves with a block or isolation valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME code,
(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure,
(10) Lowering the pressure setpoint or MSIV closure,
(11) Reducing the testing frequency of the MSIVs,
(12) More stringent valve leakage criteria, and
(13) Early removal of leaking valves.

GE should investigate the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude)."

RESPONSE

1. Nature of Study

   Puget is a member of the BWR Owners Group (BWROG) study on NUREG-0626, Item A.4.

   NUREG-0626, Item A.4, requires an evaluation of the feasibility and contraindications of reducing challenges to the safety/relief valves (SRVs) by various methods in BWRs. The study reviews potential methods of reducing the likelihood of stuck open relief valve (SORV) events in BWRs and estimates the reduction in such events that can be achieved by implementing these methods. 21

   Reducing the likelihood of SRV challenges will directly reduce the likelihood of an SORV. In addition, attention is also given to modifications which could reduce spurious SRV blowdowns and to modifications which could reduce the probability of SRVs to stick open when challenged.

2. Conduct of Study

   Although the study was precipitated by the consideration of reducing challenges to the SRVs, it was recognized that the true objective was to reduce the incidence of SORV events. In line with this approach, the study also considered reducing the causes of spurious blowdowns and reducing the probability of SRVs to stick open when challenged. The goal of the study was to identify feasible modifications to BWR design and operation, which reduce the frequency of uncontrolled blowdowns by a factor of ten relative to the BWR4 case, which was used as the base case for this evaluation.

   The details of this study are provided in Reference 1. For the BWR6 plants such as the S/HNP, it was concluded that no changes are required to achieve a

factor of ten reduction (relative to operating experience) because:

a. Design features which reduce SRV challenges are already incorporated.

b. The two-stage Crosby valves to be used in the S/HNP design are less likely to stick open due to design differences from the three-stage Target Rock valves on which the operating experience is based.

3. Completion Date

Complete. 21

4. Program for Implementation of Results

The study indicates that the required factor of ten improvement relative to operating experience is met by the present design. Thus, no changes are required to implement the results.

REFERENCES

1. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated March 31, 1981, and titled "BWR Owners Group Evaluation of NUREG 0737 Requirements II.K.3.16 and II.K.3.18."

ITEM II.K.3.18 MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the requirements set forth in Item A.7 of NUREG-0626, and perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs."

NUREG 0626 ITEM A.7

"The ADS actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme which should be considered is ADS actuation on low reactor vessel water level provided no HPCI or HPCS system flow exists and a low pressure ECC System is running. This logic would complement, not replace, the existing ADS actuation logic." 21

RESPONSE

1. Nature of Study

   Puget is a member of the BWR Owners Group (BWROG) study on NUREG-0626, Item A.7.

   This study was made to examine possible modifications to the Automatic Depressurization System (ADS) initiation logic, which would eliminate the need for manual initiation to assure adequate core cooling. For some non-line break events which are further degraded by assuming nonavailability of all high pressure injection systems, manual depressurization of the reactor is required in order to employ the low pressure injection systems. This study examines the advantages and disadvantages of a number of possible ADS initiation logic modifications.

2. Conduct of Study

   Five ADS logic alternatives were considered: Current design and four logic modifications. These four

modifications were: (a) elimination of the high drywell pressure trip; (b) addition of a timer that bypasses the high drywell pressure trip requirement after a certain length of time; (c) addition of a suppression pool temperature trip in parallel with the high drywell pressure trip; and (d) the addition of high pressure system flow measurement and logic in parallel with the high drywell pressure trip.

Each of the options is evaluated on the basis of whether it assures adequate core cooling without operator action for isolations and stuck open relief valve events. Each option is also evaluated for its capability to assure adequate core cooling without operator action. For these analyses it is assumed that all high pressure systems have failed and the ADS must depressurize the vessel to allow the low pressure systems to inject. The modeling used in these analyses is the same as that used in NEDO-24708. 21

The details of this study are provided in Reference 1.

3. Completion Date

The study is complete and was transmitted to the NRC by Reference 1.

4. Program for Implementation of Results

The BWROG concluded that an ADS modification which adds a bypass timer on the high drywell pressure trip requirement or removes the high drywell pressure trip would be beneficial. These changes would not have any major impacts on the plant design. They can be readily incorporated and will upon NRC/BWROG resolution of the item.

REFERENCES

1. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated March 31, 1981, and titled "BWR Owners Group Evaluation of NUREG-0737 Requirements II.K.3.16 and II.K.3.18."

ITEM II.K.3.21 RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATON

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the requirements set forth in Item A.10 of NUREG-0626, and perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final design."

NUREG 0626 ITEM A.10

"The core spray and LPCI system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart if required to 21 assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification."

RESPONSE

1. Nature of Study

Puget is a member of the BWR Owners Group (BWROG) study which addresses NUREG-0626, Item A.10.

In this item, the NRC suggested certain modifications to the Core Spray (CS) and the Low Pressure Coolant Injection (LPCI) Emergency Core Cooling Systems (ECCS) that are provided as part of the BWR ECCS network. The NRC suggestions center on incorporating additional control system logic to provide automatic system restart from a low reactor water level signal following actions by the operators to terminate system operation. The NRC concern is that the reactor operators may terminate ECCS operation when a high reactor water level condition exists but may neglect to reinitiate the systems if a low condition recurs. The BWROG study which is applicable to the S/HNP

design, includes the LPCI and both the low and high pressure core spray systems.

Intuitively, it might appear that additional ECCS automation would be purely beneficial since this would supposedly provide added protection against operator errors and omissions. However, these perceived benefits of extended system automation must be measured against the very real penalties of increased system complexity, reduced system reliability and restricted operator flexibility for dealing with unanticipated events. These considerations are not amenable to precise quantification, and control system design decisions must of necessity involve judgments as to relative importance of these competing influences.

2. Conduct of Study

In order to determine if any overall benefit is to be derived from the postulated design changes, it is necessary to consider the integral nature of the ECCS network and how the ECCS interacts with other Plant systems. The BWROG study provides an overview discussion of the generic GE ECCS design philosophy and design practices as they govern ECCS initiation and operator control of these systems. The need for operator override is identified and how this feature provides for improved overall system reliability. Considerable significance is attached to the complexity of logic and hardware, which would be required to deal with relatively long-term transients involving core and containment cooling on a purely automatic basis. Several long-term transient scenarios are presented to support this contention. 21

The details of the BWROG study are provided in Reference 1.

3. Completion Date

The BWROG study is complete and was transmitted to the NRC by Reference 1.

4. Program for Implementation of Results

The study concluded that while changes to the LPCI/LPCS logic would not have a net positive safety effect, modifications to the HPCS logic to assure a restart on low reactor water level would. This can be readily incorporated into the S/HNP design and will upon NRC/BWROG resolution of this item.

REFERENCES

1. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated December 29, 1980, and titled "BWR Owners Group Evaluation of NUREG-0737 Requirements." 21

ITEM II.K.3.23 CENTRAL WATER LEVEL RECORDING

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. Applicants shall implement design modifications as necessary to meet the requirements. Applicants shall submit, prior to issuance of construction permits, a general explanation of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

NUREG 0626 ITEM B.2 21

"In order to simplify the reading of the water level in the vessel and to provide the operators with a record of water level during transients, all BWRs should have the capability to record vessel water level over the range from the top of the vessel dome to the lowest pressure tap. This range of water level should be available in one location on recorders which meet normal post-accident recording requirements. The recorders should be started on a reactor trip signal."

RESPONSE

Reactor vessel water level instrumentation which meets the requirements of Regulatory Guide 1.97, Rev. 2, spanning the range from the bottom of the active fuel to the steam lines
centerline will be provided as post-accident monitoring 22
instrumentation and will be continuously recorded at one location in the control room.

ITEM II.K.3.24 CONFIRM ADEQUACY OF SPACE COOLING FOR HPCS AND RCIC SYSTEMS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the HPCS and RCIC systems requirements set forth in Item B.3 of NUREG-0626, and perform a study to determine the need for additional space cooling to ensure reliable long-term operation of these systems following a complete loss of offsite power to the plant for at least two hours. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final design." 21

RESPONSE

The Reactor Core Isolation Cooling (RCIC) System and the High Pressure Core Spray System (HPCS) are designated as safety-related systems and are designed to operate independent of off-Site power. Consequently, the RCIC and HPCS space cooling systems are also designated as safety-related, supplied with emergency power (independent of off-Site power) and serviced by the Standby Service Water System, a safety-related water supply system. Each space cooling system is designed to maintain a suitable environment for the long-term operation of the RCIC System and HPCS following a loss of off-Site power.

ITEM II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall provide information to assure that the ADS valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation while taking no credit for non-safety related equipment or instrumentation. Air (or nitrogen) leakage through valves must be accounted for to assure that enough inventory of compressed air (or nitrogen) will be available to cycle the ADS valves. Applicants shall commit that these requirements will be met in the final design at the OL stage.

In addressing this item prior to CP issuance, applicants should note that safety analysis reports claim that air (or nitrogen) accumulators for the ADS valves provide sufficient capacity (inventory) to cycle these valves open five times at design pressures. Also, General Electric has stated that the emergency core cooling systems are designed to withstand a hostile environment and still perform their functions for 100 days following an accident." 21

RESPONSE

The present Automatic Depressurization System (ADS) air accumulators are sized to cycle the ADS valves twice against 70% of containment design pressure (or five times against containment atmospheric pressure) plus accommodate component leakage for seven days. Post-accident access to replenish the air supply (assuming that the supply compressors are inoperative) is being confirmed as part of the post-accident shielding study in response to Item II.B.2. The radiation environmental qualification for the ADS air accumulators and associated components for at least 100 days will be confirmed by this study as well.

Puget is participating in the BWR Owners Group efforts to resolve Item II.K.3.28. The results of this effort will be adopted for S/HNP and design changes made if necessary. 22

ITEM II.K.3.45 EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

NUREG 0718, REV. 1, REQUIREMENT

"Applicants with BWR plants shall address the requirements set forth in Item A.15 of NUREG-0626, and provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs."

NUREG 0626 ITEM A.15

"Analyses to support depressurization modes other than full actuation of the ADS (e.g., early blowdown with one or two SRVs) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown."

RESPONSE 21

1. Nature of Study

   Puget is a member of the BWR Owners Group (BWROG) study which addresses NUREG-0626, Item A.15. This study provides an evaluation of alternate modes of reactor depressurization other than full actuation of the Automatic Depressurization System (ADS). The study includes the BWR6 product line and therefore is applicable to the S/HNP.

2. Conduct of Study

   Depressurization by full ADS actuation constitutes a depressurization from about 1050 psig to 180 psig in approximately 3.3 minutes. Such an event, which is not expected to occur more than once in the lifetime of a plant, is well within the design basis of the reactor pressure vessel. This conclusion is based on the analysis of several transients requiring depressurization via the ADS valves. Results of these analyses indicate that the total vessel fatigue usage is less than 1.0. Therefore, no change in the depressurization rate is necessary. However, to comply with NUREG-0626, Item A.15, reduced depressurization rates were analyzed and compared with the full ADS actuation. The alternate modes

considered cause vessel pressure to traverse the same pressure range in (a) depressurization case 1 (ranges from 6-10 minutes depending on plant size and ADS capacity and (b) depressurization case 2 (ranges from 15-20 minutes). The case 2 depressurization bounds the possible increase in depressurization time by producing an undesirably long core uncovered time. The case 1 depressurization gives the results of an intermediate depressurization. These modes are achieved by opening a reduced number of relief valves.

The details of this study are provided in Reference 1.

3. Completion Date 21

The study is complete and was transmitted to the NRC in Reference 1.

4. Program for Implementation of Results

The study concluded that there is no benefit to be derived from the use of reduced blowdown rates. Therefore, no changes are required to implement the results.

REFERENCES

1. Letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated December 29, 1980, and titled "BWR Owners Group Evaluation of NUREG-0737 Requirements."

ITEM III.A.1.2 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

NUREG 0718, REV 1, REQUIREMENT

"Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional criteria in NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and
criteria. Applicants shall demonstrate that the design 21
concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses."

RESPONSE

Emergency planning for the Skagit Nuclear Power Project has been previously reviewed and found acceptable by the NRC staff, as indicated by Reference 1.

As indicated in Amendment 5 to the Application for Construction Permits and Operating Licenses for the Skagit Nuclear Power Project, Units 1 and 2, the Preliminary Safety Analysis Report (PSAR) for the Project will be amended to reflect the relocation of the Project to a Site on the Hanford Reservation. The amendment of the PSAR will be filed by December, 1981, and demonstrate compliance with 10 CFR 50, Appendix E.

Emergency response facilities will be provided to meet the intent of NUREG-0696, Final Report, dated February, 1981.

The preliminary location of key Puget emergency response personnel is listed in Table III.A.1.2-1. 22

In order to address NUREG-0718, Rev. 1, preliminary information for the S/HNP emergency facilities is provided below:

21

Technical Support Center

Location

The preliminary location of the Technical Support Center (TSC) is within the Control Building on two levels, as shown on Figures III.A.1.2-1, III.A.1.2-2, III.A.1.2-2a and 22

III.A.1.2-2b. The main level at elevation 470' is adjacent to the east end of the control room. The upper level of the TSC is at the next higher elevation of the Control Building at elevation 486', and is adjacent to the east end of the auxiliary panel room. 22

Size and Function

The TSC area adjacent to the control room is approximately 860 square feet. This area would be manned by the Plant Superintendent and key Plant management personnel involved in assessing Plant status and providing support to operations personnel during emergency conditions. Prior to the time the Emergency Operations Facility (EOF) is fully manned and functional, any necessary EOF functions, including communications with off-Site agencies would also be performed at this location. This area was selected in recognition of the importance of facilitating face-to-face communications with control room personnel and providing access to information in the control room that may not be available in the TSC. As shown in Figure III.A.1.2-1, access to back row control room panels from this TSC area is possible without passing through the primary operating area (i.e., front row panel area) of the control room. The inherent advantages of this layout will be supplemented as necessary by administrative controls to minimize congestion 21 which would interfere with operations in the control room and in the TSC area.

During normal operation, the TSC area adjacent to the control room functions as the Shift Supervisor's office and a break/work area for shift personnel. The design will include features, such as movable partitions and sliding doors so that the entire area can be rapidly converted to function as a TSC. When functioning as a TSC the available area should accommodate approximately 10 to 12 individuals.

The area at elevation 486' is approximately 1040 square feet and would be used primarily for technical support personnel (Puget and NRC). It should accommodate about 10 to 15 people who would not necessarily require immediate access to control room personnel and the instantaneous knowledge of Plant status which is available through the displays in the TSC area at elevation 470'. During the preoperational test program it is anticipated that this space would be used as a work area for personnel in the startup organization. Any use of this area after the Plant becomes operational would be limited such that it would not interfere with rapid activation of the TSC for its emergency functions.

The combined TSC areas provide approximately 75 square feet of space for each of the personnel who will be manning

them. Both TSC areas are easily within two minutes walking distance from the control room. Corridors and stairways between the two TSC areas are discussed below under "Structures" and "Habitability."

TSC Equipment and Facilities

The TSC area adjacent to the control room will include the following:

a. Plant data will be available for display on the Data Acquisition System (DAS) and the SPDS. The 22
set of parameters to be displayed in the TSC has not been finalized, as Puget intends to abide by the results of the BWROG efforts in this regard when approved by the NRC. The Shift Supervisor's 22
console (which is one of the Nuclenet panels) is also located in this space and would provide access to CRT displays and data in the Nuclenet computer system.

   The SPDS and Shift Supervisor's console are not Class 1E or seismically qualified. They will be 21
   provided, however, with a reliable power supply and the capability for interconnection to the on-Site power system described in Sections 8.3.1.1.5 and 8.3.1.3.1.j(1) of the PSAR.

b. A video copier will be available for use with either the SPDS or the Nuclenet Shift Supervisor's console.

c. An open work space, which can be connected to the Shift Supervisor's office by a sliding door/wall, will be available for use by TSC personnel who are supporting operators in the control room and performing EOF functions before the EOF is activated.

d. Reliable voice communications to the TSC at elevation 486', OSC, EOF and off-Site agencies including the NRC will be provided.

e. A document storage area will be provided. Drawings, procedures and manuals in this area which are also used for reference during normal operation will be administratively controlled by a formal document control system to ensure that they are maintained up to date.

f. Kitchen and toilet facilities will be provided.

g. Although not included in the TSC designated area, an instrument repair shop is available at the west end of the control room as shown on Figure III.A.1.2-1. 21

The TSC area at elevation 486' will include the following:

a. Plant data will be available for display on the Data Acquisition System (DAS) and the SPDS;

b. An open work space area for TSC personnel which could be subdivided into individual work areas by moveable partitions;

c. A separate office which can be used for private NRC consultations if necessary (this office would be adequate for at least three persons); 22

d. A separate conference room for use by TSC personnel and/or NRC;

e. Reliable voice communications to the TSC area adjacent to the control room, and to the control room, OSC, EOF, and NRC; and

f. A document and records storage area.

Puget recognizes that additional equipment and facilities such as copying equipment, data transmission/telecopier equipment, printers for computer outputs, etc. may be necessary. The need for such equipment and its location will be determined by an analysis of the TSC function and staffing plans and will be described in the FSAR. 21

Structure

Both levels of the TSC are completely within the Control Building which is a seismic Category I structure. Direct access from one level to another is via an elevator within the Control Building or via an access corridor and stairwell on the east end of the Control Building. The access corridor and stairwell are outside of the seismic Category I Control Building structure but within a well-engineered structure.

Access between the two TSC areas is also possible via the stairway in the northwest corner of the control room. This stairway is entirely within the seismic Category I structure. As discussed above and as shown on Figure III.A.1.2-1, people can walk between this stairway and the TSC area at elevation 470' without passing through the

front row panel area thus avoiding any interference with operating personnel.

Habitability

Both TSC areas are served by the control room ventilation system which includes standby filtration with HEPA and charcoal filters. The habitability, therefore, is the same as the control room. Permanently installed radiation monitors and smoke detectors will be added in the TSC areas. 21

The access corridor, elevator and vestibules which connect the two levels of the TSC are served by the Control Building ventilation system which does not include complete standby filtration. This HVAC system is described in Section 9.4.1.2. Puget considers this aspect of the design to be adequate in view of the close proximity of the two TSC areas which would minimize the exposure of anyone walking between these areas.

Operational Support Center

The Operational Support Center (OSC) is an on-Site assembly area separate from the control room and the TSC where Puget operations support personnel will report in an emergency. 22 There will be direct communications between the OSC and the control room and between the OSC and the TSC so that the personnel reporting to the OSC can be assigned to duties in support of emergency operations.

The preliminary location for carrying out the functions of the OSC will be in the Service Building, on the first floor (elevation 415'). 22 This elevation is shown on Figures III.A.1.2-3 and III.A.1.2-3a. The OSC is common to both Unit 1 and Unit 2.

Operations support personnel will assemble in the training classroom and multipurpose area on the first floor of the Service Building. 21 Communications with the Plant, including the two TSC areas, the control room, the EOF, and off-Site locations will be available.

Emergency Operations Facilities

Puget is preposing to meet the requirements of NUREG-0696 (2/81), for an Emergency Operations Facility (EOF) through the joint utilization of the Washington Public Power Supply 22 System (Supply System) Nearsite EOF and a S/HNP backup EOF.

The Supply System Nearsite EOF is located within their Plant Support Facility approximately 5 miles from the S/HNP. The Nearsite EOF consists of a 20,000 square foot

shielded area on the lower level of the Plant Support Facility. Puget has had discussions with the Supply System concerning joint use of the Supply System Nearsite EOF. The Supply System agreed to participate with Puget in a feasibility investigation on the joint use of their Nearsite EOF. This feasibility study will be submitted to the NRC for review prior to proceeding with construction of the S/HNP portion of the joint EOF if joint use of the Supply System Nearsite EOF is proposed. In the event that this study indicates joint use of the Supply System Nearsite EOF is not feasible, a facility, meeting the requirements of NUREG-0696 (2/81), will either be constructed near the S/HNP Site, similar to the Supply System Nearsite EOF, or will be located in Richland, Washington, approximately 12 to 15 miles from the S/HNP Site and design information will be submitted to the NRC for review prior to proceeding with construction of the facility.

Should a backup EOF not be required at the time of EOF construction, Puget reserves the option of deleting the commitment to provide a backup EOF. 22

Based on discussions with the Supply System, the tentative use of the Supply System Nearsite EOF is described below.

Function

The Emergency Operations Facility (EOF) is a nearsite support facility for the management of the overall Puget emergency response (including coordination with Federal, State and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF will have appropriate technical data displays and Plant records to assist in the diagnosis of Plant conditions to evaluate the potential or actual release of radioactive materials to the environment. A senior Puget official in the EOF will organize and manage Puget off-Site resources to support the TSC and the control room operators.

Based on conceptual planning, joint Puget and Supply System utilization of the Nearsite EOF will: (a) improve emergency communications between Puget, Supply System and emergency response organization; (b) facilitate coordination of off-Site radiological monitoring; (c) improve joint dose assessment, and (d) consolidate off-Site response organizations and Puget and Supply System decision centers. This conceptual planning has indicated that an S/HNP emergency should not interfere with the Supply System emergency activities due to an accident at WNP 1, 2 or 4.

Utilization of the Nearsite EOF in the highly unlikely event of coincident WNP 1, 2 or 4 and S/HNP emergencies will be addressed in the feasibilty study and described in the report to be provided to the NRC for review. Preliminary information on joint use of the Nearsite EOF during the highly unlikely event of coincident WNP 1, 2 or 4 and S/HNP emergencies is provided below.

Joint Supply System and Puget Utilization of EOF

A preliminary floor plan of the Nearsite EOF indicating the joint utilization of the facility is shown in Figures III.A.1.2-5 and III.A.1.2-6. An elevation view of the Plant Support Facility showing the location of the EOF is provided in Figure III.A.1.2-7. A preliminary description of the joint use of the Nearsite EOF facilities is provided below:

a. Supply System and Puget Utilization of Nearsite EOF

The Supply System and Puget will have separate Decision Centers to coordinate the individual activities of each company. The Decision Centers will maintain up-to-date information on scheduling of personnel, status of Plant conditions and radiological conditions. 22

b. Off-Site Agency Decision Center

The Supply System and Puget will utilize a joint Off-Site Agency Decision Center which will be used by representatives of the various agencies to coordinate activities, make decisions, resolve problems, maintain duty rosters and provide periodic status reports to their respective agencies. This center receives information from the Meteorology and Dose Assessment and Communications Centers and initiates off-Site protective actions.

Periodic briefings will be provided by the Supply System and Puget in the off-Site Agency Decision Center to keep agency personnel up-to-date on Plant conditions and emergency measures underway.

c. Meteorology and Dose Assessment Center

The Supply System and Puget will utilize a joint Meteorology and Dose Assessment Center. The Meteorology and Dose Assessment Center will be utilized to coordinate Supply System, Puget and other radiological field monitoring teams to perform joint projected dose rate assessments

based on the Supply System and Puget facilities source terms and to record actual measured dose rates. Actual and projected doses will be available for use in making decisions concerning protective actions for the general population.

Outside agencies, particularly the Washington State Department of Social and Health Services, Benton and Franklin County Departments of Emergency Services, Oregon State Health Division, and the Department of Energy will work closely with the Supply System and Puget to evaluate dose projections. Washington State, the Department of Energy, the Supply System and Puget will utilize this information to dispatch teams and work jointly in evaluating field conditions.

d. Communications Center

The Supply System and Puget will utilize a joint Communications Center to provide radio contact with field monitoring teams, assisting agencies and other emergency centers. 22

e. Media Briefing Area

The Supply System and Puget will utilize a joint Media Briefing Area. The Media Briefing Area will be provided in a classroom in the unprotected portion of the Supply System Plant Support Facility Building. The primary Supply System and Puget media centers will be located at their respective corporate offices in Richland, Washington. Groups of media personnel will be escorted to the Nearsite EOF, as conditions permit, for briefings and tours.

f. Media Briefing Preparation Area

The Supply System and Puget will utilize a joint Media Briefing Preparation Area with sufficient equipment for the Supply System and Puget public relations personnel and off-Site agency public information personnel to develop public information media releases.

g. Nuclear Regulatory Commission Work Area

The Supply System and Puget will utilize a joint work area for Nuclear Regulatory Commission (NRC) personnel. The area will be limited to NRC personnel and will have adequate communications

including dedicated circuits to the NRC headquarters in Bethesda, Maryland.

h. Health Physics Center

The Supply System and Puget will utilize a joint Health Physics Center. The Health Physics Center will include external dosimetry, internal dose assessment, a radiological laboratory and respiratory testing facilities.

The external dosimetry area provides automated thermoluminescent dosimeter (TLD) readers which are sufficient to process the increased numbers of TLDs required during an emergency. Results are recorded in a Radiation Exposure Records System which can be accessed for information from both Supply System and Puget plant health physics areas.

The internal dosimetry area provides whole body counting facilities and a computerized internal dose assessment system. The facility is capable of accommodating about twelve whole body counts per hour. 22

The respiratory testing area consists of fitting booths and testing equipment. A large supply of respirators is available, if required, for emergency response operations.

The Radiological Laboratory and adjoining counting room provide for radiological analysis of environmenal samples as well as backup capability for Plant laboratories. If Plant analytical capabilities become unusable, Plant samples can be quickly transported to the Nearsite EOF for chemical and/or radiological analysis. The laboratory is directly accessible from the outside and can be isolated from the remainder of the building.

i. First Aid and Decontamination Area

The Supply System and Puget will utilize a joint First Aid and Decontamination Area which will be located in the unprotected portion of the Plant Support Facility adjacent to the ambulance garage. The layout of this area provides for simultaneous treatment and decontamination of injured personnel as well as independent operation of the first aid and decontamination facilities.

Locations

The Supply System Nearsite EOF is located approximately 5 miles from the S/HNP as shown in Figure III.A.1.2-4. The S/HNP backup EOF will be located in Richland, Washington, approximately 12 to 15 miles from the S/HNP Site as shown in Figure III.A.1.2-4.

Structure

The Supply System Nearsite EOF is a "well engineered" structure and will meet the requirements of the Uniform Building Code. In addition, it will be able to withstand adverse conditions of high winds (other than tornadoes) and floods with a 100-year recurrence frequency.

Habitability

The Supply System Nearsite EOF has special shielding and ventilation to maintain habitability requirements. Two feet of concrete equivalent shielding is provided to ensure that the total dose to occupying personnel is less than the Environmental Protection Agency Protective Action Guide limit of 5 rem whole body for the duration of the postulated accident. The ventilation system is designed to provide maximum habitability during an accidental radiological release. HEPA filters condition entering air during emergency conditions. Radiation detectors are strategically located in the ventilation system to detect impending infiltration of radioactive air thus allowing reconfiguration of airflows from replenishment to recirculation modes. 22

Emergency Technical Information

Equipment will be provided in the Nearsite EOF as shown in Figure III.A.1.2-5 to gather, store and display S/HNP data needed in the EOF to analyze and exchange information on Plant conditions with the designated senior Puget manager in charge of the S/HNP TSC. The EOF technical data system will receive, store, process and display information sufficient to perform assessments of the actual and potential on-Site and off-Site environmental consequences of an emergency condition. Data providing information on the general condition of the Plant will also be available for display in the EOF for Puget resource management. The EOF data set will include radiological, meteorological and other environmental data as needed to:

a. Assess environmental conditions

b. Coordinate environmental conditions

c. Recommend implementation of off-Site emergency plans

The SPDS will be available for display in the EOF. This duplication will provide Puget management and NRC representatives information about the current S/HNP reactor systems status and will facilitate communications among the control room, TSC and EOF.

The EOF will have ready access to up-to-date Plant records, procedures, and emergency plans needed to exercise overall management of Puget emergency response resources. The S/HNP EOF records will include:

a. Plant technical specifications

b. Plant operating procedures 22

c. Emergency operating procedures

d. Final Safety Analysis Report

e. Up-to-date records related to Puget, State and local emergency response plans

f. Off-Site population distribution data

g. Evacuation plans

h. Environs radiological monitoring records

i. Puget employee radiation exposure histories

j. Up-to-date drawings, schematics and diagrams showing conditions of Plant structures and systems down to the component level and in-Plant locations of these systems

REFERENCES

1. NUREG-0309, "Safety Evaluation Report: related to the construction of Skagit Nuclear Power Project, Units 1 and 2," September, 1977. 21

2. Washington Public Power Supply System Emergency Preparedness Plan, Washington Nuclear Projects 1, 2 and 4, Amendment Nos. 1 to 15, April, 1981.

ITEM III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing potential exposures to workers and public following an accident, and providing reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. Applicants shall provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844* source term radioactive materials following an accident, and submit a leakage control program, including an initial test program and a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of 21
their approach to minimizing leakage from such systems outside containment, in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses."

*TID 14844, U.S. Atomic Energy Commission, 1962.

RESPONSE

1. Immediate Leak Reduction

   Systems located outside containment that contain or might contain TID 14844 radioactive material following an accident are the:

   a. Reactor Core Isolation Cooling System (RCIC)

   b. Residual Heat Removal System (RHR)

   c. High Pressure Core Spray System (HPCS)

   d. Low Pressure Core Spray System (LPCS)

   e. Bypass leakage pathways, including Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

   f. Post-accident Sampling System (PASS)

   A defense-in-depth approach has been utilized to maintain the integrity of, and reduce potential for,

leakage from these systems. Fluid penetrations which support Engineered Safety Features Systems (RCIC, RHR, HPCS and LPCS) have isolation valves which may be closed remote-manually from the control room, if required (PSAR Section 6.2.4.1).

The above systems are designed to be low leakage systems and incorporate the following features as discussed in PSAR, Section 12.1:

a. Seamless, welded piping systems will be employed to the maximum extent practicable.

b. High quality valves, valve packings and gaskets will be used.

c. For most larger valves (2½ inches and larger) in lines carrying radioactive fluids, a double set of packing with lantern ring will be provided. A packing gland will also be provided with a leak-off connection which will be piped to a drain header.

d. Metal diaphragm valves will be utilized on those systems where essentially no leakage is permitted. 21

e. High point vents will be provided with double isolation valves or single isolation valves and pipe caps.

f. Pressure test connections for temporary or local instrumentation will be provided with double isolation valves or single isolation valves with pipe caps.

g. Heat exchangers will be provided with tubes of stainless steel or other suitable material with tube-to-tube sheet joints welded to minimize leakage.

In addition, leakage collection systems are designed to reduce exposure to radioactive materials to levels as low as practical. Design features include the following (PSAR, Section 12.1):

a. Low point drains from these piping systems and equipment vents and drains will be piped directly to a collection device connected to the collection system instead of allowing any contaminated fluid to flow across the floor to the floor drain.

b. Valves in some radioactive systems will be provided with leak-off connections piped directly to the collection system.

c. All potentially radioactive sump vents from these systems will be hard piped to the building HVAC exhaust during normal operation. The negative pressure maintained will be sufficient to ensure that backgassing will not occur. Following an accident, any gaseous releases will be treated by the Standby Gas Treatment System (SGTS).

An SGTS is provided to control exfiltration of contaminated air from the Plant following an accident which could result in abnormally high airborne radioactivity levels in the Enclosure Building, Auxiliary Building and Fuel Building. The SGTS will operate to maintain a subatmospheric pressure in these areas. Gaseous radioactive discharges from Engineered Safety Features Systems, which are not isolated from the containment following an accident, will be collected and filtered by the SGTS before release to
the environment. The SGTS will be designed to Seismic 21
Category 1 requirements. Redundant components will be used where necessary (PSAR, Section 6.5).

The MSIV-LCS is designed to minimize the release of fission products which could bypass the SGTS after a LOCA. This is accomplished by directing the leakage from the closed main steam isolation valves through a bleed line into an area served by the SGTS, eliminating direct leakage to the environment (PSAR Section 9.3.6). The additional pathways for bypass leakage, and the mechanisms for controlling and minimizing leakage to the environment, are discussed in PSAR Section 6.2.4.3.5.

The gaseous radwaste system (condenser offgas) will not contain TID-14844 sources because it is isolated from the reactor coolant boundary by means of two main steam isolation valves (MSIVs) in each line. As described above, leakage from the closed MSIVs is collected and processed by means of the MSIV-LCS.

The portions of the liquid radwaste system that might 22
contain TID-14844 sources are limited to the Containment Building by the containment liquid radwaste discharge isolation valves, and potential leakage outside containment is limited to the secondary containment by the Auxiliary Building liquid radwaste discharge isolation valve. These valves are provided with a means of leak testing. In addition, these portions of the liquid radwaste system are

routinely in use and can be periodically observed for leakage. 22

2. Continuing Leak Reduction

The design considerations discussed above are expected to reduce leakage to as low as practical levels for those systems (identified in Part 1) outside containment that would or could contain highly radioactive fluid. A program of surveillance and preventive maintenance will be implemented after the Plant goes into operation to ensure that leakage from these systems is detected and minimized. This program will consist of the following elements:

a. Pressure in the RCIC, RHR, HPCS and LPCS systems is normally maintained by jockey pumps. This system is described in Section 6.3.2.2.5 "ECCS Discharge Line Fill System" in the 251 NSSS GESSAR. The pressure monitoring and control room annunciation associated with the fill system provide a continuous monitor for excessive degradation of the pressure boundary of these systems. Annunciator response procedures for low pressure alarms in these systems will include a step to conduct visual inspections of the system components for leakage. 21

b. Radiation monitors with alarm annunciation in the control room are provided for the Standby Service Water System which removes heat from the RHR, Reactor Component Cooling Water (RCCW), and Fuel Pool Cooling heat exchangers, thereby enabling detection of radioactive fluid in the non-radioactive side due to heat exchanger tube leaks. Additional leakage detection systems are described in Section 1.A.3 of the PSAR.

c. Periodic surveys will be conducted of the RCIC, RHR, HPCS and LPCS systems when they are operating at approximately expected pressure in either normal or test mode. The extent and frequency of these surveys will be sufficient to detect significant increase in leakage from pump seals, valve stems, etc., during the period between the system leak tests described below. The frequency and scope of these surveys will also be revised as operating experience is gained in order to minimize exposure to operators consistent with the ALARA program.

d. Periodic system leak tests will be conducted for the systems identified in Part 1 above. These

leak tests will be conducted at an interval not to exceed each refueling period.

(1) Systems containing gases, i.e., the gaseous portion of the Post Accident Sampling System, will be tested by use of tracer gases, by pressure decay testing or by metered make up tests.

(2) Systems containing liquids will be tested by hydrostatic testing including a thorough visual inspection. In view of the number of boundary valves and the purpose of these leakage tests it is not intended that the leakage rates for these systems be quantified. Valve seat leakage past a minimum flow valve back to the suppression pool, for example, would be measured by such a test but would be of no real significance. Leakage which is significant, i.e., leakage which would result in leakage to the atmosphere within the secondary containment, will be detected by direct visual inspection. 21

e. Maintenance priorities will be high on leakage-related tasks.

f. Leak rate test requirements for bypass leakage pathways are addressed in FSAR Section 6.2.4.3.5 and will be established in S/HNP Technical Specifications.

Details of the testing and surveillance program will be described in the S/HNP FSAR.

ITEM III.D.3.3 IN-PLANT RADIATION MONITORING

NUREG 0718, REV 1, REQUIREMENT

"Applicants shall review their designs to assure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and accident conditions. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

RESPONSE

Portable airborne iodine samplers and sample analysis equipment as required by NUREG-0737 (11/80) Item III.D.3.3 will be available on-Site prior to the issuance of the operating license. This equipment will not be purchased for several years, but it is expected that it will be cart mounted and backup battery powered. Plant personnel will be trained in the use of this equipment under both routine and emergency conditions. Details will be provided in the FSAR.

ITEM III.D.3.4 CONTROL ROOM HABITABILITY

NUREG 0718, REV. 1, REQUIREMENT

"Applicants shall review the design of their facilities for conformance to requirements stated in the Action Plan. Applicants shall evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844* source term release and make necessary design provisions to preclude such problems.

Applicants shall address prior to the issuance of the constructions permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses." 21

*TID 14844, U.S. Atomic Energy Commission, 1962.

RESPONSE

Control room habitability of the S/HNP (previously located in Skagit County, Washington) has been reviewed and found acceptable by the NRC staff, as indicated by Reference 1.

The Control Room Heating, Ventilating, and Air Conditioning (HVAC) System design concept has not been changed from that presented in the PSAR. The control room is a pressurized design, provided with redundant Q-listed HVAC equipment, including redundant emergency filtration units with two separated air intakes (one normal and one emergency), to assure that the control room i habitable at all times. The normal outside air intake ducts and room air exhaust ducts are each provided with two fail-closed isolation valves in series and are single failure proof. On detection of high radiation in the outside air intake ducts:

a. The normal outside air intake and the exhaust ducts will be automatically closed for room isolation;

b. The emergency filtration units will be automatically started to draw a sufficient amount of outside air from the emergency air intake for room pressurization and to filter a part of the recirculated room air; and

c. The operating air conditioning unit with its associated return fan will continue in operation to maintain habitability.

As indicated in Amendment 5 to the Application for Construction Permits and Operating Licenses for the Skagit Nuclear Power Project, Units 1 and 2, the Preliminary Safety Analysis Report (PSAR) for the Project will be amended to reflect the relocation of the Project to a site on the Hanford Reservation. The amendment of the PSAR will be filed in December, 1981, and will demonstrate compliance with the following:

a. Standard Review Plans 2.2.1-2.2.2: Identification 21
of Potential Hazards in Site Vicinity.

b. Standard Review Plan 2.2.3: Evaluation of Potential Accidents.

c. Standard Review Plan 6.4: Habitability Systems.

The following documents shall be used for guidance:

a. Regulatory Guide 1.78: "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

b. Regulatory Guide 1.95: "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

c. In addition, the control room will be reviewed by consideration of the guidance in "K. G. Murphy and K. M. Campe, Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19," 13th AEB Air Cleaning Conference, August, 1974. More recent information on habitability analyses will also be considered.

During the TMI-2 accident, the control room was contaminated via internal pathways. It is observed that the

causes of contamination at TMI-2 were: (a) lack of adequate control room access control, (b) access by contaminated personnel, (c) doors that were left open, and (d) the inability to accurately monitor the control room atmosphere in the recirculation mode.

The S/HNP should not have the above listed difficulties as the Plant will be provided with a dedicated Technical Support Center (TSC) and an on-Site Operational Support Center (OSC) to be used as staging areas for emergency support personnel as discussed in the response to Item III.A.1.2. 21

A three stage continuous air monitor will be provided inside the control room to check accurately on possible control room airborne contamination at all times. Portable iodine monitors (see Item III.D.3.3) will be available for use in the control room for checking on that specific and important type of airborne contamination.

REFERENCES

1. NUREG-0309, "Safety Evaluation Report: Related to the Construction of Skagit Nuclear Power Project," September, 1977.

TABLE I.A.4.2-1

MANPOWER ESTIMATE DURING CONSTRUCTION

MONTHS PRIOR TO FUEL LOAD (FL)	CUMMULATIVE PLANT STAFF(1)	CUMMULATIVE SRO AND RO CERTIFICATIONS(2)	
Start of Construction/96 mos. prior to FL	2	0	
84	4	0	
72	6	0	
60	9	0	21
48	29	1	
36	55	8	
24	104	26	
12	139	34	
Unit 1 FL/Total	168	50(3)	
FL + 12 mos	175	58	

NOTES:

(1) Ref. Table II.J.3.1-1

(2) Based on the use of the Black Fox Simulator

(3) Includes approximately 16 Unit 2 operators and supervisors certified prior to the Unit 2 test program

Table I.D.1-1

Relationship Between S/HNP 6 Human Factors Concepts and NUREG-0659 10 Human Engineering Topics

NUREG 0659 Topic	S/HNP Human Factor Concepts (1)	(2)	(3)	(4)	(5)	(6)
(1) Control Room Workspaces	X	X		X		X
(2) Workplace Environment	X	X				X
(3) Annunciators and Auditory Signals		X	X	X	X	
(4) Controls		X		X		X
(5) Visual Displays		X	X		X	X
(6) Panel Layout	X		X	X		X
(7) Control/Display Integration		X	X	X	X	
(8) Labels and Location Aids	X	X	X	X		
(9) Process Computers		X	X		X	X
(10) Data Recording & Retrieval		X	X	X	X	X

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Table II.B.2-1

Post-Accident Source Terms Bases

Source No.	Core Inventory Releases	Source Medium	Approximate Dilution Volume	Systems Containing Sources	
A	Noble gases 100% Halogen 25% Solids 0%	Air	Containment and Drywell 1.8×10^6 ft^3	Containment, Containment Gas Sample Line	
B	Noble gases 100% Halogen 25% Solid 0%	Air	Drywell 3.0×10^5 ft^3	MSIV Leakage Control, Drywell Gas Sample Line	
C	Noble gases 100% Halogen 25% Solid 0%	Steam	Steam Space 1.4×10^4 ft^3	RHR in Steam Condensing Mode, RCIC Steam Supply	21
D	Noble gases 0% Halogen 50% Solid 1%	Water	Reactor Coolant System 1.1×10^4 ft^3	RHR in Shutdown Cooling Mode.	22
E	Noble gases 100% Halogen 50% Solid 1%	Water	Reactor Coolant System 1.1×10^4 ft^3	Reactor Coolant Sample Line	
F	Noble gases 0% Halogen 50% Solids 1%	Water	Suppression Pool with Makeup & RCS 2.1×10^5 ft^3	RHR in Suppression Pool Cooling, LPCI, and Containment Spray Modes, LPCS, HPCS, RCIC, Suppression Pool Sample Line	
G	Containment leakage rate is 0.25% per day. SGTS flow rate is one air change per day			Standby Gas Treatment System (SGTS)	22

Table II.B.2-2

Post-Accident Source
Terms for Gas-Containing Systems 22

ISOTOPE	CURIES	
KR-85M	3.0+07	
KR-85	1.2+06	
KR-87	4.9+07	
KR-88	7.4+07	
KR-89	9.0+07	
KR-90	1.0+08	
KR-91	7.8+07	
XE-133	2.3+08	
XE-135M	4.0+07	
XE-135	3.0+07	
XE-137	2.0+08	
XE-138	1.9+08	
XE-140	1.0+08	
XE-141	4.1+07	
BR-84	5.5+06	21
BR-85	6.9+06	
BR-86	1.0+07	
BR-87	1.2+07	
BR-88	1.5+07	
BR-89	1.5+07	
BR-90	1.5+07	
I-131	2.8+07	
I-132	3.9+07	
I-133	5.6+07	
I-134	6.0+07	
I-135	5.2+07	
I-136	2.9+07	
I-137	3.5+07	
I-138	2.2+07	
I-139	9.2+06	
Total	1.7+09	

Note: Source terms are based on 100% noble gases and 25% halogens of the core inventory, with no credit taken for decay. 22

Table II.B.2-3

Post-Accident Source Terms for
Depressurized Liquid-Containing Systems 22

ISOTOPE	CURIES	ISOTOPE	CURIES	ISOTOPE	CURIES
BR-84	1.1+07	MO-101	1.9+06	TE-134	2.0+06
BR-85	1.4+07	MO-102	1.8+06	TE-135	1.0+06
BR-86	1.8+07	MO-103	1.5+06	CS-138	2.1+06
BR-87	2.5+07	MO-104	1.2+06	CS-139	2.0+06
BR-88	3.1+07	TC-99M	1.8+06	CS-140	1.8+06
BR-89	3.1+07	TC-100	1.9+05	CS-141	1.4+06
BR-90	5.5+07	TC-101	1.9+06	CS-142	8.6+05
I-131	5.5+07	TC-103	1.8+06	CS-143	4.1+05
I-132	7.8+07	TC-104	1.4+06	BA-139	2.1+06
I-133	1.1+08	TC-105	9.4+05	BA-140	2.0+06
I-134	1.2+08	TC-107	4.1+05	BA-141	1.9+06
I-135	1.0+08	RU-103	1.8+06	BA-142	1.6+06
I-136	5.7+07	RU-105	9.8+05	BA-143	1.3+06
I-137	7.0+07	RU-106	6.6+05	BA-144	7.8+05
I-138	4.3+07	RU-107	6.2+05	LA-140	2.1+06
I-139	2.3+07	RU-108	4.9+05	LA-141	1.9+06 21
SE-84	2.2+05	RH-103M	1.8+06	LA-142	1.7+06
SE-85	2.6+05	RH-104	7.4+05	LA-143	1.6+06
SE-87	2.7+05	RH-105M	2.1+05	LA-144	1.4+06
RB-88	7.4+05	RH-105	9.8+05	CE-141	1.9+06
RB-89	9.4+05	RH-106	7.0+05	CE-143	1.7+06
RB-90	1.2+06	RH-107	6.6+05	CE-144	1.4+06
RB-91	1.2+06	RH-108	5.3+05	CE-145	1.1+06
SR-89	9.8+05	RH-109	3.2+05	CE-146	9.0+05
SR-91	1.2+06	PD-109	3.4+05	CE-147	6.2+05
SR-92	1.4+06	SN-130	3.7+05	CE-148	3.6+05
SR-93	1.6+06	SN-131	3.3+05	PR-143	1.6+06
SR-94	1.6+05	SN-132	2.9+05	PR-144	1.4+06
Y--91	1.3+06	SB-127	1.1+05	PR-145	1.1+06
Y--92	1.4+06	SB-129	3.7+05	PR-146	9.4+05
Y--93	1.6+06	SB-130	5.3+05	PR-147	7.0+05
Y--94	1.7+06	SB-131	9.0+05	PR-148	5.7+05
Y--95	1.8+06	SB-132	9.4+05	PR-149	3.7+05
Y--96	1.7+06	SB-133	1.0+06	ND-147	7.4+05
ZR-95	1.8+06	SB-135	1.7+05	ND-149	4.1+05
ZR-97	1.9+06	TE-127	1.5+05	ND-151	2.1+05
ZR-98	1.8+06	TE-129	3.5+05	PM-147	2.7+05
NB-95	1.8+06	TE-131M	1.8+05	PM-149	6.2+05
NB-97	1.9+06	TE-131	9.8+05	PM-151	2.2+05
NB-99	1.8+06	TE-132	1.6+06	SM-153	3.6+05
NB-100	1.1+06	TE-133M	1.3+06	EU-156	2.0+05
MO-99	2.1+06	TE-133	9.8+05		
				Total	9.4+08

Note: Source terms are based on 50% halogens and 1% solid fission products of the core inventory, with no credit taken for decay. 22

TABLE II.B.2-3a
Post-Accident Source Terms for Pressurized Liquid - Containing System

ISOTOPE	CURIES	ISOTOPE	CURIES	ISOTOPE	CURIES	
KR-85M	3.0+07	Y-96	1.7+06	TE-132	1.6+06	
KR-85	1.2+06	ZR-95	1.8+06	TE-133M	1.3+06	
KR-87	4.9+07	ZT-97	1.9+06	TE-133	9.8+05	
KR-88	7.4+07	ZP-98	1.8+06	TE-134	2.0+06	
KR-89	9.0+07	NB-95	1.8+06	TE-135	1.0+06	
KR-90	1.0+08	NB-97	1.9+06	CS-138	2.1+06	
KR-91	7.8+07	NB-99	1.8+06	CS-139	2.0+06	
XE-133	2.3+08	NB-100	1.1+06	CS-140	1.8+06	
XE-135M	4.0+07	MO-99	2.1+06	CS-141	1.4+06	
XE-135	3.0+07	NO-101	1.9+06	CS-142	8.6+05	
XE-137	2.0+08	NO-102	1.8+06	CS-143	4.1+05	
XE-138	1.9+08	MO-103	1.5+06	MO-101	1.9+06	
XE-140	1.1+08	MO-104	1.2+06	BA-139	2.1+06	
XE-141	4.1+07	TC-99M	1.8+06	BA-140	2.0+06	
BR-84	1.1+07	TC-100	1.9+05	BA-141	1.9+06	
BR-85	1.4+07	TC-101	1.9+06	BA-142	1.6+06	
BR-86	1.8+07	TC-103	1.8+06	BA-143	1.3+06	
BR-87	2.5+07	TC-104	1.4+06	BA-144	7.8+05	
BR-88	3.1+07	TC-105	9.4+05	LA-140	2.1+06	
BR-89	3.1+07	TC-107	4.1+05	LA-141	1.9+06	
BR-90	5.5+07	RU-103	1.8+06	LA-142	1.7+06	
I-131	5.5+07	RU-105	9.8+05	LA-143	1.6+06	22
I-132	7.8+07	RU-106	6.6+05	LA-144	1.4+06	
I-133	1.1+08	RU-107	6.2+05	CE-141	1.9+06	
I-134	1.2+08	RU-108	4.9+05	CE-143	1.7+06	
I-135	1.0+08	RH-103M	1.8+06	CE-144	1.4+06	
I-136	5.7+07	RH-104	7.4+05	CE-145	1.1+06	
I-137	7.0+07	RH-105M	2.1+05	CE-146	9.0+05	
I-138	4.3+07	RH-105	9.8+05	CE-147	6.2+05	
I-139	2.3+07	RH-106	7.0+05	CE-148	3.6+05	
SE-84	2.2+05	RH-107	6.6+05	PR-143	1.6+06	
SE-85	2.6+05	RH-108	5.3+05	PR-144	1.4+06	
SE-87	2.7+05	RH-109	3.2+05	PR-145	1.1+06	
RB-88	7.4+05	PD-109	3.4+05	PR-146	9.4+05	
RB-89	9.4+05	SN-130	3.7+05	PR-147	7.0+05	
RB-90	1.2+06	SN-131	3.3+05	PR-148	5.7+05	
RB-91	1.2+06	SN-132	2.9+05	PR-149	3.7+05	
SR-89	9.8+05	SB-127	1.1+05	ND-147	7.4+05	
SR-91	1.2+06	SB-129	3.7+05	ND-149	4.1+05	
SR-92	1.4+06	SB-130	5.3+05	ND-151	2.1+05	
SR-93	1.6+06	SB-131	9.0+05	PM-147	2.7+05	
SR-94	1.6+06	SB-132	9.4+05	PM-149	6.2+05	
Y-91	1.3+06	SB-133	1.0+06	PM-151	2.2+05	
Y-92	1.4+06	SB-135	1.7+05	SM-153	3.6+05	
Y-93	1.6+06	TE-127	1.5+05	EU-156	2.0+05	
Y-94	1.7+06	TE-129	3.5+05			
Y-95	1.8+06	TE-131M	1.8+05			
		TE-131	9.8+05	TOTAL	2.2+09	

Note: Source terms are based on 100% noble gases, 50% halogens and 1% solid fission products of the core inventory, with no credit taken for decay.

Table II.B.2-4

Post-Accident Vital Areas

Description	Building	Occupancy
Control Room	Control	Continuous
Technical Support Center	Control	Continuous
Operation Support Center	Service	Continuous
Post-Accident Sampling Station	Turbine	Infrequent
Sample Analysis Area	Service	Continuous
Health Physics Area	Service	Continuous
Secondary Alarm Station	Control	Continuous

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Table II.B.2-5

Potential Post-Accident Support Areas*

Description	Location
Class 1E Switchgear, Load Center, and Motor Control Center Areas	Control Building
Class 1E Load Centers and Motor Control Centers, and Standby Service Water Pumps	Ultimate Heat Sink Building
Class 1E Motor Control Centers (1B-231, 1B-236, 1B-131, 1B-136, 1B-137, 1B-235, 1B-237, 1B-135)	Auxiliary Building
Battery Rooms	Control Building
Diesel Generator Areas	Diesel Generator Building
ESF HVAC Mechanical Rooms	Control Building
Radwaste Control Room	Radwaste Building
Remote Shutdown Panel Room	Control Building
Computer Room	Control Building
ADS Air Bottle, Div. 1 and 2	Auxiliary Building

*Note: Occupancy for all of these areas is not anticipated.

21

ABLE II.B.8(1)-1

Initiating Events for Risk Study

1. LOCA
2. Transients
3. Steam/Feedwater line breaks
4. Failures during cold shutdown operation
5. Fire
6. Earthquakes* 22
7. Explosions and missiles, internal and external*
8. Fl ods*
9. Tornadoes, hurricanes*
10. Station blackout, loss of AC/DC

* The best available methodology will be utilized where applicable and will be consistent with IEEE/ANS efforts where appropriate.

TABLE II.B.8(1)-2

Outline of Risk Study Report
Sheet 1 of 2

I. INTRODUCTION

II. SUMMARY

III. METHODOLOGY OVERVIEW

A. Event Trees

B. Fault Trees

C. Quantification of Accident Sequences

D. Containment Failure Analyses

E. Fission Product Release Analyses

F. Treatment of Uncertainties 22

IV. SYSTEM DESCRIPTIONS

A. Performance Requirements

B. Actuation

C. Environment Considerations

D. Dependency Diagrams for Support Systems - Power, Cooling, Lubrication

V. CORE MELT PROBABILITIES

A. Dominant Sequences

B. Dominant Cut-Sets

VI. PLANT MODIFICATIONS THAT ADDRESS DOMINANT SEQUENCES

A. Improvement in Reliability Expected

B. How Factored into Design, Equipment Purchase, Fabrication, Procedures, Operations, etc.

C. Basis for Not Implementing More Reliable Alternatives

TABLE II.B.8(1)-2

Sheet 2 of 2

VII. FISSION PRODUCT RELEASE ANALYSIS

A. Release Groups

B. Containmnt Failure Probabilities 22

C. Fission Product Release Fractions

D. Total Radioactive Release from Containment to Environment for the Various Release Groups

VIII. APPENDICES (DETAILS OF STUDY)

TABLE II.B.8(4)-1

S/HNP AND GRAND GULF
CORE COMPARISON

	GRAND GULF	S/HNP
R P V Size (In.)	251	251
Rating (MWe)	1,331	1,335
Number of Fuel Assemblies	800	848

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TABLE II.B.8(4)-2

S/HNP AND GRAND GULF
COMPARISON OF CONTAINMENT PARAMETERS

	PARAMETER	GRAND GULF	S/HNP
A. CONTAINMENT			
	Diameter (I.D.) (Ft.)	124	130
Wetwell	Volume ($Ft.^3$)	151,644	167,170
Upper Cont.	Volume ($Ft.^3$)	1,248,588	1,296,105
Total Cont.	Volume ($Ft.^3$)	1,400,232	1,463,275
	Initial Temp. ($^0F.$)	80	80
B. DRYWELL			
	Diameter (Ft.)	73	75
	Volume ($Ft.^3$)	270,128	303,455
	Initial Temp. ($^0F.$)	135	135
C. SUPPRESSION POOL			
	Surface Area in Drywell ($Ft.^2$)	533	570
	Surface Area in Cont. ($Ft.^2$)	6,667	7,599
	Volume (Max.) ($Ft.^3$)	138,851	183,960
	Volume (Min.) ($Ft.^3$)	135,245	153,930
	Depth (Ft.)	18.6	18.75
	Initial Temp. ($^0F.$)	85	100
Vents	Number	135	135
	Size and Location	Same	Same
	Drywell Holdup ($Ft.^3$)	50,010	39,000
	Drywell Holdup Surface ($Ft.^2$)	3,145	2,700

Table II.B.8(4)-2

	PARAMETER	GRAND GULF	S/HNP
D. SPRAY SYSTEM			
Cont./Wetwell	Flow Rate (GPM)	11,300	10,500
	Temp. (^0F.)	135	135
Drywell (Break Flow)	Flow Rate (GPM)	14,518	Similar
	Temp. (^0F.)	185	185
Containment to Wetwell Carry Over Fraction	Ratio	0.45	Similar
Upper Pool	Volume Dumped (Ft.^{3})	36,380	40,170
	Temp. (^0F.)	125	125
	Dump Flow Rate (Ft.^{3}/Min)	7,276	5,217
E. PASSIVE HEAT SINKS			
In the Containment, Wetwell and Drywell are Similar to Grand Gulf with Skagit often being slilghtly higher because of the larger containment.		Similar	Similar
F. POST LOCA COMBUSTIBLE GAS CONTROL MIXING SYSTEM*	Flow Rate (CFM)	700	500
	Head (Includes cf H_2O)	467	6.5

*The difference in heads is due to the fact that the Grand Gulf design, the drywell atmosphere is bubbled through the suppression pool, whereas in the S/HNP design, the drywell atmosphere is directly exhausted to the containment at a location above the wetwell.

TABLE II.F.3-1

Preliminary Type A Variables

Variable	Manually Initiated Action
1) Containment Hydrogen Concentration	1) a. Initiate hydrogen mixing recirculation fan
	b. Initiate hydrogen recombiners
	c. Initiate containment spray
2) Suppression Pool Water Temperature	2) a. Initiate suppression pool cooling
	b. Initiate reactor vessel depressurization
3) Ultimate Heat Sink Level	3) Initiate makeup water to ultimate heat sink

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TABLE II.J.3.1-1
SKAGIT/HANFORD NUCLEAR PROJECT
PUGET SOUND POWER AND LIGHT COMPANY AND
NORTHWEST ENERGY SERVICES COMPANY
TECHNICAL MANPOWER ESTIMATE DURING CONSTRUCTION
(IN EQUIVALENT NUMBER OF MEN)

	Years after Receipt of Construction Permit								
	0	1	2	3	4	5	6	7	8
Milestone	CP								FL CO
Staff									
Licensing and Engineering	33	33	35	33	35	30	35	14	9
QA	9	12	15	16	16	16	15	14	9
Construction	6	6	8	10	10	14	15	14	13
Administration	26	2[illegible]	29	29	28	26	23	21	20
Operations	2	4	6	9	29	55	104	139	168
Total	76	83	92	97	118	141	192	202	219

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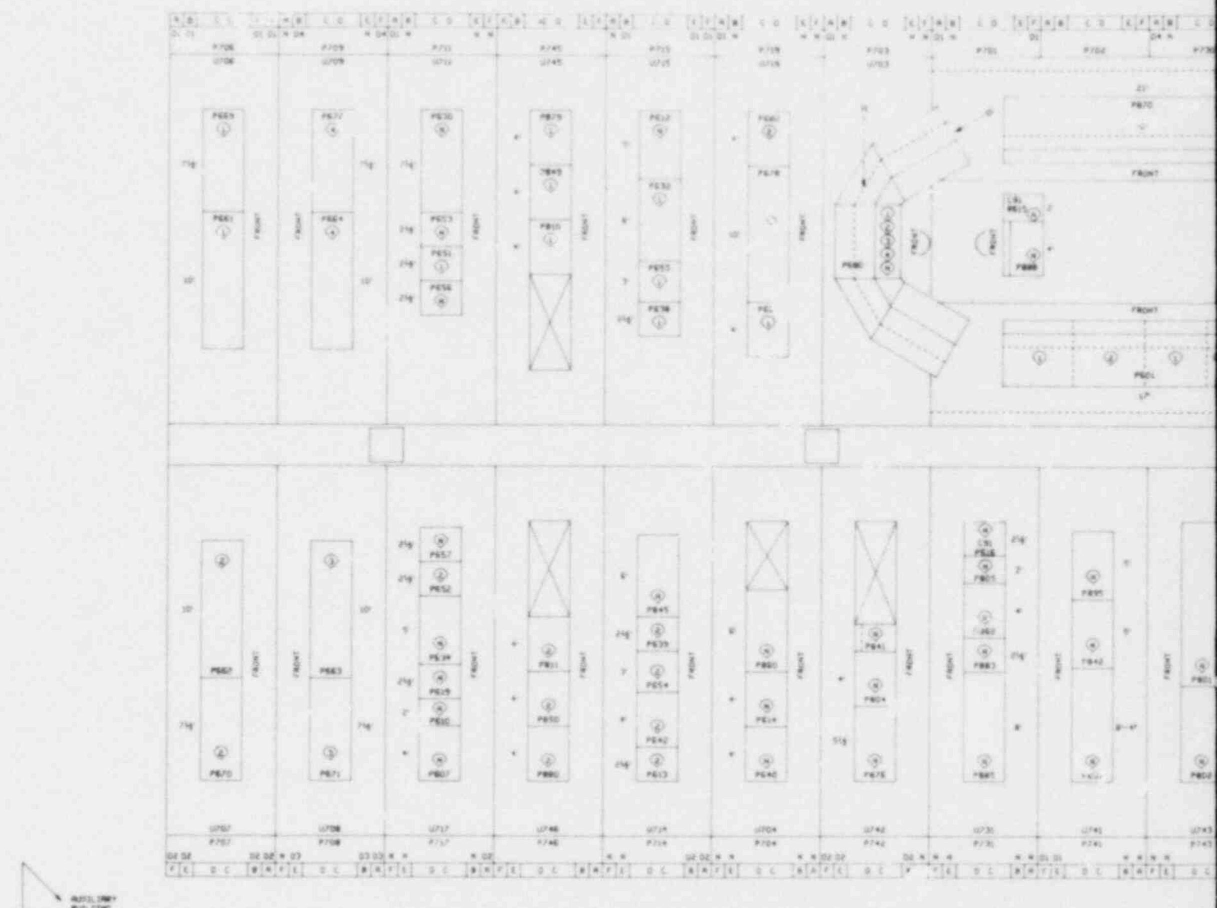
Notes: CP = Construction Permit
FL = Fuel Load
CO = Commercial Operation

TABLE III.A.1.2-1

Location of Key Puget Emergency Response
Personnel During Alert or Greater Emergencies

Personnel	Alert	Site Area Emergency	General Emergency
	Emergency Class		
Emergency Director	TSC	TSC	TSC
Radiation Protection Manager	TSC	TSC	TSC
Recovery Manager	TSC	TSC	TSC
Shift Technical Advisor	TSC	TSC	TSC
Emergency Communications Coordinator	TSC	EOF	EOF
TSC Supervisor	TSC	TSC	TSC
OSC Supervisor	OSC	OSC	OSC
On-Site Emergency Teams	OSC	OSC	OSC
Public Affairs Spokesperson	TSC	EOF	EOF
Off-Site Agency Liaison	TSC	EOF	EOF
Radiologcal Emergency Manager	TSC	EOF	EOF

22

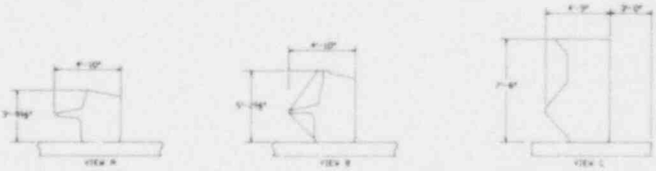


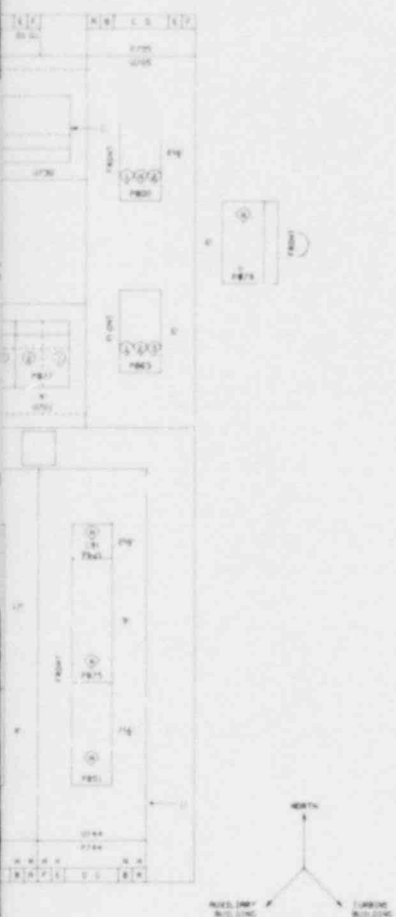
CONTROL ROOM ARRANGEMENT

PGCC FLOOR EL. 470'-1"
CONCRETE FLOOR EL. 469'-0"

NOTES:

1. UNLESS OTHERWISE SPECIFIED, ALL CONTROL ROOM PANEL DESIGNATIONS ARE UNDERSTOOD TO BE PREFIXED BY H13 (E.G. H13-P601)
2. DELETED.
3. TERMINATION CABINET BAY ASSIGNMENTS ARE BASED ON PRELIMINARY INFORMATION FROM ... REF BLP-6338
4. PANELS CONTAINING SAFETY RELATED EQUIPMENT AND/OR CABLES ARE INDICATED BY THE SYMBOL (X), WHERE "X" CORRESPONDS TO THE SAFETY RELATED DIVISION, EITHER 1,2,3 OR 4. NON SAFETY RELATED PANELS ARE INDICATED BY THE SYMBOL (N)
5. SPDS LOCATION WILL BE DETERMINED BASED ON AN ANALYSIS OF THE OPERATOR'S NEEDS AND A FUNCTIONAL ANALYSIS OF THE USE OF THE SPDS.





PANEL NO.	DESCRIPTION
H13-P601	REACTOR CORE COOLING BENCHBOARD
H13-P607	TIP CONTROL AND MONITORING
H13-P610	CONTROL ROD DRIVE TEST PANEL
H13-P612	FEEDWATER AND RECIRC INSTRUMENT PANEL
H13-P613	NSSS PROCESS INSTRUMENT PANEL
H13-P614	NSSS TEMPERATURE RECORDERS
H13-P619	JET PUMP INSTRUMENT PANEL
H13-P630	REMOTE ANNUNCIATOR ELECTRONICS PANEL (NSSS)
H13-P632	LEAK DETECTION DIV 1
H13-P634	RECIRC FLOW CONTROL PANEL
H13-P638	CONTAINMENT ATMOSPHERIC MONITORING PANEL A
H13-P639	CONTAINMENT ATMOSPHERIC MONITORING PANEL B
H13-P640	TRANSIENT TEST PANEL
H13-P642	LEAK DETECTION DIV 2
H13-P651	ROD ACTION CONTROL CABINET DIV 1
H13-P652	ROD ACTION CONTROL CABINET DIV 2
H13-P653	ROD DRIVE CONTROL CABINET
H13-P654	MSIV LEAKAGE CONTROL DIV 2
H13-P655	MSIV LEAKAGE CONTROL DIV 1
H13-P656	DCS RAVS AND RDVS (NSSS)
H13-P657	DCS RAVS AND RDVS (NSSS)
H13-P661	NSPS DIV 1 CABINET
H13-P662	NSPS DIV 2 CABINET
H13-P663	NSPS DIV 3 CABINET
H13-P664	NSPS DIV 4 CABINET
H13-P669	NEUTRON / MSL RADIATION MONITORING CABINET A
H13-P670	NEUTRON / MSL RADIATION MONITORING CABINET B
H13-P671	NEUTRON / MSL RADIATION MONITORING CABINET C
H13-P672	NEUTRON / MSL RADIATION MONITORING CABINET D
H13-P678	STANDBY INFORMATION PANEL
H13-P679	SUPERVISORS CONSOLE
H13-P680	NUCLEAR CONSOLE
H13-P681	POST ACCIDENT MONITORING DIV 1
H13-P682	POST ACCIDENT MONITORING DIV 2
H13-P800	AUXILIARY LONGTERM RESPONSE BOARD
H13-P801	GENERATOR AND TRANSFORMER PROTECTIVE RELAYING
H13-P802	PLANT ELECTRICAL METERING BOARD
H13-P804	METEOROLOGICAL MONITORING PANEL
H13-C805	MASTER SWITCHYARD SUPERVISORY CONTROL CABINET
H13-P810	RADIATION MONITORING DIV 1
H13-P811	RADIATION MONITORING DIV 2
H13-P841	ATS AUTOMATIC TURBINE STARTUP TYPER
H13-P842	BOP TEMPERATURE RECORDERS
H13-P845	OFF GAS LOW TEMPERATURE CONTROL PANEL
H13-P849	BOP ESF SYSTEMS DIV 1
H13-P850	BOP ESF SYSTEMS DIV 2
H13-P851	REMOTE ANNUNCIATOR ELECTRONICS PANEL (BOP)
H13-P860	PROCESS AND AREA RADIATION MONITORING PANEL
H13-P862	FIRE PROTECTION CONTROL AND ALARM PANEL
H13-P863	HVAC CONTROL PANEL
H13-P870	BOP LONGTERM RESPONSE BOARD
H13-P875	BOP AUXILIARY RELAYS AND ELECTRONICS PANEL
H13-P876	RFPTS SUPERVISORY INSTRUMENTS
H13-P877	DIESEL GENERATORS NO. 1 AND 2 CONTROL BOARD
H13-P879	ESF SYSTEMS AUXILIARY RELAY PANEL DIV 1
H13-P880	ESF SYSTEMS AUXILIARY RELAY PANEL DIV 2
H13-P881	RAW WATER SUPPLY CONTROL PANEL
H13-P883	...URING PANEL
H13-P88	...GUING PANEL
H13-P	...TIONS CONSOLE
H13-...	...ELECTRONICS
C91-R615	VIDEO COPIER
C91-P616	PMS ANALOG I/O CABINET (BOP)
C91-P621	PMS DIGITAL I/O CABINET (BOP)
	PANELS LOCATED IN
	UPPER CABLE SPREADING ROOM
H13-P637	PRESSURE REGULATOR PANEL (STEAM BYPASS)
H13-C807	SECURITY COMPUTER
H13-P821	DEH COMPUTER
H13-P822	TURBINE SUPERVISORY INSTRUMENTS PANEL
H13-P887	REHEAT TEMPERATURE CONTROLLER CABINET
H13-P851	RFPTS ELECTRONICS CABINET
H13-P899	EMERGENCY TRIP SYSTEMS CABINET
	SAFETY PARAMETER DISPLAY SYSTEM (NOTE 5)

PUGET SOUND POWER & LIGHT COMPANY
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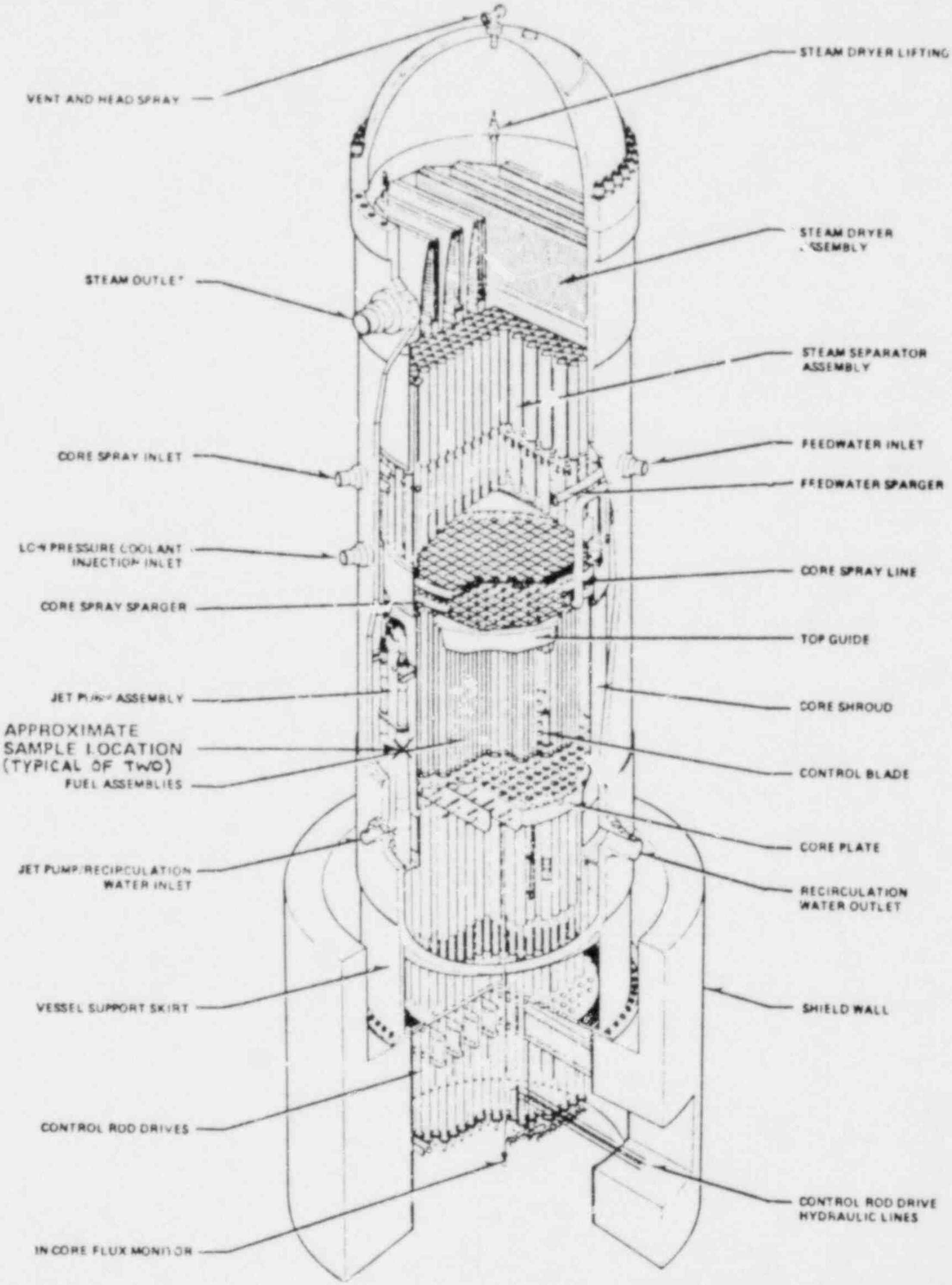
CONTROL ROOM LAYOUT

FIGURE I.D.1-1



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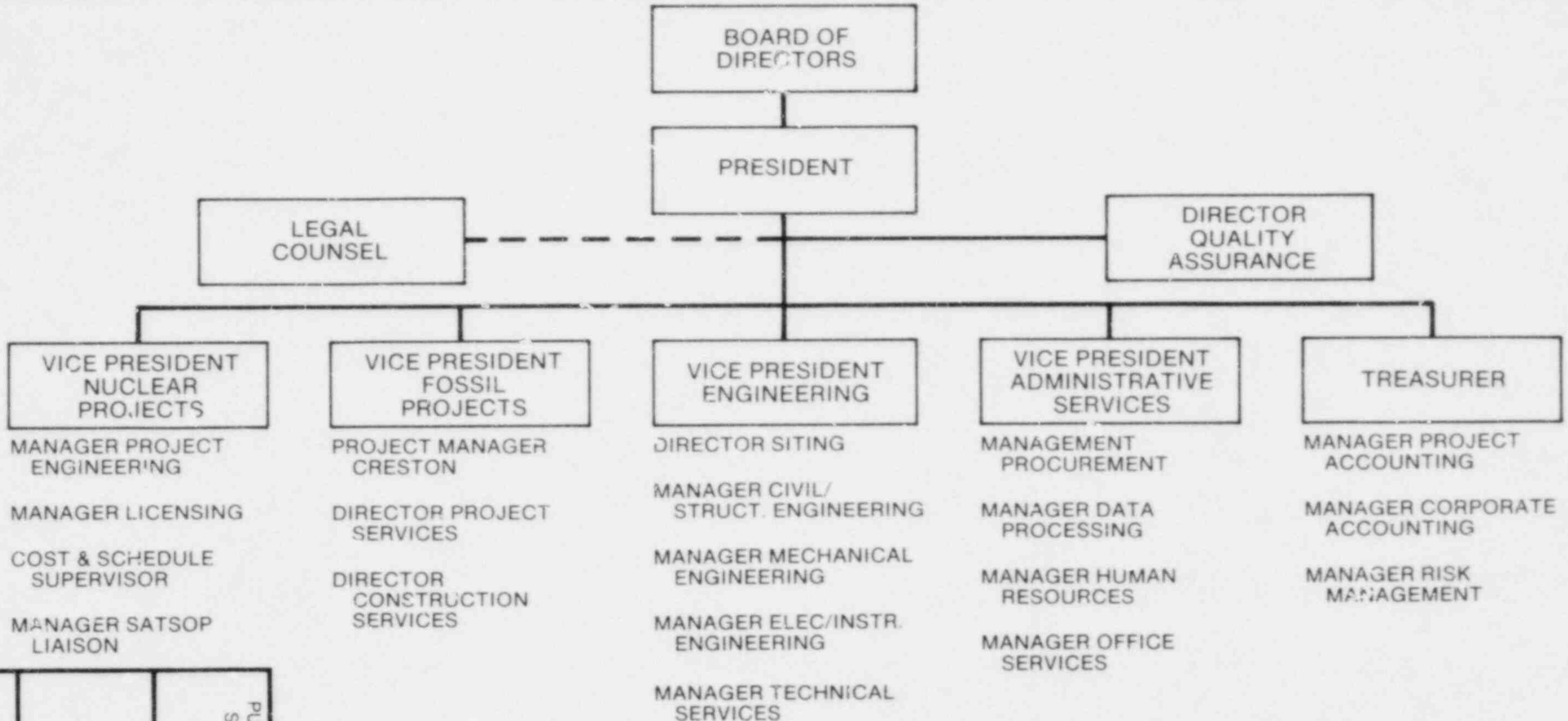
LOCATION OF POST-ACCIDENT RCS SAMPLE POINT

FIGURE II.B.3-1



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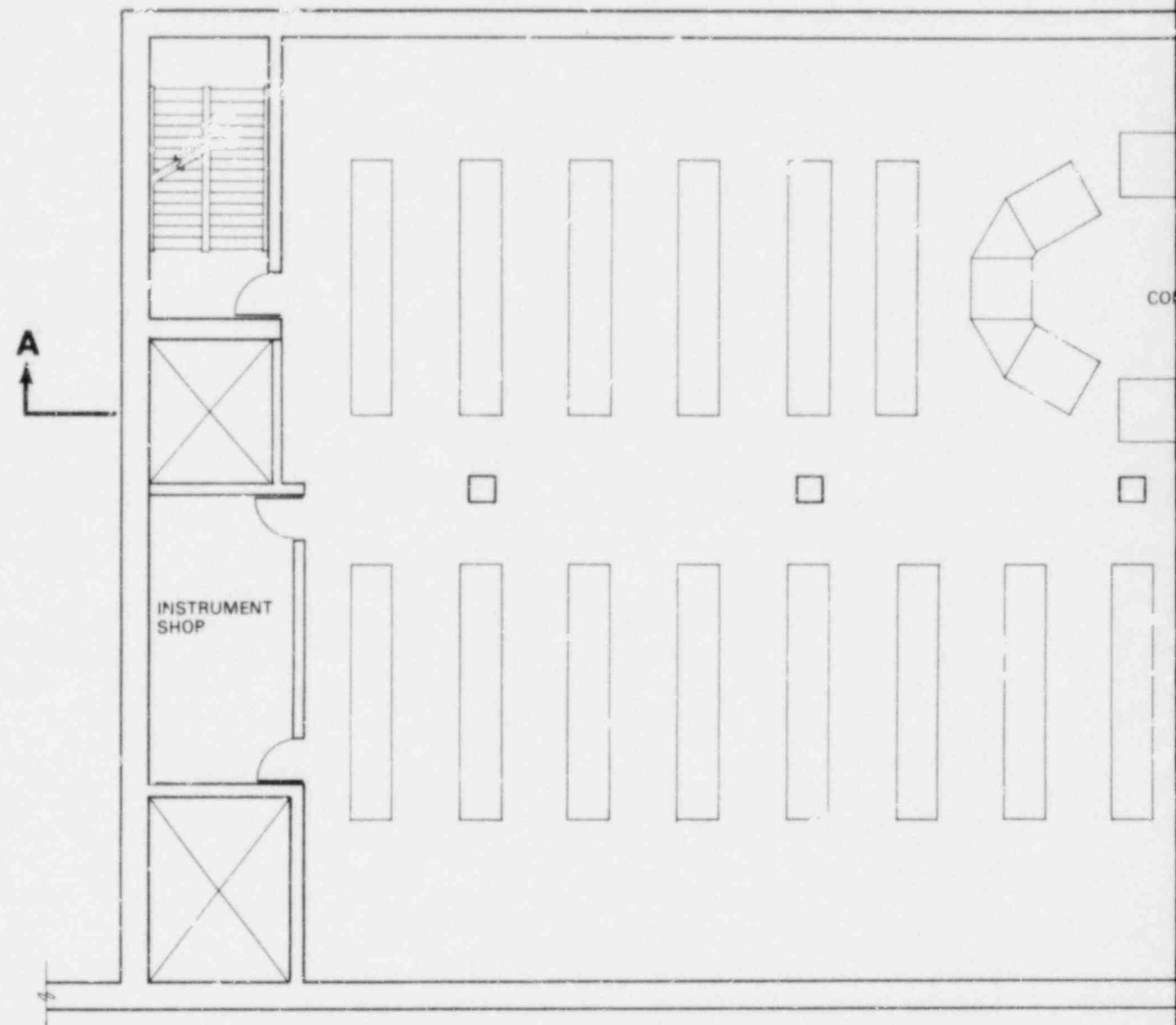
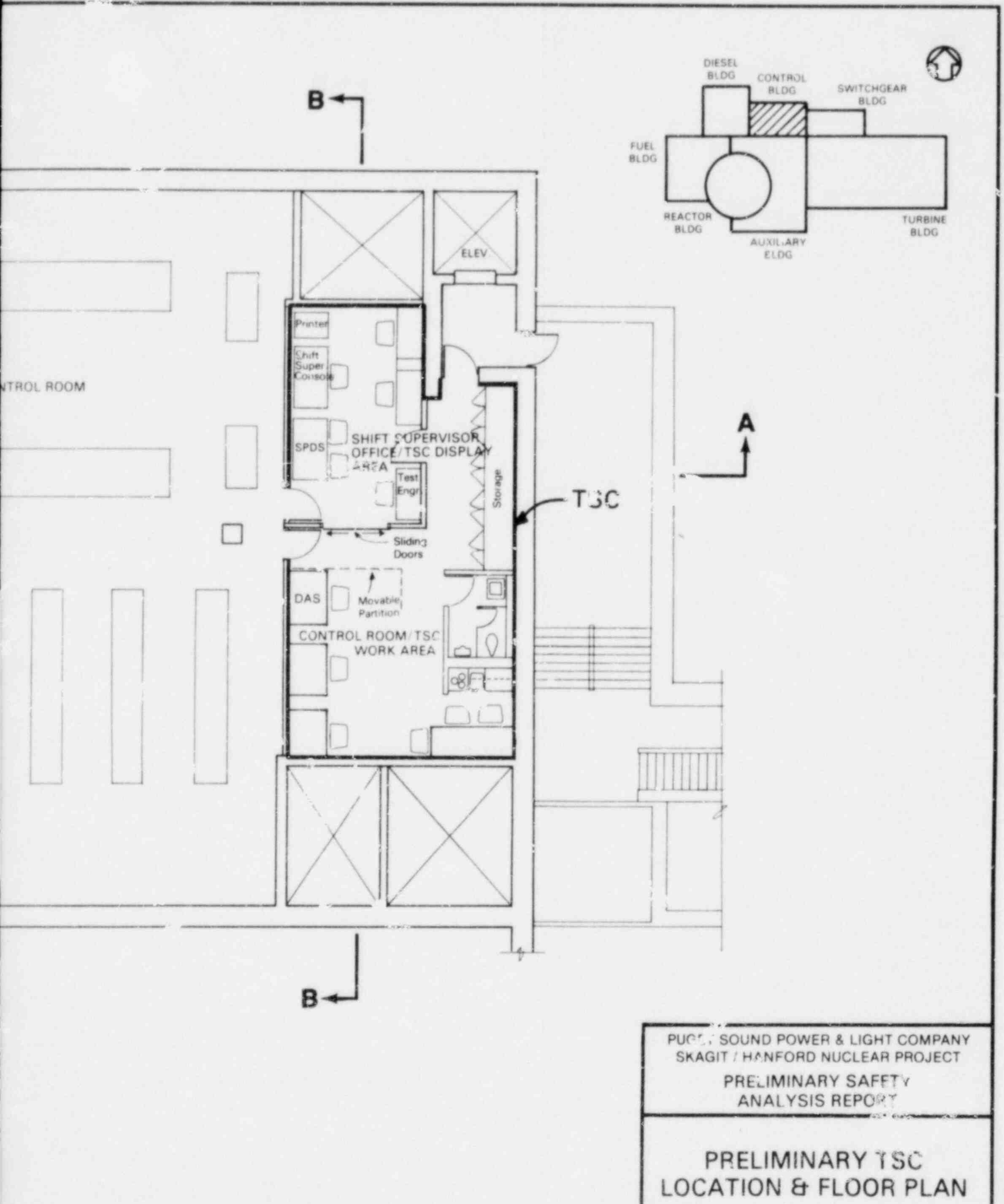
NESCO ORGANIZATION

FIGURE II.J.3.1-1



CONTROL BUILDING EL. 470'-0"



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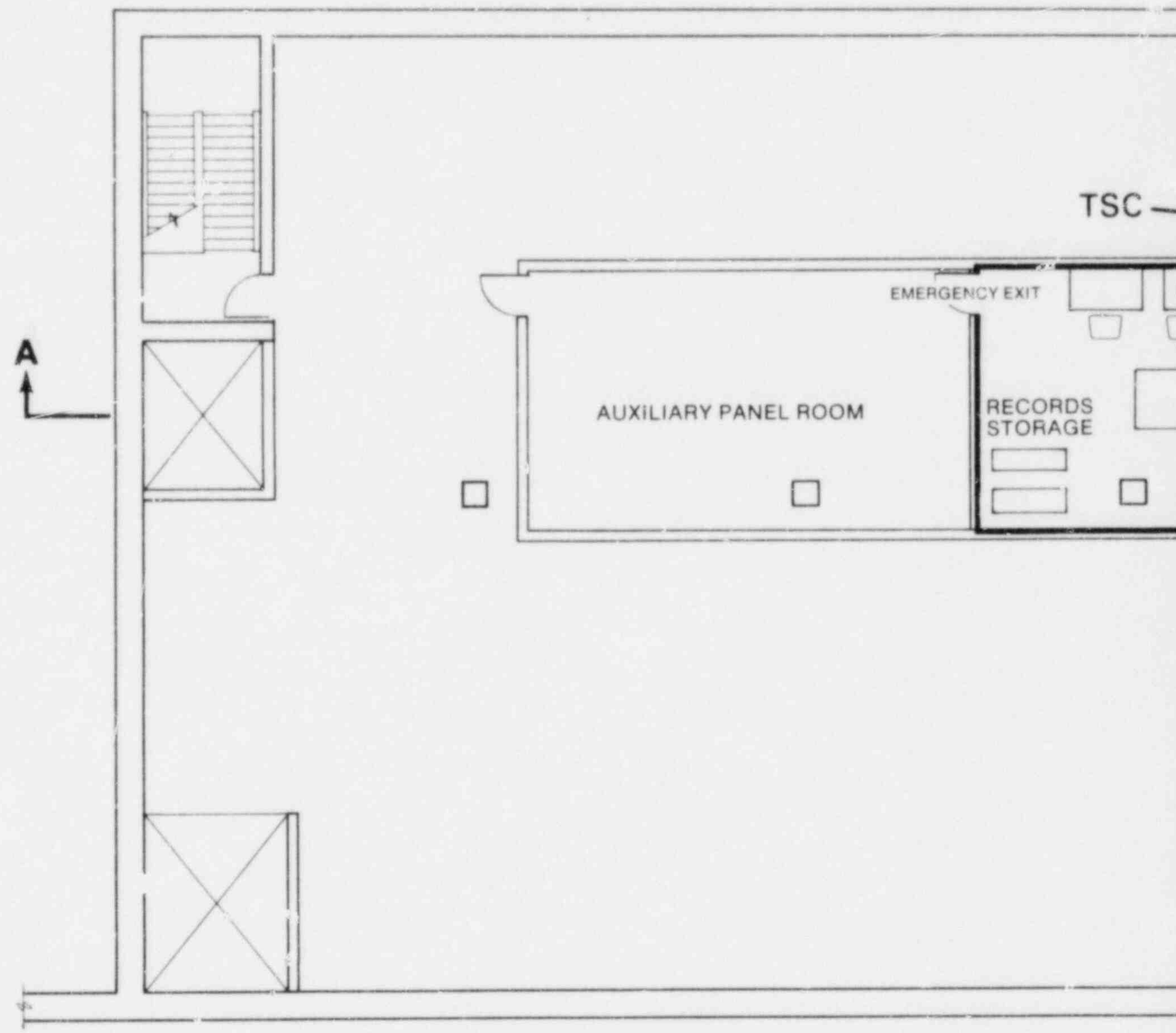
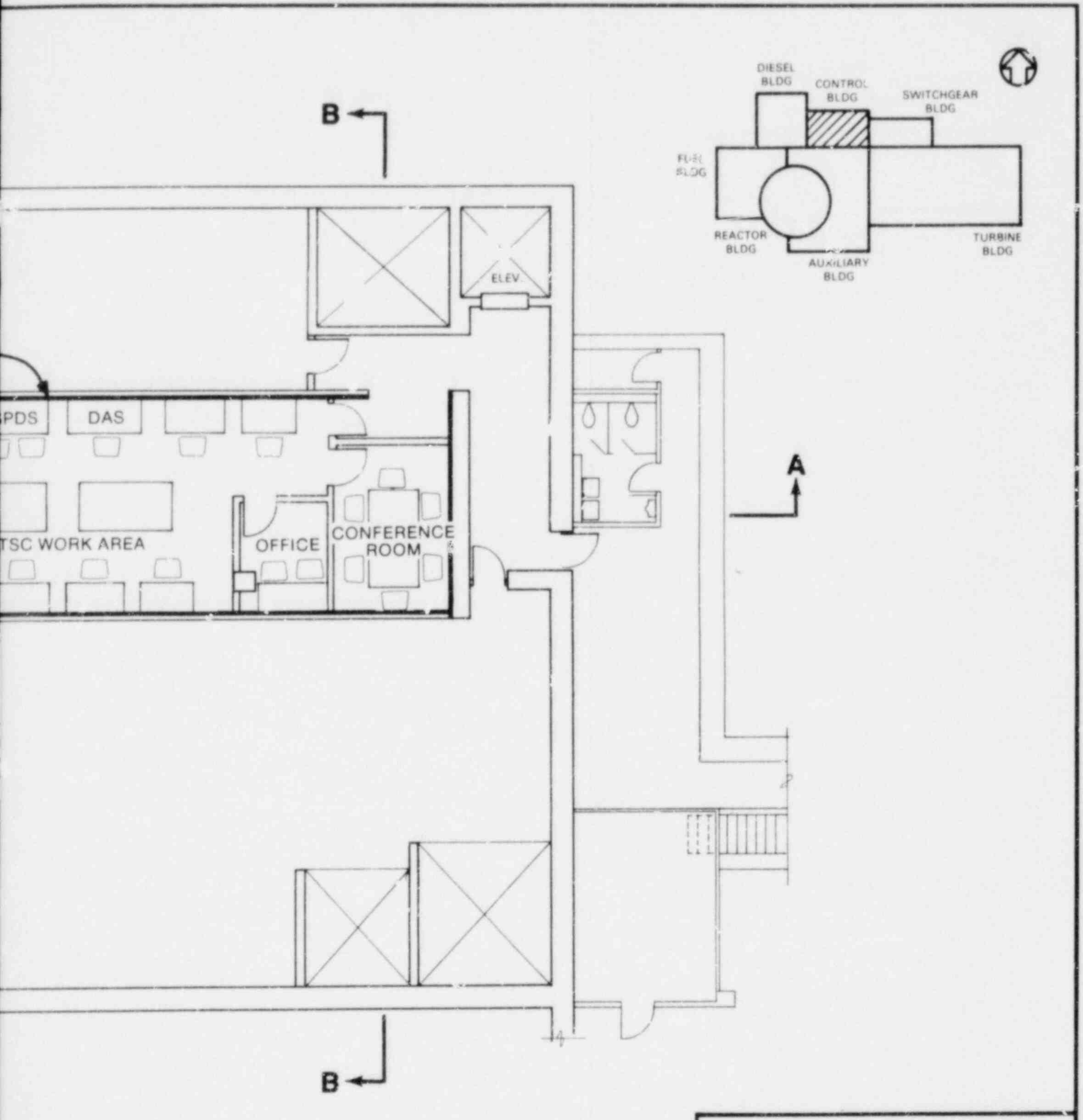
PRELIMINARY TSC LOCATION & FLOOR PLAN

FIGURE III.A.1.2-1



CONTROL BUILDING EL. 486′-0″



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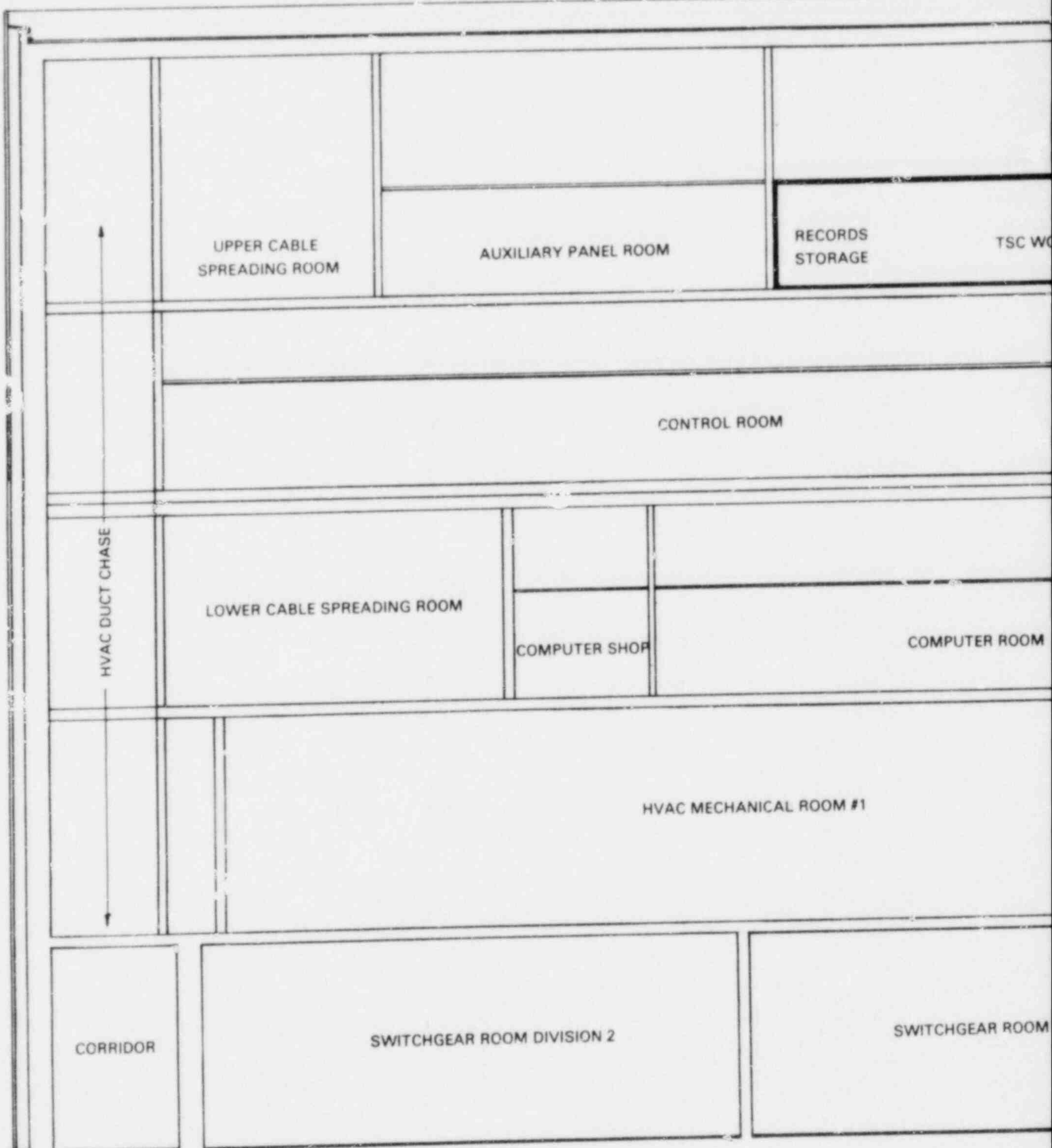
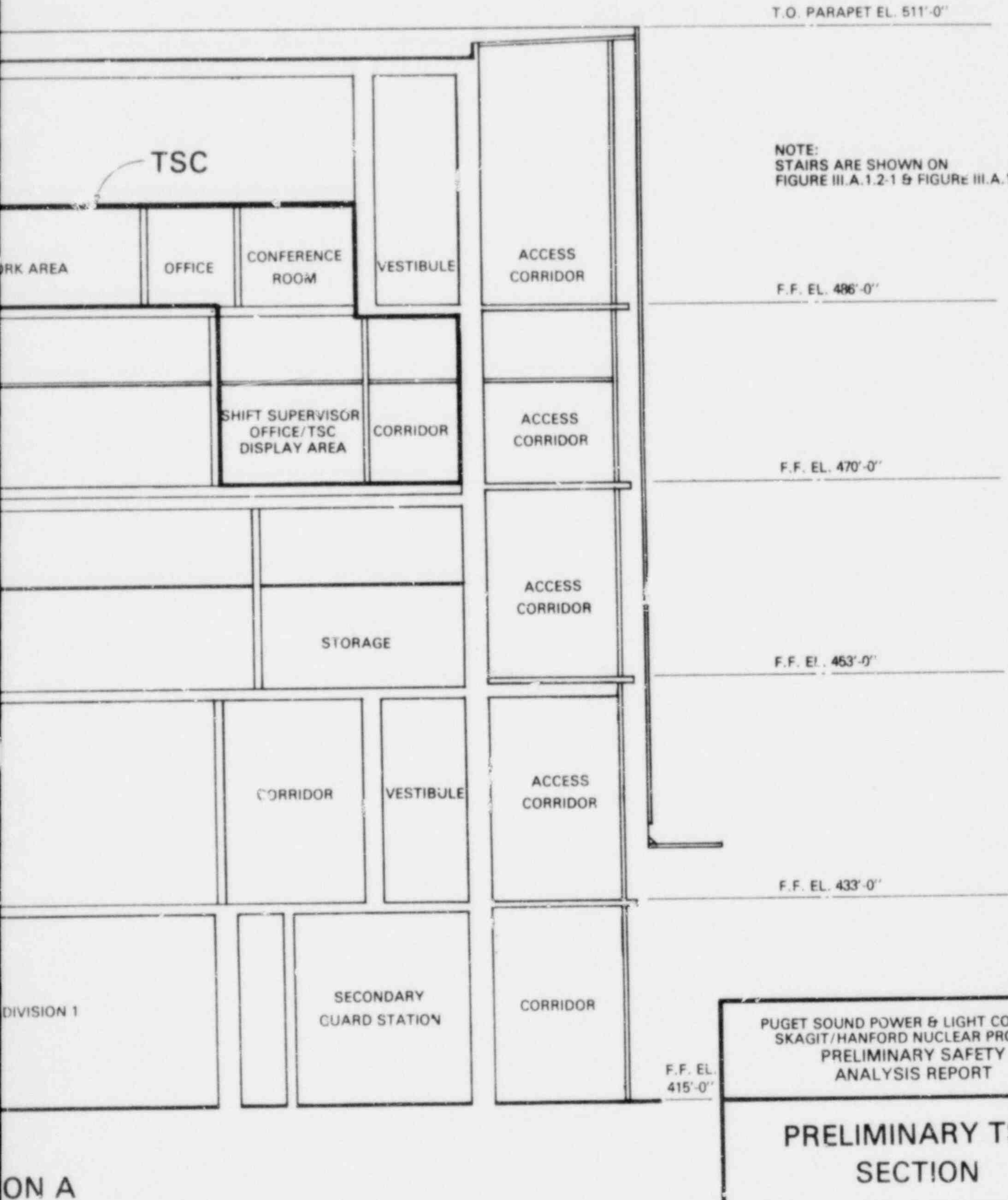
PRELIMINARY SAFETY ANALYSIS REPORT

PRELIMINARY TSC LOCATION & FLOOR PLAN

FIGURE III.A.1.2-2



CONTROL BUILDING SECT



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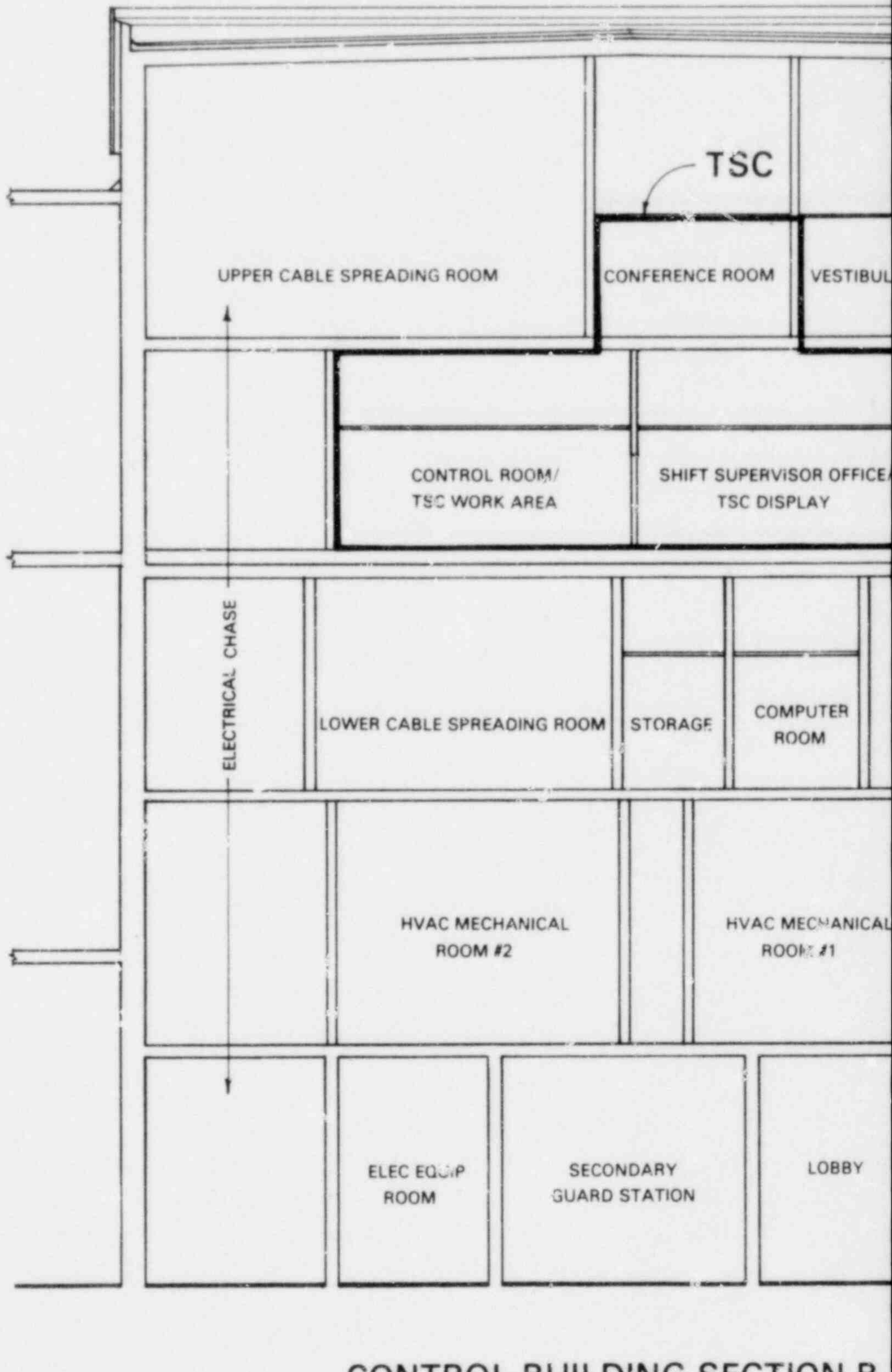
ANALYSIS REPORT

PRELIMINARY TSC SECTION

FIGURE III.A.1.2-2a



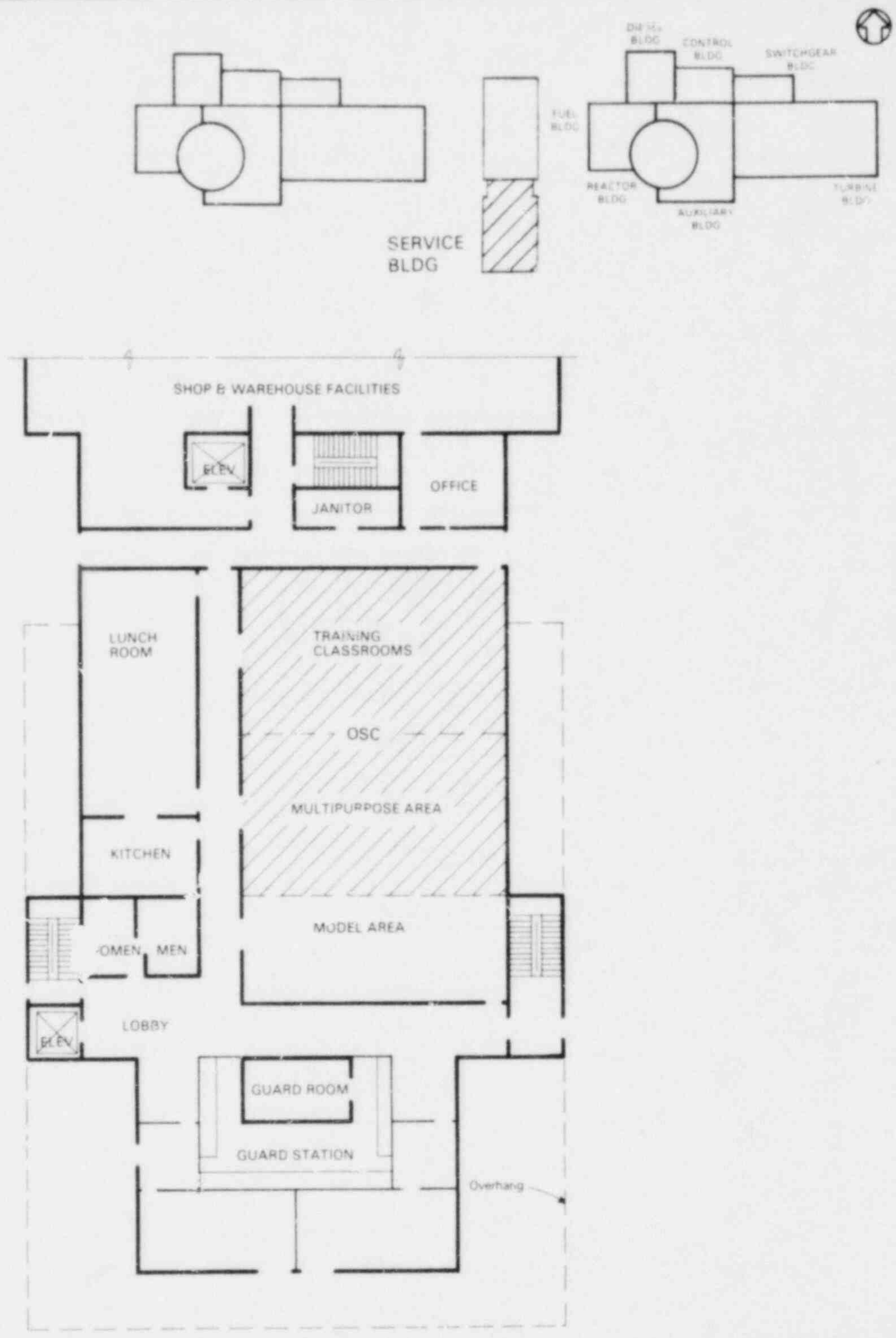
CONTROL BUILDING SECTION B

NOTE:
STAIRS ARE SHOWN ON
FIGURE III.A.1.2-1 & FIGURE III.A.1.2-2

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

PRELIMINARY TSC
SECTION

FIGURE III.A.1.2-2b

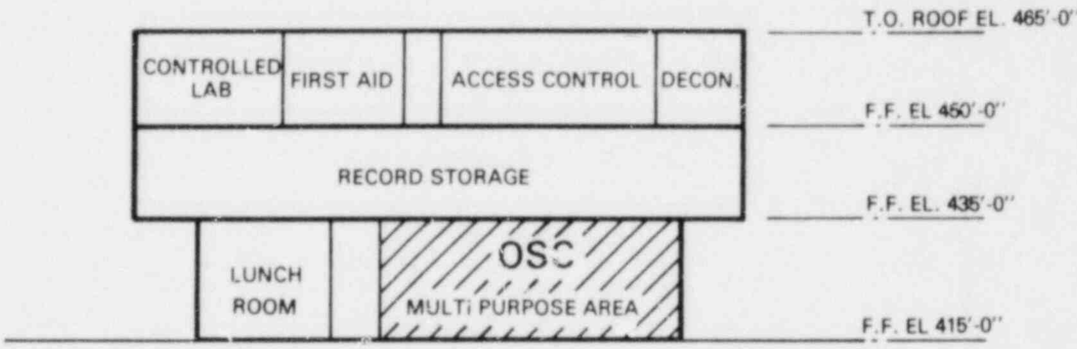


SERVICE BUILDING
FIRST FLOOR — EL. 415'-0"

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

PRELIMINARY OSC
LOCATION

FIGURE III.A.1.2-3



SERVICE BUILDING SECTION A

PUGET SOUND POWER & LIGHT COMPANY
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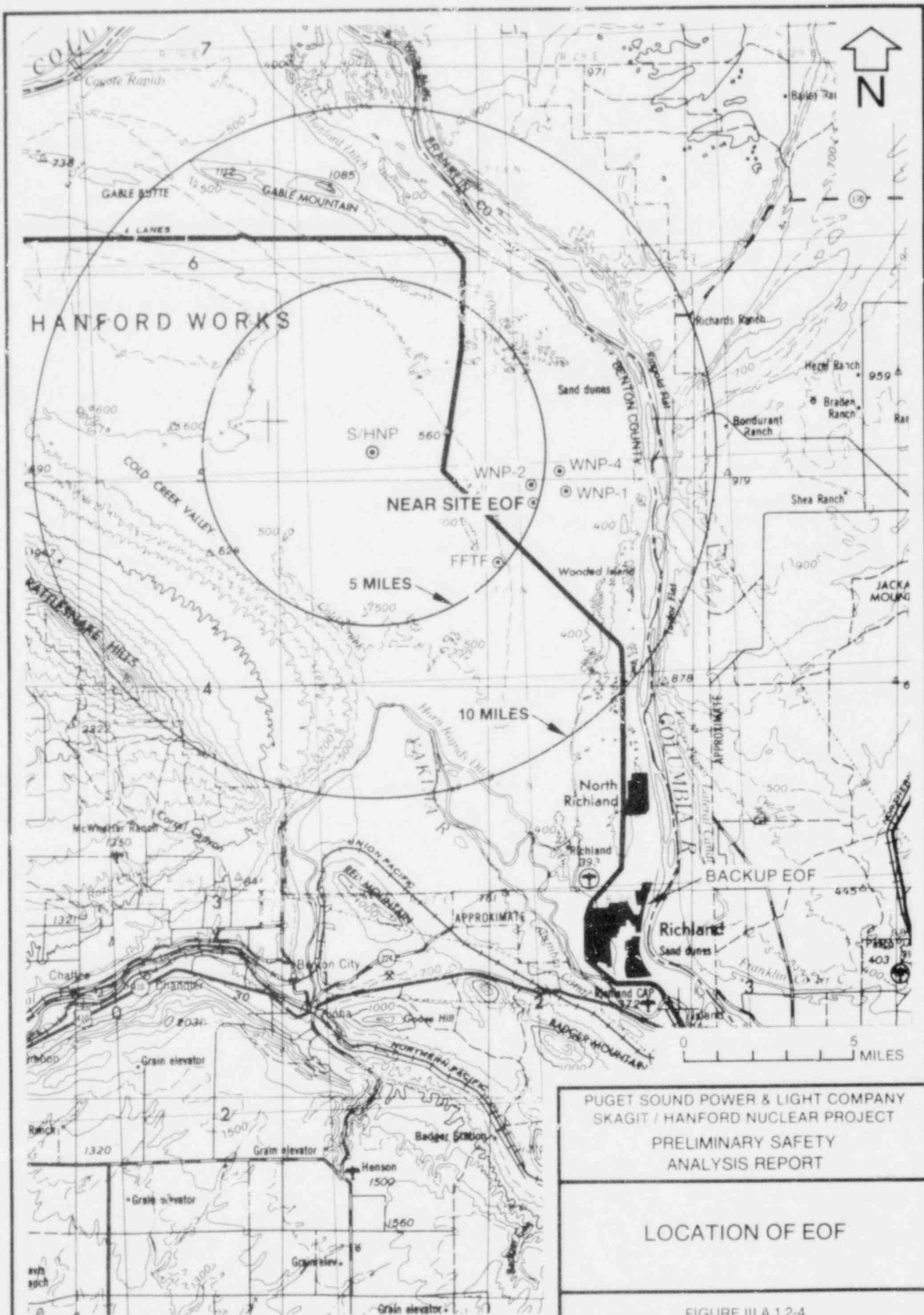
PRELIMINARY SAFETY ANALYSIS REPORT

PRELIMINARY OSC SECTION

FIGURE III.A.1.2-3a



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SKAGIT / HANFORD NUCLEAR PROJECT
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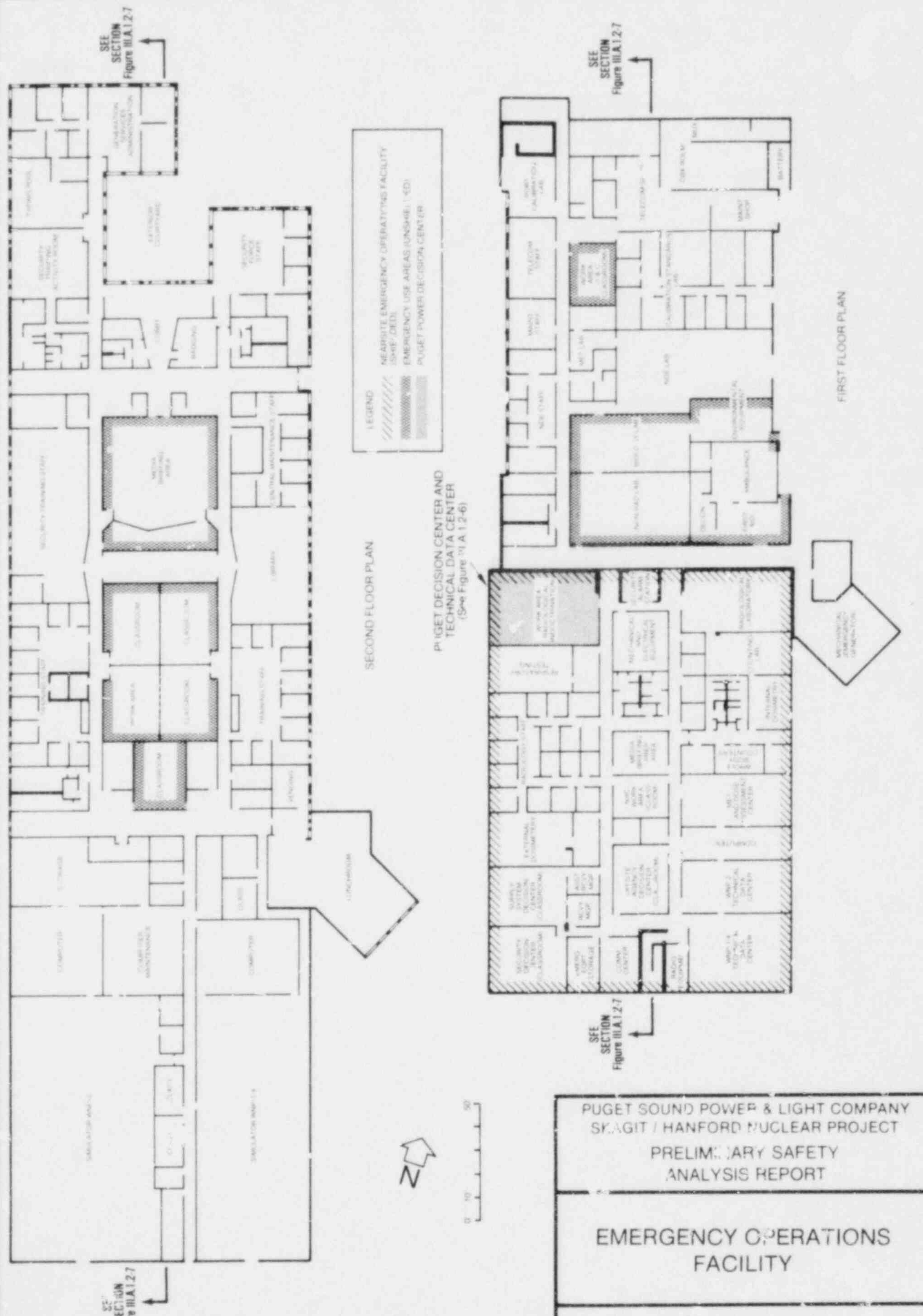
LOCATION OF EOF

FIGURE III.A.1.2-4



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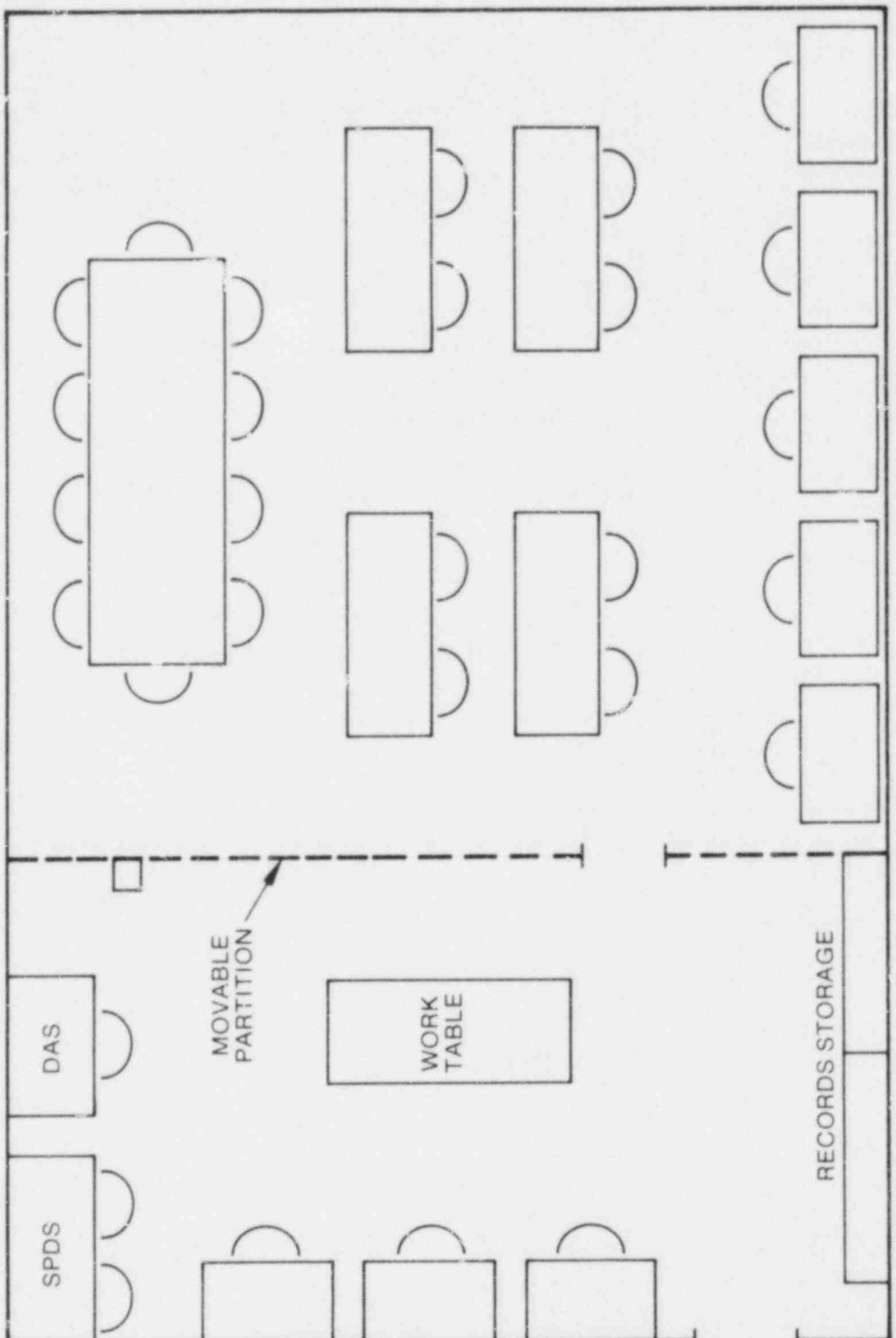
EMERGENCY OPERATIONS FACILITY

FIGURE III.A.1.2-5



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
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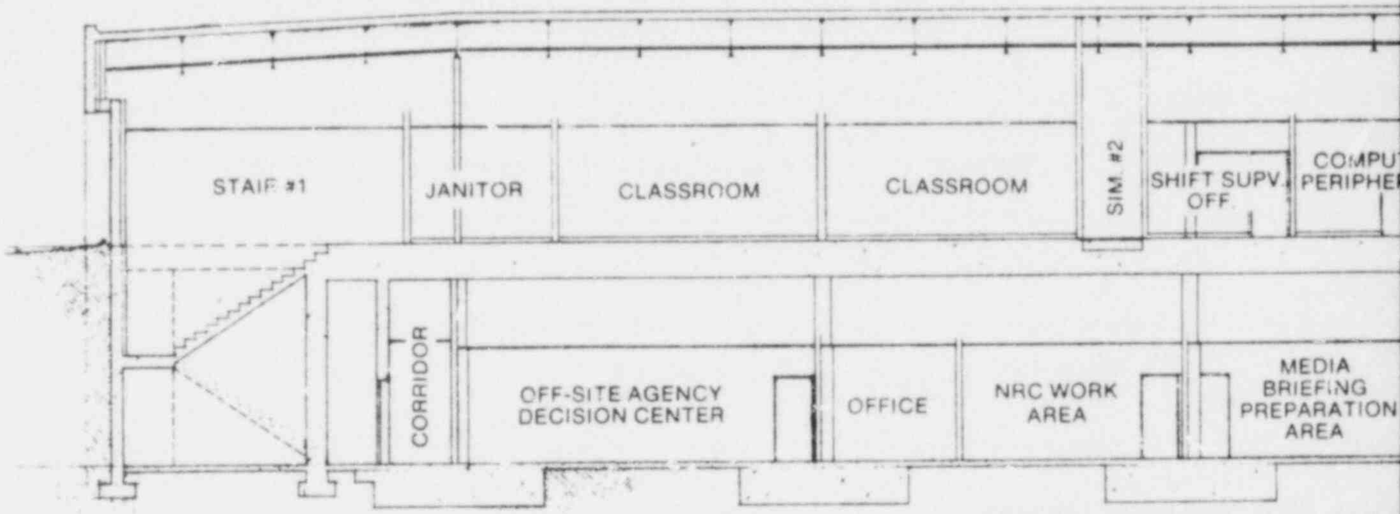
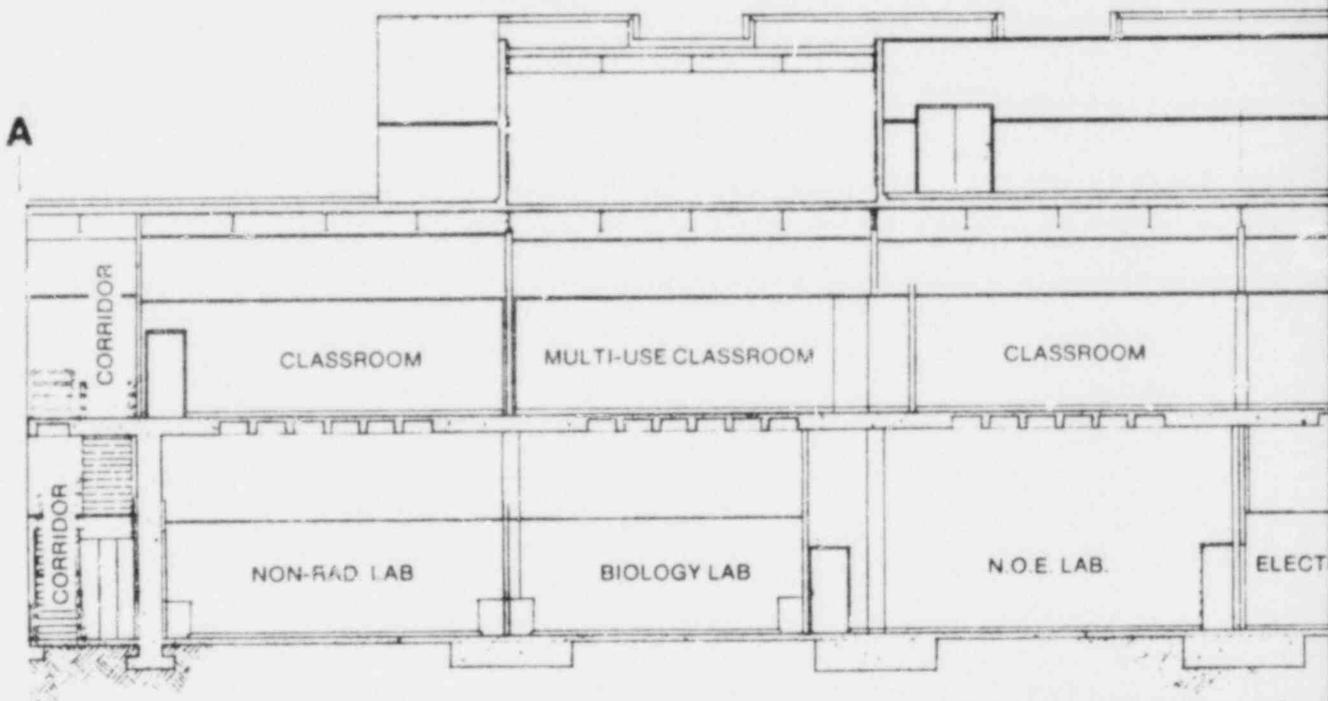
PUGET DECISION CENTER AND TECHNICAL DATA CENTER (EOF DETAIL)

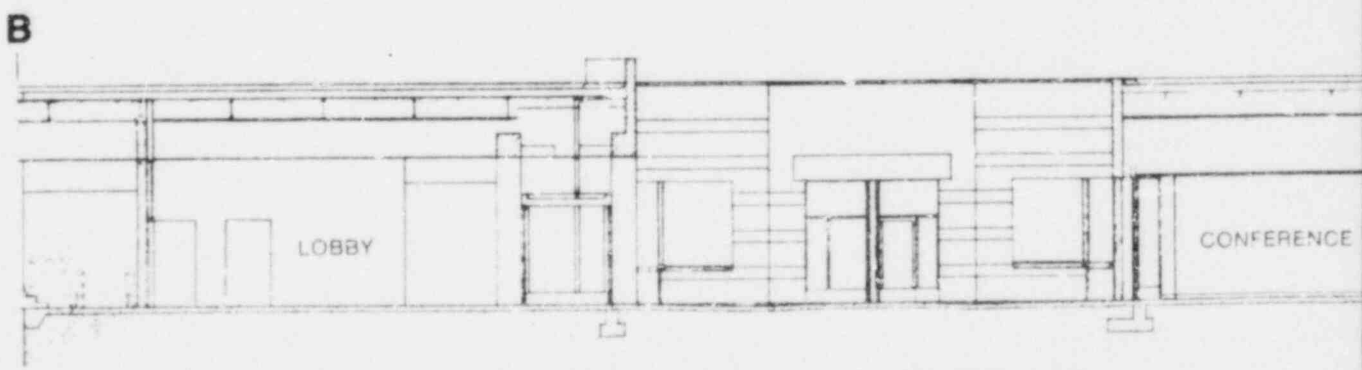
FIGURE III.A.1.2-6



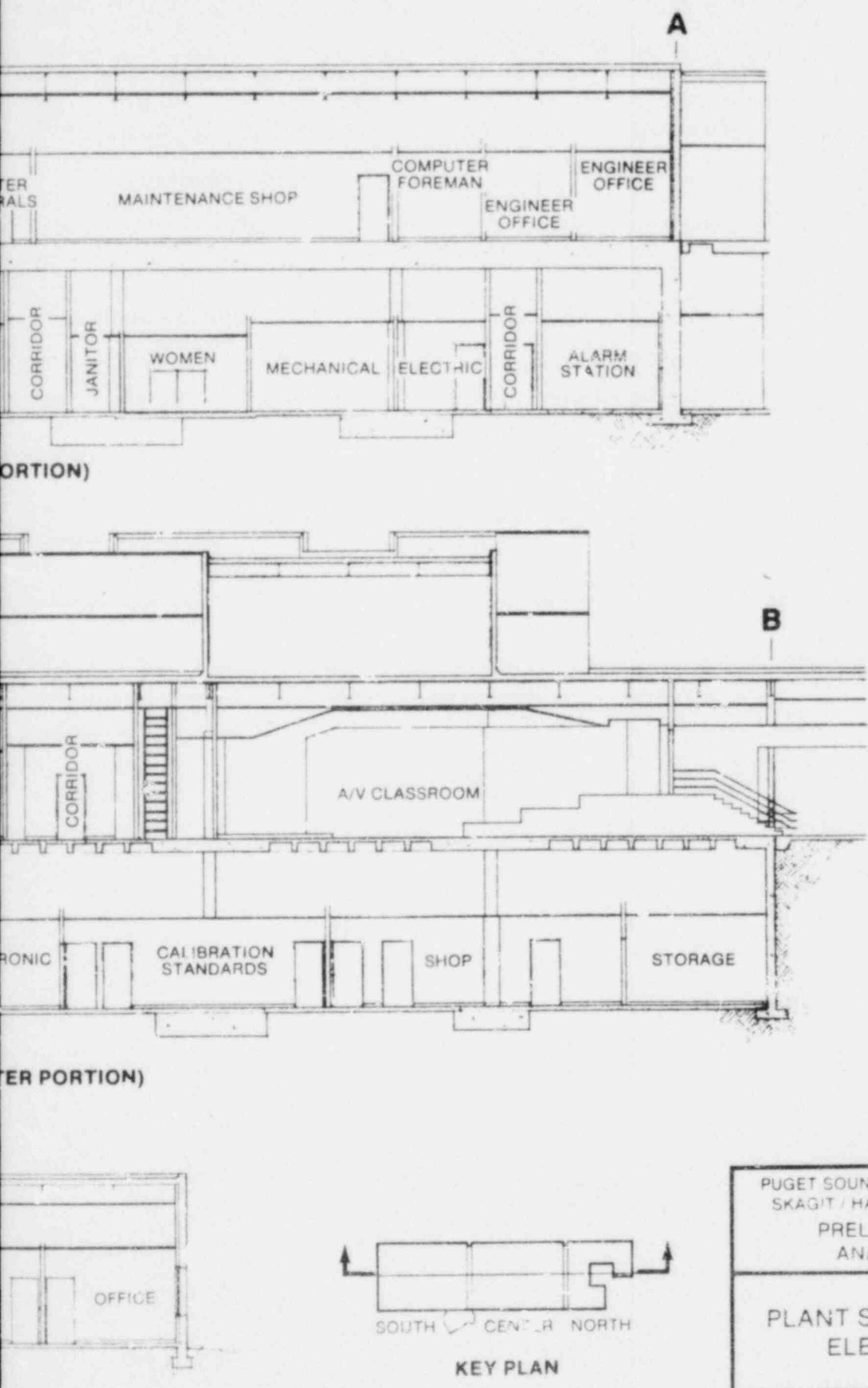
PARTIAL SECTION (@ SOUTH P



PARTIAL SECTION (@ CENT



PARTIAL SECTION (@ NORTH PORTION)



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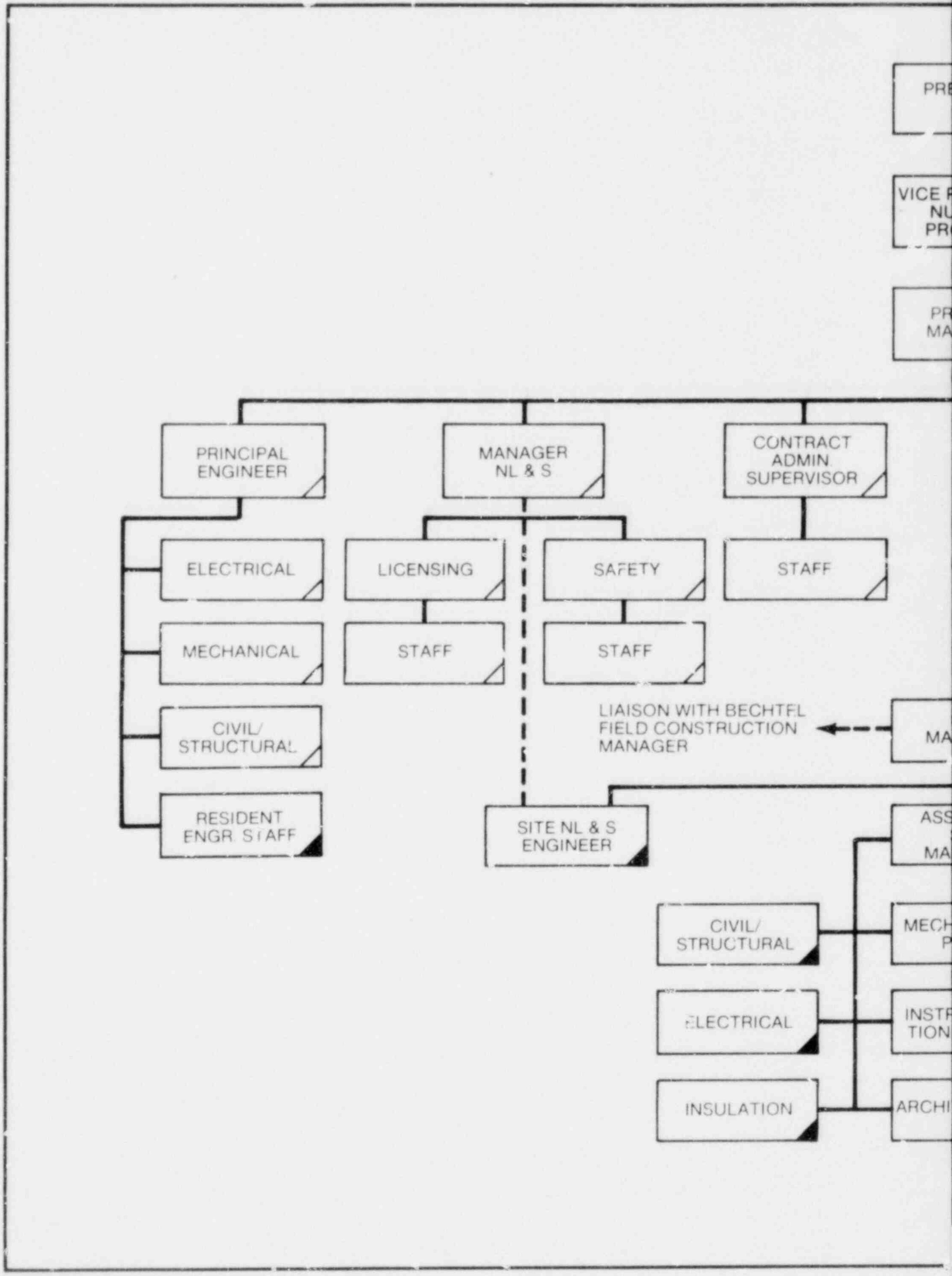
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

PLANT SUPPORT FACILITY
ELEVATION VIEW

FIGURE III.A.1.2-7



PRE
VICE
NU
PR
PR
MA
PRINCIPAL ENGINEER
MANAGER NL & S
CONTRACT ADMIN. SUPERVISOR
ELECTRICAL
LICENSING
SAFETY
STAFF
MECHANICAL
STAFF
STAFF
CIVIL/ STRUCTURAL
LIAISON WITH BECHTEL FIELD CONSTRUCTION MANAGER
MA
RESIDENT ENGR. STAFF
SITE NL & S ENGINEER
ASS
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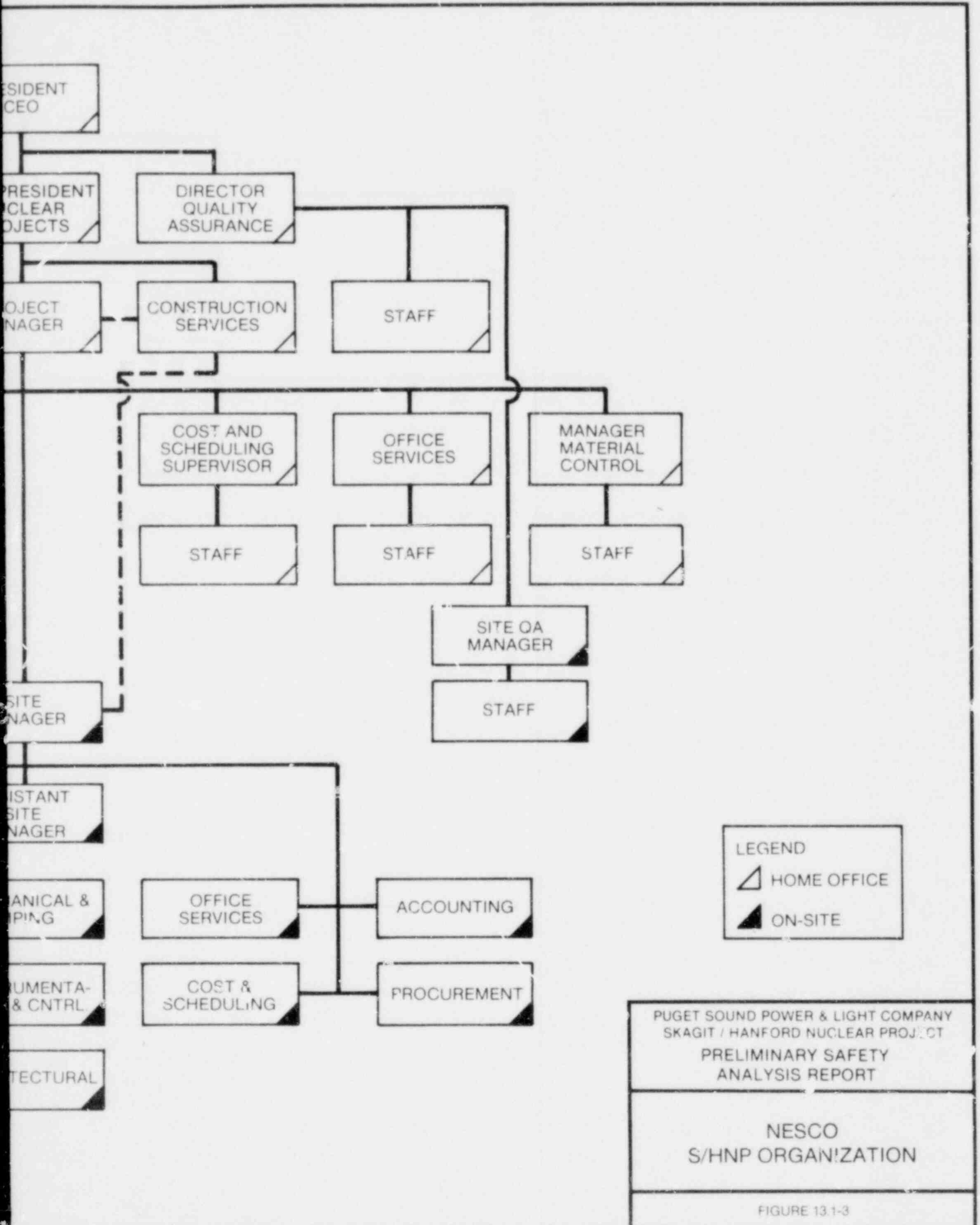

ESIDENT
CEO
PRESIDENT
CLEAR
OJECTS
DIRECTOR
QUALITY
ASSURANCE
OJECT
NAGER
CONSTRUCTION
SERVICES
STAFF
COST AND
SCHEDULING
SUPERVISOR
OFFICE
SERVICES
MANAGER
MATERIAL
CONTROL
STAFF
STAFF
STAFF
SITE QA
MANAGER
STAFF
SITE
NAGER
ISTANT
SITE
NAGER
ANICAL &
IPING
OFFICE
SERVICES
ACCOUNTING
UMENTA-
& CNTRL
COST &
SCHEDULING
PROCUREMENT
TECTURAL
LEGEND
HOME OFFICE
ON-SITE
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT
NESCO
S/HNP ORGANIZATION

FIGURE 13.1-3

PUGET
SENIOR VICE PRESIDENT
OPERATIONS

VICE PRESIDENT
GENERATING RESOURCES

SKAGIT/HANFORD OWNERSHIP CONTRACT

SKAGIT/HANFORD PROJECT OWNERS COMMITTEE(1)

- PROJECT STATUS REVIEW
- BUDGET AND SCHEDULE REVIEW AND APPROVAL
- MAJOR COMMITMENT AUTHORIZATION
- COMPREHENSIVE PROJECT DIRECTION AND GUIDANCE

DIRECTOR
NUCLEAR
PROJECTS

ENGINEERING AND OPERATIONS
- REVIEW SPECIFICATIONS AND DRAWINGS FOR OPERATIONS SUITABILITY
- SPARE PARTS REQUIREMENTS
- START-UP

COST AND SCHEDULE
- BUDGET AND CASH FLOW MONITORING
- SCHEDULE MONITORING
- FINANCIAL PLANNING

PUBLIC RELATIONS
- ALL INTERFACES RELATED TO PUBLIC RELATIONS

QUALITY ASSURANCE(2)
- QUALITY ASSURANCE PROGRAM
- COMPLIANCE WITH CORPORATE AND REGULATORY REQUIREMENTS

LICENSING
- APPLY FOR LOCAL, STATE AND FEDERAL PERMITS
- MONITOR LICENSING COMMITMENTS

ENVIRONMENTAL SCIENCES
- MONITOR DEVELOPMENT AND IMPLEMENTATION OF ENVIRONMENTAL PROGRAM FOR PRE-CONSTRUCTION PERMIT AND CONSTRUCTION
- DEVELOP AND IMPLEMENT ENVIRONMENTAL PROGRAM FOR OPERATION

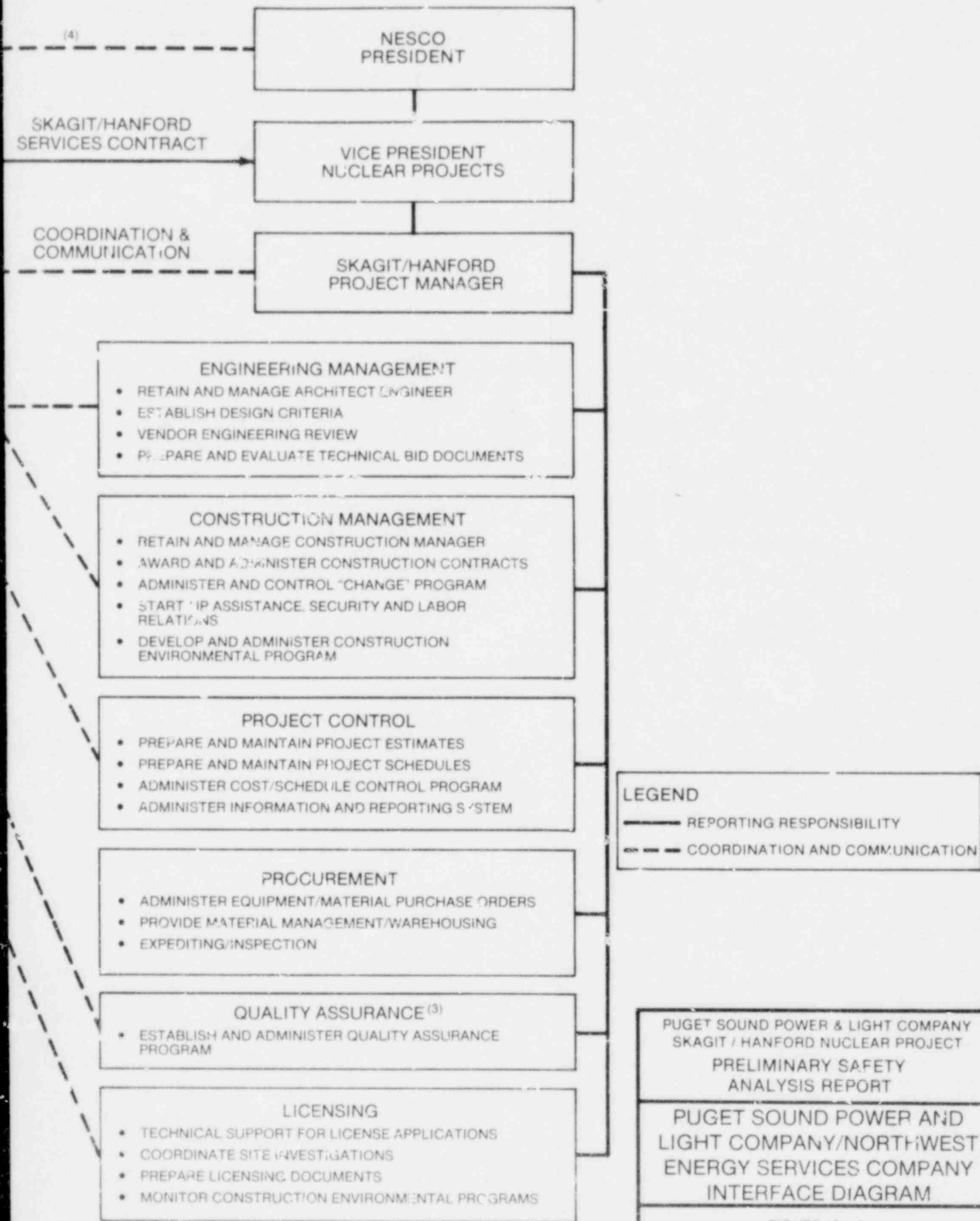
NOTES:

(1) • COMMITTEE MEMBERSHIP CONSISTS OF PROJECT OWNERS
• COMMITTEE CHAIRMAN IS PROJECT SPONSOR — PUGET SOUND POWER AND LIGHT

(2) • REPORTS TO VICE PRESIDENT, GENERATION RESOURCES

(3) • REPORTS TO PRESIDENT

(4) • ULTIMATE RESOLUTION OF QUALITY PROBLEMS THAT CANNOT BE RESOLVED AT A LOWER LEVEL



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

PUGET SOUND POWER AND LIGHT COMPANY/NORTHWEST ENERGY SERVICES COMPANY INTERFACE DIAGRAM

FIGURE 13.1-4

3.8.1.4.8 Test Pressure Load

The test pressure load is 1.15 x 15 = 18 psig.

3.8.1.4.9 Relief Valve Discharge Pressure Load

See Appendix 6C of this PSAR. 12

3.8.1.4.10 Suppression Pool Dynamic Pressure Loads

See Appendix 6C of this PSAR. 12

3.8.1.5 Design and Analysis Procedures

This section describes the analytical and design procedures used in designing the containment.

3.8.1.5.1 Analytical Methods

Computer programs are relied upon to perform many of the computations required for the containment analysis. However, in many cases, classical methods and manual techniques are used for analysis of localized areas of the containment and or preliminary proportioning. Manual calculations are generally used for:

a. The initial proportioning of the dome, wall, and base slab

b. Evaluation of the effects of locally applied loads such as pipe rupture or crane loads

c. The preparation of input for the computer analyses, and 21

d. Areas which do not lend themselves to computer applications.

Classical methods as described by Timoshenko, Roark, Bijlaard, and others are used in these analyses. Section 7.0 of BC-TOP-5-A(4) describes the analytical methods in more detail. Only portions of this section of BC-TOP-5- 12

A[4] that apply to non-prestressed concrete are used. For a summary of analytical methods, see Table 3.8-4. 12

3.8.1.5.2 Design Methods

Design methods incorporate several phases. Experience based on completed design or parametric studies of other structures of a similar nature is used as well. Only those design methods of section 6.0 of BC-TOP-5-A[4] that apply 12 to non-prestressed concrete are used.

The final design phase incorporates and refines information gained in earlier phases. It also incorporates closer approximations of the equipment and piping and related loads based on completion of detailed engineering design. Improved assumptions as to material properties including the effects of cracking of concrete are made. The method of analyzing the effect of penetrations, the thickening, the reinforcements, the embedments, etc., are discussed in Sections 7.3 and 7.4 of BC-TOP-5-A[4]. The design of the liner and its anchorage system and analyses for computation of seismic loas is provided in Section 3.7 of this PSAR, BC- 12 TOP-1[5] and Sections 7.5 of BC-TOP-5-A[4]. Comparisons of predictions are made, as appropriate, within allowable values of stresses, strains, deformations, and capacities. This procedure is used for both phases of design.

During normal operation of the plant, periodic actuation of the safety relief valves will produce pressure and thermal fluctuations on the containment liner plate in the 21 suppression pool areas. The suppression pool liner plate 130.20 will be analyzed and designed for fatigue and cyclic loads in accordance with ASME, Section III. Division 2, Subsubarticle CC-3760.

Design of the concrete containment for a degraded core accident shall comply with the ASME Code, Section III, Division 2, Article CC-3000. General yielding of cross- 21 sections shall not be permitted, however, local yielding will be permitted as long as serviceability and containment integrity are maintained. The liner plate will be considered as a load carrying member. Liner plate strains shall not exceed the allowables in Table CC-3720-1 for 22 factored load combinations. Dynamic effects of the pressure time history will be considered in the design of the containment, locks, hatches and penetrations.

CHAPTER 17.0

QUALITY ASSURANCE

CONTENTS

The Puget QA organization is responsible for surveillance and enforcement of all quality-related activities and is free from direct pressures for cost/schedule. The primary tool for carrying out this responsibility is a well-planned system of auditing. Although other departments are also given responsibilities and authorities to ensure that effective corrective action is taken, this does not mitigate the basic responsibilities of the QA organization for ensuring that the overall intent of the QA Program is carried out and maintained. 21 16

The Manager, Quality Assurance, is responsible for the development, administration and implementation of an overall QA Program to assure management of a completed Plant which will meet specified quality standards.

The Manager, Quality Assurance, has the responsibility and the authority to take whatever action is necessary to ensure that the items or systems identified as nonconforming or any other situations that adversely affect quality are dealt with promptly and effectively. The Manager, Quality Assurance, has stop work authority and will use all of the tools described in Puget's Quality Assurance Manual to ensure that corrective action concerning quality deficiencies are prompt and effective.

The Manager, Quality Assurance, is responsible to the Vice President, Generation Resources, for:

a. Developing and maintaining a QA Program that meets the requirements of 10 CFR 50, Appendix B, ANSI N45.2 and ASME Section III, Boiler and Pressure Vessel Code.

b. Auditing to verify that the QA programs of Puget and NESCO are implemented and effective.

c. Developing and maintaining the S/HNP Quality Assurance Manual. 21

d. Conducting QA indoctrination and orientation sessions for Puget personnel performing safety-related activities.

e. Reviewing and accepting the QA programs of Bechtel and NESCO.

f. Issuing and reviewing nonconformance and corrective action reports and evaluating conditions that may pertain to conditions outlined in 10 CFR 50.55(e).

g. Stopping work or controlling further processing, delivery, installation or use of nonconforming items until proper disposition has been approved.

h. Accompanying NESCO in sufficient number of audits to maintain confidence in NESCO's audit effective-
ness. 21

i. Identifying quality problems and initiating recommendations or providing solutions and verifying implementation of solutions.

j. Acting as Puget's contact with the NRC, Region V, and ASME Authorized Inspection Agency with respect to matters pertaining to QA.

The organization structure is such that the Manager, Quality Assurance, has the authority to carry out his responsibilities. The Manager, Quality Assurance, reports
directly to the Vice President, Generation Resources. 21
Within the scope of assigned duties, the Manager, Quality Assurance, shall exercise the authority of the Vice President, Generation Resources. The Manager, Quality Assurance, meets a minimum of once a month with the Vice President, Generation Resources, to discuss the status and adequacy of the QA Program. The Manager, Quality Assurance, has no other duties or responsibilities unrelated to
QA that prevent full attention to QA matters. The Manager, 21
Quality Assurance, has the authority to bring quality problems to the attention of the Senior Vice President, Operations, and/or Puget's President and Chief Executive Officer for resolution if they cannot be solved at a lower level. Ultimate resolution of quality related problems at
this level will involve the President and CEO of NESCO. 22
This reporting level is to assure organizational freedom to identify problems affecting quality and to ensure that solutions are determined and implemented. The qualification requirements for the position of the Manager, Quality Assurance, are as follows:

Six years' experience in the field of quality assurance, preferably nuclear equipment manufacture or
nuclear plant construction or construction supervisory 21
experience. At least one year of this six years' experience shall be nuclear power plant experience in the overall implementation of the Quality Assurance Program. (This experience shall be obtained within the QA organization.) A minimum of one year of this six years' experience shall be related technical or academic training. A maximum of four years of this six years' experience may be fulfilled by related technical or academic training.

Puget plans to establish a strong disciplined QA management
organization staffed with well-qualified individuals 21
knowledgable in QA/QC principles with sufficient authority
and responsibility to carry out the QA/QC function. The
number of personnel in the organization will depend on the
status of the Project. Long range matching of QA/QC
resources with work load will be accomplished by QA review
of projected work forces of the utility and its
contractors. This review will permit recruiting and
training activities to be carried out in such a manner as
to provide trained QA personnel necessary to assure the
quality of work.

QA/QC personnel will reevaluate staffing levels periodically (i.e., monthly) to assure they are adequate and modify as necessary.

Effectiveness of the staffing program will be assured by QA
participation in the work planning, surveillance and audit
and by the authority to stop work when it appears staffing
is inadequate. 22

Criteria for determining the QA/QC staffing needs include (1) work load and schedule and (2) personnel efficiency.

The methods used to establish the work load and schedule include work packaging; development of specific work plans (including inspections); identification of areas of inspection (QA/QC) expertise needed; and participation in day-to-day staff, planning and scheduling meetings.

The factors considered in determining personnel efficiency include experience and training; quality of work coming to the inspector; quantity of work coming to the inspector (workflow); procedure effectiveness; and productivity of personnel.

The criteria for determining if QA/QC staff is adequate include the degree of: uninspected work; quality problems detected after inspection; inspector preparation time; close-out of nonconformances and audits; follow-up of corrective actions; and complaints.

Prior to initiation of safety-related Site construction
activities, Puget plans to add a QA Specialist and Site QA
Engineer. The QA Specialist will be responsible for
providing assistance to the Manager, Quality Assurance, 21
maintaining the QA Manual, providing QA training and
performing off-Site audits. The Site QA Engineer will be
responsible for performing routine Site audits and surveillance.

During construction Puget requires NESCO's Director, Quality Assurance, and Bechtel's Project Quality Assurance Engineer submit a weekly report to Puget's Manager, Quality Assurance, summarizing quality problems and corrective action being taken.

17.1.1.2 Northwest Energy Services Company

As described in Section 13.1.1.1.1 NESCO, as Puget's agent, provides Project management services, which include:

a. Engineering management services.

b. Licensing management services.

c. Contract administration services.

d. Procurement services. 21

e. Construction management services.

f. Preoperation testing services.

g. Warehousing and material services.

h. QA services which include:

(1) Reviewing and approving design and procurement documents to assure adequate QA requirements are included.

(2) Reviewing Bechtel's and major contractors' QA programs.

(3) Auditing and performing surveillance of Bechtel, GE and major contractors.

NESCO's organization that will be located on-Site and off-Site is discussed in Chapter 13.

The responsibilities of NESCO's organizations performing activities affecting quality, excluding the QA organization, are contained in Chapter 13.

NESCO's QA organization currently consists of a Director, Quality Assurance. NESCO does not plan to develop a separate QA program until design and procurement activities for the S/HNP resume. In the interim, NESCO activities come under Puget's QA Program. Prior to resuming

significant design and procurement activities, NESCO plans to add a QA Specialist and QA Auditor and develop and implement their own QA program. NESCO's QA program will contain the equivalent controls and requirements of Puget's QA program. 21 22

NESCO's Director, Quality Assurance, reports directly to the President and is responsible for the NESCO QA Program. The Director, QA, formulates or reviews overall quality policies for NESCO S/HNP, provides technical guidance to the Project and evaluates the effectiveness of the total QA Program. He is responsible for coordinating the activities of the off-Site and on-Site quality groups to assure maintenance of a common approach.

The following is a specific list of the primary responsibilities of the Director, QA:

a. Formulating quality policies for use by NESCO where necessary to implement or supplement basic QA requirements prescribed by Puget's QA organization.

b. Developing and maintaining the NESCO S/HNP Quality Assurance Manual.

c. Approving QA procedures and instructions which define responsibilities and functions of QA personnel. 21

d. Concurring with quality-related procedures and manuals prepared by departments within the Project for conformance to QA policies.

e. Formulating audit programs and conducting audits and reviews to assure NESCO management and Puget management that the QA programs of NESCO and contractors conform with policies and requirements.

f. Identifying quality problems; initiating, recommending or providing solutions; and verifying implementation of solutions. Identifying the need for corrective action and assuring follow-up.

g. Providing and maintaining a qualified and suitably trained staff of QA and technical discipline engineers to carry out required Project and staff functions.

h. Formulating programs for maintaining the professional competence of personnel within the QA

organization and providing assistance in training, indoctrination and orientation programs for NESCO management, engineering and construction personnel whose activities affect quality.

i. Providing periodic reports to the President on the status and effectiveness of NESCO and principle contractors' programs and advising of any problems requiring special attention.

j. Serving as the focal point for Project communication on matters relating to the Project QA Program.

k. Stopping work or controlling further processing, delivery, installation or use of nonconforming items until proper disposition has been approved.

NESCO's Director, Quality Assurance, has no other duties or responsibilities unrelated to QA that prevent full attention to QA matters, and he is free from direct pressures for cost and schedule.

NESCO's QA Specialist will be responsible for providing administrative assistance to the Director, Quality Assurance, maintaining the QA Manual, developing necessary QA procedures, providing QA training, reviewing design and procurement documents to assure that adequate QA requirements are included. NESCO's QA Auditor will routinely perform procedural audits of NESCO, Bechtel, GE and other contractors to verify compliance to their procedures and regulatory requirements. Audits will be performed in accordance with ANSI N45.2.12-1977. Puget's QA Manager will perform an overview of NESCO's activities by performing routine audits to determine and evaluate their effectiveness in carrying out the QA Program. 21 |22

Prior to initiation of safety-related Site construction activities, NESCO plans to add a Site QA Manager, Site QA Systems Engineer and Civil Engineer. Quality Assurance Record Control and Vendor and Material Control personnel will be added prior to the arrival of QA records and/or material at the Site. The various engineering disciplines and NDE Engineer will be added prior to the start of construction work in those disciplines. Prior to preoperational testing, QA Engineers will be assigned to provide surveillance of startup testing. Emphasis will be placed on selecting QA personnel with QA/QC nuclear plant construction experience. Discipline QA Engineers will be required to have a degree in their engineering discipline or equivalent experience. Audit personnel will be qualified in accordance with the appropriate requirements of

ANSI N45.2.23-1978. QA staffing will be based on a long- 22
range projected work schedule and will be periodically reevaluated and adjusted as necessary. Puget's and NESCO's anticipated QA organizations, when construction is at its peak, are illustrated in Figures 17.1-1 and 17.1-3.

NESCO's Site QA Manager will report to NESCO's Director, Quality Assurance, and will be responsible for planning, directing and implementing the QA activities at the construction site. The Site QA Manager will have appropriate responsibilities and authority to exercise proper control over the Site QA Program and will be free from non-QA duties and can thus give full attention to assuring that the QA Program at the Site is being effectively implemented. The Site QA Manager will also maintain liaison 21
with NESCO's Site Construction Manager. NESCO's QA activities at the construction site will be in addition to and independent of Bechtel's QA/QC activities. NESCO's Site QA/QC organization will include personnel who will be responsible for QA Records Control, Vendor and Material Control, Site QA Systems and NDE Evaluation as shown in Figure 17.1-3.

An independent surveillance system is essential if Puget is to effectively exercise its responsibilities to assure that construction contractors and Bechtel are properly carrying out their responsibilities during construction. For that reason, NESCO's Site QA organization will be staffed by various engineering disciplines as shown in Figure 17.1-3. They will be required to review documentation relating to their discipline, including specifications, contractor procedures, field drawings, field change requests and change orders.

Site QA personnel are involved in Plant activities important to safety and are kept abreast of work schedule and construction activities by periodically attending construction status meetings. In addition, Site QA 22
personnel actively participate in day-to-day planning, scheduling and construction status meetings to review problem areas and evaluate these areas to determine if they are chronic and/or are developing a trend. Problem areas are also evaluated to determine the extent corrective action is taken and its effectiveness. Site QA personnel ensure that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments.

During construction, nonconformance reports will normally 21
be issued by Bechtel and/or the various construction

contractors, but may also be issued by Puget and NESCO. Nonconformance reports and corrective action requests will be reviewed and analyzed by NESCO's QA organization and the results routinely reported to Puget.

Planned and documented surveillance will be performed by NESCO's Site QA discipline engineers in their areas of expertise. Planning will include documenting in procedures mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector. In addition to monitoring the work, the construction contractor's ability to effectively perform the work from a quality standpoint will be reviewed, as well as the contractor's first line inspection system. Where practical, surveillance by NESCO's Site personnel will be performed after the construction contractor's work has been accepted by first line inspection. Thus the construction contractor's inspection capability will also be evaluated. Surveillance will also be aimed at evaluating Bechtel's surveillance inspection of the construction contractor's work. 21

Although NESCO, as Puget's agent, will perform QA services, responsibility for the overall program is retained and exercised by Puget.

NESCO's QA Program is subject to the acceptance of Puget's Manager, Quality Assurance.

Puget fulfills its QA responsibilities by overviewing NESCO's quality-related activities. The overview consists of performing routine audits and surveillance of NESCO.

17.1.1.3 Bechtel Power Corporation (San Francisco Power Division)

Bechtel Power Corporation has been contracted to provide the architect/engineer and procurement services for the Project. Bechtel is responsible for performing engineering work as required to properly describe and detail the work to be constructed. These engineering responsibilities include:

a. Preparation of specifications for construction contracts, assembling bid packages, analyzing bids and submission of recommendations to NESCO and 21

placement of construction contracts as directed by Puget.

b. Providing NESCO assistance in obtaining licenses, permits, and certificates required by Puget from regulatory agencies. 21

c. Providing a QA Program consistent with the requirements of NRC regulations covering Bechtel's activities for the Project. This QA Program is subject to the acceptance of Puget's Manager, Quality Assurance. (See Section 17.2.) 21

d. Providing home office engineering assistance in planning and scheduling of Puget's startup activities.

Bechtel is responsible, acting as Puget's agent, to procure (with the exception of the NSSS, Nuclear fuel and turbine-generator) materials, machinery, apparatus and supplies required for permanent and temporary construction. Bechtel will provide the following services:

a. Prepare inquiries, solicit quotations, analyze proposals and make purchase recommendations.

b. Prepare purchase orders.

c. Inspect materials and equipment and witness manufacturer's tests before shipment, as required, to assure that material and equipment meet specifications

d. Expedite manufacturer's engineering, shop production and shipment of materials and equipment.

21

Bechtel has responsibility to provide construction management services for the Project. These responsibilities include:

a. Development of an overall construction schedule and organization, planning and management of the construction program.

b. Administration of contracts.

c. Materials receiving, storage and warehousing. 21

d. Quality surveillance and auditing of on-Site contractor activities and advising Puget and NESCO on acceptability of completed work.

The Director, Nuclear Projects, will be notified by NESCO's 21
Project Manager when the Project is completed and ready for
final acceptance. The acceptance, identification of any 411.9
remaining work items, and performance of such work will be
coordinated between the Director, Nuclear Projects, and the 21
Plant Superintendent.

17.1.2 QUALITY ASSURANCE PROGRAM

Puget's QA Program is described in Section 17.1 and is 21
currently in effect for ongoing activities affecting
quality. The structures, systems and components to be
covered by the Puget QA Program are those which have a role
in preventing accidents or in mitigating the consequences
of accidents which could cause undue risk to the health and
safety of the public. The QA Program provides control over
activities affecting the quality of the identified structures, systems and components to an extent consistent with
their importance to safety. A detailed description of the
systems and components within the scope of this program is
summarized in Section 3.2. The formulation of policy and
technical direction of the QA Program is assigned to the
Manager, Quality Assurance, who reports to the Vice
President, Generation Resources. Management is routinely
informed of quality through receipt of monthly reports
which illustrate problem areas, reveal quality trends and
note corrective action taken. Management is also informed
through receipt of Puget issued audit and nonconformance
reports and corrective action requests. The above
information is provided to management so they can routinely 21
assess the effectiveness of the QA Program.

The Puget Quality Assurance Manual is approved by the Manager, Quality Assurance, and contains the written policies and procedures which apply to activities affecting the quality of Q-Listed items. These quality policies and procedures are mandatory requirements which must be implemented and enforced. A matrix of Puget's QA Program procedures, the corresponding criteria of 10 CFR 50, Appendix B, and QA Program synopsis is presented in Table 17.1-1.

The Quality Assurance Manual pertains to the design and construction phase of the Project and contains the written policies and procedures by which Puget will perform its quality assurance activities. The Quality Assurance Manual has been issued. Applicable sections of the manual have been implemented in advance of the activity to be controlled. Additional sections of the manual will be implemented as the activity proceeds.

The review and concurrence of the Vice President, Generation Resources, are required on the Quality Assurance 21
Manual. He reviews, on a quarterly basis, selected organizations participating in the QA Program to evalute the 411.5
status and adequacy of their part in the program. The review is designed to include all applicable organizations at least once a year.

The Vice President, Generation Resources, resolves corrective action items involving quality arising from a 21
difference of opinion between QA personnel and other department personnel. If it cannot be resolved at this 411.6
level, it may be taken to the Senior Vice President, Operations, and/or the President and Chief Executive 21
Officer.

Management and the NRC are immediately notified when 21
deficiencies, as defined in 10 CFR 50.55(e), are detected and evaluated.

Puget's QA Program requires that personnel and organizations performing QA functions have direct access to management levels which will assure the ability to identify 21
quality problems, initiate, recommend or provide solutions through designated channels and verify implementation of solutions.

Puget and NESCO's organization and the necessary technical organizations are participating early in the QA definition stage to identify the extent QA controls are to be applied to specific structures, systems, and components and will continue to do so.

Bechtel Engineering considers the importance of design features and characteristics when defining technical,
inspection, and test requirements in the technical 22
specifications. Bechtel Engineering, with QA participation, utilizes a unique ordering approach when specifying the QA criteria for procurements and contracts. Bechtel's Quality Control and Procurement Supplier Quality Representatives consider the specification requirements when preparing inspection instructions. The "graded approach" has been utilized by Bechtel for applying QA criteria to non-Q-listed items when formalized QA programs are required.

For items determined to be important to safety where specific QA controls cannot be imposed in a practical manner, an evaluation will be made to determine special quality verification requirements to be applied during installation or testing to provide the necesssary assurance that the item(s) meet project requirements.

Puget's QA Program requires that activities affecting quality shall be accomplished under suitable, controlled conditions. NESCO, Bechtel and GE, as major contractors 21
retained by Puget, are responsible for assuring that activities affecting quality will be accomplished under suitable, controlled conditions, including use of appropriate equipment; suitable environment such as adequate cleanliness; and compliance with necessary prerequisites.

Although some functions of the QA Program are delegated to contractors, subcontractors and suppliers, this does not mitigate Puget's responsibility for the quality of delegated work. Puget's primary tool for ensuring that delegated portions of the QA Program are being effectively 21
carried out is a well-planned system of auditing and surveillance.

Puget's QA Program requires that verification of conformance to established requirements be accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified.

Puget's QA Program requires that inspections are performed 21
with appropriate equipment under suitable environmental 411.8
conditions.

The details of the Bechtel and GE programs are described in Sections 17.2 and 17.3, respectively.

It is the responsibility of Puget's QA organization to review and document concurrence with the QA Program of 21
NESCO. It is also the responsibility of Puget's QA organization to conduct (or have conducted) audits of the major contractors' QA program activities.

A QA Program will be submitted by the major construction contractors for Bechtel's evaluation and concurrence and 411.3
Puget's and NESCO's information prior to the start of construction. The program must meet, as a minimum, the applicable provisions of requirements specified in 10 CFR 21
50, Appendix B, and ANSI N45.2-1971.

Puget has issued Quality Assurance Procedure No. 19, "Indoctrination and Training." This procedure establishes a system for the indoctrination and training of personnel performing activities affecting quality to assure that suitable proficiency is achieved and maintained. This procedure states that it is the responsibility of each functional manager to implement a system for the indoctrination and training of personnel assigned to perform activities affecting quality. The procedure recognizes

that training is a continual process and outlines training areas that should be considered to assure that employees remain knowledgeable and proficient in their fields. The procedure provides forms for documentation of individual training.

The training, qualification and certification programs are established such that:

a. Personnel responsible for performing quality affecting activities are insructed as to the purpose, scope, and implementation of the quality related manuals, instructions and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs,
documentation includes the objective, content of 22
the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria and developed to deterine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (1) the specific functions personnel are qualified to perform, and (2) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Revision 1.

Puget's QA Program requires that personnel performing
activities affecting quality attend QA orientation ses-
sions. These sessions consist of a presentation by QA
personnel concerning the purpose, history and philosophy of 21
QA including a review of 10 CFR 50, Appendix B, (18 cri-
teria) and the Quality Assurance Manual. The sessions also
emphasize management's support of the QA Program, and
stress the importance of quality and how everyone's con-
tribution to the program is needed for the program to
succeed.

The QA Program complies with the requirements of 10 CFR 50, Appendix B, 10 CFR 50, Appendix A, ANSI N45.2-1971, ASME Section III, Subsection NCA 4000 and Puget's need to assure a safe, reliable and economical nuclear power plant. The QA Program complies with WASH-1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," Revision 1, dated May 24, 1974, (Gray Book) and WASH-1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," Revision 0, dated May 10, 1974, (Green Book), with the exception of Bechtel's and GE's exceptions specified in Sections 17.2 and 17.3. The program also complies with the following NRC Regulatory Guides including Bechtel's and GE's exceptions specified in Sections 17.2 and 17.3: 21

- 1.28 QA Program Requirements (Design and Construction) (formerly Safety Guide 28) (6-7-72)
- 1.29 Seismic Design Classification (2-76)
- 1.30 QA Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment (formerly Safety Guide 30) (8-11-72)
- 1.37 QA Requirement for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3-16-73)
- 1.38 QA Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants (3-16-73) 21
- 1.39 Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3-16-73)
- 1.54 Quality Assurance Requirements for Protective Coatings applied to Water-Cooled Nuclear Power Plants (6-73) 6
- 1.58 Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel (Revision 1, 9-80)
- 1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants (10-73)

1.74 QA Terms and Definitions (2-74)

1.88 Collections, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records (8-74) 21

1.144 Auditing of Quality Assurance Programs for Nuclear Power Plants (Revision 1, 9-80)

1.146 Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (8-80) 22

NESCO, as Puget's agent, is responsible for preparation of the PSAR and coordinating its review. 21

Puget and NESCO have procedures for preparation of the PSAR, which outline the responsibilities for preparation and review. These procedures specified the manner in which PSAR review is to be conducted and documented, to ensure that the PSAR includes sufficient information to serve as the basis for Puget's and NESCO's conclusion that the Plant can be built and operated without undue risk to the health and safety of the public. These procedures have provisions for: 21 21

a. Verification that technical comments on safety-related structures, systems and components described in the PSAR are properly resolved.

b. Listing of PSAR Sections and identification of the group having prime review responsibility.

c. General guidelines applicable to PSAR sections.

The appropriate Puget or NESCO QA organization reviews and documents concurrence with these procedures. 21

Puget, as the applicant, approves and submits the PSAR to the NRC.

17.1.3 DESIGN AND DESIGN CHANGE CONTROL

The applicable requiremnts of ANSI N45.2-1971, Section 4.0 and ANSI N45.2.11-1974 are applied to Design and Design Change Control. 21

The design of systems, structures and components is controlled to ensure that design requirements such as design bases, regulatory requirements, codes and standards are correctly translated into specifications, drawings, procedures, and instructions.

The design and design review function has been delegated to Bechtel as Architect/Engineer and Construction Manager and GE as Nuclear Steam Supply System and Nuclear Fuel Supplier to assure that design documents comply with applicable regulatory requirements, quality standards, codes and good engineering practices.

Puget requires that Bechtel and GE conform to 10 CFR 50, Appendix B, requirements in the QA procedures including:

a. Design activities and documents are developed, controlled and processed in a manner to assure the proper translation of applicable quality,

regulatory, code and design basis requirements, and that deviations are controlled.

b. Materials, parts, equipment and processes are selected and reviewed for suitability.

c. Design interfaces and coordination between participating design organizations are identified and controlled.

d. Design control measures are applied to items such as the following: Reactor physics, stress, thermal, hydraulic, and accident analysis, compatibility of materials and accessibility for inservice inspection, maintenance, and repair and delineation of acceptance criteria for inspections and tests.

e. Design activities and documents are prepared, checked, reviewed, verified, approved, distributed, filed and maintained in a controlled manner.

f. The verifying or checking process is performed by individuals or groups other than those who performed the original design and shall be by a procedure requiring a documented check to verify 21
the dimensional accuracy and completeness of the design drawings and specifications.

g. Design verification methods are identified.

h. Specifying when and how often independent reviews of design activities from conceptual designs through final approval drawings and specifications are performed.

i. Design changes, including field changes, are subject to the same design controls that were applied to the original design and are reviewed and approved by the organization which performed the original design unless the originating organ- 21
ization designates another responsible organization.

j. Preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design. "As-built" drawings is an item on the check list prior to 22
fuel loading.

NESCO reviews design[1] and design criteria[2] documents, submitted by Bechtel, to carry out its responsibilities of assuring that the above requirements are being met. NESCO's design review supplements but does not replace the design control measures implemented by Bechtel to meet the requirements of 10 CFR 50, Appendix B. It is not necessarily the intention of NESCO to review all design and design criteria documents and changes to these documents. NESCO may (through sampling) review those documents it feels necesssary to determine the extent that Bechtel is carrying out its responsibilities. The extent of NESCO's reviews will be determined by what is found during the review. NESCO takes whatever action is necessary to promptly correct deficiencies found as a result of design and design criteria document reviews. 21

To facilitate this sampling process, design and design criteria documents generated by Bechtel which pertain to the Project are sent to NESCO for review and approval and are received by document control for logging, printing and distribution.

The NESCO Princial Engineer is responsible for designating the documents to be reviewed and sending them to the applicable disciplines for review. Project Engineering, Nuclear Licensing and Safety, and Project Management are responsible for reviewing designated design criteria documents. S/HNP Project Management, Nuclear Licensing and Safety, Project Engineering, and Quality Assurance are responsible for reviewing designated design and procurement documents.

NESCO QA personnel review designated design documents for consideration of inspection, testing, records and other QC provisions associated with the specifications. They also
review to assure that design characteristics are correctly 21
stated, can be controlled, inspected, and tested, and there
are adequate acceptance and rejection criteria and they 411.10
have been prepared, reviewed, and approved in accordance
with QA Program requirements. They also check to ensure 16

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1. Design Documents - Documents specifying the characteristics of structures, systems and components of a facility such as specifications and drawings.
2. Design Criteria Documents - Documents specifying the functional, reliability, maintainability and safety requirements of the structures, systems and components of a facility. They usually consist of design bases in the SAR, AE Design Criteria and system descriptions.

that the required independent design reviews were performed 21 411.10
and that the documents contain the necessary quality
assurance requirements such as inspection and test
requirements, acceptance requirements, and the extent of 22
documenting inspection and test results.

To provide a thorough review of the designated design and design criteria documents, a design review procedure is used by NESCO which provides design review instruction.
These instructions specify that the applicable reviewer 21
checks for suitability of material, parts, equipment, and
processes and that valid industry standards and specifi- 411.13
cations, material testing programs, and design reviews are used. Each review discipline conducts training sessions as necessary to ensure that review engineers are adequately trained in their area of design review.

Errors and deficiencies found during the review process 21
that adversely affect safety-related structures, systems, 411.12
and components are documented by NESCO and appropriate corrective action is taken.

Details of NESCO's review of designated design and design
criteria documents are specified in NESCO's Project Pro- 21
cedures Manual.

NESCO approves design documents and design criteria after comments have been resolved and NESCO is satisfied that the document meets requirements.

Bechtel and GE are responsible for development and preparation of design documents and the controls for changing design documents. Within Bechtel and GE, design changes (including field changes) are subject to design control measures commensurate with those applied to the original design and are approved by the organization which performed the original design.

NESCO controls PSAR/FSAR revisions or amendments as speci-
fied in Nuclear Licensing and Safety procedures. In 21
addition, NESCO may initiate design changes by letter.

Within Bechtel, design changes (both internal and external) are controlled by Bechtel's control system. Supplier and contractor design changes are requested through Bechtel's established Procurement or Contractor Document Control Program.

Bechtel submits changes to NESCO for review and approval 21
for any revisions of prior approved design documents or design characteristics changes whose limits are controlled by:

Documents are prepared, revised and issued in a controlled manner with a copy sent to Records.

NESCO audits and performs [illegible] surveillance of Bechtel, GE and major contractors to ensure that they have documented instructions and procedures related to quality activities.

The details of how NESCO, Bechtel and GE implement the development and issuance of instructions, procedures and drawings are described in their procedures. 21

17.1.6 DOCUMENT CONTROL

The applicable requirements of ANSI N45.2-1971, Section 7, are applied to document control.

Instructions, procedures and drawings, including changes thereto which prescribe activities affecting quality, are documented and controlled to assure that they are reviewed for adequacy. Only appropriate revisions are in use, and changes are reviewed and approved by the same organization that performed the original review and approval. 22
Controlled documents include:

a. Design specifications.

b. Design, manufacturing, construction and installation drawings. 21

c. Procurement documents. 411.12

d. PSAR and related design criteria documents.

e. Manufacturing, inspection and testing instructions.

f. Test procedures.

g. As-built documents.

Puget requires that procedures be established for review, approval and issuance of documents and changes thereto to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation and the QA organization reviews and documents concurrences with these documents with regard to QA-related aspects. 21

Instructions and procedures issued by Puget are controlled by the issuer as follows: 21 411.55

a. The names of intended recipients are reported on control forms. This also serves, and is maintained, as a current distribution list.

b. Document recipients are assigned a controlled copy which is given a control number. This number is recorded on the document and on the control form.

c. When the document is sent to the recipient, a receipt is sent with the document and the delivery date is recorded on the control form. 21 411.15

d. When the recipient receives the document, he signs the receipt to verify the document was received.

e. If the document is a revision, the recipient also signs to verify that he has destroyed or marked obsolete the superseded material.

f. The receipt is returned to the sender who records the date of receipt on the control form which closes the control loop.

Approved changes are promptly included in instructions, procedures, drawings and other appropriate documents associated with the change.

Obsolete or superseded documents are controlled to prevent inadvertent use. 21 411.17

Documents are available at locations where the activity will be performed prior to beginning the work.

Design documents (and revision thereto) are distributed to responsible NESCO individuals in a timely manner and controlled to prevent inadvertent use of superseded material. 21 411.11

The following Puget and NESCO documents and the organizations responsible for development, issuance and control of the documents are presented below: 21

Document	Responsibility	
Engineering Department Procedures Manual	Generating Plant Engineering	16
Project Procedures Manual	Project Management	
Quality Assurance Manual	Quality Assurance	
Quality Assurance Instructions	Quality Assurance	
Records and Information Manual	Micrographics and Records	
Puget S/HNP Licensing Procedures Manual	Licensing & Regulations	21

Design documents are distributed to NESCO's review disciplines in a controlled manner, and records are maintained of current revision and review status for control purposes. 21 411.17

Puget and NESCO do not generate specifications, procurement documents or drawings. This responsibility is delegated to Bechtel. 21

A master list is provided by Bechtel for Puget's and NESCO's use which identifies the current revision number of specifications, drawings and procurement documents pertaining to the S/HNP. This list is updated as changes are made and are available, on a timely basis, to responsible personnel. 21 411.17

Aperture cards for all drawings are marked obsolete to indicate when they have been superceded. 411.55 16

NESCO audits and performs Site surveillance of Bechtel, GE and other major contractors to ensure that their documents are adequately controlled. 21

Details of document control measures implemented by NESCO, Bechtel and GE are described in their procedures.

17.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES

The applicable requirements of ANSI N45.2-1971, Section 8, are applied to control of purchased material, equipment and services. 21

Verification of supplier's activities during fabrication, inspection, testing, and shipment of material, equipment, and components is planned and performed with QA organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of 22 documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

Puget requires that Bechtel provide, as appropriate:

a. Evaluation and selection of sources of supply.

b. Surveillance at the supplier's facility in 21 accordance with written procedures during manu- 4 facture, inspection and testing of the procured item or service to verify compliance with quality requirements.

c. Source and/or receipt inspection, in accordance 21 with written procedures and acceptance criteria, 4 of procured items furnished by supplier.

d. Documentary evidence at the Site from suppliers that items Bechtel and GE procured meet procurement quality requirements such as codes, standards or specifications. Bechtel requires that this documented evidence be available at the 21 Site prior to installation or use of the procured 411.19 item and that documentation is retained at the Site. In lieu of required documentation, a system of certification verifying that documentation is available and that responsible personnel have reviewed the documentation is acceptable providing that certifications are explicit as to what is being certified (specifically identified by

purchase order number). The certification system
also requires that procurement document require-
ments which have not been met are identified
together with a description of those noncon-
formances dispositioned "accept as is" or 21
"repair". The certification program is admin-
istered by Bechtel. Puget exercises its respon- 411.19
sibilities through a system of periodic audits by
NESCO to insure the PSAR and Puget's QA require-
ments are not violated. Procured items received
at the Plant Site without either complete docu-
mented evidence that quality requirements are met
or certification are controlled as nonconforming
in accordance with Subsection 17.1.15, which
includes control of further processing, instal-
lation or use of the item.

Bechtel is required to audit and evaluate the effectiveness of control of quality-related activities of contractors, consistent with the importance to the safety, complexity, and quality of the item or service being furnished.

Details of how NESCO, Bechtel and GE assure the control of purchased material, equipment, and services are described in their procedures.

NESCO reviews procurement documents to verify that 21
requirements are included that assure that purchased material, equipment and services conform to procurement document requirements.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that purchased material, equipment and services are being adequately controlled.

17.1.8 IDENTIFICATION AND CONTROL OF MATERIAL, PARTS AND COMPONENTS

The applicable requirements of ANSI N45.2-1971, Section 9, are applied to identification and control of materials, parts and components.

Puget requires that measures be established to identify and
control items such as material, parts and components, 21
including partially fabricated assemblies, to prevent use of incorrect or defective items.

Puget requires that identification of the item (i.e. heat number, part number, serial number, or other appropriate

marking) is maintained either on the item or on records
traceable to the item and verified and documented by the QA 22
organization, as required, throughout fabrication,
erection, installation and use of the item, and that the 21
method and location of the identification does not affect
the function or quality of the item being identified.

Puget requires that provisions are made for handling and storing items to retain identification and to prevent intermixing.

NESCO reviews procurement documents to assure that requirements for identification and control of material, parts and components are included.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure they have adequate identification and control of material, parts and components.

Details of how NESCO, Bechtel and GE assure adequate identification and control of material, parts and components are described in their procedures.

17.1.9 CONTROL OF SPECIAL PROCESSES

The applicable requirements of ANSI N45.2-1971, Section 10, are applied to control of special processes.

Puget requires that measures are established to control special processes such as welding, heat treating, non-destructive testing and electrochemical machining, and to assure that they are accomplished by qualified personnel using written procedures qualified in accordance with applicable codes, standards, specifications or other special requirements. The QA organization verifies the recorded evidence and documents the results. 21

Puget requires that measures are described that assure the qualifications of special processes, personnel performing special processes, and equipment are kept current and that record files are maintained.

NESCO reviews procurement documents to assure that requirements for control of special processes are included. NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that special processes are satisfactorily controlled.

The details of how NESCO, Bechtel and GE assure the control of special processes are described in their procedures.

17.1.10 INSPECTION

The applicable requirements of ANSI N45.2-1971, Section 11, are applied to inspection.

Puget requires inspection of activities affecting quality to verify conformance with documented instructions, procedures and drawings. 21

Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA aspects and other technical organizations, as appropriate:

a. A qualification program for inspectors (including NDT personnel) is established under directon of the QA organization and documented, and the qualifications and certifications of inspectors are kept current. 22

b. Modifications, repairs, rework and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives. Management personnel responsible for inspection and test in the area involved will determine the provisions that constitute acceptable alternatives. 21 411.21,56 This decision will be based on the particular circumstance involved after consulting with appropriate engineering personnel. In all cases, the alternative inspection method must be at least equivalent to the original method and shall be sufficient to ensure that the product, item or system being inspected will function and perform as originally required.

c. Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA aspects and other technical organizations, as appropriate:

   (1) Identification of characteristics and activities to be inspected. 22

   (2) A description of the method of inspection.

   (3) Identification of the individuals or groups responsible for performing the inspection operation.

   (4) Acceptance and rejection criteria.

   (5) Identification of required procedures, drawings, and specifications and revisions.

(6) Recording inspector or data recorder and the results of the inspection operation.

(7) Specifying necessary measuring and test equipment including criteria for determining accuracy requirements. 22

d. Inspections are performed by individuals other than those who performed the activities being inspected. Individuals performing inspections shall report to the appropriate QA/QC organization. 21

e. Inspection procedures or instructions are available with necessary drawings and specifications and revisions for use prior to performing inspection operations. 21 411.20

f. Source inspection is considered when it is not feasible or possible to verify conformance to specification after delivery, or when the time to replace or repair non-conformance would be prohibitive if detected after delivery.

g. Indirect control by monitoring processing methods, equipment and personnel is used if direct inspection of the product is impossible or disadvantageous.

h. Inspection or tests are performed for each work operation as necessary to verify quality. The results of the inspection operation are recorded as well as the identification of the inspector or data recorder. 21

i. Procedures shall be established which will identify mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

j. Both inspection and process monitoring are used when control is inadequate without both.

k. Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA organization as a minimum evaluates, verifies and documents completeness of this activity.

Receiving inspection is performed by the Bechtel QC organization at the construction site to assure: 22

a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment and acceptance records satisfy the inspection instructions prior to installation or use. 22

c. Specified inspection, test and other records (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use as a conforming item.

Puget requires that Bechtel develop, document and implement receiving inspection plans prior to receiving inspection. This includes a review of documentation to assure that items conform to specifications and contract requirements. Certification of conformance issued in lieu of required documentation is reviewed to determine the validity and adequacy of the information presented.

Puget requires that Bechtel transmit receiving inspection plans to NESCO for review and concurrence.

Puget requires that Bechtel develop, document and implement construction surveillance plans and transmit the plans to NESCO for review and concurrence.

21

NESCO reviews procurement documents to assure that inspection requirements are included.

NESCO audits and performs surveillance of Bechtel, GE and major contractors to verify conformance to inspection requirements.

Details of how NESCO, Bechtel, and GE assure that a program for inspection is established are described in their procedures.

17.1.11 TEST CONTROL

The applicable requirements of ANSI N45.2-1971, Section 12, are applicable to Test Control.

Tests required to demonstrate that structures, systems, and components will perform satisfactorily in service are

defined in procurement documents, engineering drawings,
specifications, or test procedures. Suppliers and con-
tractors are required to perform tests in accordance with 21
written test procedures. Test procedures or instructions
shall provide for the following as reviewed and concurred
with by the QA organization for QA aspects and by their
technical organizations for technical aspects.

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumen-
tation, adequate test equipment and instrumenta- 22
tion including their accuracy requirements,
completeness of item to be tested, suitable and
controlled environmental conditions, and
provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

Test results will be documented and evaluated to assure that test requirements have been satisfied. Test plans, procedures, hold and witness point controls, test reports and records are used to demonstrate that completed tests have met test objectives.

Procedures are established to control altering the sequence 21
of required tests, inspections and other operations impor-
tant to safety. Such actions shall be subject to the same
controls as the original review and approval. The QA
organization reviews and documents concurrence with these
procedures.

Proof tests prior to installation, such as supplier pump performance tests, equipment seismic testing, and environmental testing, are identified in procurement documents. This testing will be performed under the construction permit.

Construction tests are conducted as the final construction
activity and prior to preoperational testing. Construction
testing is conducted to demonstrate that equipment instal-
lation is complete and is identified in the construction
contract documents. Construction testing involves such
testing as hydro and piping leak tests; instrument, con-
trol, and stroke valve tests; control and power connections 21
polarity tests; alarm circuit test per schematics; and pump
motor rotation test. This testing will be performed under
the construction permit.

Preoperational and startup testing will be under the control of Puget and is described in Chapter 14 of the PSAR.

Major contractors and suppliers are required to develop
test plans and submit them to Bechtel for review and
concurrence. These test plans provide criteria for
determining the accuracy requirements of test equipment and 22
criteria for determining when a test is required or how and
when testing activities are performed.

Puget requires that Bechtel and GE plan, implement and coordinate construction testing in accordance with approved written test procedures.

NESCO reviews procurement documents to assure that test control requirements are included.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that test requirements are followed.

Details of how test control requirements are implemented by
NESCO, Bechtel and GE are described in their procedures. 21

17.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

The applicable requirements of ANSI N45.2-1971, Section 13, are applied to Control of Measuring and Test Equipment.

Puget requires that procedures be established for the calibration (technique and frequency), maintenance and control of measuring and test equipment (instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection and monitoring of structures, systems, and components. The review and documented concurrence of these procedures shall be described and the

organization responsible for the function shall be identified.

Puget requires that measuring and test equipment is properly identified, controlled, adjusted and calibrated at specific periods to maintain accuracy within necessary limits. 21

Puget requires that measuring and test equipment is adjusted and calibrated against certified equipment or reference or transfer standards having known valid

relationships to nationally recognized standards, or if no national standards exist, the basis for calibration is documented.

Puget requires that measures be established to assure that the error of calibration standards is less than the error of production measuring and test equipment.

Puget requires that if measuring and test equipment is found out of calibration, measures be established for 21
evaluating the validity of previous inspection or test results and the acceptability of items inspected or tested since the last calibration check and for repeating original inspections or tests using calibrated equipment where necessary to establish acceptability of suspect items.

Puget requires that measures be established to assure the maintenance of records that indicate the calibration status of all items under the calibration system and that identify the measuring and test equipment.

Contractors at the Site shall be responsible for their on- 21
Site calibration and shall meet the above requirements. 411.23

Puget requires that its contractors have procedures for 21
control, calibration and adjustment of measuring and test equipment which enact the above requirements and include:

a. Identification of measuring and test equipment.

b. How measuring equipment is routinely monitored and the action required when limits are exceeded. 21

c. Establishment of calibration frequencies. 411.24

d. Disposition and reporting of damaged or out-of-calibration measuring equipment.

e. How calibration data is recorded and reported.

NESCO reviews procurement documents to assure that requirements for control of measuring and test equipment are included.

NESCO audits and performs Site surveillance of Bechtel, GE 21
and major contractors to assure control of measuring and test equipment.

Details of how NESCO, Bechtel and GE assure control of measuring and test equipment are described in their procedures.

17.1.13 HANDLING, STORAGE AND SHIPPING

The applicable requirements of ANSI N45.2-1971, Section 14, and ANSI N45.2.1-1973, ANSI N45.2.2-1972 and ANSI N45.2.3-1973 are applied to handling, storage and shipping.

Puget requires that handling, storage, packaging, shipping, cleaning and preservation of material, parts, components and equipment be controlled in acccordance with work and inspection instructions to prevent damage, loss, or deterioration and, where necessary, a suitable protective environment is provided and maintained. The QA organization shall review and document concurrence with these procedures. 22

Puget requires that Bechtel procedures to control Site handling, storage and shipping be transmitted to NESCO for review and concurrence.

NESCO provides warehouse management for Puget's transitory storage facility. 21

NESCO provides surveillance of the transitory storage facility warehousing services contractor to assure conformance to specified requirements.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that handling, storage and shipping are adequately performed.

Details of how NESCO, Bechtel and GE assure control of cleanliness, handling, storage and shipping are described in their procedures.

17.1.14 INSPECTION, TEST AND OPERATING STATUS

The applicable requirements of ANSI N45.2-1971, Section 15, are applied to inspection, test and operating status.

Puget requires that procedures be established to indicate, by application and removal of marking such as inspection and welding stamps, tags, labels, routing cards or other suitable means, the status of inspections and tests performed on individual items of the nuclear power plant throughout fabrication, installation and test. The QA organization shall review and document concurrence with these procedures.

Puget requires that measures be provided for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of such inspection and test.

Puget requires that measures be established for indicating the operating status of structures, systems and components of the nuclear power plant such as tagging valves and switches to prevent inadvertent operation.

NESCO reviews procurement documents to assure that requirements for indicating inspection, test and operating status are included.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure conformance to inspection, test and operating status requirements.

Details of how NESCO, Bechtel and GE assure implementation of an inspection, test and operating status are described in their procedures.

21

17.1.15 NONCONFORMING MATERIALS, PARTS OR COMPONENTS

The applicable requirements of ANSI N45.2-1971, Section 16, are applied to nonconforming materials, parts or components and as applicable to services (including computer codes). 22

Puget's QA Program requires measures that:

a. Control materials, parts or components which do not conform to requirements in order to prevent their inadvertent use or installation.

b. Provide for, as appropriate identification, documentation, segregation, disposition and notification to affected organizations.

c. Assure that nonconforming items are reviewed and that they are accepted, rejected or repaired or reworked in accordance with documented procedures.

d. Control further processing, delivery or installation pending proper disposition of the deficiency.

e. Periodic analysis of nonconformance reports are performed to show quality trends, and such analyses are forwarded to management.

Puget and NESCO controls nonconforming materials, parts or components as follows:

a. A nonconformance is defined as any deficiency in characteristic, documentation or procedure which renders the quality of an item unacceptable or indeterminate.

b. Nonconformances are documented and reported on a nonconformance report.

c. QA personnel are responsible for administering the nonconformance report system.

d. A nonconformance report may be initiated by anyone associated with the S/HNP and are usually identified during audits, surveillance and inspection as well as general observation while performing other tasks.

e. Nonconformance reports include, as a minimum, the following:

(1) A description of the nonconformance. 21

(2) A disposition recommendation including corrective action to prevent recurrence when applicable.

(3) Disposition approval/disapproval.

(4) Verification of disposition completion.

f. QA and Nuclear Projects approval/disapproval is required on nonconformance report dispositioning.

g. Nonconformance reports are assigned a report number and the number is entered in a nonconformance log for control purposes.

h. Nonconforming items are identified with a hold tag and prevented from being used or installed. Where feasible, nonconforming items are removed to a controlled hold area, roped off or otherwise segregated. Generation of a hold tag does not necessarily require issuance of a nonconformance report.

i. QA personnel review nonconformance reports to assure:

e. Classifying records as lifetime or nonpermanent.

f. Establishing procedures for receipt of records and reporting on their status.

g. Establishing storage procedures indicating:

(1) Description of the storage area.

(2) Filing system used.

(3) Method for verifying that records are in agreement with transmittal documents.

(4) Rules governing access to and control of files (security).

(5) Methods for maintaining control of and accountability for removed records.

(6) Method for filing supplemental information and disposing of superseded records.

h. Listing authorized personnel who shall have access to the files.

i. Temperature and humidity control.

NESCO reviews Bechtel's Site QA record control procedures. This review is to assure that the Bechtel proposed system will be compatible with Puget's and NESCO's systems. 21

Puget reviews NESCO's record management and turnover procedures to assure they are compatible with Puget's system.

Puget requires that Bechtel QA records are turned over to Puget at the completion of the Project in a planned and systematic manner to be maintained by Puget.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that an effective QA records control program has been implemented. 21

Details of how NESCO, Bechtel and GE assure control of QA records are described in their procedures.

17.1.18 AUDITS

The applicable requirements of ANSI N45.2-1971, Section 19, ANSI N45.2.12-1977, and ANSI N45.2.23-1978 are applied to audits. 22

Puget requires its contractors and their suppliers to develop and implement a comprehensive system of planned and documented audits which require that: 21

a. Audits are planned and scheduled to assure they are regularly scheduled on the basis of the status and safety importance of the activities being performed and are initiated early enough to assure effective QA during design, procurement, manufacturing, construction and installation, inspection and test. As a minimum, safety-related activities shall be audited at least annually or at least once within the lifetime of the activity, whichever is shorter, although longer cycles are acceptable with other evaluations of individual elements.

b. Audits are performed in accordance with written procedures or checklists by qualified personnel not directly responsible for the activities being audited.

c. Audits are documented and distributed for review to those in management responsible for the activities audited.

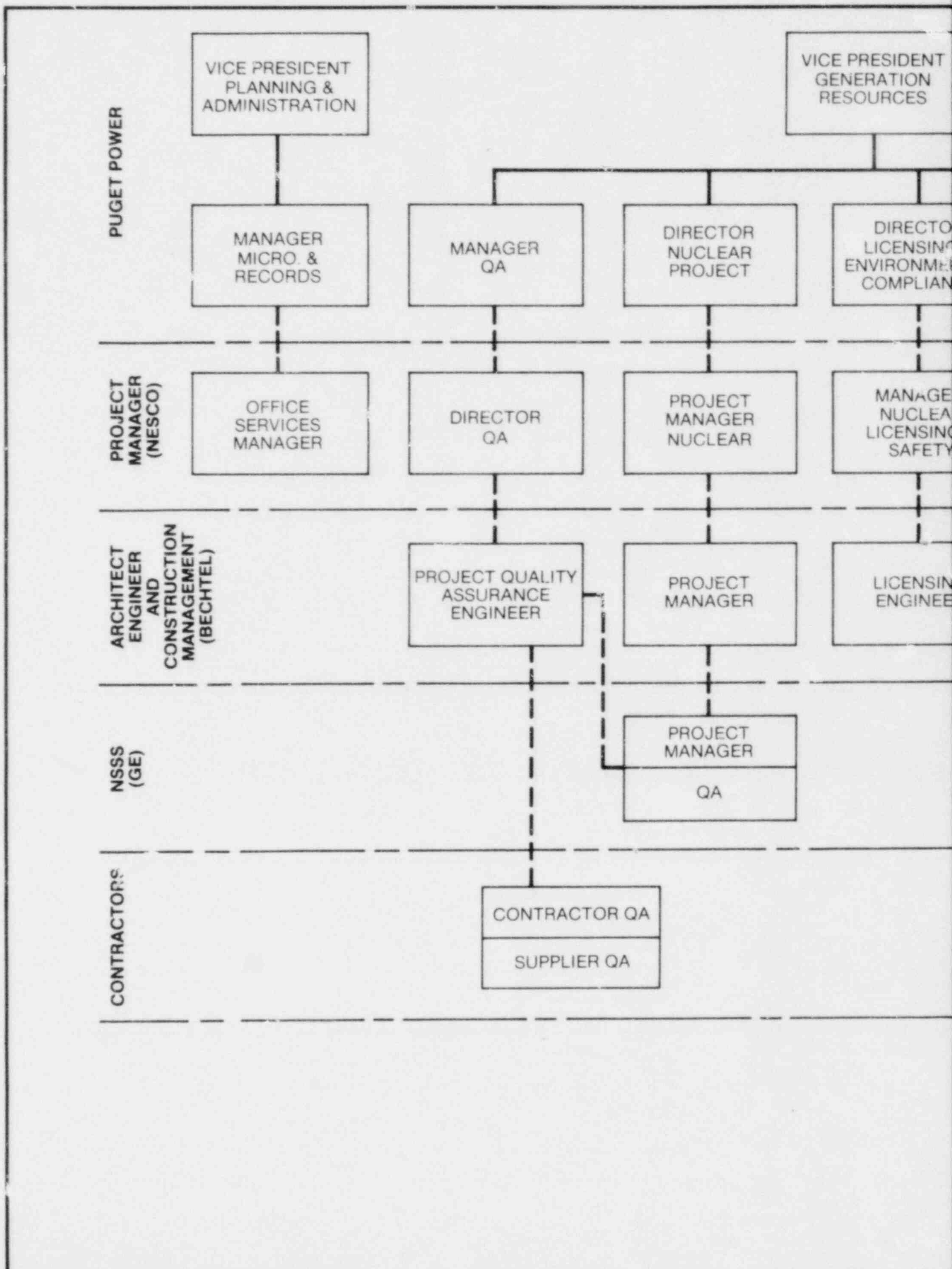
d. Follow-up audits are performed concerning audits identifying nonconforming incidents to determine if effective corrective action was taken.

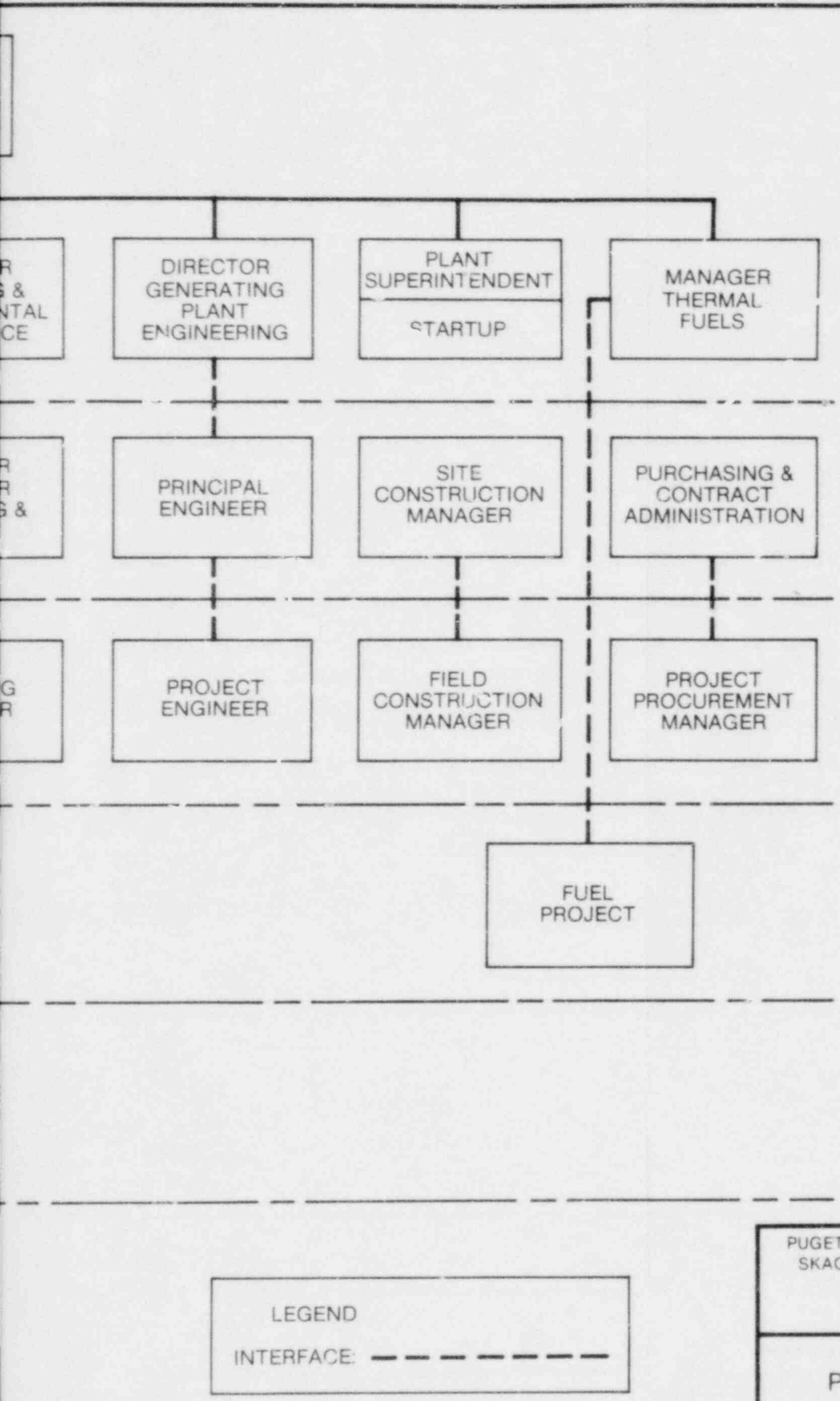
Puget's and NESCO's audits, internal and external, are designed to meet the above requirements. 21

Audits are performed in these areas where the requirements of 10 CFR 50, Appendix B, are being implemented including, as a minimum, those activities associated with:

a. The determination of Site features which affect Plant safety (e.g., core sampling, Site preparation and meteorology). 21 411.27

b. The preparation, review, approval and control of the PSAR, designs, specifications, procurement documents, instructions, procedures and drawings.


PUGET POWER
PROJECT MANAGER (NESCO)
ARCHITECT ENGINEER AND CONSTRUCTION MANAGEMENT (BECHTEL)
NSSS (GE)
CONTRACTORS
VICE PRESIDENT PLANNING & ADMINISTRATION
VICE PRESIDENT GENERATION RESOURCES
MANAGER MICRO. & RECORDS
MANAGER QA
DIRECTOR NUCLEAR PROJECT
OFFICE SERVICES MANAGER
DIRECTOR QA
PROJECT MANAGER NUCLEAR
PROJECT QUALITY ASSURANCE ENGINEER
PROJECT MANAGER
PROJECT MANAGER
QA
CONTRACTOR QA
SUPPLIER QA



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

PUGET/CONTRACTOR INTERFACE

FIGURE 17.1-2

17.2 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION BECHTEL POWER CORPORATION

17.2.1 SCOPE OF RESPONSIBILITY

This section describes Bechtel's responsibilities in providing quality-related services for engineering design, procurement, and construction management to Puget on the S/HNP. The scope of responsibility differs from that indicated in BQ-TOP-1 in that Bechtel does not perform construction.

Construction management provisions for quality-related services include receiving and storage of owner-furnished materials delivered to the job site prior to the arrival of the installation contractor. Quality surveillance and audit are provided over on-Site contractor activities for implementing their QA Program including inspection responsibilities. Bechtel advises Puget on acceptability of completed work. 6

17.2.2 QUALITY ASSURANCE PROGRAM DOCUMENTATION APPLICABLE ON SKAGIT HANFORD NUCLEAR PROJECT

See Tables 17.2-1, 17.2-2, and 17.2-3 which document the QA Program for the S/HNP.

17.2.3 QUALITY ASSURANCE TOPICAL REPORT

The Bechtel QA Program plan for use by the Bechtel Power Corporation during design, procurement and construction management of the S/HNP Units 1 and 2 is described in Bechtel Topical Report BQ-TOP-1, "Bechtel Quality Assurance Program for Nuclear Power Plants," subject to the following modifications and additions: 21

Page 1, et al:

Replace all references to "Atomic Energy Commission (AEC)" with "Nuclear Regulatory Commission (NRC)."

Replace all references to "Bechtel Thermal Power Organization (TPO)" with "Bechtel Power Corporation (BPC)."

Page 2:

Add the following paragraphs at the bottom of the page:

"Regulatory Guide 1.144, Rev. 1 (9-80), 'Requirements for Auditing of QA Programs' ANSI N45.2.12-1977." 22

"Regulatory Guide 1.146, 'Qualification of QA Audit Personnel' (1978)"

Delete the words "(8/73)" from the paragraph which begins with "Regulatory Guide 1.58 Rev. 1 9-80 ANSI N45.2.6-1978..." and replace with the word "(1978)." 22

Page 6, et al:

Replace all references to "Materials, Fabrication and Quality Control Services (MF&QCS)" with "Material and Quality Services and Codes and Standards (M&QS)." 21

Replace all references to "Procurement Inspection" with "Supplier Quality."

Replace all references to "Procurement Inspection Department" with "Procurement Supplier Quality Department."

Page 8, et al:

Replace all references to "Manager of Procurement Inspection" with "Manager of Supplier Quality."

Replace all references to "procurement inspection (or inspector) personnel" with "supplier quality representative(s)."

Replace all references to "Procurement Inspection Department Manual" with "Procurement Supplier Quality Manual."

Replace all references to "Inspection Manager" with "Supplier Quality Manager."

Replace all references to "Procurement Inspection Department" with "Procurement Supplier Quality Department."

Page 21:

Add the following paragraph after the third paragraph:

> "QA staffing is based on the long range projection work schedule and is periodically reevaluated and adjusted as necessary." 21

Page 22:

Replace the word "1973" with "1978" after the words "... ANSI N45.2.6" in subparagraph (2) of the first paragraph.

Delete subparagraph (3) and replace with: 22

> "Auditor Qualifications - Personnel performing audits will be qualified in accordance with the appropriate requirements of ANSI N45.2.23 - 1978. (Regulatory Guide 1.146)."

Page 28:

Add the following paragraph after the sixth paragraph:

> "For commercial 'off-the-shelf' items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practical manner, special qualify verification requirements shall be 21 established and described to provide for an 'acceptable' item."

Page 30:

Add the following paragraph after the second paragraph:

> "Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable and that there are adequate acceptance and rejection criteria."

Page 31:

Add only the word "Q-List" after the word "specifications" in the second sentence of the fourth paragraph.

Page 40:

Add the following after the words "planning document" in the first sentence of the fourth paragraph:

> ", these documents provide for inclusion of mandatory hold points when applicable."

Page 42:

Add only the words, "which include designation of applicable witness and hold points," after the words "Test plans and procedures," in the third sentence of the third paragraph.

Add the following paragraph after the fourth paragraph:

> "Procedures are established and described to control altering of the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. The QA organization reviews and documents concurrence with these procedures." 21

Page 43:

Add the following to the fifth paragraph:

> "Procedures provide for the selection of measuring equipment compatible with the type and accuracy requirements of the operations to be performed."

Page 49:

Add the following after the first sentence of the fifth paragraph:

> "An installation shall be considered to be in an 'as constructed' condition if it is installed within tolerances established by Project engineering as indicated in the design output documents. Completed quality verification records which correctly identify the "as built" condition of the Plant, including material certification and test data for traceability, quality verification records such as inspection and test reports evidencing conformance to design documents, and nonconformance reports for repair and 'use-as-is' dispositions are placed in quality record files."

Page 50:

Delete the remainder of subparagraph 3) of the second paragraph after the word "personnel" and replace with the following:

> "audits of Bechtel suppliers performing continuing work for one or more Bechtel projects are conducted as a minimum on an annual basis; audits of suppliers performing limited duration assignment are conducted at least once during the life of the contract. The 21
> requirement may be waived when evidence exists of continuing satisfactory performance including surveillance by Procurement Supplier Quality Department. This waiver is based on an annual review by Procurement Supplier Quality with concurrence of the Project Quality Assurance Engineer. Results of these reviews are placed in supplier quality history files."

The annual audit requirement shall not apply to standard off-the-shelf items and bulk commodities where required quality can adequately be determined by receipt inspection or post-installation checkout of test.

Add to BQ-TOP-1 Appendix A new page A-10:

Regulatory Guide 1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS (Revision 1, September 1980)

Response

The requirements of ANSI N45.2.12-1977 as modified and interpreted by the regulatory position will be applied to the Bechtel quality program for safety-related items except as modified or interpreted below: 2

1. Reference: Standard Section 4.3.2.4 and 4.5.1 (Investigation). As an equivalent alternative to the requirement for the audited organization to investigate any adverse audit finding to determine and schedule appropriate corrective action, Bechtel's auditing organization may determine the investigatory action and corrective action including action to prevent recurrence pertinent to adverse audit finding. These actions are agreed to by the audited organization. Further, in Section 4.5.1, as equivalent alternative to the 30-day response time, a response time appropriate to the finding is agreed to by the audited and auditing organizations.

2. Reference: Regulatory Section C.7 Standard Section 5.2 (Audit Records). Audit records shall include documents as defined in the standard and other documents if necessary to support audit findings. 22