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 21 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
CHAPTER 3	
3.8-8a/8b through 3.8-9/10	3.8-8a/8b through 3.8-9/10
3.8-24c/24d	3.8-24c/24d
Table 3.8-1/2	Table 3.8-1/2
CHAPTER 13	
13-i/ii through 13-v/blank	13-i/ii through 13-v/vi
13.1-1/2 through 13.1-7/8	13.1-1/2 through 13.1-71/8
Table 13.1-1 (sheet 1 of 25)/ (sheet 2 of 25) through (sheet 25 of 25)/ (sheet 25a of 25)	Table 13.1-1 (sheet 1 of 14)/ (sheet 2 of 14) through (sheet 13 of 14)/ (sheet 14 of 14)
----	Table 13.1-1a (sheet 1 of 8)/ (sheet 2 of 8) through (sheet 7 of 8)/ (sheet 8 of 8)
Figure 13.1-1 through 13.1-4	Figure 13.1-1 through 13.1-4b

Discard Old Sheet (Front/Back)	Insert New Sheet (Front/Back)

CHAPTER 17

17-i/ii through 17-v/blank	17-i/ii through 17-iv
17.1-1/2 through 17.1-47/48	17.1-1/2 through 17.1-45/46
Table 17.1-1 (sheet 1 of 3)/ (sheet 2 of 3) through (sheet 3 of 3)/blank	Table 17.1-1 (sheet 1 of 3)/ (sheet 2 of 3) through (sheet 3 of 3)/blank
Figure 17.1-1 through 17.1-4	Figures 17.1-1 through 17.1-3
17.2-1/blank	17.2-1/2 through 17.2-5/ blank

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. 21

APPENDIX 1B

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APPENDIX 1B

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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 21
Training of Nuclear Power Plant Personnel." The S/HNP licensed operator training program will also meet the requirements of 10 CFR 55, "Operators Licenses."

The simulator used in the S/HNP operator training program, will meet RG 1.149, 4/81, "Nuclear Power Plant Simulators 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.

The S/HNP will construct a Plant-unique simulator if either: 21

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; (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." 21

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. <u>Organizational Responsibilities</u>

   Witnin 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. <u>Administrative and Technical Review Steps</u>; and

3. <u>Recipients of Information</u>

   a. <u>General</u>

      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

(a) License Event Reports
(b) Regulatory Guides 21
(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 21
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.

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

(2) Licensee Event Reports 21

(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)

(d) Safety and Licensing Reports

(e) Reports and Instructions prepared by GE engineering organizations 21

(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. <u>Avoidance of Extraneous and Unimportant Information</u>; and

5. <u>Avoidance of Conflicting or Contradictory Information</u> 21

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. <u>Practical Interim Audits</u>

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 plan will be developed and implemented by the Plant Superintendent who will also be responsible for training of operators. 21

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.

2. The results of the program, described in the response to Item II.B.8.1.

3. Results from applicable portions of generic efforts on procedures, such as those being sponsored by the B Owners Group and currently nderway, efforts by INP. 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 NURFG-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 plant-specific procedures. (The use of procedures during the preoperational testing program is discussed further in S/HNP PSAR Section 14.1.3.4). 21

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.

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

The response to this question is arranged in the order suggested in NUREG-0718, Rev. 1, requirement. The sections are: 21

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 sufficient information for NRC review prior to fabrication of the main control boards.

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 Nuclenet 1000

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.

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.

(2) Integrate planned operations functions into a single operator's station for steam supply and power conversion systems. 21

(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 Control

Room 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

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. 21

(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. Design Concept is Technically Feasible, within the State of the Art, and Will Be Properly Implemented

a. Technically Feasible

The Nuclenet 1000 control room concept was 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 21

(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. State of the Art

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.

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 21 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 21
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 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 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. 21

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. <u>Commitment to Control Room Designs Complying with Human Factors Principles</u>

The S/HNP commitment to incorporate human factor engineering into the design of the control room has been addressed. As described above, this commitment and the review work performed satisfies the requirements of NUREG-0718, Rev. 1, and the guidelines presented in NUREG-0659 even though S/HNP's program was initiated in 1976 several years prior to the requirement. 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 review will encompass the guidelines and requirements applicable at the time the detailed final design process is resumed. 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.

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.

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: 21

(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 Ir' . ation
Steam Bypass ·d Pressure Regulation
Turbine Controls
Generator and Main Transformer
Performa Monitoring System
Display .ntrol System

(2) Reactor Core Cooling Benchboard P601, Diesel Generator Control Boarc 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.

System Assignments:

Reactor Core Cooling Benchboard P601 21
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
Ste..a 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

Service Water System
Turbine Building Closed Cooling Water 21
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.

c. The detailed layout of the individual controls and instruments is not yet final. The design finalization and human factors review, prior to insert fabrication, will be performed as described above. 21

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

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. 21

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 21
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).

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 <u>SAFETY SYSTEM STATUS MONITORING</u>

<u>NUREG 0718, REV. 1, REQUIREMENT</u>

"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 implemented properly prior to the issuance of operating licenses."

<u>RESPONSE</u>

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 21
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.

To add further verification to the Q-List, it will be 21
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 CFR 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.

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

d. Containment stresses exceeding those allowed for 21
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 21
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.

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.

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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.

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.

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." 21

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:

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. 21

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.

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. 21

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.

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 21
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 contruction status meetings. 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 assignments.

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.

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. 21

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 check-lists 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.

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.

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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.

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. 21

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.

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. 21

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.

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. 21

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.

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 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 Procedures are established indicating an effective corrective action program has been established for contractors and suppliers performing safety-related activities. The appropriate QA organization reviews and documents concurrence with the procedures. 21

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.

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. 21

2D6 (18B1) 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 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 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 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. 21

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.

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. 21

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.

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. 21

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.

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.

RESPONSE (Puget, NESCO, and Bechtel) 21

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. The criteria for determining staffing for the QA organization includes:

Continually evaluating the size and adequacy of the QA organization, including the inspection staff, as activities change and evolve to assure that the QA organization is of sufficient size to effectively evaluate the performance of primary contractors and the work they perform. The number of personnel in the organization will depend on the status of the project. QA staffing will be based on a long-range projection work schedule and will be periodically re-evaluated and adjusted as necessary.

2F2 The QA organization is involved in establishing long-range projected work schedules. Adequate QA/QC staffing will be available to prevent QA/QC

personnel from being required to perform inspections or evaluations without adequate preparation time or under pressure to complete inspections within a scheduled time period. Adequate QA/QC staff will be available to allow for prompt closeout of open nonconformances and proper followup to ensure corrective action has been taken.

NRC ACCEPTANCE GUIDANCE

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

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) 21

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.

NRC ACCEPTANCE GUIDANCE

Providing a QA role in design and analysis activities.

The QA program provides provisions to assure that:

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 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. 21

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.

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."

RESPONSE

Venting capability of the S/HNP reactor vessel is addressed 21
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 noncon-densible 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 ve..ced 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.

   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. Procedures for the use of these lines will be summarized in the FSAR. 21

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.

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.

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."

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*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 and II.B.2-3.

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.

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: 21

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 taking into account the frequency and duration of the activities anticipated for that area.

Potential support areas will be reviewed to determine under what circumstances they will be accessible, consistent with GDC 19 limits. 21

Average area dose rates will be used to determine personnel doses, although local hot spots may exist. 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 identified above to establish the integrated dose, including 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 21
provide pertinent data to the operator in order to describe radiological and chemical status of the Reactor Coolant System.

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 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." 21

RESPONSE

Introduction

Recognizing the importance of seeking improvements in the reliability of core and containment heat removal systems, the S/HNP commits to a program which will improve reliability but which is also practical, considering two-thirds* of S/HNP engineering design is already completed.

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*Subsequent setback in overall engineering completion to less than 50% has resulted from Project relocation to a Hanford site. However, such relocation has negligible impact on the design of NSSS/ECCS systems and the control room; and design of these systems remains more than 2/3 complete. In addition, most of the NSSS/ECCS components have already been fabricated and delivered into storage.

Although this commitment is for a program to ensure high reliability of critical safety functions such as core and containment heat removal systems, this program does not include a plant/site specific probabilistic risk assessment (PRA). Rather, it takes advantage of results obtained from several PRA studies in order to assess and improve reliability of highlighted systems. The program described below meets the intent of the requirement 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.

This program plan has been developed to take advantage of work completed or already underway in the PRA area and emphasizes early feedback of important conclusions into definition of areas where the greatest opportunity for risk reduction may exist. It also offers a program better oriented toward total operations reliability improvement. A full plant/site specific PRA requires a large effort using services that could be better applied in actual reliability improvement activities.

The following discussion is provided in the format of the NRC clarification statement of issues:

1. <u>Description of Reliability Improvement Program</u>

   Taking advantage of existing BWR plant PRAs and identifying unique S/HNP features, limiting systems and components affecting core and containment heat removal capability will be identified. Then parallel studies will be performed to identify new Plant systems, and improvements to existing Plant systems. Recommendations from each study would be evaluated with respect to reliability gains by means of quantitative techniques where applicable. 21

   The first study will be to identify the most promising additional Plant systems which could substantially improve core and containment heat removal capability. The second study, a design review, will be to assess, for existing identified systems, those features that can lead to substantial reliability improvements that would nevertheless have minimum impact on the existing design. The second study will include consideration of potential improvements which might be realized through component selection, additional startup verification, improved procedures, improved operator training, and improved component surveillance.

   Potential for systems interaction will be considered by examining the potential for common-mode failure and

negative impact of non-safety systems on the ability of safety systems to perform their intended functions.

As indicated in the response to Item I.D.1, a control room design review will assure that the final control room design implements applicable human factors engineering principles to maximize operator reliability.

The above studies will include active participation of S/HNP personnel (i.e., Puget and NESCO) to: (a) assure maximum value in operational planning; and (b) to facilitate the incorporation of identified additional Plant systems which will substantially improve core and containment heat removal capability.

2. Performance of Reliability Improvement Program

a. Review Existing PRAs

Following receipt of the Construction Permit for the S/HNP, the available completed PRAs for BWRs will be reviewed. These studies will include WASH-1400, the IREP for Grand Gulf, the PRA for Limerick, and the IDCOR generic BWR study. Rather than expend the resources to complete a new PRA for each individual project, it is felt that the work completed on other projects can be extended to S/HNP and others by identifying those features unique to S/HNP and adjusting the conclusions accordingly for areas leading to potential risk reduction. The dominant sequences will be identified and the areas in the existing design having the greatest risk reduction potential will be identified. Also, from these studies, the areas requiring the greatest attention to operator training, procedure generation, and maintenance and surveillance practices will be identified. 21

b. Study Designs for Risk Reduction

Without significantly impacting existing system design, a special study will be completed to identify the most promising systems for core and containment heat removal reliability improvement. This study will be used as the basis for future design provisions in the event that future rulemaking should determine that additional reduction in the risk of degraded cores is required. The benefit of this approach is that the existing design will be minimally impacted and the benefit of continuing with standard design

concepts is preserved. Included in the study will be the identification, for example, of the risk reduction benefit of new independent systems such as an isolation condenser for containment heat removal.

The study seeks to determine those design features that have the highest potential for significiant risk reduction for the S/HNP. This study, employing the principles of probabilistic risk assessment, identifies and evaluates those features that could reduce the incidence of degraded cores. Each feature is evaluated as to feasibility, effectiveness, and relative risk reduction.

This study will quantify, on a relative basis, the magnitude of the risk reduction associated with selected new design features.

Rather than expending resources and time to first complete an S/HNP specific PRA which results would not be available in time for incorporating into the S/HNP final design, prior results of generic BWR risk assessments such as WASH-1400 are examined to look for dominant event sequences.

PRAs of BWRs such as WASH-1400 have shown that 21
three sequences of events are predominant:

(1) Failure to remove decay heat. The reactor core is covered with water but decay heat cannot be removed from the suppression pool, eventually leading to containment failure from overpressure.

(2) Failure to shut down reactor. The reactor scram and the standby liquid control poison systems are not effective, leading to containment failure from overpressure caused by the depositing of energy to the suppression pool through safety/relief valves.

(3) Failure to provide makeup water to the reactor. During break and non-break accidents, such failure leads to core damage and its associated impact on the integrity of the containment and the release of fission products.

This study presumes that the issue of Anticipated Transients Without Scram (ATWS) will be resolved satisfactorily between General Electric, Puget, and the NRC and that the risks associated with failure to shut down the reactor will be low after such resolution. On that basis, the approach taken in the study is to examine a variety of design features which might reduce the risks associated with failure to remove decay heat and failure to provide makeup water.

This study will include a definition of these features and investigations into their technical feasibility, design impact on S/HNP, advantages and disadvantages, and an estimate of risk reduction.

The impact of each new feature on Plant risk will be estimated by a simplified analysis. The results reported in WASH-1400 will be modified to represent the S/HNP BWR6 plant. The initiating events (anticipated transients, loss-of-coolant accidents (LOCA), and anticipated transient without scram (ATWS)) can be consolidated into a limited number of dominant event tree sequences leading to core damage. Event trees and transient analysis are used to place all of the initiating events into a few classes from which the reduction in event probabilities leading to core damages can be assessed. 21

The reduction in core damage probability will be estimated for each of the new features and compared to the S/HNP "base case."

Following completion of this study, decisions will be made either to proceed with implementation of the recommended new features or delay for future review depending upon the status of regulatory actions requiring adoption of additional risk reduction systems.

c. Design Review Program

Under the direction of the S/HNP, a design review will be conducted to examine the basis for component selection and potential for undesirable systems interaction.

This study will utilize the experience of a group of senior engineers from within Puget and NESCO and outside the utility not directly involved in

the Plant and system design process to critically assess the design principles applied to achieve high systems reliability.

One member of this team will have experience in probabilistic risk assessment. The work from study item 2a, above, will be used to identify the dominant event sequences and limiting components which will be given highest priority for review by the Design Review Team.

The Design Review Team will also examine the Systems Analysis described in the response to I.F.1.

The System Analysis provides a methodological approach to safety system requirements and leads to a firm understanding of those most necessary systems or those systems important to safety. These systems will be examined by the Design Review Team to assure that the necessary reliability is being designed into the systems (considering single active component failures and potential operator errors).

Equipment selection will be examined against
available equipment failure data and recommenda- 21
tions may be made toward improving reliability through improved operational planning or additional equipment qualification through startup testing. Modifications in system configuration may also be suggested for further evaluation of benefits and impact on design.

Particular attention will be given to identifying those systems and features which should be given emphasis in operating procedures and operator training. Feedback from other operating plants will also be factored into this study where applicable.

Utilizing the experience in probabilistic risk assessment and engineering experience contained in the Design Review Team, efforts will be made to quantify the relative reliability improvement to be gained with component modifications, system modifications, improvements in operator actions, or other operational planning recommendations made for those systems examined.

In addition to system by system reviews, the potential for system interaction will also be

identified. The possible interaction of non-safety systems with safety systems leading to possible loss of core cooling or containment heat removal will be identified. Past studies of systems interactions for this Project will be reviewed for completeness.

Results of the Design Review Team work will be documented and available for NRC review prior to the FSAR submittal.

d. Control Room Design Review

As indicated in the response to I.D.1, the S/HNP control room utilizes the Nuclenet design which is an advanced, computer-driven display concept. A control room model has been constructed and examined for proper application of human factors principles. Prior to initial operation, a verification of the adequacy of the control room design to include those elements that cannot be evaluated on the mock-up will be completed. This review will assure an even higher level of operator reliability.

3. Factoring Program Results into Design, Fabrication and Operation 21

Recognizing that most of the NSSS/ECCS components have already been fabricated and delivered into storage, the following describes how the Reliability Improvement Program will be factored into design, fabrication and operation of the S/HNP. Because the S/HNP personnel will have participated in the four phases of the program described in 2 above, the potential for reliability improvement will be quickly recognized and retained for incorporation into operational planning, future configuration control decisions, and exercising any remaining options on component selection, specifications and testing.

a. Configuration Control

Based on existing system layout the study results will be used to optimize the reliability of the core and containment cooling functions. This may include (to the extent that it remains practical to modify designs at the conclusion of the studies) piping interconnections, separation, number of maintenance valves, local manual versus remotely operated valves, dependencies on auxiliaries, etc. Evaluation of separate diverse

systems to accomplish these funtions, including design trade-off studies, is embodied in the study item 2b. In addition, study item 2c (Design Review Program) results of system modification studies will be considered.

b. Component Selection, Specifications and Testing

The results of the study item 2c will include recommendations with respect to reliability improvements for selection, specification, and pre-installation testing of components. These recommendations will be considered in view of the advanced stage of design, manufacture, and delivery of many components. The recommendations will be incorporated where practical.

c. System Interaction

One of the elements of the study item 2c will be a systems interaction study. The identification and recommended corrections of potential systems interaction which reduce the reliability of the core and containment cooling functions, is part of this study. The Review Team recommendations (for example, non-safety grade control systems interfering with safety functions) will be examined for 21
practical modifications.

d. Maintenance

The study item 2c will consider and recommend specific maintenance considerations for critical components to improve system reliability. The Design Review Team recommendations will be reviewed at the time that maintenance planning begins and will particularly include attention to periodic replacement of parts and components and attention to concerns for long-term unattended operation following a severe accident.

e. Procedures

Recommendations toward those procedural areas that can lead to reliability improvements from study item 2c will be identified by the Design Review Team. These recommendations will be reviewed and incorporated where practical into the development of operational, maintenance, surveillance, emergency and administrative procedures.

f. Operator Training

Where the study item 2c and 2d indicate critical operator interface areas, the operator training program plan will be developed tc address these concerns.

g. Operating Feedback

The study item 2c will identify those systems most limiting in the dominant event sequence studies. The surveillance testing program will include consideration of these critical systems.

h. Design Review

The study item 2c embodies the review of critical systems and components for reliability improvements. Where substantial reliability improvements can be gained through special quality assurance considerations, these considerations will be reviewed for possible incorporation into the quality assurance program.

4. Schedule

Tne reliability improvement program will commence at the issuance of the Construction Permit and is expected to take approximately two years as shown in Figure II.B.8.1-1. By the end of the study, design and fabricacion will be essentially complete for core and containment heat removal systems and support systems. However, the results will be used on a case by case basis to determine whether major redesign, repurchasing, and refabrication are warranted. 21

5. Acceptance Criteria

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

6. Mitigation Features

The S/HNP agrees with the NRC guidelines that additional mitigation features will not be a consideration of this study.

7. Radioactive Release Estimate

Although the primary focus of this reliability improve-
ment program is directed at reducing the potential for
a degraded core condition, the need for excessive
design or operation modifications will be examined in
view of the expected benefit of the fission product
retention capability of the Mark III Suppression Pool.
In addition, the remote siting of the S/HNP will be
considered in assessing the benefits of substantial
design modifications. In the event that the effect of
improvements to the reliability of the core and
containment cooling systems cannot be evaluated with a
qualitative assessment of relative reduction of 21
fission product releases discussed above, total
radioactive releases to the environment will be
estimated for the contemplated improvements.

8. Summary

In summary, the above program is established in order to provide a balanced approach toward assuring the low level of risks achievable with a BWR6 Mark III design concept. Full advantage is being taken of quantitative risk assessments performed generically and on other units in more advanced stages of design and operational planning.

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."

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. 21

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

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.

Ignition systems to control hydrogen are presently being utilized on Sequoyah 1 and McQuire (PWRs), and will be utilized on Pilgrim 2 (PWR) and Grand Gulf (BWR Mark III Containment similar to the S/HNP). The S/HNP is monitoring 21
the development of hydrogen igniter systems throughout the industry, including the testing programs being performed by EPRI and others in the industry. As the results of these analyses and test programs become available they will be considered for possible incorporation into the S/HNP design.

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.

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 and after being exposed to the environmental conditions created by activating the hydrogen igniter system or will be protected from these environmental conditions. 21

A report of the S/HNP hydrogen igniter system will be furnished within 2 years of receipt of a Construction Permit. As a minimum, this report will include: (a) analyses of various accident scenarios that can lead to 100% cladding-water reactions and the consequential responses of the containment; (b) analyses of hydrogen releases, mixing, and distributions within the containment; (c) responses of the containment structures and essential equipment to local detonations and to the environmental conditions resulting from the combustion of the hydrogen; and (d) igniter performance and endurance characteristics.

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 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. 21
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

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

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 servicability and containment integrity are maintained. Liner plate 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.

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. 21

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.

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. 21

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.

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 21
would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion."

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 hydrogen will not collect in areas where 21
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:

(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." 21

RESPONSE

The equipment of the systems required to achieve and maintain safe shutdown will be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system.

The location of components associated with these systems and the method of protection if required) 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 21
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."

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 and two-phase flow safety valve tests for the conditions which can 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.

A description is given in Reference 3 for those tests which 21
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 two-phase and solid water flow conditions for which the valves are being tested.

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 21
September 17, 1980, and titled "NUREG Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves."

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 21
the NRC by letter, T. Keenan (BWROG) to D. G. Eisenhut (NRC) dated October 17, 1979.

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 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 21
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

<u>RESPONSE</u>

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. <u>Compliance with SRP 6.2.4, Rev. 1</u>

   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

Essential systems are those critical to the immediate mitigation of any event that results in automatic containment isolation. Essential systems are not automatically isolated by accident signals.

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 21
an accident, while still maintaining the effectiveness of the containment.

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.

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

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 21

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. <u>Purging Consistent with ALARA</u>

   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. <u>Performance of Purge and Vent Valves Against Accident Pressure</u>

   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.

   The purge and vent containment isolation valves are
   not expected to have to close against the containment 21
   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. <u>Interim NRC Guidance on Valve Operability</u>

   The purge and vent containment isolation valves will meet the applicable portions of the October 23, 1979, interim NRC guidance, "Guidelines for Demonstration of Operability of Purge and Vent Valves."

4. <u>Procedures and Restrictions Consistent with Revised Requirements</u>

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

5. <u>Assurance of Purge and Vent Isolation Reliability</u>

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 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 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:

Range: 10^{-6} microcuries/cc to 10^{3} microcuries/cc
10^{-6} microcuries/cc to 10^{4} microcuries/cc
(drywell or SGTS purge is included)

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

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.

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:

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. 21

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 indicators of coolant level in the reactor vessel and incore thermocouples in PWR's and BWR's.

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 21
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 instrumentation which will be incorporated in the S/HNP design is addressed in the response to Item II.F.3.

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 [illegible] staff. The S/HNP design will incorporate the results of these efforts as appropriate.

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 has appropriate flow instrumentation).

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 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 incorporates appropriate HPCS flow instrumentation. 21

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 incorporates the appropriate flow instrumentation.

d. <u>Standby Liquid Control System (SLCS) Flow Type D</u>

Due to the relatively low flow rate involved and the wide pressure range from atmospheric to maximum reactor vessel pressure involved, a credible direct flow measurement is not possible. A qualitatively derived flow measurement is possible using the 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.

e. High Radioactivity Liquid Tank Level Type D

The S/HNP BWR6 design uses sumps as the primary collection point. The principal sumps - the drywell equipment drain sump, drywell floor drain sump and the containment equipment and floor drain system - 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 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

These sensors are addressed under Type C as Primary Containment Area Radiation, and Radiation Exposure Rate; and under Type E as Primary Containment Area Radiation, High Range; Reactor Building or Secondary Containment Area Radiation; and Radiation Exposure Rate. The S/HNP Area Radiation Monitor design is presented in the S/HNP PSAR, Section 12.4 and in particular Figures 12.1-1 through 12.1-6 and 12.1-11 through 12.1-14. In addition to the installed area radiation monitors, specific operator actions will be planned based on exposure rates determined for the task utilizing portable dose rate instrumentation. 21

The S/HNP design also recognizes the Reg. Guide 1.97, Rev. 2, criteria 1.5d, "The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc. to give anomalous indications potentially confusing to the operator. Human factors analysis should be used in determining type and location of displays." Consequently the S/HNP radiation monitoring design emphasis is on accurate presentation of this data to the reactor operator. Of particular concern, when human factors are considered, is the Reg. Guide 1.97, Rev. 2, concept that an area monitor is a detector or indicator of a breach in the reactor coolant pressure boundary or the containment boundary. The area monitors will, under accident conditions, read significantly different

than during normal operation even without a breach. Instructions to the operator that these devices are an indicator of a breach could lead to the improper isolation of a safety system. The S/HNP design incorporates other adequate and appropate indicators of a breach in containment. Area monitors are not appropriate for this purpose. After the construction permit is issued, the detailed ranges of the area monitors will be reviewed for appropriate conformance to Reg. Guide 1.97, Rev. 2, requirements.

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 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. System status inputs to the effluent monitoring computer and values determined during preoperational testing are used to derive the necessary flow
rates. S/HNP airborne radioactive materials 21
release monitoring provisions meet the intent of Reg. Guide 1.97, Rev. 2. Final design information will be presented in the FSAR.

h. BWR Core Thermocouples, Type B and Type C

At the present time there is no qualified design for BWR core thermocouples. The S/HNP agrees with the stated purposes "to provide diverse indication of water level" and "to monitor core cooling." The S/HNP intends to implement the optimum qualified and approved system available at the time final design commitment is required. Significant questions exist as to the final acceptability by the NRC of the use of core thermocouples for the stated purpose. The S/HNP is committed to achieving the objective of diverse water level indication and monitoring of core cooling, thus being in compliance with this requirement.

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 shall be used by CP and ML applicants in addressing this 21
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 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 organizational 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 organization for management of design and construction of the S/HNP. 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. 21

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, 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. Adjustments to the manpower plans are made periodically as required by actual workload.

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. 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.

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 signficantly 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. <u>Details of the Interaction of Design and Construction Activities</u>

a. <u>General</u>

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.

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.

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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.

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.

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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 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; 21
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 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.

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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. .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. Periodic meetings are held by the Vice President, Generation Resources, to discuss Project status and problems.

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 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. Periodic meetings are held by the Vice President, Nuclear Projects, to discuss 21
Project design and construction status and problems.

The NESCO S/HNP Project Manager provides periodic reports to 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 communication and hold regular Executive Review Meetings. NESCO's Vice President, Nuclear Projects, is in frequent communication and holds regular meetings with executives of Bechtel and GE thus enabling Bechtel and GE management to be informed regularly of the Project status, management and technical issues, and plans for the future.

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 immediately. 21

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 lowest set relief valve which will open to limit the 21
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 21
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.

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.

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. 21

2. <u>Conduct of Study</u>

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 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. 21

3. Completion Date

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

4. Program for Implementation of Results

The study concluded that the leakage through a grossly failed recirculation pump seal is of no consequence to any of the loss-of-coolant accident analyses. Therefore, no changes are required to implement the results.

REFERENCES

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

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 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 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." 21

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.

2. Conduct of Study 21

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. 21

REFERENCES

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. 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: 21

(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 provid d no HPCI or HPCS system flow exists and a low pressure ECC system is running. This 21
logic would complement, not replace, the existing ADS actuation logic."

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.

The details of this study are provided in Reference 1. 21

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 hi 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 MODIFICATION

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 designs.

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 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." 21

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

"In order to simplify the reading of the water level in the 21
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 core support plate to the steam lines centerline will be provided as post-accident monitoring instrumentation and will be continuously monitored.

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 agree with the NRC on a uniform design basis for ADS air accumulator sizing. The results of this effort will be adopted for S/HNP and design changes made if necessary.

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

1. Nature of Study 21

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

The study is complete and was transmitted to the NRC in Reference 1. 21

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 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. 21

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.

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

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 and III.A.1.2-2. 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.

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 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. 21

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. The 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. As a minimum, the SPDS (see Item I.D.2) will be available. The Shift Supervisor's 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 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. 21

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.

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

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

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

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

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

e. 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.

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 preliminary location for carrying out the functions of the Operational Support Center (OSC) will be in the Service Building, on the first floor (elevation 415'). This elevation is shown on Figure III.A.1.2-3. 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. Communications with the Plant, including the two TSC areas, the control room, the EOF, and off-Site locations will be available. 21

Emergency Operations Facility

The preliminary location of the S/HNP Emergency Operations Facility (EOF) is the Washington Public Power Supply System (Supply System) Nearsite EOF. Puget has had discussions with the Supply System concerning the joint use of the Supply System Nearsite EOF. The Supply System has indicated its willingness to conduct a feasibility investigation on the joint use of their Nearsite EOF. This feasibility study will be submitted as part of the S/HNP FSAR if joint use of the Supply System Nearsite EOF is proposed. In the event that this study indicates joint use of the Supply System's Nearsite EOF is not feasible, a facility meeting the requirements of NUREG-0696 (2/81) will either be constructed at the S/HNP Site, similar to the Supply System's Nearsite EOF, or will be located in Richland, Washington, approximately 12 to 15 miles from the S/HNP Site. Based on the above discussions with the Supply System, the tentative S/HNP joint use of the Supply System's Nearsite EOF is as described below.

The Supply System Nearsite EOF is located approximately 5 miles from the S/HNP as shown in Figure III.A.1.2-4. The Nearsite EOF meets the functional requirements of NUREG-0696 (2/81). Detailed design information on the Supply

System Nearsite EOF is provided in Reference 2. A floor plan of the Nearsite EOF, indicating utilization of the facility in the highly unlikely event of a WNP 1, 2 or 4 emergency and a S/HNP emergency, is shown in Figure III.A.1.2-5.

REFERENCES 21

1. NUREG-0309, "Safety Evaluation Report: related to the construction of Skagit Nuclear Power Project, Units 1 and 2," September, 1977.

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 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." 21

*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 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. 21

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.

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. 21

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 PSAR 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 21
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.

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 is 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 of Potential Hazards in Site Vicinity.

b. Standard Review Plan 2.2.3: Evaluation of Potential Accidents. 21

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
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

21

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

21

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. Reactor Coolant Sample Line	
E	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	
F	Containment leakage rate is 0.25% per day. SGTS flow rate is one air change per day			Standby Gas Treatment System (SGTS)	

Table II.B.2-2

Post-Accident Containment Atmosphere, Drywell Atmosphere and Steam Source Terms

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
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

21

Note: Source terms are based on 100% noble gases and 25% halogens of the core inventory.

Table II.B.2-3

Post-Accident Reactor Coolant and Suppression Pool Source Terms

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	
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	21
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+06	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+0[illegible]	EU-156	2.0+05	
MO-99	2.1+06	TE-133	9.8+0[illegible]			
				Total	9.4+08	

Note: Source terms are based on 50% halogens and 1% solid fission products of the core inventory.

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	21
Post-Accident Sampling Station	Turbine	Infrequent	
Sample Analysis Area	Service	Continuous	
Health Physics Area	Service	Continuous	
Secondary Alarm Station	Control	Continuous	

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.

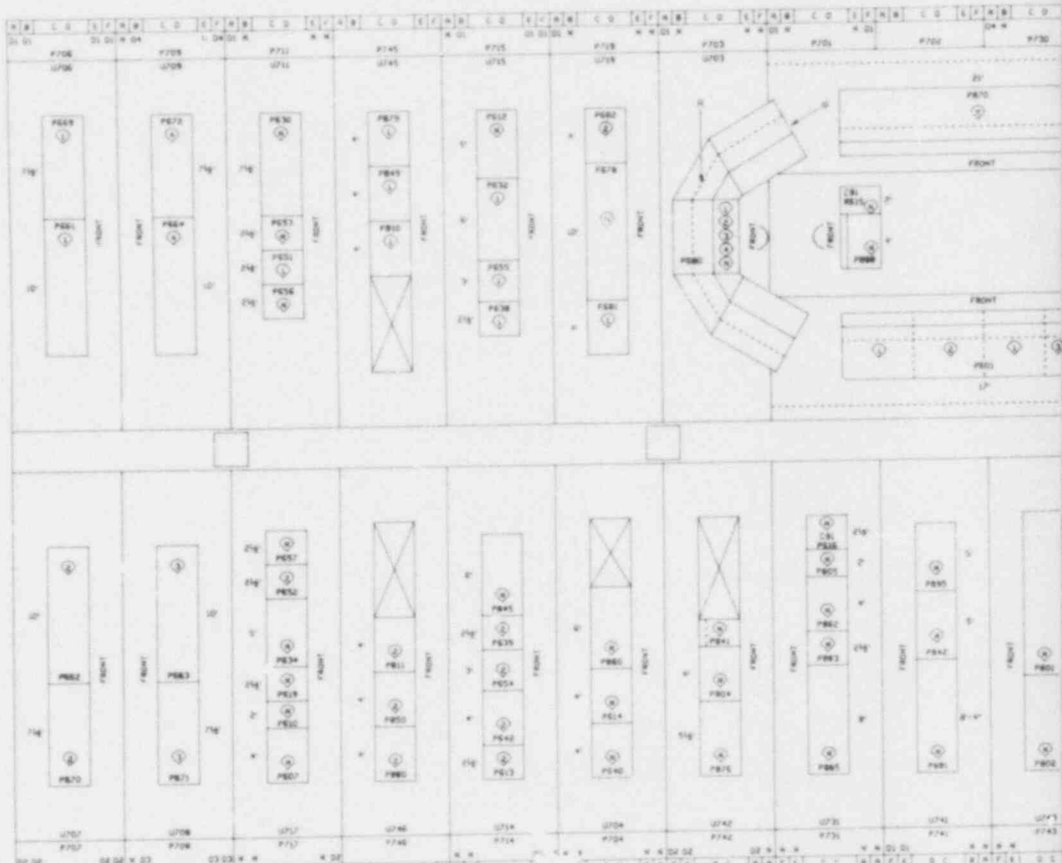
21

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		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	28	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

21

Notes: CP = Construction Permit
FL = Fuel Load
CO = Commercial Operation

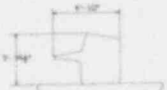


CONTROL ROOM ARRANGEMENT

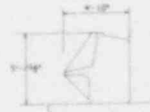
PGCC FLOOR EL. 470'-1"
CONCRETE FLOOR EL. 469'-0"

NOTES:

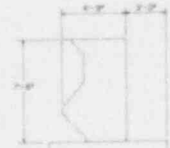
1. UNLESS OTHERWISE SPECIFIED, ALL CONTROL ROOM PANEL DESIGNATIONS ARE UNDERSTOOD TO BE PREFIXED BY H13 (E.G. H13-P800)
2. DELETED.
3. TERMINATION CABINET BAY ASSIGNMENTS ARE BASED ON PRELIMINARY INFORMATION FROM GE [illegible]
4. PANELS CONTAINING SAFETY RELATED EQUIPMENT AND/OR CABLES ARE INDICATED 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).



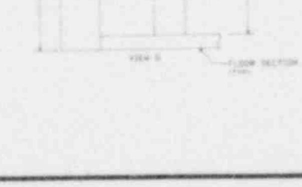
VIEW A



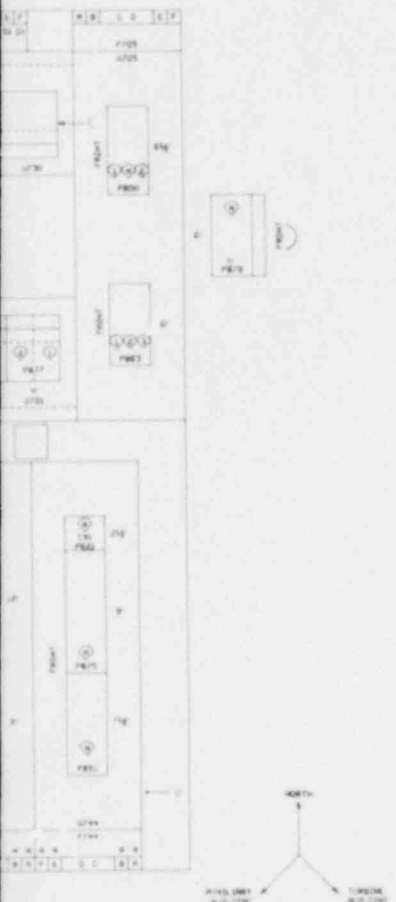
VIEW B



VIEW C



VIEW D



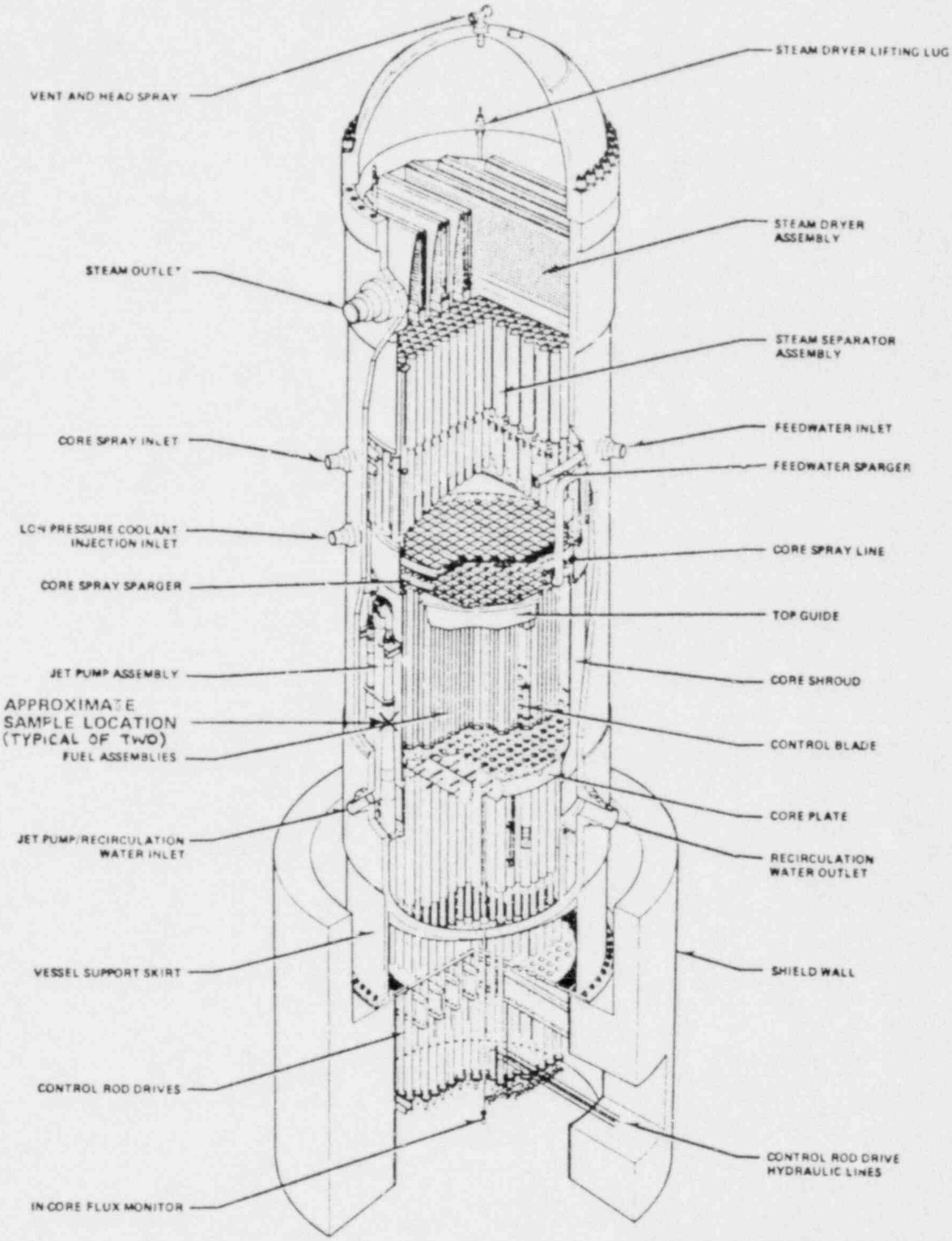
PANEL NO.	DESCRIPTION
H13-P601	REACTOR CORE COOLING BENCHBOARD
H13-P607	TIP CONTROL AND MONITORING
H13-P610	CONTROL ROD DRIVE TEST PANEL
H13-P612	FEEDWATER AND RECIRC INSTRUMENT PANEL
H13-P613	NSSS PROCESS INSTRUMENT PANEL
H13-P614	NSSS TEMPERATURE RECORDERS
H13-P619	JET PUMP INSTRUMENT PANEL
H13-P630	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 RAVMS AND ROVMS (NSSS)
H13-P657	DCS RAVMS AND ROVMS (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	SUPERVISOR'S CONSOLE
H13-P680	NUCLENET 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-P805	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	RFPT'S 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	SEISMIC MONITORING PANEL
H13-P885	VIBRATION MONITORING PANEL
H13-P888	PLANT COMMUNICATIONS CONSOLE
H13-P895	WOP ELECTRONICS
C91-R615	VIDEO COPIER
C91-P614	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-P807	SECURITY COMPUTER
H13-P821	DEH COMPUTER
H13-P822	TURBINE SUPERVISORY INSTRUMENTS PANEL
H13-P867	REHEAT TEMPERATURE CONTROLLER CABINET
H13-P891	RFPT'S ELECTRONICS CABINET
H13-P898	EMERGENCY TRIP SYSTEMS CABINET

PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CONTROL ROOM LAYOUT

FIGURE I.D.1-1



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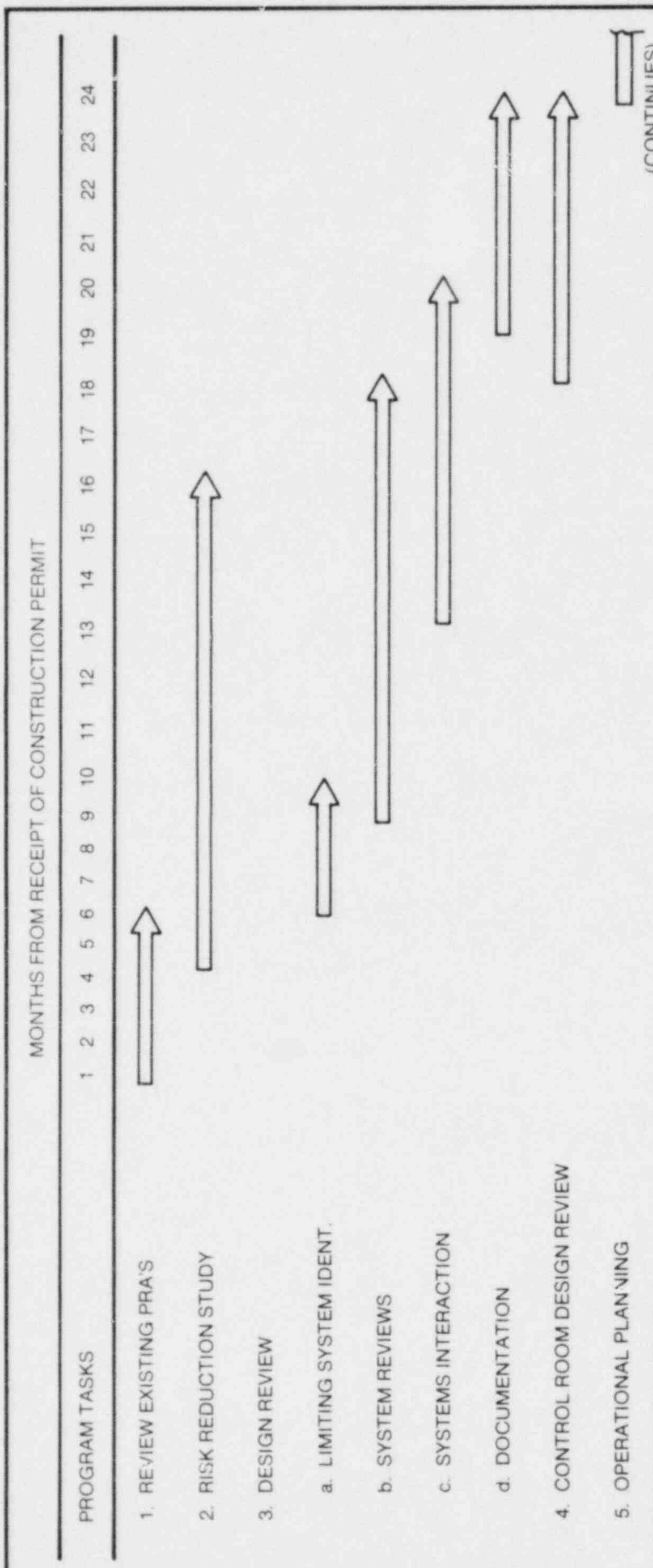
SKAGIT/HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

LOCATION OF POST-ACCIDENT RCS SAMPLE POINT

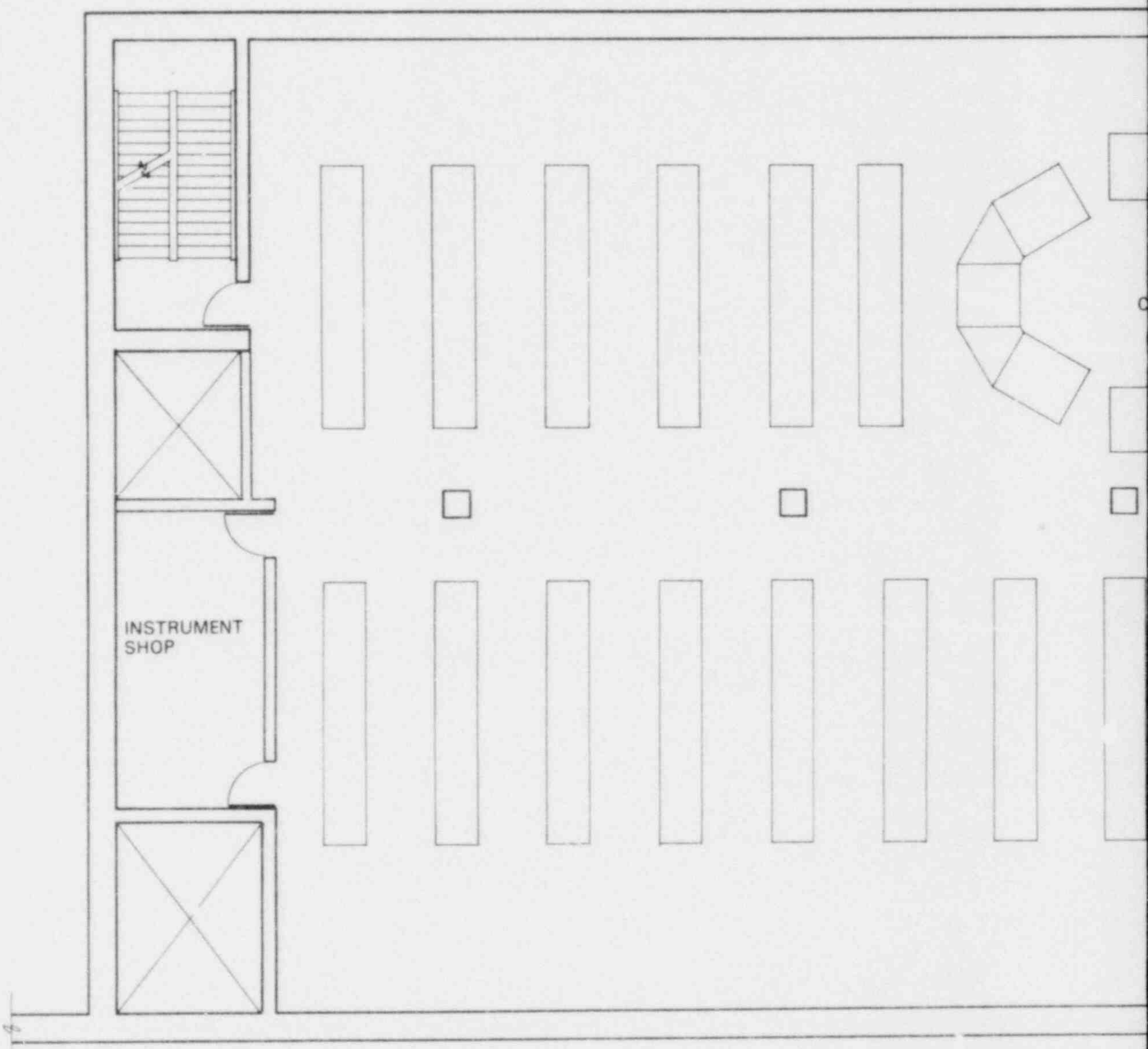
FIGURE II.B.3-1



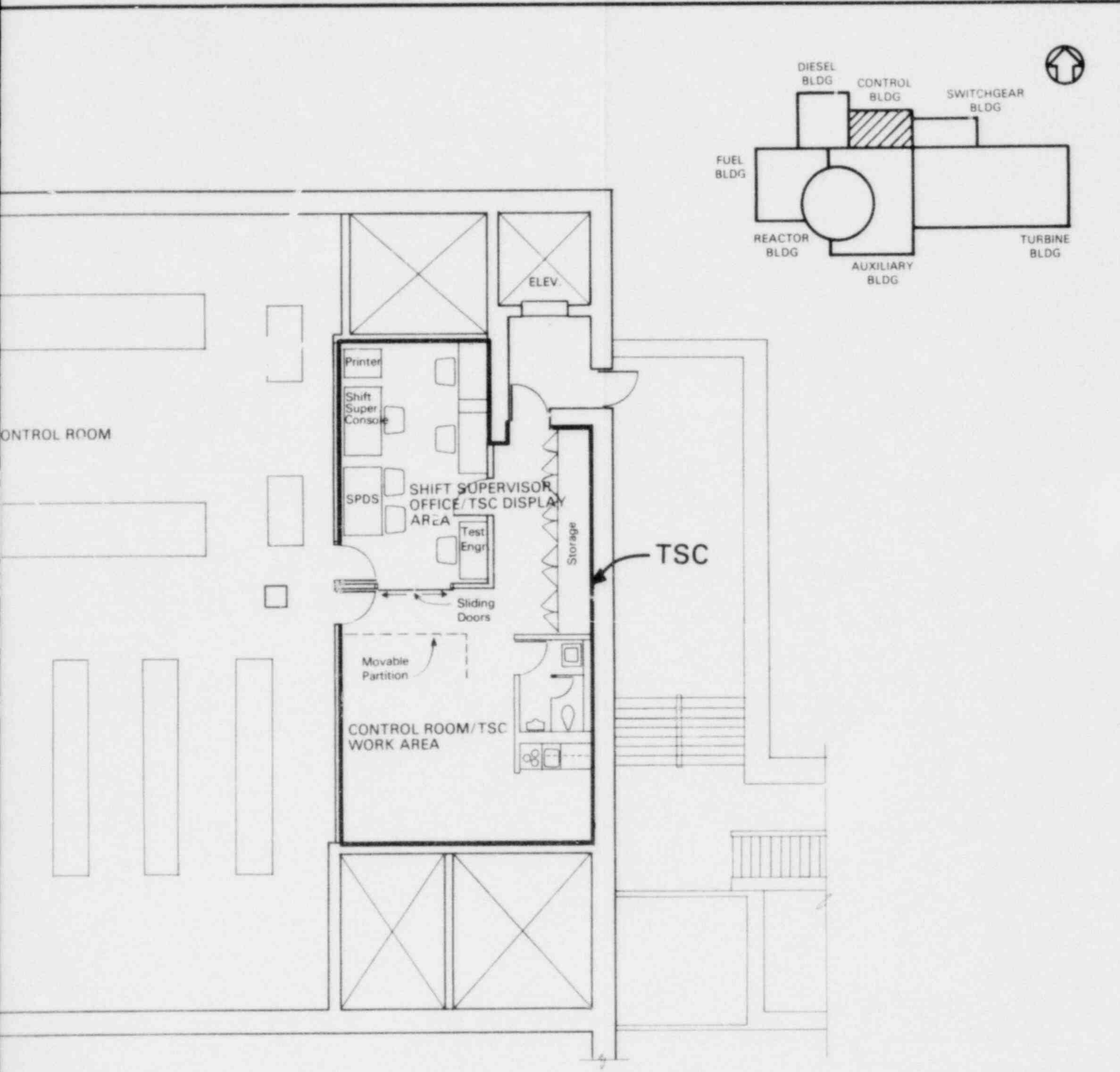
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

SCHEDULE FOR RELIABILITY IMPROVEMENT PROGRAM

FIGURE II.B.8.1-1



CONTROL BUILDING EL. 470'-0"



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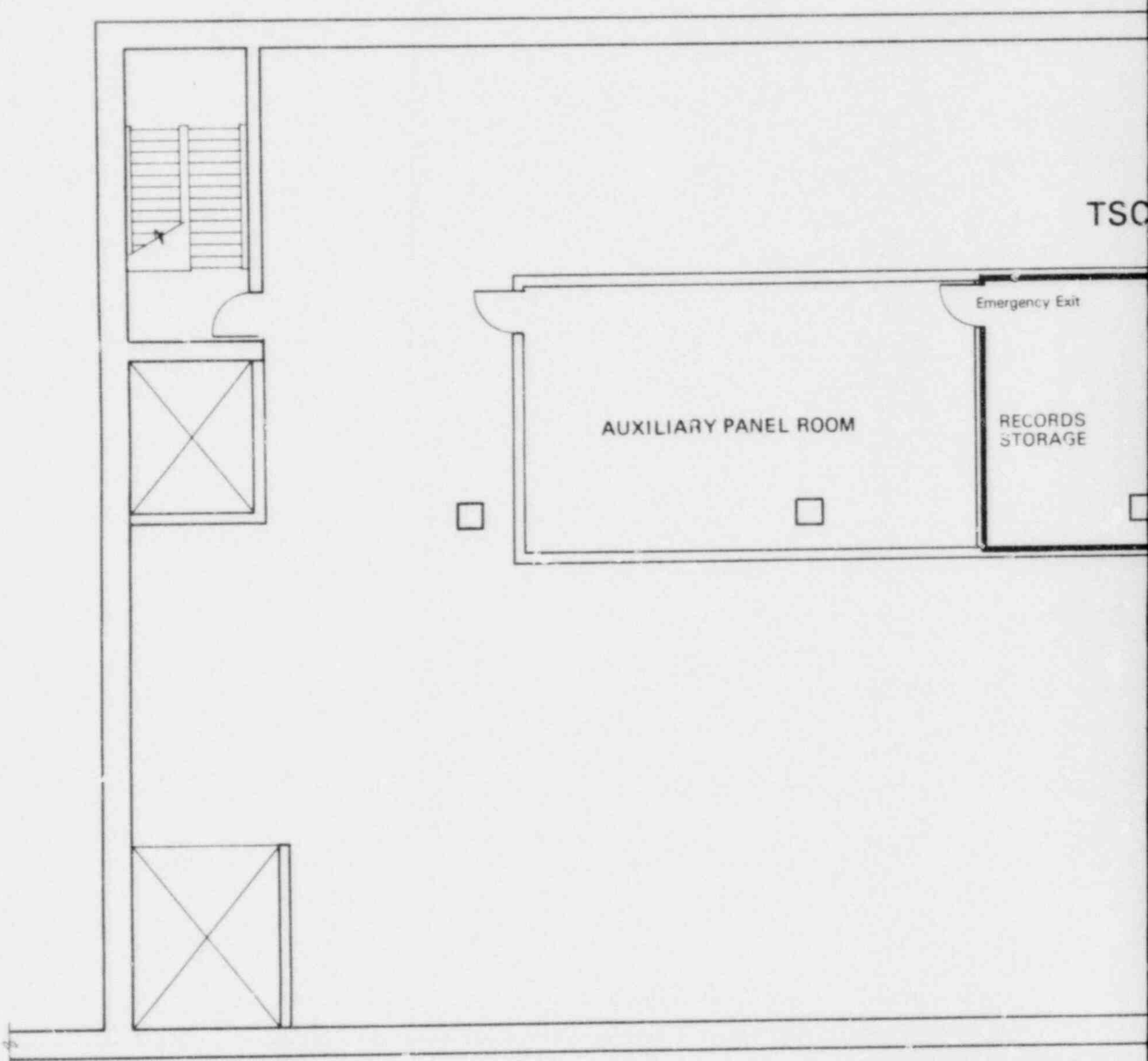
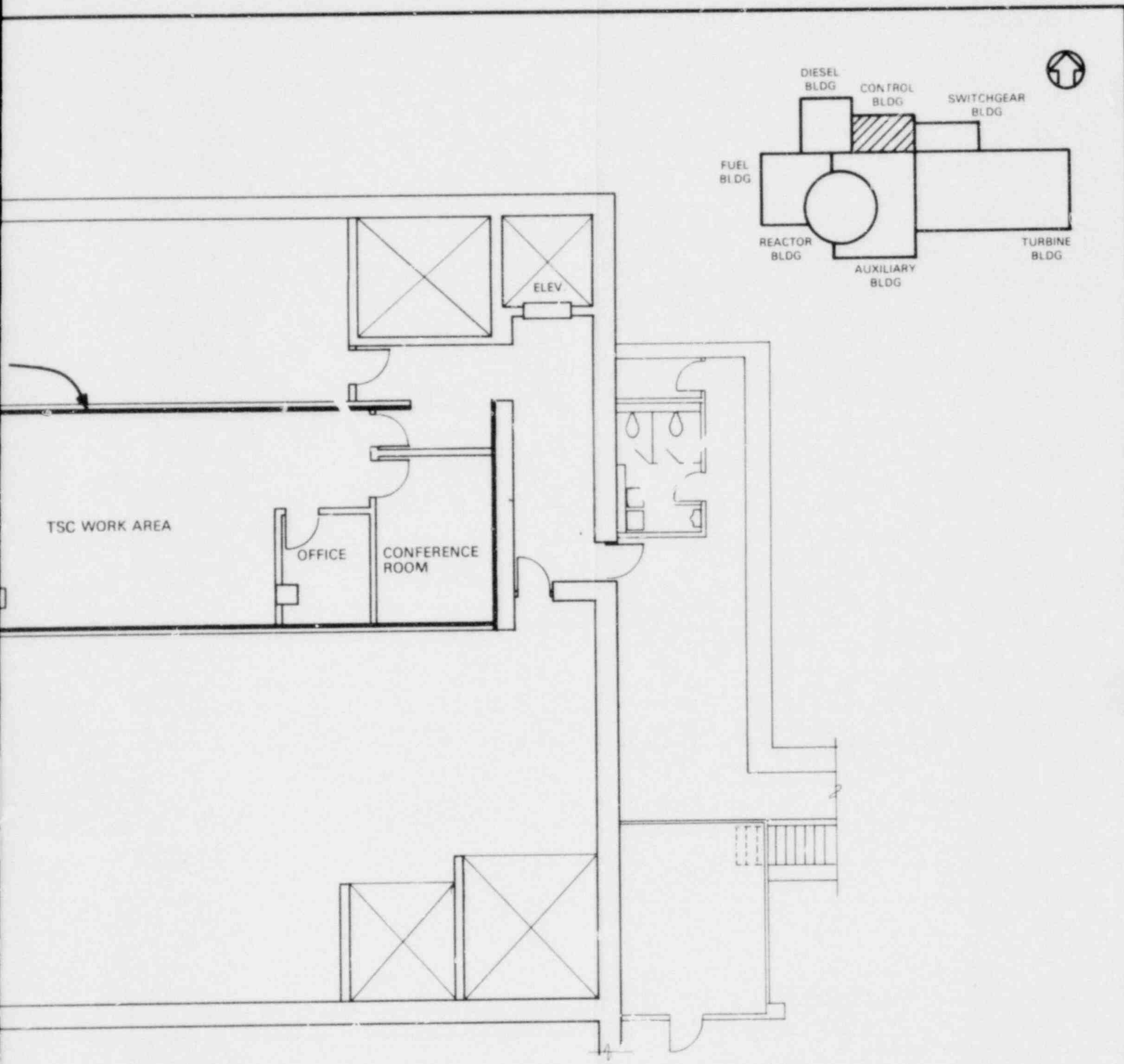
PRELIMINARY SAFETY ANALYSIS REPORT

PRELIMINARY TSC LOCATION & FLOOR PLAN

FIGURE III.A.1.2-1



CONTROL BUILDING EL. 486'-0



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

PRELIMINARY TSC LOCATION & FLOOR PLAN

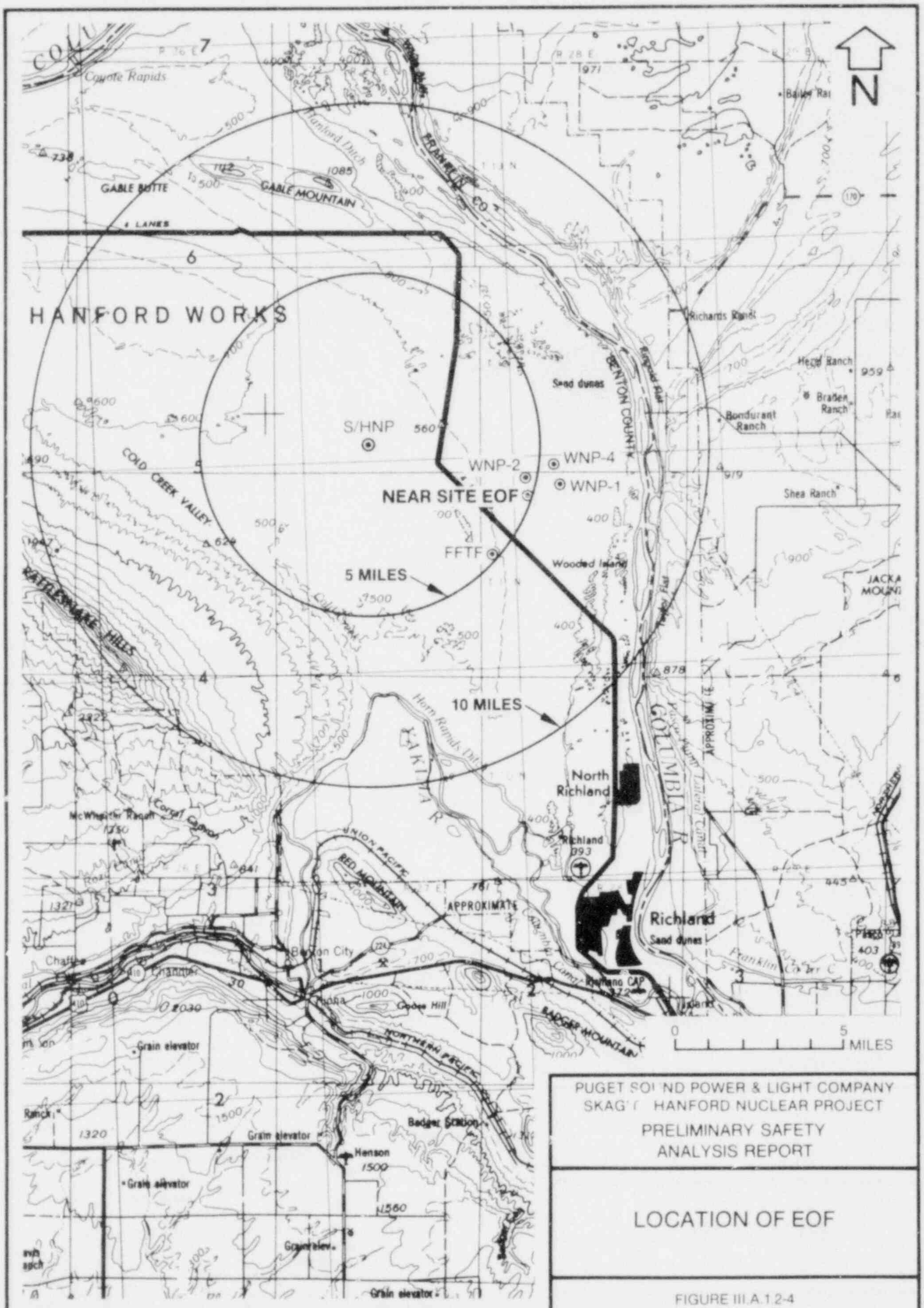
FIGURE III.A.1.2-2

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



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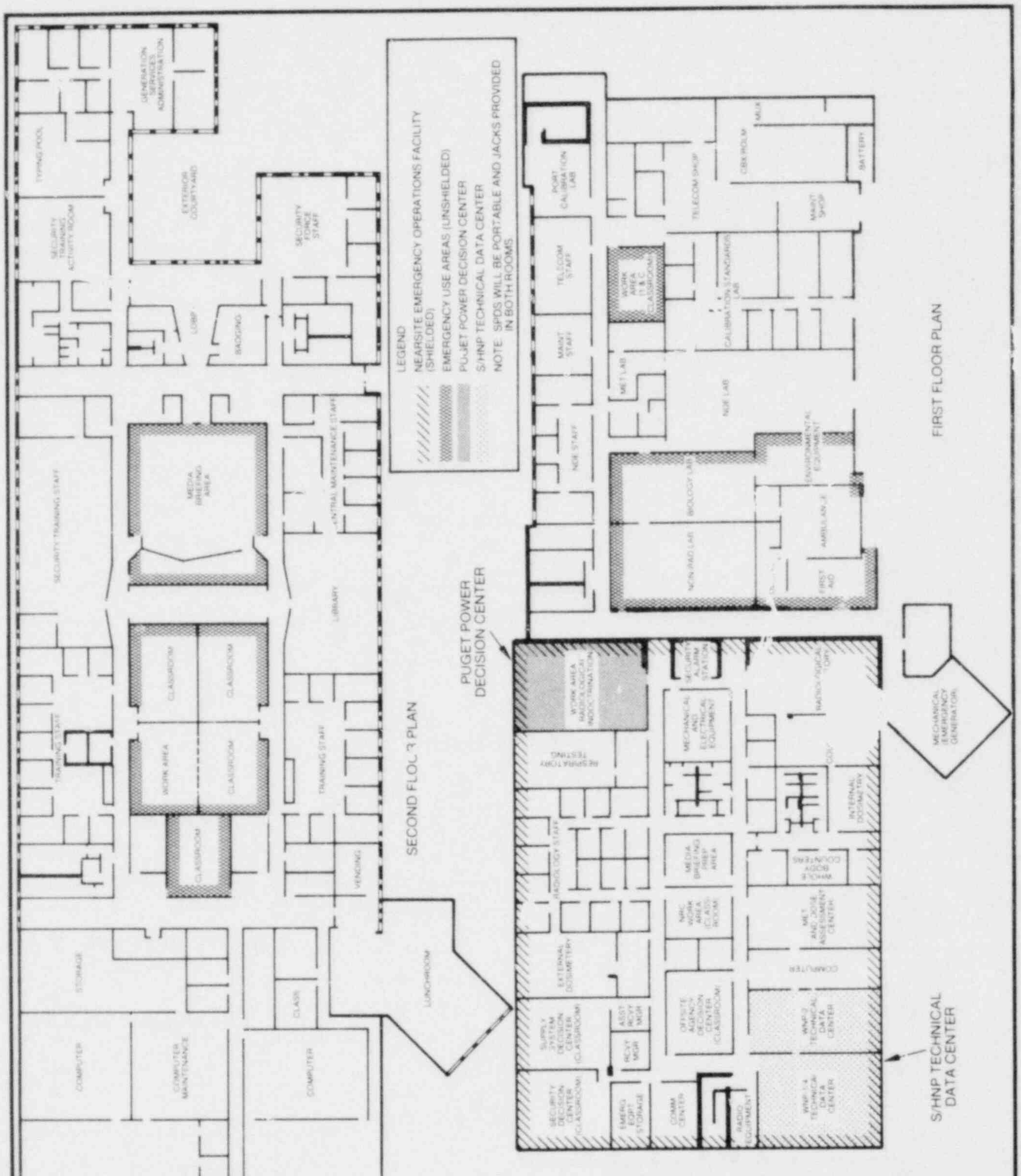
PRELIMINARY SAFETY ANALYSIS REPORT

LOCATION OF EOF

FIGURE III.A.1.2-4



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SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

EMERGENCY OPERATIONS FACILITY

FIGURE III.A.1.2-5

- W_{tq} - The loads due to tornado wind pressure.
- W_{tp} - The differential pressure loads due to rapid atmospheric pressure change.
- W_{tm} - The tornado generated missile impact effects. The type of impact, i.e., plastic, elastic, etc., together with the ability of the structure to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact. 130.5

(3) Abnormal Loads. Abnormal loads are loads generated by the design basis accident (DBA) and by a degraded core accident. 21

- P_a - Design pressure load within the structure generated by the DBA including pool swell phenomenon (see Appendix 6C of this PSAR). 12
- 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. 21
- T_a - Thermal effects and loads generated by the DBA including T_o.
- R_a - Pipe reaction from thermal conditions generated by the DBA including R_o.
- H_g - The internal pressure resulting from post-accident recovery flooding of the drywell and containment. 12 042.32
- R_r - The local effects on the containment due to the DBA. The local effects shall including the following:
  - R_{rr} - The effects resulting from the reaction of a ruptured high energy pipe during the postulated event of the DBA. The time dependent nature of the load and the ability of the structure to deform beyond yield shall be

considered in establishing the structural capacity necessary to resist the effects R_{rr}.

- R_{rj} - The effects resulting from jet impingement from a ruptured high energy pipe during the postulated event of the DBA. The time dependent nature of the load and the ability of the structure to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of R_{rj}. 130.5

- R_{rm} - The effects resulting from the impact of a ruptured high energy pipe during the DBA. The type of impact, i.e., plastic, elastic, etc., together with the ability of the structure to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact.

- T_{mw} - Thermal effects and loads resulting from a degraded core accident. 21

c. Load Values

(1) The live load shall be considered to vary from zero to full value for all load combinations.

(2) The maximum effects of P_a, T_a, R_a, and R_r shall be combined unless a time-history analysis is performed to justify lower combined values.

d. Load Definitions 12

(1) Static and Seismic Loads. Static loads are defined as those loads which are considered to remain constant with respect to time or which have a long period of application or rise time relative to the response period of the containment. This category also includes seismic load for which the dynamic effects have been included in their determination.

The following are examples of loads in this category: 12

- o Dead load (D) and live load (L)
- o Accident pressure (P_a) 130.5
- o Pipe reactions during normal and postulated accident conditions (R_o and R_a)
- o Design wind (W) and tornado wind pressure (W_{tq}) and differential pressure (W_{tp})
- o Operating and safe shutdown earthquake (E_o and E_{ss}), except when combined with impulse loading and impact effects

(2) Impulse Loads. Impulse loads are time dependent and include the following: 12

- o The dynamic effects of accident pressure (P_a) where rate of loading effects the response of the structure
- o The effects of pipe rupture reactions (R_{rr}) and jet impingement loading (R_{rj})

(3) Impact Effects. Impact effects are those that can be specified in terms of kinetic energy at impact. These include the impact energies resulting from tornado missilies (W_{tm}), pipe rupture generated missiles (R_{rm}), and any other specific site dependent missiles, including the case where a gap exists between the pipe and its structural restraint.

3.8.1.4.1 Design Accident Pressure Load

Transients resulting from the design basis accident (DBA) and other lesser accidents are presented in Subsection 6.2.1 of this PSAR and serve as the basis for the containment design pressure of 15 psig.

3.8.1.4.2 Post-Accident Flooding of Containment

To cover undefined damages resulting from the postulated loss-of-coolant accident, the containment is designed for flooding to a level 6 ft-10 in. above the top of the active fuel in the core.

Flooding would not be initiated until all other approaches are considered.

3.8.1.4.3 Thermal Loads

The temperature gradients through the containment wall are shown in Figures 3.8-7, 3.8-8 and 3.8-9 for the operating condition and the postulated design accident condition.

3.8.1.4.4 Wind and Tornado Loads

Wind and tornado loads are in accordance with Section 3.3 of this PSAR.

3.8.1.4.5 Earthquake Loads

Earthquake loads are in accordance with Section 3.7 of this PSAR.

3.8.1.4.6 External Pressure Load

External pressure loading with a differential of 3 psig from outside to inside is considered.

3.8.1.4.7 Missile and Postulated Pipe Rupture Effects

Missile and postulated pipe rupture loads are in accordance with Sections 3.5 and 3.6 of this PSAR, respectively.

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

d. Areas which do not lend themselves to computer applications. 21

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 12
a summary of analytical methods, see Table 3.8-4.

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-
sections shall not be permitted, however, local yielding 21
will be permitted as long as serviceability and containment integrity are maintained. Liner plate strains shall not exceed the allowables in Table CC-3720-1 for factored load combinations.

3.8.1.6 Structural Acceptance Criteria

The fundamental acceptance criterion for the completed containment is the successful completion of the structural integrity test with measured responses within the limits predicted by analyses. The predicted limits are determined on the basis of test load combinations and code values for stress, strain, or gross deformation for the range of material properties and the specified construction tolerances. In this way the margins of safety associated with the design and construction of the containment are, as a minimum, the accepted margins associated with nationally recognized codes of practice.

The structural integrity test is planned to provide information on both the overall response of the containment and the response of localized areas, such as major penetrations which are important to its design functions. This information makes possible the assessment of the margins of safety available locally.

The design and analysis methods, as well as the type of construction and construction materials, are chosen to allow assessment of the structure's capability throughout its service life. Additionally, surveillance testing as described in Subsection 3.8.1.8 provides further assurances of the structure's continuing ability to meet its design functions.

3.8.1.7 Materials, Quality Control, and Special Construction Techniques

Materials used in construction of the containment and internal structures are as described in this section. The containment and internal structures are constructed of concrete and steel using proven methods common to heavy industrial construction. 12

3.8.1.7.1 Concrete

a. Concrete with the following design compressive strengths (f'_c) is used in the containment structure:

(1) Containment wall and dome........... f'_c = 4000 psi 12

(2) Containment base slab and weir wall. f'_c = 5000 psi 12

(3) Containment internal structures
except weir wall.................... f'_c = 4000 psi 12

All structural concrete is batched and delivered in accordance with ASTM C-94.

b. Cement

Cement is Type II having moderate heat of hydration. Appropriate tests are repeated periodically during construction to check storage environmental effects on cement characteristics. The tests supplement visual inspection and material storage procedures.

c. Aggregates

All aggregates conform to ASTM C 33-74a, "Standard Specifications for Concrete Aggregate." 12

d. Drywell Closure Head

The steel drywell closure head is bolted to the roof slab of the drywell structure and is removable. The head forms part of the drywell pressure boundary and also supports the water in the upper pool. The head consists of a right vertical cylinder with a 2:1 ellipsoidal head and a flange which connects to a mating flange anchored to the drywell roof slab. The head will be leaktight and provision is made to permit pressurized leak testing of the gaskets. Lifting lugs are provided to allow the head to be removed during refueling operations. 130.6

The drywell closure head is designed for uniform subcompartment pressure and for local jet impingement resulting from a postulated rupture of the RCIC head spray line and a main steam line. Design of the drywell closure head for jet impingement follows the analytical techniques given in BN-TOP-2, Revision 2, "Design for Pipe Breaks" and the stress limits of ASME, Section III, Division 1, Subsection NE-3000. 130.6

The bellows and seal plate are permitted to yield when subjected to jet impingement forces from the spray line or the main steam line. However, the reactions on the drywell closure head due to the jet loads are considered when verifying the pressure retaining integrity of the closure head.

3.8.2.2 Description of Class 2 Components

Cold piping penetration assemblies which penetrate the containment consist of a penetration nozzle, and a process pipe. Cold piping penetration assemblies which penetrate the drywell will consist of an embedded pipe sleeve or an embedded sleeve, a process pipe and a head fitting. The cold penetrations have the process pipe welded directly to the penetration nozzle. The piping configuration and support on either side of the penetration are designed to preclude overstressing the containment nozzle under any condition, including postulated accidents. For details see Figure 3.8-4. 12

3.8.2.3 Applicable Codes, Standards and Specifications

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Nuclear Power Plant Components", Section III, Division 1 (1974 Edition and applicable Addenda) Subsections NA and NE for class MC components and NA and NC for class 2 components. 130.6 12

Regulatory Guide 1.57 - Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components. 130.6

3.8.2.4 Loads and Loading Combinations 12

The locks, hatches, closure head and penetrations in the containment are designed in accordance with the loads given in Subsection 3.8.1.4 and for the loading combinations shown in Table 3.8-1, except that the load factor for all loads will be 1.0. 130.6

The locks, hatches, closure head and penetrations in the drywell are designed in accordance with the loads given in Subsection 3.8.3.3 and for the loading combinations shown in Table 3.8-1, except that the load factor for all loads will be 1.0.

3.8.2.5 Design and Analysis Procedures 12

Components are designed according to the documents listed in Subsections 3.8.2.2 and 3.8.2.3 and BC-TOP-1, Rev. 1, Dec. 130.6
1972, Part II. 12

3.8.2.6 Structural Acceptance Criteria 12

Structural acceptance will be in accordance with Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components and BC-TOP-1, Rev. 1, Dec. 1972, Part II. Stresses are limited to those 130.6
permitted by Figures NE-3221-1 and NE-3222-1 of ASME Code, Section III, Div. 1 for Class MC components and Sub-article 12
NC-3112.4 for Class 2 components. For combinations of loads caused by a degraded core accident (Abnormal Load Combination No.3 of Table 3.8-1), Class MC and Class 2 components shall be 21
designed to Level C Service limits, per Subparagraph NCA-2142.2. Deformations are limited so that leak tightness of gaskets and seals will be maintained for all design conditions.

3.8.2.7 Materials, Quality Control and Special Construction Techniques 12

Materials for Class MC components will conform to ASME Code 12
Section III, Div. 1 Article NE-2000 (see Subsection 3.8.1.7.4) and a quality control program conforming to NE-4000 and NE-5000 130.6
of the same code will be formulated. The equivalent Articles relating to Class 2 components are NC-2000, NC-4000 and NC- 12
5000. There will be no special construction techniques.

TABLE 3.8-1

LOAD COMBINATIONS AND LOAD FACTORS

CATEGORY		D	L(1)	H_q	P_t	P_a	T_t	T_o	T_a	E_o	E_{ss}	W	W_t	R_o	R_a	R_r	P_v	P_{mw}	T_{mw}
Service:																			
Test	1.	1.0	1.0	---	1.0	---	1.0	---	---	---	---	---	---	---	---	---	---	---	---
Construction	1.	1.0	1.0	---	---	---	---	1.0	---	---	---	---	---	---	---	---	---	---	---
Normal	1.	1.0	1.0	---	---	---	---	1.0	---	---	---	---	---	1.0	---	---	1.0	---	---
Severe	1.	1.0	1.0	---	---	---	---	1.0	---	1.0	---	---	---	1.0	---	---	1.0	---	---
Environmental	2.	1.0	1.0	---	---	---	---	1.0	---	---	---	1.0	---	1.0	---	---	1.0	---	---
Factored:																			
Severe	1.	1.0	1.3	---	---	---	---	1.0	---	1.5	---	---	---	1.0	---	---	1.0	---	---
Environmental	2.	1.0	1.3	---	---	---	---	1.0	---	---	---	1.5	---	1.0	---	---	1.0	---	---
Extreme	1.	1.0	1.0	---	---	---	---	1.0	---	---	1.0	---	---	1.0	---	---	1.0	---	---
Environmental	2.	1.0	1.0	---	---	---	---	1.0	---	---	---	---	1.0	1.0	---	---	1.0	---	---
Abnormal	1.	1.0	1.0***	---	---	1.5	---	---	1.0	---	---	---	---	---	1.0	---	---	---	---
	2.	1.0	1.0	---	---	1.0	---	---	1.0	---	---	---	---	---	1.25	---	---	---	---
	3.	1.0	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---	1.0	1.0
Abnormal/	1.	1.0	1.0	---	---	1.25	--	---	1.0	1.25	---	---	---	---	1.0	1.0	---	---	---
Severe	2.	1.0	1.0	---	---	1.25	---	---	1.0	---	---	1.25	---	---	1.0	1.0	---	---	---
Environmental	3.	1.0	1.0	1.0	---	---	---	1.0	---	1.0	or	1.0	---	---	---	---	---	---	---
Abnormal/ Extreme Environmental	1.	1.0	1.0	---	---	1.0	---	---	1.0	---	1.0	---	---	---	1.0	1.0	---	---	---

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[1]Includes all temporary construction loading during and after construction of containment.

* Concrete tangential shear not to exceed 40 psi for the containment structure described in Subsection 3.8.1
** Concrete tangential shear not to exceed 60 psi for the containment structure described in Subsection 3.8.1
*** For Main Steam Safety Relief Valve Loads, a load factor of 1.25 shall be used. For all other live loads, a load factor of unity shall be used.

TABLE 3.8-2

COMPUTER PROGRAMS FOR USE ON SEISMIC CATEGORY I STRUCTURES OTHER THAN CONTAINMENT

SAR IDENT NO.	CODE NO.	NAME	DOCUMENTATION TRACEABILITY	REMARKS
1	None	Classical Methods	a. Roark, Formulas for Stress and Strain, McGraw-Hill b. M. Hentenyi, "Beams on Elastic Foundation, The Univ. of Michigan Press, 1946. c. ACI-Standard 318-71 d. AISC-Steel Const:-ction :anual, 1970.	The classical methods are for use ir. analyses of beams, pla:ed, frame and shells. They are given in the standard text book and reference handbooks as use universities and engineer:ng pi.ctice.
2	CE309	Structural Engineering System Solver	Pacific Internatior al Computer Corporation (PICC)	A method formulated for digital computer solution and based on a computer program widely known as SMISS.
3	CE548	Symbolic Matric Interpretive System	PICC	A problem solving method formulated digital computer solution and based on program widely known as SMISS.
4	CE779	Structural Analysis Program	PICC	A method formulated for digital computer solution and based on a program commonly called SAP as developed at the University of California, Berkeley.
5	CE901	Structural Design Language	PICC	A method formulated for digital computer solution and based on a program commonly called ICES-STRUDL
6	ME620	Heat Conduction	PICC	A heat transfer analysis method formulated for digital computer solution using finite elements.
7	None	MRI/STARDYNS	Control Data Corporation	A multipurpose method formulated for digital computer solution.
8	None	Marc CDC	Control Data Corporation	Formulated for digital computer solution.
9	None	Ease	Control Data Corporation	Formulated for digital computer solution.
10	CE668	Plate Beinding 3 Deg.	PICC	A linear elastic analysis of plates on elastic subgrade formulated for digital computer solution using finite elements.

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CHAPTER 13.0

CONDUCT OF OPERATIONS

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CHAPTER 13.0

CONDUCT OF OPERATIONS

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CHAPTER 13.0

CONDUCT OF OPERATIONS

FIGURES

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

The Skagit/Hanford Nuclear Project will be jointly owned by Puget Sound Power & Light Company (Puget), Pacific Power & Light Company, Portland General Electric Company and The Washington Water Power Company. Puget, as the Project sponsor, retains overall responsibility for the design, construction and operation of the Project.

For the Skagit/Hanford Nuclear Project, the Northwest Energy Services Company (NESCO) has the responsibility to perform Project management services and Bechtel Power Corporation (Bechtel) has been selected to perform the architect/engineer, procurement and construction management services. General Electric Company (GE) will design, fabricate and deliver the nuclear steam supply system (NSSS) and nuclear fuel sufficient for the initial core loading. Westinghouse Corporation will design, fabricate and deliver the turbine generator and its auxiliaries. Puget will retain full responsibility for the functional performance of the design of all systems in the Plant. Puget plans to perform startup and testing with Puget employees with technical assistance from GE, Bechtel and NESCO. 21

13.1.1 Management and Technical Support Organization

13.1.1.1 Design and Operating Responsibilities

The following paragraphs describe the corporate functions and their specific responsibilities for the activities.

13.1.1.1.1 Design and Construction Activities (Project Phase)

13.1.1.1.1.1 Principal Site-related engineering work. NESCO's S/HNP Project Manager has overall responsibility for Site-related engineering work. To provide technical expertise in specific areas, the following consultants have been retained:

Meteorology	Battelle Pacific Northwest Laboratories NUS Corporation
Geology	Golder Associates
Seismology	Golder Associates
Hydrology	Battelle Pacific Northwest Laboratories NUS Corporation
Demography	NUS Corporation
Environmental Effects	Battelle Pacific Northwest Laboratories NUS Corporation

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13.1.1.1.1.2 Design of Plant and auxiliary systems. Bechtel has been selected to perform as the architect/engineer. GE will design the NSSS. Westinghouse will design the turbine generator and its auxiliaries.

All correspondence between Puget, NESCO, Bechtel and GE which affect the design interfaces is received and processed by each of the parties according to internal procedures. Periodic reviews between NESCO, Bechtel and GE are held to provide assurance that design interface requirements are met. Puget participates in these reviews.

13.1.1.1.1.3 Review and approval of Plant design features. Puget, as the Applicant, is responsible for the overall design and engineering of the S/HNP and has assigned the responsibility for providing Project management services to NESCO. Although Bechtel, the architect/engineer, and GE, the NSSS supplier, perform the design, engineering and design verification of the facility, NESCO accomplishes its responsibilities by reviewing key elements of the design. Puget oversees the design effort and reviews selected areas of design (e.g. affecting operational aspects). 21 412.1

The basic design criteria are developed by Bechtel under the overview of NESCO. In developing these criteria, Bechtel uses information from the following sources:

a. Basic design and interface criteria supplied by GE as part of the NSSS.

b. Previous design and construction experience of Bechtel.

c. Previous operations and maintenance practices of Puget.

d. Applicable codes, standards and regulations.

e. Industry experience and practice.

Bechtel uses this information to develop a System Description (SD) for each major system and preliminary drawings or sketches to define the preliminary design. 21
NESCO reviews and approves design criteria and preliminary design documents such as P&IDs, general arrangement drawings, and electrical single line diagrams as prescribed by the Puget QA Manual and Project procedures. These documents form the basis for design drawings and procurement specifications. 412.1

The Quality Assurance Program for S/HNP, to assure compliance with 10 CFR 50, Appendix B, is discussed in Chapter 17.

<u>13.1.1.1.1.4 Site layout with regard to environmental effects and security provisions</u>. The Site layout for the S/HNP was developed by Bechtel under the direction of NESCO and Puget. Site layout considerations included environmental effects and security provisions. 21

<u>13.1.1.1.1.5 Development of safety analysis reports</u>. Overall responsibility for preparation of the Preliminary Safety Analysis Report (PSAR) is assigned to NESCO Nuclear Licensing and Safety. All sections of the PSAR are reviewed by Puget's Licensing and Environmental Compliance Department. Preparation of individual sections was assigned to: cognizant technical groups within Bechtel and GE for the BOP and NSSS systems respectively; outside consultants as described in PSAR Section 13.1.1.1.1.1; and appropriate departments within Puget.

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<u>13.1.1.1.1.6 Review and approval of material and component specifications</u>. Safety-related Project specifications are reviewed in accordance with the Quality Assurance Program for the S/HNP as discussed in Chapter 17. 412.1

During the design and construction phases of the S/HNP, several organizations participate in the procurement of material and components. However, NESCO does not relinquish the overall responsibility for assuring adequacy for the procurement program. The NESCO S/HNP Project Manager is responsible for review and approval of the procurement

schedule and the control of procurement documents. GE is responsible for the procurement of equipment, material, and services related to the NSSS scope of supply. Bechtel's scope of responsibility relates to the equipment, material, and services for the balance of plant (BOP) systems not included in the GE scope of supply.

For procurement purposes the system requirements related to each component specification are developed in technical specifications as noted in Section 13.1.1.1.1.3. Requirements related to the particular piece of equipment are contained in the PSAR, industry codes and standards, and regulatory guides. Other requirements are based on Bechtel's design practices, NESCO experience, and industry experience.

NESCO reviews designated specifications prepared by Bechtel in accordance with internal procedures. The personnel required to review such specifications are also described in procedures. NESCO conducts audits of specification review activities within GE, Bechtel, and NESCO. The Quality Assurance Program in relation to this activity is discussed in Chapter 17.

<u>13.1.1.1.1.7 Procurement of materials and equipment</u>. 21 412.1
Engineered equipment inquiries, including standard commercial terms and conditions, schedule information and the approved specifications are assembled by Bechtel and distributed to NESCO approved bidders. Bidders for safety-related equipment are expected to have an acceptable quality assurance plan and applicable stamps and licenses. Upon receipt of bids, technical adequacy and specification compliance are reviewed by engineering; quality assurance plans and testing methods are reviewed by quality assurance personnel; and pricing and terms are evaluated by purchasing. Bechtel provides an evaluation and recommendation as to a vendor after resolving with each bidder, to the extent possible, unacceptable or unfavorable bid exceptions. NESCO reviews the recommendation, making the ultimate selection with the approval of Puget. Bechtel then prepares and issues the purchase order.

Safety-related materials procured in the field by the Bechtel Site organizations must also be reviewed by Bechtel quality assurance personnel.

<u>13.1.1.1.1.8 Management and review of construction activities</u>. NESCO will oversee the management and review of construction activities and will review selected design changes.

NESCO construction management will perform the following management and control activities at the construction site:

a. Construction review and approval of construction-related documents prior to implementation.

   (1) Field requests for design changes are reviewed for potential cost and schedule impacts and for assurance of proper engineering review.

   (2) Construction-originated drawings are reviewed for cost impact, economical design, and conformance with the overall construction plan.

   (3) Recommendations for field purchases are reviewed to ensure conformance with engineering specifications and drawings and general Project procurement guidelines.

   (4. Construction procedures are reviewed to ensure conformance with engineering specifications and NESCO standard policies. Additionally, the procedure is analyzed to ensure that the most efficient and economical method of performing the work has been selected.

b. NESCO construction engineers monitor field activities. This includes monitoring of ongoing construction activities for conformance to specifications, drawings, and procedures. Construction plays an active part in resolution of problems identified during field monitoring. Review of the security, safety, and environmental programs for compliance with established Project construction guidelines is also included in the general monitoring program. 21 412.1

c. The contractor's cost and schedule performance is monitored to keep construction management and Project management informed of Project status. The construction engineers evaluate ongoing construction activities for cost, schedule, and quality problems and actively participate in identifying and selecting resolutions to eliminate or reduce these problems.

13.1.1.1.2 Preoperational Activities

<u>13.1.1.1.2.1 Human engineering design objectives and control room layout</u>. The S/HNP design will incorporate

GE's Nuclenet Control Room which represents the state of the art in human engineering. Specific panel layouts for balance of Plant equipment will be developed by Bechtel and integrated into the Nuclenet design. To verify that human engineering design objectives are met, Puget reviews the layouts on a full scale mock-up of the front row control room panels and conducts panel layout review meetings with the design organizations. The control room review process is described in more detail in Appendix 1B in the response to Item I.D.1 of NUREG-0718. 21 412.6

13.1.1.1.2.2 Staff recruiting and training programs. The Plant Superintendent will be responsible for the development and implementation of staff recruiting and training programs. As key personnel are appointed to Plant operating staff positions, they will provide input to the development of training programs. 21

13.1.1.1.2.3 Plans for initial testing. An integrated startup group will be formed under Puget's direction which will include representatives from NESCO, Bechtel, GE and Westinghouse. Preliminary planning for startup activities will be performed by NESCO with assistance from Bechtel as necessary during the design and construction phases. 21 412.6

13.1.1.1.2.4 Plant maintenance programs. The Plant Superintendent will be responsible for the development of Plant maintenance programs including preventive maintenance. As key personnel are appointed to positions in the maintenance department, they will provide input to the development of these programs.

13.1.1.1.3 Technical Support for Operations.

Detailed plans for providing technical support, both on-Site and off-Site, during operation will be presented in the FSAR. 21

13.1.1.2 Organizational Arrangement

The organizational responsibilities for S/HNP activities are described below. Figure 13.1-1 shows the overall S/HNP organization and indicates the interface relationship between Puget, NESCO, Bechtel, GE and Westinghouse.

13.1.1.2.1 Puget Sound Power and Light

The Puget corporate organization is shown in Figure 13.1-2. The Vice President, Generation Resources, reports to the Senior Vice President, Operations, who reports to the President and Chief Executive Officer. The Vice President, Generation Resources, assigns responsibilities to various Puget organizations described below.

13.1.1.2.1.1 Quality Assurance. The Manager, Quality Assurance, is responsible for providing the programmatic direction, and administering policies, goals, objectives and methods which are described in the Puget QA Program. The Puget QA Program describes specific controls to be established by Puget, NESCO and the major contractors on the S/HNP. The Manager, Quality Assurance, reports on all technical and administrative matters directly to the Vice President, Generation Resources. This organizational arrangement provides independence from cost and scheduling influences. Information on the responsibilities of the Manager, Quality Assurance, is provided in Section 17.1.1.1.1.

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13.1.1.2.1.2 Licensing and Environmental Compliance. The Director, Licensing and Environmental Compliance, reports to the Vice President, Generation Resources, and is responsible for acquisition of local, State, and Federal permits and approvals; review of PSAR and ER; development and implementation of the environmental program for operation; and monitoring of compliance with licensing commitments.

13.1.1.2.1.3 Generating Plant Engineering. The Director, Generating Plant Engineering, reports to the Vice President, Generation Resources, and is responsible for providing engineering support services in the mechanical, civil/structural, electrical and control disciplines as requested by the Director, Nuclear Projects.

13.1.1.2.1.4 Nuclear Projects. The Director, Nuclear Projects, reports to the Vice President, Generation Resources, and is responsible for coordinating all internal and external matters pertaining to the S/HNP. He has the responsibility and authority to:

a. Establish S/HNP objectives and plan, budget, and schedule requirements.

b. Identify internal organizational services required for the S/HNP and make necessary arrangements for these services with associated departments.

c. Interface directly with the NESCO S/HNP Project Manager, to provide Puget input, review and approval in areas of licensing, engineering, construction, operation and budget control for the S/HNP.

13.1.1.2.1.5 Thermal Fuels. The Thermal Fuels Administrator reports to the Vice President, Generation Resources, and is responsible for direction of activities pertaining to the management of Puget's nuclear fuel requirements and for special studies as assigned. He has the responsibility for establishing fuel specifications and evaluating bids from suppliers of uranium, conversion services, fuel fabrication and fuel reprocessing. He is also responsible for recommending the services of consultants for nuclear fuel services as needed and supervises the activities of these consultants.

13.1.1.2.1.6 Plant Operations. The Plant Superintendent reports to the Vice President, Generation Resources, and is responsible for S/HNP operation and maintenance. Initially, he is responsible for all operations planning activities, including development of the detailed Plant organization, hiring and training the Plant Staff and development of plans to facilitate the transition from the design and construction phase to operations. 21

13.1.1.2.1.7 Micrographics and Records. The Manager, Micrographics and Records, reports to the Manager, Office Services, and is responsible for control and maintenance of all records required to document activities affecting safety-related activities.

13.1.1.2.2 Northwest Energy Services Company

The NESCO organization chart which shows the NESCO S/HNP organizational structure is shown in Figure 13.1-3. Figure 13.1-4 shows Puget's interface relationship with NESCO. The NESCO S/HNP Project Manager under the overview of the NESCO Vice President, Nuclear Projects has responsibility for design, procurement, construction, testing and quality of work. The S/HNP Project Manager assigns

responsibilities to the various NESCO organizations described below.

13.1.1.2.2.1 Quality Assurance. The Director, Quality Assurance, is responsible for providing the programmatic direction, and administering policies, goals, objectives and methods as described in the NESCO QA Program. The NESCO QA Program will interface with the Puget QA Program objectives. This plan will describe the specific QA controls to be established by NESCO and the major contractors on the S/HNP to meet the requirements of the Puget QA Program. The Director, Quality Assurance, reports on all technical and administrative matters directly to the President of NESCO. This organizational arrangement provides independence from cost and scheduling influences. Information on the major responsibilities of the Director, Quality Assurance, is provided in 17.1.1.2.

13.1.1.2.2.2 Project Engineering. The NESCO Principal Engineer is responsible for overseeing, coordinating and administering the NESCO Project Engineering effort. Responsibilities of Project Engineering include management of the engineering interface with Bechtel and major equipment suppliers. 21

13.1.1.2.2.3 Nuclear Licensing and Safety. The Manager, Nuclear Licensing and Safety (NL&S), is responsible for licensing of the S/HNP, for ensuring compliance with all licensing commitments. The Manager, NL&S, is responsible for the review of the Project design, construction procedures and schedules for consistency with commitments, obligations, licenses, permits and authorizations, and for review of unreviewed safety questions. The Manager, NL&S, is also responsible for review and identification of important experiences and feedback of pertinent information to those responsible for designing and constructing the S/HNP.

13.1.1.2.2.4 Site Construction. The Site Construction Manager, is responsible for on-Site construction management oversight and maintaining on-Site liaison between NESCO, Bechtel Construction Management and the other Principal Contractors, GE and Westinghouse.

13.1.1.2.2.5 Material Control. The Manager, Material Control, is responsible for the receipt, storage, protection, maintenance and inventory of S/HNP material and

equipment during the transition period following transfer of title to Puget and shipment to the S/HNP Site.

13.1.1.2.3 Interrelationships with Contractors and Suppliers.

Puget has complete responsibility for S/HNP licensing, design, construction, quality assurance, testing, and operation of the S/HNP. Puget has retained various contractors, suppliers, and consultants to provide equipment and services for the S/HNP as described in this section. The principal interface with these parties is through Puget's Director, Nuclear Projects, although the administration of specific contracts may be assigned to other organizations.

NESCO has been designated by Puget to perform certain Project management services for the S/HNP including:

a. Licensing management services including:

   (1) Preparing licensing documents

   (2) Appearing and acting on behalf of Puget in connection with licensing proceedings. 21

   (3) Monitoring design, construction and operational commitments for compliance with licensing commitments, including the establishment of safety-related criteria required to be incorporated into the design to ensure licensability; and reviewing design documents for conformance to licensing commitments.

b. Cost and scheduling services:

   (1) Developing, reviewing, and updating S/HNP cost estimates, budget and cash flows.

   (2) Developing, reviewing and monitoring payment estimates, cash requirements, engineering progress, construction progress, labor productivity and other major S/HNP activities.

c. Engineering management services including directing the architect/engineer in the determination of S/HNP siting and design criteria; directing investigations and studies by the architect/engineer for the formulation of S/HNP design and specifications; reviewing detailed design, plans, specifications and drawings prepared by the architect/engineer.

d. Procurement overview services including monitoring of A/E preparation of bidders lists for procurement of facilities, equipment and services; preparation of bid award recommendations; awarding of purchase orders and contracts and performance of inspections and tests for purchased facilities, equipment and materials.

e. Contract administration services including administering, as Puget's agent, specific contracts for materials, equipment and services.

f. Quality assurance services including establishing and administering a QA Program which meets the requirements of Puget's QA Program; auditing the architect/engineer's and contractors' procedures and methods for conformance to S/HNP design, plans, drawings and specifications; reviewing contractors' QA programs for conformance to applicable laws, ordinances, rules and regulations.

g. Construction management services:

(1) Monitoring, evaluating, and recommending changes to the Construction Program for the S/HNP to ensure compliance with S/HNP schedules, performance and budget requirements; providing Puget with the information necessary to ensure complete Construction Program overview. 21

(2) Monitoring and evaluating S/HNP engineering and procurement activities as they relate to construction to help ensure compliance with S/HNP schedules, performance and budget requirements.

(3) Providing interface between Puget and the architect/engineer to ensure compliance with Puget requirements during all phases of construction.

h. Preoperational testing services including: preparing deficiency lists; assisting Puget in preparation of preoperational testing procedures and startup testing procedures; and assisting Puget in conducting preoperational, startup and performance testing.

<u>13.1.1.2.3.1 General Electric</u>. GE has been awarded the contracts to design, fabricate, and deliver the direct-cycle boiling water nuclear steam supply sytem, and to provide technical direction for installation and startup of this equipment. GE has engaged in the development, design,

construction and operation of boiling water reactors since 1955. Thus GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors.

Through personnel located at its headquarters in San Jose, California, GE will provide the necessary management organization and on-Site representatives to support the S/HNP. The GE Project personnel will control all efforts within its scope of responsibility including those of its suppliers. Interface matters and schedules will be closely coordinated with the S/HNP organizations of NESCO and Bechtel. Figure 13.1-4a shows the GE organization and interface with NESCO, respectively.

13.1.1.2.3.2 Bechtel Power Corporation. Bechtel Power Corporation, has been awarded contracts for engineering, procurement, and construction management of the S/HNP. Bechtel has been engaged in construction and engineering since 1898. For the last 30 years, Bechtel has been active in the fields of pipelines, petroleum, power generation and distribution, harbor development, mining and metallurgy, and chemical and industrial processing. The Bechtel organization has grown progressively to become one of the world's largest engineer-constructors for industrial facilities. Since the close of World War II, Bechtel has designed over 230 thermal power-generating units, representing more than 100,000 megawatts of new generating capacity of which more than 50,000 megawatts are nuclear. Bechtel is qualified to provide the required services for S/HNP design, construction, equipment procurement and startup. 21

Bechtel Power Corporation is responsible for all Bechtel nuclear and fossil power activities and is one of the three principal operating companies in the Bechtel Group. The President of Bechtel Power Corporation, a member of the Office of the President of the Bechtel Group, is executive sponsor of Bechtel Power Management. Bechtel Power Management consists of four operating power divisions; San Francisco, Los Angeles, Ann Arbor and Gaithersburg. Each power division is a separate operating center with complete responsibility and authority for its assigned projects. The manager of each power division reports to the General Manager (executive vice president) of Bechtel Power Management, who in turn reports to the President of Bechtel Power Corporation. The Bechtel Power Management provides communication between the four power divisions. Bechtel Power Management includes staffs, in the functional areas of engineering, construction, business development, quality assurance and services. Each staff is responsible for providing guidance to its power division counterpart in

such matters as interpreting Bechtel policy, establishing standards and procedures, and defining preferred methods for fulfilling their responsibilities. The staffs do not provide direction of day-to-day activities.

The San Francisco Power Division departmental managers report directly to the division manager and his deputy and include the Manager of Project Operations and the managers of the functional departments: Engineering, Construction, Business Development, Quality Assurance, and Services. All San Francisco Power Division departments are physically located in the division office and receive functional support from the corporate service departments (e.g. Labor Relations, Security and Safety, Procurement, Finance and Accounting, etc.)

Bechtel's corporate, divisional, and land-departmental resources are mobilized for the S/HNP in the form of a S/HNP team. Each member of the S/HNP team maintains a dual working relationship in which he receives technical and administrative direction and control from his department manager and overall S/HNP direction from the S/HNP Project Manager. The Project Manager provides the focal point within Bechtel for Project direction as well as the primary contact with NESCO, GE and Westinghouse. The Project Manager reports to the Manager of Projects and then to the Division Manager of Project Operations. A diagram illustrating the organization of the Project teams' basic relationship to Bechtel management and NESCO is presented in Figure 13.1-4b. 21

13.1.1.2.3.3 Westinghouse. Westinghouse has been awarded the contract to design, fabricate, and deliver the turbine-generator as well as to provide technical assistance for installation and startup of this equipment. Westinghouse will work with Bechtel, GE, and NESCO to integrate the design of the turbine-generator into the Project.

13.1.1.2.3.4 Technical Support. Technical support is provided by various independent consultant firms and individuals specializing in various disciplines to address pertinent aspects of the Project. The principal firms retained by Puget to provide these services are discussed in Section 13.1.1.1.1.1. The primary interface between these consultants and Puget is through the NESCO S/HNP Project Manager and the NESCO Manager-Nuclear Licensing and Safety.

13.1.1.3 Qualifications

13.1.1.3.1 Puget Sound Power and Light

Engineering and support services are provided to the S/HNP by the staffs of various Puget departments. Within Puget, the person responsible for the coordination of these services is the Director, Nuclear Projects. The educational and experience qualifications of the Puget personnel providing services to the S/HNP are shown in Table 13.1-1.

Resumes of key Puget personnel involved in the S/HNP are provided below:

Vice President, Generation Resources. Robert V. Myers, graduated from the University of Washington with a B.S. in Mechanical Engineering. Upon graduation he joined GE and spent five years in work associated with the operation of the Production Reactors at Hanford, the government reservation near Richland, Washington. When GE left Hanford as the prime contractor, he stayed on for three more years with the new contractor, Douglas United Nuclear, where he rose to the position of Manager, KW Reactor, responsible for the operation and maintenance of that reactor. 21

In 1968 he rejoined GE as Operations Manager on the startup and operation of the Southwest Experimental Fast Oxide Reactor (SEFOR), a 20 MW_t sodium-cooled breeder demonstration reactor.

In 1973, he joined Puget as Manager, Nuclear Licensing & Safety, responsible for licensing of Skagit. He held the positions of Manager and Director Nuclear Plant Operations Planning and Director Resource Planning. In 1980, he was appointed Vice President, Generation Resources, responsible for construction and operation of Puget's electrical generating resources.

Director, Nuclear Projects. Ronald D. Hill received a B.S. in Mechanical Engineering from California State Polytechnic University in 1955. He joined GE upon graduation and received additional training in the GE engineering program. In 1956, he was assigned to GE's Atomic Power Equipment Department as design engineer for systems and equipment associated with research and test reactors. From 1957 to 1959, he was assigned to GE's Flight Test Department as Project Engineer on the J-79 aircraft gas turbine flight test program for the F-104 aircraft. In 1959, he was

reassigned to GE's Atomic Power Equipment Department and continued with that department until 1973 in various functions including design engineer, technical leader, Project engineer and program manager. During this period, he worked on various GE nuclear projects including Dresden I, Kahl, Japan Power Demonstration Reactor, Tarapur, Tsuruga, Fukushima I, Oyster Creek, Nine Mile Point and Dresden II.

In 1973, Mr. Hill joined Transfer Systems Inc. in New Haven, Connecticut, as Manager, Engineering. During this period, he directed the design and engineering of specialized reactor refueling equipment.

In 1974, Mr. Hill joined Exxon Nuclear Company in Bellevue, Washington, and was assigned as a consultant to Puget's Skagit Project. During this assignment, he held several positions including Senior Nuclear Project Engineer, Manager, Mechanical Engineering, and Principal Engineer.

In 1978, Mr. Hill joined Puget and has functioned as Director, Nuclear Projects, since 1980.

<u>Director, Generating Plant Engineering</u>. L. J. Finnegan, received a B.S. in Mechanical Engineering from Gonzaga University in 1959, and an M.S. in Mechanical Engineering from the University of Notre Dame in 1961. He was employed with the Martin Marietta Company from 1961 to 1962 in the aerospace industry. From 1962 through 1964 he was employed with Westinghouse Electric Corporation at the National Reactor Test Station in Idaho; there he was engaged in testing of Naval reactor systems. From 1964 through 1969 he was employed with Phillips Petroleum Company at the National Reactor Test Station and was involved in the analytical and experimental work associated with the LOFT Program. During this time he coauthored the CONTEMPT Computer Program which is used to predict the reactor building containment response to a loss of cooling accident. He then joined Consumers Power Company in Jackson, Michigan, and was responsible for the supervision of the design review of mechanical aspects of fossil, nuclear and pump storage Projects. He left Consumers Power in April of 1973 to join Puget. Since that time, he has served as Project Manager for the Colstrip Units 1 and 2, which are fossil plants jointly owned with Montana Power, and was promoted to his present capacity in 1975. 21

<u>Director, Licensing & Environmental Compliance</u>. W. J. Finnegan received a B.S. in Civil Engineering from Seattle

University in 1954 and an M.B.A. from Seattle University in 1971.

From 1954 to 1960, Mr. Finnegan was employed by the U.S. Army Corps of Engineers in Seattle as a Hydraulic/Civil Engineer where he engaged in general planning studies relating to the design and operation of multipurpose water resources projects in the Pacific Northwest.

Mr. Finnegan began at Puget in 1960 as a hydrologist, becoming Staff Engineer in Power Resources in 1965, Manager of Environmental Affairs in 1972, Director of Conservation and Environmental Affairs in 1976 and Director of Licensing and Environmental Compliance in August, 1980.

In his career at Puget, Mr. Finnegan has held a broad range of responsibility in directing Company policy with regard to air and water pollution, resource management, energy conservation and land-use planning. He has administered and coordinated Federal Power Commission (now known as the FERC) matters.

In his present capacity, Mr. Finnegan is responsible for liaison with all levels of government in the energy policy, environmental, project licensing and outdoor recreation fields. He is frequently Company spokesman to the media for combustion turbine programs, small hydro development programs, environmental and other matters. He is also a frequent witness in congressional, legislative and administrative proceedings. 21

Manager, Quality Assurance. R. N. Hettinger was employed by GE as a Quality Control Engineer at the Hanford Nuclear Facility in 1957. He assisted in the development and implementation of a QC program for a nuclear reactor pressure monitor system.

In 1964, Mr. Hettinger was promoted to Process Control Engineer and performed process control activities at GE's nuclear fuel manufacturing facility. In 1968 Mr. Hettinger was promoted to Quality Systems Engineer. His duties consisted of developing QC policy and procedures for nuclear fuel manufacturing.

Mr. Hettinger was employed as a Senior Process Control Engineer with United Nuclear Corporation in 1970. His responsibilities included developing and directing process control activities for reactor fuel and assembly manufacturing in the Naval Reactors Program.

Mr. Hettinger was employed as a Quality Assurance Specialist by Puget in 1974. His responsibilities included designing, preparing and implementing a QA program for the Skagit Project. Mr. Hettinger was promoted to his present position as Manager, Quality Assurance, in 1979.

Mr. Hettinger is a senior member and past chairman of the American Society for Quality Control, Richland, Washington Section, is certified as a Quality Engineer by the ASQC and is a Registered Professional Engineer in Quality Engineering.

Thermal Fuels Administrator. Ralph S. Olson has a B.S. in Chemical Engineering and an M.S. in Nuclear Engineering from the University of Washington in Seattle, Washington.

Prior to joining Puget, Mr. Olson worked as a Shift Refueling Engineer (Nuclear) at the Puget Sound Naval Shipyard and as a Project Engineer in the Power Division of Kaiser Engineers, Oakland, California. Mr. Olson joined Puget in 1973 as a Project Engineer, Nuclear Licensing and Safety, and was responsible for coordinaton of production of the Skagit PSAR. In 1976, he became Manager of Nuclear Fuel and Special Studies and was responsible for economic and technical evaluation of nuclear fuel proposals and nuclear fuel design review. In 1979, he was promoted to Thermal 21
Fuels Administrator and acquired responsibility for fossil fuel activities and fuel resource exploration in addition to nuclear fuel activities.

13.1.1.3.2 Northwest Energy Services Company

Engineering and support services are provided by various NESCO organizations. Within NESCO, the person responsible for coordination of these services is the S/HNP Project Manager. The educational and experience qualifications of the NESCO personnel providing services to the S/HNP are listed in Table 13.1-1a.

Resumes of key NESCO personnel involved in the S/HNP are provided below:

Vice President, Nuclear Projects. Gordon K. Jacobsen, received a Bachelor of Science in Electrical Engineering from the University of Washington in 1954. He began at Puget as an engineering aide in 1954 and continued with Puget as Junior and Distribution Engineer, acquiring supervisory responsibilities in 1958. He was subsequently

promoted to Head Distribution Engineer. After acting as Coordinator of Operations and Supervisor of Transmission, he attended Nuclear Schools at GE, Westinghouse, Atomics International, and the University of Michigan. At that point, one half of his time was directed toward advising the President of Puget on all matters dealing with nuclear power. This eventually formalized into the position of Assistant Manager of Nuclear Planning, from there extending into Project Manager for Nuclear Siting and Design. His last position with Puget was Manager of the Skagit Project, encompassing all areas of supervision including licensing, procurement, construction and design.

In 1980 he joined NESCO as Vice President of Nuclear Projects where he is involved in total Project management for the Skagit/Hanford Nuclear Project, supervising all facets of design, construction, licensing and procurement.

Principal Engineer, Project Engineering. John R. Fishbaugher received a B.S. in Chemical Engineering from Iowa State University in 1951. On graduation, he joined GE at the Hanford Atomic Production Works. He was a production reactor supervisor from 1951 to 1958. During this period, he also developed methodology for fuels management and production scheduling at the reactors. 21

From 1958-1962, he was an operations engineer assigned to the Plutonium Recycle Test Reactor (PRTR) prototype nuclear power plant. He provided project oversight and operational preparations during design and construction, and served as an operating supervisor during startup. From 1962-1965, he was Manager of the Plutonium Recycle Critical Facility (PRCF), a zero-power test facility, during its design, construction, startup and operation. (During this time, Battelle Northwest Laboratories became the successor contractor for this work.) From 1965-1968, he was facility manager for the PRTR for Battelle Northwest.

From 1968-1972 he provided technical staff support services including liquid metal test program planning, nuclear licensing, early QA planning, industrial safety and nuclear materials management.

From 1972-1974, he was a reactor inspector for the AEC, serving as principal inspector for the Quad Cities Nuclear Plant.

In 1974, he joined Puget. Since then, he has been responsible for the engineering component of the Skagit Project from its outset to an engineering completon of

approximately 67%. Since 1980, he has continued with NESCO in the same capacity.

Manager, Nuclear Licensing and Safety. James E. Mecca received a B.S. degree in Mechanical Engineering from the Milwaukee School of Engineering in 1959 and an M.S. in Radiological Sciences from the University of Washington in 1962. From 1963 to 1967 he was Radiation Safety Officer with the Milwaukee School of Engineering. In this capacity he was in charge of updating chemistry, physics and engineering curricula. He was also involved in the design of new laboratory facilities and was responsible for all facility licenses and safe usage procedures.

With Douglas United Nuclear, Inc., 1967-1972, he was Senior Engineer in the Nuclear Technology Section where he was directly involved in N-Reactor regulatory work and in developing system test procedures, inspection methods, specifications and criteria of systems. He acted as Principal Investigator on several R & D projects and directed associated work with consultants, technicians and engineers.

With Babcock & Wilcox, 1972-1975, he was Manager of Special Licensing, responsible for coordinating and administrating all Standard Plant Safety Analysis Reports and review of Regulatory Guides. He handled all generic technical issues associated with licensing of Babcock & Wilcox reactors. 21

With Puget 1975-1980, he was Manager, Nuclear Licensing & Safety, responsible for preparation of all licensing documents for the purpose of obtaining Federal, State and local permits and authorizations for the Skagit Project. With NESCO, he continues in the same capacity.

Director, Quality Assurance. E. V. Padgett, Jr., graduated from Oklahoma State University with a B. S. in Chemical Engineering. Upon graduation he joined GE and spent nine years in work associated with process engineering and process control engineering in the manufacture of fuel elements for the production reactors at Hanford. When GE left Hanford as the prime contractor, he stayed on for three more years with the new contractor, Douglas United Nuclear, where he became Manager, Quality Assurance, in the fuels manufacturing section.

In 1968, he joined United Nuclear Corporation, as Manager, Quality Control, in the Naval Products Division at New Haven, Connecticut. His responsibilities included developing and implementing QA/QC program activities for

manufacturing reactors and fuel assemblies for the Naval Reactor Program.

In 1971, Mr. Padgett was promoted to Manager of Operations in United Nuclear Corporation's Chemical Operations plant in Hematite, Missouri. He was responsible for the manufacture of UO_2 powder and pellets for the power reactor industry as well as fuel materials for the Naval Reactor Program. When the plant ownership was changed to Gulf United Nuclear, Mr. Padgett rejoined United Nuclear as Manager of Structural Procurement Control. During this period, he was responsible for design, procurement specifications and supplier quality control in the manufacture of fuel and reactor components. 21

In 1973, Mr. Padgett joined Puget as Manager, Quality Assurance, on the Skagit Project. His responsibilities included the developing and implementing of a Quality Assurance Program for the Project. In 1979, he was promoted to Director of Compensation and Benefits in the Human Resources Department.

Mr. Padgett joined NESCO in his present position as Director, Quality Assurance, in 1981.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Plant Organization

The Skagit Nuclear Power Plant organization (Figure 13.1-5) will consist of approximately 100 full-time employees prior to startup of the first unit. This organization is expected to increase in size to approximately 150 for two-unit operation as shown in Figure 13.1-6.

The Plant Superintendent reports to the Vice President-Power Supply and has the overall responsibility for the safe and reliable operation of the Plant. He is responsible for direct management of the Plant including planning, coordination,

TABLE 13.1-1
PUGET TECHNICAL STAFF
PROJECT PERSONNEL

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
R. V. Myers Vice President, Generation Resources. Responsible for construction, and operation of Puget's thermal power generation program	BSME (1960) U of WA Reactor Operator Certificate-Hanford	ANS	General Electric Company 1960-61 Technical Graduate Program, Richland, Washington planning, engineering, 1961-62 Reactor Specialist (Sr. Reactor Operator) 1962-63 Analyst (Asst. Reactor Mgr.) 1964-65 Supervisor, Coolant and Component Testing (nondestructive testing of fuel elements and reactor hardware) Douglas United Nuclear 1965-68 Reactor Manager, KW Reactor, Richland, Washington General Electric Company 1968-71 Operations Manager, SEFOR Reactor (LMFBR Test Reactor), Fayetteville, Arkansas 1971-73 Project Engineer, LMFBR Proposals, San Jose, California Puget Sound Power & Light Company 1973-75 Manager, Nuclear Licensing & Safety 1975-77 Manager, Nuclear Plant Operations Planning 1977-80 Director, Resources Planning 1980-present Vice President, Generation Resources
R. A. Newk[illegible] Sr. Staff Engineer on loan to INPO	BS (1964) U.S. Naval Academy, US Naval Nuclear Power School Bainbridge, Maryland U.S. Naval Nuclear Prototype Training, Windsor, Conn. (1965) GE Boiling Water Reactor Training Ctr. 12 week operators course, Morris, Illinois (1971)	Sr. Reactor Operator Licenses Dresden Units 2 & 3, Quad Cities 1 & 2	U.S. Navy 1964-70 Qualified on S1C and S5W reactor plans as Engineering Office of the Watch. Reactor Control Div. Officer on USS Plunger (SSN 595). Aux. Div. Officer, EOOW USS Puffer (SSN 652) during initial core loading, startup testing and sea trials. Commonwealth Edison Company 1970-74 Tech. Staff Engineer, Dresden Nuclear Power Station, Startup Engineer, Dresden 3. Technical Staff Supervisor, Quad-Cities Nuclear Power Station Puget Sound Power & Light Company 1974-79 Sr. Nuclear Project Engineer Responsible for coordinating design review activities and assisting the Principal Engineer, Nuclear, in engineering matters related to the Project. 1979-80 Manager, Skagit Operations (Plant Superintendent). Responsible for planning, staffing and operation of the Skagit Nuclear Plant. 1980-present Sr. Staff Engineer Temporary assignment to INPO Evaluation and Assistance Division in Atlanta, Georgia.

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TABLE 13.1-1

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
R. S. Olson Manager, Fuels Resources. Responsible for the direction of all activities pertaining to the management of the Company's thermal fuel requirements, including nuclear fuel.	BS Chem E (1967) MS Nuc E (1970) Nuc E Phd Studies through Qualifying Exams U of WA Radiation Control, Puget Sound Naval Shipyard (1971)	ANS, AICHE	<u>Puget Sound Naval Shipyard</u> 1971 Refueling Engineer, Core Components Section, Nuclear Power Branch. Responsible for maintenance and overhaul of major components of naval reactors including RV Heads and CRDM <u>Kaiser Engineers,</u> Oakland, California 1971-72 Nuclear Engineer, Licensing. Assisted in preparation of a computerized commercial nuclear power plant licensing program <u>Puget Sound Power & Light Company</u> 1973-74 Project Engineer, Licensing & Safety 1974-76 Project Engineer, PSAR 1976-7' Manager, Nuclear Fuels and Special Studies 1979-present Manager, Fuels Resources
L. J. Finnegan Director, Generating Plant Engineering. Responsible for design, development of equipment and system specifications and engineering support for both new and existing power plants	BSME (1959) Gonzaga U. MSME (1961) U of Notre Dame U of Denver Graduate Courses in Nuc E (NRTS Extension) (1961-62) U of Idaho (1965-69) Graduate courses in Chem. E. MIT (1971) Engineering Aspects of Waste Heat Disposal (2 weeks) GE BWR Design Orientation (1975)	ASME	<u>Boeing Company</u>, Seattle, Washington 1959 Engineer. Analytical heat transfer work <u>Martin-Marietta Company</u>, Denver, Colorado 1961-62 Eng. Analytical heat transfer work <u>Westinghouse Electric Corp., NRTS</u> Idaho 1964-67 Staff Engr. Analytical and experimental work in the fields of heat transfer, fluid flow, and thermodynamics as applied to Nuclear Reactor Safety - LOFT Program 1967-69 Group Leader - Responsible for group which acted as consultants to LOFT Program for program planning and AEC-DRL for reactor safety. <u>Consumers Power Company</u>, Jackson, Michigan 1969-71 Sr. Engr.-Design Responsible for review of the mechanical aspects of fossil, nuclear and pumped storage plants. 1971 Staff Engr. Responsible for the supervision of design review of the mechanical aspects of fossil nuclear and pumped storage plants. <u>Puget Sound Power & Light Company</u> 1973-present Project Manager (Colstrip). Responsible for all aspects of Puget's participation in this project. 1974-75 Chief Mechanical Engineer. Responsible for the supervision of design review of the mechanical aspects of fossil and nuclear plants 1975-76 Manager, Generating Plant Engineering 1976-present Director, Generating Plant Engineering (1978-79) Program Director, Regional Siting Program

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TABLE 13.1-1

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
W. D. Porter Manager, Civil/Structural Engineering, Generating Plant Engineering. Responsible for civil/structural aspects related to system specifications and engineering support for thermal power plants	BSCE (1969) MSCE (1972) U of WA	PE Washington Level II Inspector (Nuclear) AWWA ASCE ACI	J. J. Millegan & Associates, Inc. 1972-76 Consulting Engineer Puget Sound Power & Light Company 1976-77 Associate Engineer 1978 Engineer 1979-80 Sr. Engineer 1981-present Manager, Civil/Structural Engineering, Generating Plant Engineering
M. P. Blanchette Associate Engineer, Generating Plant Engineering Responsible for performing engineering design and review of civil/structural work. Monitor construction activity.	BSCE (1978) U of W	Washington State EIT	Puget Sound Power & Light Company 1978-80 Assistant Engineer 1980-present Associate Engineer
B. M. Turner Associate Engineer, Generating Plant Engineering Responsible for reviewing civil technical specifications and drawings.	BSCE (1978) U of Idaho	ASCE	Puget Sound Power & Light Company 1978-80 Assistant Engineer 1980-present Associate Engineer
G. R. Reid Manager, Mechanical Engineering, Generating Plant Engineering. Responsible for mechanical engineering aspects related to system specifications and engineering support for thermal and hydro power plants.	BSME (1973) U of WA U. S. Navy Reactor Operator Training GE BWR Design Orientation (1975)	ASME EEI Prime Movers	U. S. Navy 1966-71 Nuclear Power Plant Operator 1971-72 Engineering Watch Supervisor Puget Sound Power & Light Company 1973-76 Mechanical Engineer, Engineering services 1976-80 Mechanical Engineer, Generating Plant Engineering 1980-present Manager, Mechanical Engineering, Generating Plant Engineering.
A. W. Brocha Associate Engineer, Generating Plant Engineering Reviews mechanical technical specifications and drawings.	BS in NES (1977) U of Florida	ASME	Puget Sound Power & Light Company 1977-80 Assistant Engineer 1981-present Associate Engineer

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
K. R. Olin Mechanical Engineer, Generating Plant Engineering	BSME (1972) Gonzaga U GE BWR Power Plant Training Course (1975) Short Courses in HVAC and Pump Design and Application	ASME PE-ME-WA	Pacific Gas Transmission 1972-73 Drafting and Field Engineering work for Natural Gas Company Puget Sound Power & Light Company 1973-75 Performed design review work on Colstrip Units 1-4 1975 Worked on construction and startup of Colstrip Unit #1 1975-78 Performed design review work on Skagit and Colstrip Projects 1978-80 Combustion Turbine Power Plant Design 1980-present Site and startup engineer for Combustion Turbine Power Plant
M. Powell Associate Engineer, Generating Plant Engineering Design and review mechanical technical specifications and drawings for power plants	BSME (1977) Fresno State University U. S. Navy Nuclear School	ASME	U.S. Navy 1967 Qualified mechanical operator on Naval Nuclear Plant Puget Sound Power & Light Company 1977-present Associate Engineer
S. J. St. Clair Associate Engineer, Generating Plant Engineering	BSME (1978) U of Portland	EIT (1977) NACE	Puget Sound Power & Light Company 1978-79 EIT 1979-81 Assistant Engineer 1981-present Associate Engineer
R. R. Van Valin Senior Engineer, Generating Plant Engineering engineering aspects of thermal Responsible for mechanical generating facilities.	BSME (1955) U of WA BSEE (1955) U of WA MBA (1976) Pacific Lutheran U	PE Washington and Montana	Westinghouse 1955-58 Sales Engineer, Utility Apparatus 1965-67 Sales Engineer, Utility Apparatus Seattle Pacific College 1958-65 Asst. Professor - Engineering Puget Sound Power & Light Company 1967-71 Supervisor, Technical Services Marketing 1971-76 Managerm Technical Services Marketing 1976-present Sr. Engineer Generating Plant Engineering
J. F. Watkins Manager, Electrical Engineering, Generating Plant Engineering Responsible for directing and reviewing Puget Generating Facilities Electrical Design.	BS in EE & Applied Science (1973) Portland State U GE BWR Design Orientation (1975)	IEEE PES	Puget Sound Power & Light Company 1973-present Engineer, Electrical

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TABLE 13.1-1

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
G. Chvoj Engineer, Generating Plant Engineering Instrumentation and Control Design Review	MS in ME and I&C (1968) Technical University Prague, Czechoslovakia GE BWR Design Orientation (1975)		Collins Radio 1969-71 Components Appl. Engineer Iowa Electric 1971-74 Instrumentation and Control Engineer Puget Sound Power & Light Company 1974-present Engineer, Instrumentation and Control
S. L. Muchlinski Associate Engineer, Generating Plant Engineering	BS in EE (1978) WSU BSEE Pullman, WA	IEEE	Bonneville Power Administration 1976-77 Electrical Engineer Technician. Duties included: Computer programming, computer systems, startup and testing and data base management. Puget Sound Power & Light Company 1978-present Duties included: Design/review of all computer systems for the Skagit Project. Current duties include: Generating Station - electrical systems design, construction monitoring and startup.
R. N. Hettinger Quality Assurance Manager. Responsible for surveillance and enforcement of all quality related activities.	GE Sponsored Quality Control Courses (1964-65) Puget sponsored Supervision and Leadership Courses (1979) Lead Auditor Training Course (1980)	Sr. member and past chairman of local section, ASQC ASQC Certified Quality Engr. Registered Professional Quality Engineer	E. I. DuPont Company 1946 Laboratory Assistant, Hanford Project General Electric - Hanford 1946-50 Atomic Products Operation Laboratory Assistant 1950-57 Engineering Technician 1965-68 Quality Control Engineer Douglas Unit Nuclear Corp. - Hanford 1965-68 Quality Control Engineer 1968-70 Quality Systems Engineer United Nuclear Corp. - Naval Products Div. 1970-74 Senior Process Control Engineer Puget Sound Power & Light Company 1974-79 Quality Assurance Specialist 1979-present Quality Assurance Manager

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
R. D. Hill Director, Nuclear Projects Dept. Directs the Company's input, review and approvals of engineering, licensing, operation, budget and schedule for nuclear power projects.	BSME (1955) California State Polytechnic University	California State EIT ASME, ANSI N45 Committee	General Electric Company 1955-56 Test Engineer, Aircraft Gas Turbines, combustion research and test reactors, Schenectady, New York 1957-59 Project Engineer, Flight Test Department, Edwards AFB, California 1959-73 Design Engineer, Tech. Leader, Project Engineer; Program Manager in Advance Engineering, Atomic Power Equipment Dept., San Jose, California Transfer Systems, Inc. 1973-74 Manager, of Product Planning, Manager of Engineering, reactor refueling equipment New Haven, Connecticut Exxon Nuclear Company 1974-78 Assigned to Puget as Sr. Nuclear Project Engineer, Manager, Mechanical Engineering, Principal Engineer, Mechanical. Puget Sound Power & Light Company 1978-present Principal Engineer, Generating Plant Engineering Director, Nuclear Projects Department.
S. W. Martsolf Staff Engineer, Nuclear Projects. Resident representative of the Skagit/Hanford Project Owners at NESCO offices. Responsible for liaison and monitoring of the the licensing and engineering activities.	U. S. Navy Nuclear Power School Vallejo, California (1962) Nuclear Power Prototype Schenectady, NY (1963) BSEE (1972) U of WA	IEEE	U. S. Navy 1960-69 Nuclear Reactor operator and reactor technician USS Sargo SSN 583, USS Triton SSN-586, USS Halibut SSN-587. Two years as Electrical Tech. at Naval Communication Station. Puget Sound Power & Light Company 1973-present Held various positions related to the Skagit Nuclear Project as Asst. Engineer, Associate Engineer, Construction and Staff Engineer.
W. J. Finnegan Director, Licensing and Environmental Compliance. Directs the planning, development and implementation of the Company's licensing, regulation and environmental policies and programs, including environmental research and monitoring, siting, and obtainment of necessary approvals for all generation resources, T & D installations and other Company facilities.	BSCE (1954) Seattle U MBA (1971) Seattle U Westinghouse School of Environmental Management (1971)	ASCE, EEI, USCOLD, WSCC	U. S. Army Corps of Engineering 1954-60 Hydraulic Engineer, Planning Division, Seattle Puget Sound Power & Light Company 1960-65 Hydrologist 1965-72 Staff Engineer, Power Resources 1972-76 Manager, Environmental Affairs 1976-80 Director, Conservation & Env. Affairs 1980-present Director, Licensing & Env. Compliance.

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
M. V. Stimac Manager, Licensing & Regulation. Responsible for the planning, development and implementation of the Company's licensing & regulation policies, programs and procedures, including siting and obtainment of necessary approvals for all generation resources, T & D installations and other Company facilities.	BSEE (1968) U of WA U.S. Naval Nuclear Power School (1969) U.S. Naval Nuclear Prototype Training (1969) HTGR Advanced Technology (1973) BWR6 Design (1975) MS Fisheries (Radiation Ecology) U of WA Complete except thesis	EIT, State of Washington ANS, IEEE Health Physics Society	U. S. Navy 1968-72 Engineering Officer of the Watch. Responsible for supervision, direction and maintenance of Naval nuclear propulsion plan. Qualified on S1W and S5W plants. Supervised operation, maintenance, refueling and startup testing of S5W reactor. Puget Sound Power & Light Company 1972-74 Project Engineer, Environmental Studies, Skagit Nuclear Power Project 1974-76 Project Engineer, NRC Environmental Report, Skagit Nuclear Power Project 1976-78 Senior Nuclear Licensing Engineer 1978-79 Task Force Member, Regional Siting Program 1979-80 Manager, Environmental Regulations 1980-present Manager, Licensing & Regulation
C. T. Van Decar Sr. Environmental Engineer. Responsible for the planning, development and implementation of the Company's licensing and regulation policies, programs and procedures, including siting and obtainment of necessary approvals for thermal generation resources, selected T & D installations, and other Company facilities. Also, administers the Company's special substance control programs.	BSME (1958) U of WA MS (1975) Air Resources	PE - Colorado PNWIS-APCA AWRA, NELPA	The Boeing Company 1958-60 Design & Test Engineer Pacific Car & Foundry 1960-62 Design Engineer The Boeing Company 1962-70 Design & Test Engineer Colorado Department of Health 1971-72 Air Quality Engineer Puget Sound Power & Light Company 1972-80 Environmental Engineer 1980-present Senior Env. Engineer
R. W. Clubb Manager, Environmental Sciences	BS - Zoology & Entomology (1976) U of Utah Ph.D. (1974) U of Utah	American Society of Limnology & Oceanography Phi Kappa Phi Phi Sigma Sigma Xi North American Benthological Society American Association for the Advancement of Science Association of Power Biologists	U of Utah 1967-70 Research Assistant 1974 Post Doctoral Research Camp Dresser & McKey, Inc. 1975-77 Director, Aquatic Sciences Puget Sound Power & Light Company 1977-present Manager, Environmental Sciences
R. S. Barnes Water Quality Specialist	BS - Chem (1970) U of WA MSC Water Resources (Civil Engineering) (1976) U of WA Ph.C Water Resources (Civil Engineering) (1977) U of WA Ph.D. Water Resources U of WA (Civil Engineering)	Sigma Xi, WPCS	U. S. Army 1965-69 Intelligence Analyst University of Washington 1973-78 Research Assistant, Water & Air Resources Division, Research Engineer, Water & Air Resources Division Puget Sound Power & Light Company 1978-present Water Quality Specialist

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
C. L. Feldmann Fisheries Biologist	BS Fisheries & Science (1973) U of WA MS (1974) U of WA	American Fisheries Society World Mariculture Society Assoc. of Power Biologists	University of Washington 1973-74 Research Assistant Quinalt Indian Nation 1974-78 Fisheries Scientist/Aquaculturist Puget Sound Power & Light Company 1978-present Fisheries Biologist
J. F. Thielke Air Quality Specialist/ Meteorologist	BSME (1965) Stevens Institute of Technology MSME (1967) Stanford U Ph.D. (1973) U of WA	APCA PNWIS	U. S. EPA 1967-69 Air Pollution Engineer U.S. Peace Corps 1973-76 Consultant - Brazil University of Washington 1976-78 Research Associate URS Company 1978-79 Consultant Puget Sound Power & Light Company 1979-present Air Quality Specialist/Meteorologist
M. L. Walters Associate Environmental Scientist	BS Biology (1971) Graceland College	National Wildlife Society American Society of Mammalogists Audubon Society Assoc. of Power Biologists	Dames & Moore, Consultants 1972-76 Associate Ecologist Wildlife Biologist 1976-79 Consultant Puget Sound Power & Light Company 1979-80 Field Biologist 1980-present Associate Environmental Scientist
J. W. Richardson Director, Transm. & Distribution Engineering. Responsible for directing engineering activities as they relate to the entire scope of the Company's activities	BSEE (1945) U of WA Gulf General Atomic NP for Utilities GE Course in Nuclear Generation Westinghouse Course in Nuclear Generation GE BWR Design Orientation (1975)	IEEE (Past Section Chairman) PE EE Wash. Electric League Electric Club NELPA, NSPE	Allis-Chalmers, Milwaukee, Wisconsin 1945-47 Puget Sound Power & Light Company 1947-55 General Engineering 1955-64 Division Engineering 1964-67 Division Management 1967-70 Director, General Services 1970-73 Chief Engineer 1973 Director, Engineering Services 1975-76 Director, Engineering 1976-present Director, Transmission & Distribution Engineering
L. E. Karlsen Manager, System Planning and Design. Responsible for expansion planning of the transmission and distribution systems	BSEE (1957) Mich. Tech. WSU Graduate Work (1975-76)	PE EE Wash., IEEE, PES	Commonwealth Associated, Inc. Jackson, MI 1957-61 Substation & Transmission Design Puget Sound Power & Light Company 1961-66 Distribution Engineer 1966-67 Sr. Distribution Engineer 1968-72 Division Engineer 1972-75 Asst. Superintendent 1975-77 Sr. System Planning Engineer 1977-78 Group Supv., System Planning

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TABLE 13.1-1

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
F. L. Wilton Sr. Engineer, System Planning. Assists in system planning and other duties as assigned.	BSEE (1958) U of WA	PE EE	Puget Sound Power & Light Company 1966-67 Engineer-in-Training 1967-68 Assistant Engineer 1968-73 Distribution Engineer 1973-74 Sr. Distribution Engineer 1974-77 Associate Engineer 1977-79 Engineer, System Planning 1979-present, Sr. Engineer, System Planning
A. J. Tomac Group Supervisor, System Planning. Responsible for short and long-range transmission and distribution planning	BA in Math (1964) WA St. College BSEE (1964) Wash St. College GE BWR Design Orientation (1975)	IEEE	General Dynamics 1964-69 Research Engineer Philco-Ford 1969-73 Engineer Specialist Puget Sound Power & Light Company 1973-74 Assistant Engineer, I 1974-76 Assistant Engineer, II 1976-78 Associate Engineer 1978-79 Engineer 1979-80 District Engineer 1980-81 Division Engineer 1981-present Group Supervisor System Planning
M. B. Buchanan Group Supervisor, Analytical Methods. Resp. for powerflow and dynamic stability studies for design of the transmission system. Responsible for short circuit current, subsynchronous resonance and electromagnetic transient studies for specifications on the generator to integrate the plant into the transmission network.	BSEE (1953) U of Illinois MSEE (1954) U of W Additional graduate work in Math and Computer Applications (1966-67)	IEEE	Douglas Aircraft Company, Inc. 1954-55 Associate Engineer, El Segundo, Ca. Central Illinois Electric & Gas Co. 1955-59 Distribution Engineer. Also, substation design transmission line design system analog studies. Eaton Manufacturing Company Kenosha, Wisc. 1959-60 Development Engineering Puget Sound Power & Light Company 1961-present Planning Section
W. G. Torgerson Associate Engineer, Analytical Methods. Responsible for compilation of data bases and actual running of powerflow, short curcuit current and dynamic stability studies.	BSEE (1976) Seattle U	IEEE	Puget Sound Power & Light Company 1976-present Associate Engineer.
J. C. Reese Associate Engineer, Analytical Methods. Responsible for subsynchronous resonance and electromagnetic transients studies for the transmission system and generator design.	BSEE (1973) U of Utah MSEE (1979)	PE Utah IEEE, Nat. Geog. Society	Boeing Company 1973-75 Engineer Utah Power & Light 1976-79 Sr. Engineer Puget Sound Power & Light Company 1979-present Associate Engineer

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
W. S. Lenox Engineer, Analytical Methods Prepares powerflow, short circuit and stability studies.	BSEE (1948) Ill. Inst. of Tech.	PE EE Washington WSPE, NSPE, IEEE	Sola Electric Company 1952-61 Electrical Engineer Universal Oil Products Co. 1961-65 Programmer The Boeing Company 1965-69 Research Engineer Burgman & Sons 1970-71 Estimator-Designer J. L. Duncan Engr. 1971-72 Electrical Engineer Puget Sound Power & Light Company 1972-74 Asst. Engineer, I 1974-78 Asst. Engineer, II 1978-80 Associate Engineer 1980-present Engineer
L. W. Pananen Engineer, Metering. Specifies and recommends new metering equipment design for substations and transmission lines.	BSEE (1964) U of WA	PE EE Washington IEEE	Seattle City Light 1964-66 Engineering Production U. S. Navy 1966-67 Naval Flight School 1967-71 Operational Flying Seattle City Light 1971-73 Engineering Production 1974-78 Project Engineering 1978-79 Production Supervisor Puget Sound Power & Light Company 1979-present Metering Engineer
C. A. Prior System Protection Engineer Responsible for planning protective schemes for Generation, Transmission & Distribution Systems	BASc EE (1948) U of British Columbia GE BWR Design Orientation (1975)	PE EE Washington IEEE, WSPE	Puget Sound Power & Light Company 1955-58 Engineering, Standards 1959-67 Engineer, Analog & Digital Computers 1968-present Engineering, Protection
R. W. Nelson Manager - Standards & Performance Engineering. Responsible for directing programs of personnel in continuous review and updating of T & D design standards, checking performance of materials installed, inspection of purchased materials and equipment, establishing operating criteria, with equipment representatives, design engineers and division superintendents	BSEE (1948) Washington State U GE BWR Design Orientation (1975)	PE Washington & Oregon IEEE, PES, ANSI, C37 Comm. NFPA	California Pacific Utilities Co. 1948-54 District Engineer Puget Sound Power & Light Company 1954-56 Distribution Engineer 1956-58 Distribution Supervisor 1958-63 Asst. Superintendent 1963-68 Land & Right-of-Way Engineer. 1968-70 General Distribution Engineer. 1970-present Manager, Standards & Performance Engineering

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TABLE 13.1-1

Sheet 11 of 14

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
J. Jacobs Group Supervisor, Standards & Development. Supervises Standards & Development Group	BSCE (1953) Univ. of Manitoba	PE CE Washington (1967) IEEE	A. R. Grimwood Ltd. 1956-57 Layout Engineer Puget Sound Power & Light Company 1957-59 Jr. Distribution Engineer 1959-60 Distribution Engineer 1960-66 Sr. Distribution Engineer 1966-67 Associate Engineer 1967-69 Engineer 1969-79 Standards & Development Engineer 1979-present Group Supervisor, Standards & Development
D. A. Hindman Manager, Transmission and Distribution Substation & Control Engineering. Directs the personnel of the T & D Substation and control engineering design, studies and plans for construction, maintenance and operation of Company substations.	BSEE (1961) U of Colorado IEEE Seminar on Reliability in Nuclear Power Generation Stations (1974) GE BWR Design Orientation (1975)	PE EE Washington & Oregon IEEE Power Engr. Society NELPA	California Electric Power Company 1961-62 Engineer Distribution Puget Sound Power & Light Company 1964-present Engineer, Transmission and Distribution
D. R. Schafer Group Supervisor T & D Substation Engineering T & D Engineer	BS EE Gonzaga U GE BWR Design Orientation (1975)	PE Washington IEEE	U. S. Navy 1957-61 In-flight Electronic Technician, maintained all electronic equipment on the Lockheed Radar Super Constellation Puget Sound Power & Light Company 1966-present Engineer, Transmission and Distribution
W. L. Pagel Group Supervisor T & D Substation Controls. Responsible for substation control design.	BSEE (1966) U of WA	IEEE, PE Washington	Puget Sound Power & Light Company 1965-present Engineer, Transmission and Distribution
R. E. Brown Group Supervisor Graphics. conventional drafting, computer graphics & document storage retrieval systems	BSCE (1971) St. Martin's College Ballistic Missile Inertial Guidance & Control Repairman School (1964) USA SF Engineer School (1980)	ASCE	U. S. Army 1964-67 Puget Sound Power & Light Company 1971 Jr. Engineering Aide 1971-72 Associate, Engineering Aide 1972-74 Assistant Engineer I 1974-78 Assistant Engineer II 1978-present Group Supervisor Graphics

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
R. G. Jacobsen Apparatus Engineer. Provides advice to Division, Staff, and Engineering personnel, as a consulting expert in the field of electrical equipment. Research equipment problems and prepare equipment specifications to minimize problems. Assignments can be unique and complex in nature.	BSEE (1947) U of WA GE BWR Design Orientation (1975)	PE Washington IEEE, NELPA	University of Washington, Audio-Visual Dept. 1947-52 Specified equipment and supervised its installation Puget Sound Power & Light Company 1952-54 Jr. Engr. & Jr. Distribution Engineer in Engineering Standards 1954-66 Assistant Engineer & Assoc. Engineer in Substation Design Group 1966-76 Electrical Equipment and Controls Engineer and Substation & Equipment Engineer in charge of Substation Design Group 1976-present Staff Engineer, Apparatus
M. Shahla Group Supervisor, Civil/Structural Design. Responsible for the activities of the activities of the personnel and the work in the structural group as it pertains to the Civil and structural T & D Engineering Dept. and other assigned Company projects.	BSCE U of WA GE BWR Design Orientation (1975)	PE Washington, ASCE	State of Washington, Dept. of Highways 1961-62 Highway design and construction inspection Tudor Engineering Company 1963-66 Civil & Structural design of Bay Area Rapid Transit Adibi, Haris & Associates 1966-68 Highway design Puget Sound Power & Light Company 1968-present Struct. design of substations and transmission structures
J. W. Hipke Manager, Transmission & Distribution Line Engineering. Responsible for the activities of the personnel and work of the Transmission & Distribution section in engineering high voltage and distribution line projects referred to the Department.	BSEE (1958) Wash State U	PE Washington IEEE, NELPA	Puget Sound Power & Light Company 1958-61 Engineer Distribution 1961-62 Assistant Engr., Transm. Line Design 1962-64 Assistant Engr., Distribution Line Design 1964-68 Head Distr. Engr. (No. Div.) coordinated all engineering in division 1968-71 Div. Engineer, Coord. all engineering in No. Central Division 1971-73 Asst. Supt., Coord. construction and engineering in line headquarters 1974-76 Sr. Transmission Engineer 1976-present Manager, Transmission & Distribution Line Engineering

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TABLE 13.1-1

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
L. H. Bollinger Group Supervisor, Transmission Engineering. Supervises and coordinates activities of the transmission group in engineering high voltage line projects referred to Transmission and Distribution Engineering; furnishes assistance, recommendations and directions to division personnel on location, design and construction of high voltage line projects engineered in the divisions; and prepares specifications for special projects.	Studied ME, U of Idaho	IEEE, NELPA	Puget Sound Power & Light Company 1941 Maint. Man Class B, Rock Island Plant 1942-43 Engineer Trainee 1943-45 Associate Draftsman 1945-47 Draftsman 1947-53 Sr. Draftsman 1953-56 Design Draftsman 1956-59 Assistant Engineer 1959-73 Associate Engineer 1973-78 Engineer Transmission 1978-present Group Supervisor, Transmission Engineering
W. J. Kotsogean Distribution Group Supervisor. Supervises the personnel of the Distribution Group in special studies which support distribution field engineering.	BSEE (1969) U of W	IEEE, PE EE Washington	Novar Corp. Californ a 1970-71 Technician Puget Sound Power & Light Company 1973-79 Distribution Engineer 1979-80 Associate Engineer, Distr. Group 1980-present Distr. Group supervisor
D. J. Murphy Manager, Security. Responsible for Asset Protection/Loss Prevention for entire corporation, including Nuclear Power Program. Coordinates the fulfillment of government requirements and regulations during the construction and operational phases of the Nuclear Power Project.	BS in Police Administration (1961) Michigan State University	ASIS - American Society for Industrial Security Certified Protection Professional (CPP conferred by ASIS)	Oakland California Police Department 1961-62 Police officer assigned to patrol division Lockheed Missiles & Space Co., Inc. 1962-71 Security Sprvsr. - Direction of security staff assigned to various classified programs, security on each program. The Boeing Company 1971-75 Security Admin. - Formulation and implementation of a security procedure which encompassed physical and internal security requirements. Puget Sound Power & Light Company 1976-present Security

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TABLE 13.1-1

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
J. K. Simpson Security Supervisor. Responsible for security plans, procedures and operations to fulfill government, corporate and department requirements. Coordinates all corporate efforts to meet security and operational phases of the Nuclear Power Project	AA in Administration of Criminal Justice (1979) Bellevue Community College In pursuit of BA, Law and Legal Systems, City College of Seattle	ASIS - American Society for Industrial Security IABTI - Int'l. Assoc. of Bomb Technicians & Investigators	<u>United States Navy</u> 1951-76 Retired as Lieut. Commander, USN, Aviation Weapons Systems Specialist. Qualified Nuclear Emergency Team Commander. Held numerous assign. in nuclear weapons program. Graduate Nuclear Weapons Training Cntr. San Diego. Resp. for base and nuclear weapon site security, weapons recovery plan, officer-in-criteria for design, construction tactical-command, counter-terrorist plan. and team execution, and Nuclear Weapons Logistics Movement Coordinator. <u>Puget Sound Power & Light Company</u> 1977-present Security

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TABLE 13.1-1a
WESCO TECHNICAL STAFF
PROJECT PERSONNEL

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Warren J. Ferguson President	BSEE (1951) University of Washington	Chairman, Edison Electric Institute Construction Committee Member, Atomic Industrial Forum Member, American Nuclear Society Past Section Chairman and Board Member, IEEE	General Electric Company 1951-53 Engineer on GE's test engineering program 1953-56 Test Engineer, Reactor Plant Start-up, Richland 1956-58 Superintendent Special Reactor Maintenance 1959-63 Manager, B-C Reactor Operation 1964-66 Manager, Manufacturing, Hanford Fuel Fabrication Operations 1966-67 Division Manager, N-Reactor Project Hanford United Nuclear Corporation 1967-73 Vice President and General Manager, Naval Products Division two plants - New Haven and Montville, Connecticut Puget Sound Power & Light Company 1973-74 Director, Major Project Construction 1974-75 Vice President, Construction 1975-80 Vice President, Engineering & Construction
Joseph P. Burn Vice President - Engineering	MSME (1965) Carnegie Institute of Technology BSME (1954) Clarkson College of Technology Welding and nondestructive testing conference (1954) Course in nuclear equipment design and reactor systems engineering (1958-1960) Numerous management development	American Society of Mechanical Engineers	Westinghouse Company 1957-62 Supervisor and Engineer 1961-62 Supervisor, Quality Control 1962 Senior Engineer, Control Rod Drive Mechanisms 1962-69 Supervisor, Main Coolant Pump Engineering 1969-70 Engineering Manager 1970-72 Manager, Plant Mechanical Components, Reactors 1972-74 Manager, Plant Engineering, Hanford Nuclear Projects, Inc. 1974-80 Director of the Lead Architect Engineering

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Gordon W. Jacobsen Vice President - Nuclear	BSEE (1954) University of Washington Extended training programs at General Atomic, Westinghouse and the University of Michigan	American Nuclear Society Atomic Industrial Forum	U.S. Army Air Corps 1946-49 Puget Sound Power & Light Company 1954 Engineering Aide 1955 Junior Engineer 1956 Junior Distribution Engineer 1956-58 Distribution Engineer 1958-62 Distribution Supervisor 1962-63 Head Distribution Engineer 1963-65 Coordinator of Operations 1965-68 Supervisor, Transmission 1968-73 Assistant Manager of Nuclear Planning 1973-75 Project Manager, Nuclear 1975-77 Manager, Skagit Nuclear Project 1977-80 Director, SNP Skagit
William L. Heilman RSP Director	MSCE (1964) University of Wisconsin BSCE (1963) University of Wisconsin Additional courses in Engineering aspects of heat disposal from power generation	International Water Resources Association American Society of Civil Engineers National Society of Professional Engineers Washington Society of Professional Engineers Tau Beta Pi Chi Epsilon Registered Professional Engineer - Michigan and Iowa Authored numerous publications	University of Michigan 1964-67 Research Assistant J. D. Calvert 1964-67 Associate Consulting Engineer Gilbert Commonwealth 1967-72 Associate Engineer, Engineer and Senior Engineer 1972-75 Manager, Water Resources 1975-77 Manager, Environmental Sciences and Civil/Hydraulic Services 1977-79 Manager, Engineering, Environmental Systems Div. Puget Sound Power & Light Company 1979-80 Director, RSP

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TABLE 13.1-1a

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Wesley W. Aardal Senior Engineer Startup - Satsop	Engineering Curriculum Denver University and University of Washington (1947-50) Chief Operator Training Course Westinghouse	Certified as Engineering Officer of the Watch	United States Army Chemical Corps 1952-57 Westinghouse Electric Corporation 1957-62 Operations Engineer Todd Shipyards Corporation 1962-63 Technical Assistant to Manager Westinghouse Electric Corporation 1963-64 Scheduling Engineer 1964-65 Documentation Engineer 1965-70 Systems Engineer 1970-73 Systems Engineer and Chief Test Operator 1973-74 Startup Engineer, Japan, Korea Nuclear Station 1974-77 Chief Site Engineer, Korea Nuclear Station 1976-77 Startup Manager, Korea Nuclear Station Pacific Power & Light Company 1977-78 Site Manager 1977-78 Startup Consultant 1978-80 Senior Construction Engineer

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Thomas A. Caracciolo Manager - Material Control	BA Pre-Law (1952) San Diego State College Graduate Courses in Law University of San Diego (1960-61)	Commercial Pilots License	General Dynamics Corporation 1960-62 Construction Contracts Administrator Rockwell International Corporation 1962-63 Contracts Administrator International Harvester Corporation 1963-64 Contracts Administrator Rockwell International Corporation 1964-70 Senior Contracts and Proposal Specialist Westinghouse Hanford Company 1970-73 Senior Procurement Specialist Burns & Roe, Inc. 1973-75 Manager, Construction Contracts HUILO Incorporated 1975-76 Director, Contracts and Procurement Bechtel Power Corporaton 1976-78 Manager, Field Procurement Puget Sound Power & Light Company 1978-80 Contracts Administrator
John L. Elmer Cost & Schedule Supervisor	BA Fine Arts and Education (1973) Washington State University Selected mathematics and science courses San Francisco State University		Bechtel Power Corporation 1972-74 Assistant Project Scheduler 1974-76 Project Construction Planner 1976-78 Lead Field Scheduler 1978 Lead Estimator 1978-79 Lead Cost Engineer, Fossil Puget Sound Power & Light Company 1979-80 Project Cost & Schedule Engineer
Susan D. Fesser Contract Administration Supervisor - Nuclear	BA Business Administration (1972) University of Washington Graduate level business courses Joint Center for Graduate Studies Seminars in Construction Law, Negotiating and Management		Westinghouse Hanford Company 1973-75 Buyer Teltone Corporation 1975-77 Senior Buyer Puget Sound Power & Light Company 1977-80 Contract Administrator, Skagit Project

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TABLE 13.1-1a

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
John R. Fishbaugher Principal Engineer - Nuclear	BS Chem E (1951) Iowa State University Selected short courses in nuclear technology and management	Member American Nuclear Society EPRI Advisory Committee, Nuclear Power Division	General Electric Company 1951-53 Production Reactor Supervisor 1954 Fuel Management for Production Reactors 1955-58 Production Reactor Supervisor 1958-62 Operations Engineer for Construction/ Startup of Prototype Power 1962-65 Manager of Zero-Power Reactor Test Facility Battelle Northwest and Westinghouse 1965-68 Reactor Plant Manager 1968-72 Technical Staff Support Services Atomic Energy Commission 1972-74 Reactor Inspector Puget Sound Power & Light Company 1974-78 Chief Engineer, Skagit Project 1978-80 Principal Engineer, Nuclear
Vern Grayhek Senior Staff Engineer - Nuclear	BSME (1955) Gonzaga University Architecture (2 years) University of Washington (1951)	Registered Professional Nuclear Engineer - California Puget Section of American Nuclear Society Puget Power Association of Supervisors Northwest Electric Light and Power Association	Westinghouse Atomic Power Division 1955-56 Mechanical Design and Thermal Analysis General Electric, Nuclear Energy Division 1956-62 Design Engineer 1962-66 Resident Engineer 1966-68 Manager, Operating Plant Engineering 1968-69 Project Engineer 1969-71 Operations Manager 1971-72 Warranty Representative 1972-76 Manager, Performance Analysis and Service Communications Puget Sound Power & Light Company 1976-78 Senior Nuclear Project Engineer 1978-80 Senior Staff Engineer Nuclear

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TABLE 13.1-1a

Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Eugene Normand Senior Staff Engineer - Nuclear	PhD Nuc E (1970) University of Washington MS Nuc E (1966) University of Washington BS Chem E (1964) Polytechnic Institute of Brooklyn	American Nuclear Society Authored Numerous Publications	Aerojet Nuclear Systems Company 1969-71 Physics Specialist Sargent and Lundy Engineers 1969-71 Fusion Project Coordinator 1971 Group Leader, Project Shielding 1971-72 Project Shield Engineer 1972 Radiation Analysis 1973 Supervisor, Shielding and Radiological Safety Section Puget Sound Power & Light Company 1979-80 Staff Engineer
Dennis B. Hacking Senior Staff Engineer - Nuclear	Graduate Work in Physics Washington State University (1968-69) BS Physics (1968) University of Utah		Litton Industries 1964-68 Electronic Test Technician and Inertia Instrument Test Technician Douglas United Nuclear 1968-70 Nuclear Engineer Sargent and Lundy Engineers 1972-73 Shielding Project Engineer 1973-76 Mechanical Project Engineer Puget Sound Power & Light Company 1976-77 Senior Mechanical Engineer 1977-78 Senior Engineer Projects 1978-80 Staff Engineer, Nuclear
Gary F. Weidinger Senior Staff Engineer	BS Aerospace and ME (1971) Montana State University	Member, American Society of Mechanical Engineers Registered Professional Mechanical Engineer - California (1974) Washington (1978)	Bechtel Power Corporation 1971-77 Engineer and Supervising Engineer, Nuclear and Coal Fired Generating Stations Puget Sound Power & Light Company 1977-81 Senior Mechanical Engineer, Nuclear and Coal Fired Projects Northwest Energy Services Company 1981-Present Senior Staff Engineer, Skagit Nuclear Project

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
James E. Mecca Manager Nuclear Licensing and Safety	MS Rad. Sci. (1962) University of Michigan BSME (1959) Milwaukee School of Engineering Courses in variety of disciplines including computer programming and business management	Health Physics Society EPRI Task Force on Nuclear Safety and Analysis Past Member of ANSI and ASTM Standards Committees Registered Professional Engineer Washington and Wisconsin Licensed Power Plant Engineer (1st class, no limit) - City of Milwaukee Authored numerous publications	Oregon State Board of Health 1961-63 Radiation Physicist and Head of Radiological Control Unit Milwaukee School of Engineering 1963-67 Radiation Safety Officer Douglas United Nuclear 1967-70 Senior Engineer Babcock & Wilcox Company 1970-75 Manager of Special Licensing Puget Sound Power & Light Company 1975-80 Manager, Nuclear Licensing and Safety
Terence L. Grebel Licensing Engineer	MSEE (1971) University of Southern California BSEE (1968) California State College at Long Beach	Institute of Electrical and Electronic Engineers (IEEE) American Nuclear Society Professional Engineer - Electrical California	Los Angeles Department of Water and Power 1968-79 Supervising Associate Engineer Stafco, Inc. 1979-80 Senior Engineer Northwest Energy Services Company 1981 Licensing Engineer
Matthew Lyon Licensing Engineer	BA Physical Science (1965) San Jose State College	Member Health Physics Society (National and Cascade Chapter) U.S. Coast Guard Auxiliary	General Electric 1961-65 Radiation Control Technician Controls for Radiation 1965-67 Shift Health Physicist, Plum Brook Nuclear Facility Wisconsin Michigan Power Company 1967-71 Plant Health Physicist, Point Beach Nuclear Plant Tennessee Valley Authority 1971-74 Supervisor, Health Physics 1974-76 Supervisor, Radiation Control Section Puget Sound Power & Light Company 1976-79 Health Physicist 1979-80 Licensing Engineer

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Name, Assignment Responsibility	Pertinent Educational Background	Professional Licenses and Organizations	Prior Technical Experience
Benjamin F. Burton Associate Engineer	BSME (1976) University of Washington AA Pre Professional (1974) Green River Community College Selected courses in nuclear power, health physics, radiation protection, reactor operator and system safety and reliability analysis	American Society of Mechanical Engineers Health Physics Society (ICRP) First Class Stationary Engineer Maryland (1971) Reactor Operator (AEC) (1970) Engineer In Training - Fundamentals Washington (1976) Professional Engineer - Mechanical Washington (1979) and Ohio (1980)	United States Navy 1961-68 Leading Engineering Laboratory Technician Baltimore Gas & Electric Company 1968-72 Senior Control Room Operator Puget Sound Power & Light Company 1976-77 Assistant Project Engineer 1977-80 Associate Engineer
Jeanette LaFleur Coordinator - Nuclear Licensing & Safety	General Curriculum Pomoma Junior College (1951)	Energy Advocates	Whidbey Island Naval Air Station 1951-52 Secretary Puget Sound Power & Light Company 1960-65 Senior Stenographer 1965-68 Executive Secretary 1968-72 Secretary to Director of General Services 1972-77 Secretary to Director/Manager of Major Project Construction 1977-80 Coordinator, Nuclear Licensing & Safety

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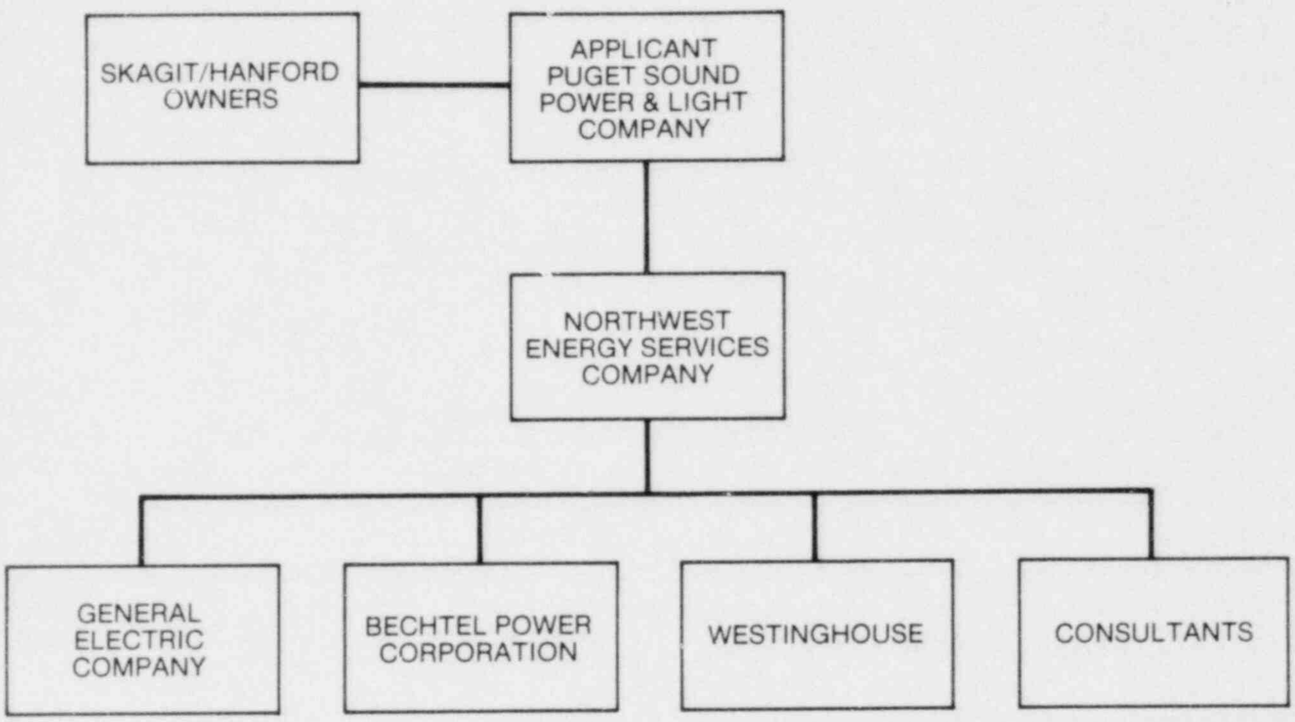
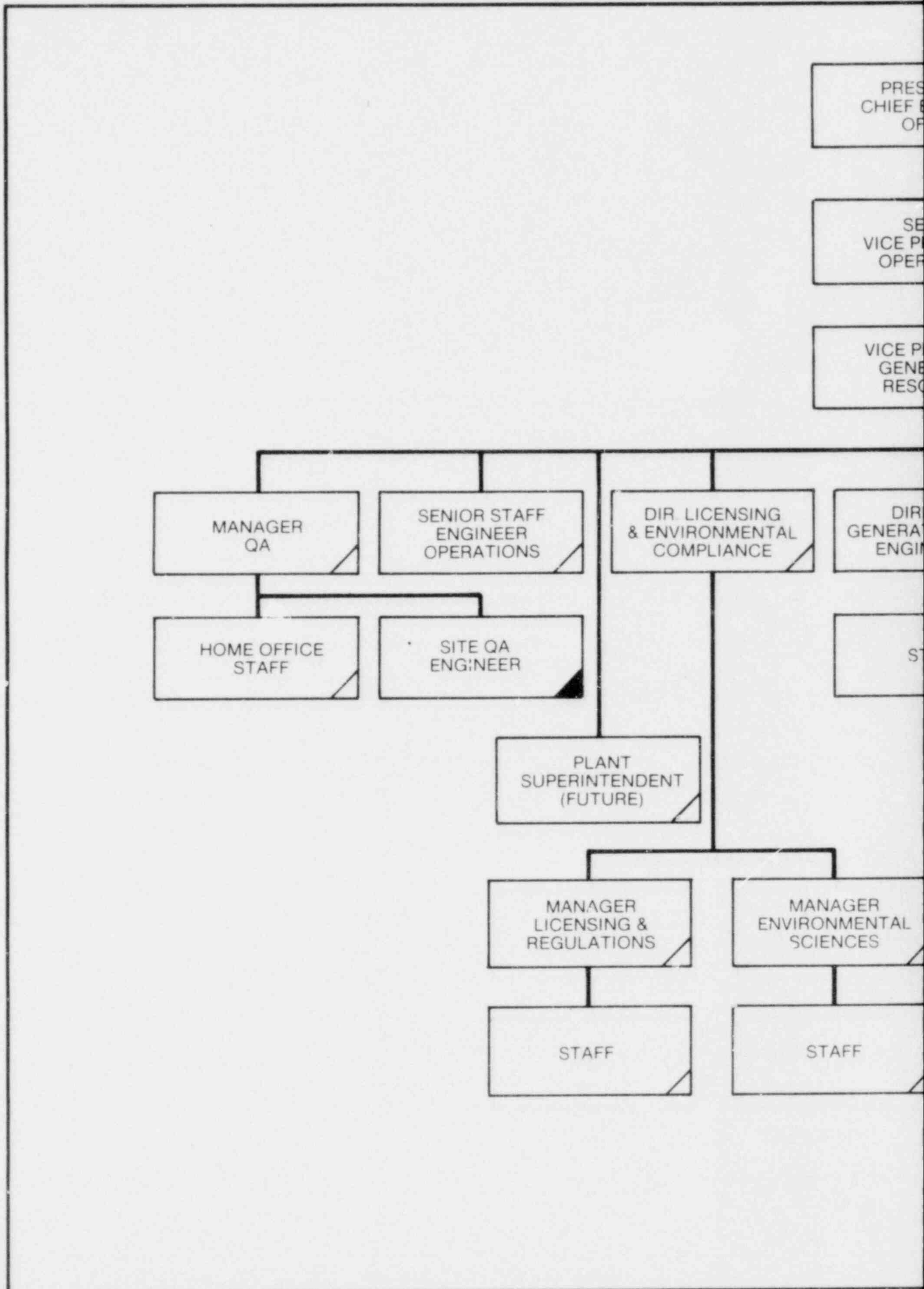


PUGET SOUND POWEP & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

S/HNP
PROJECT ORGANIZATION

FIGURE 13.1-1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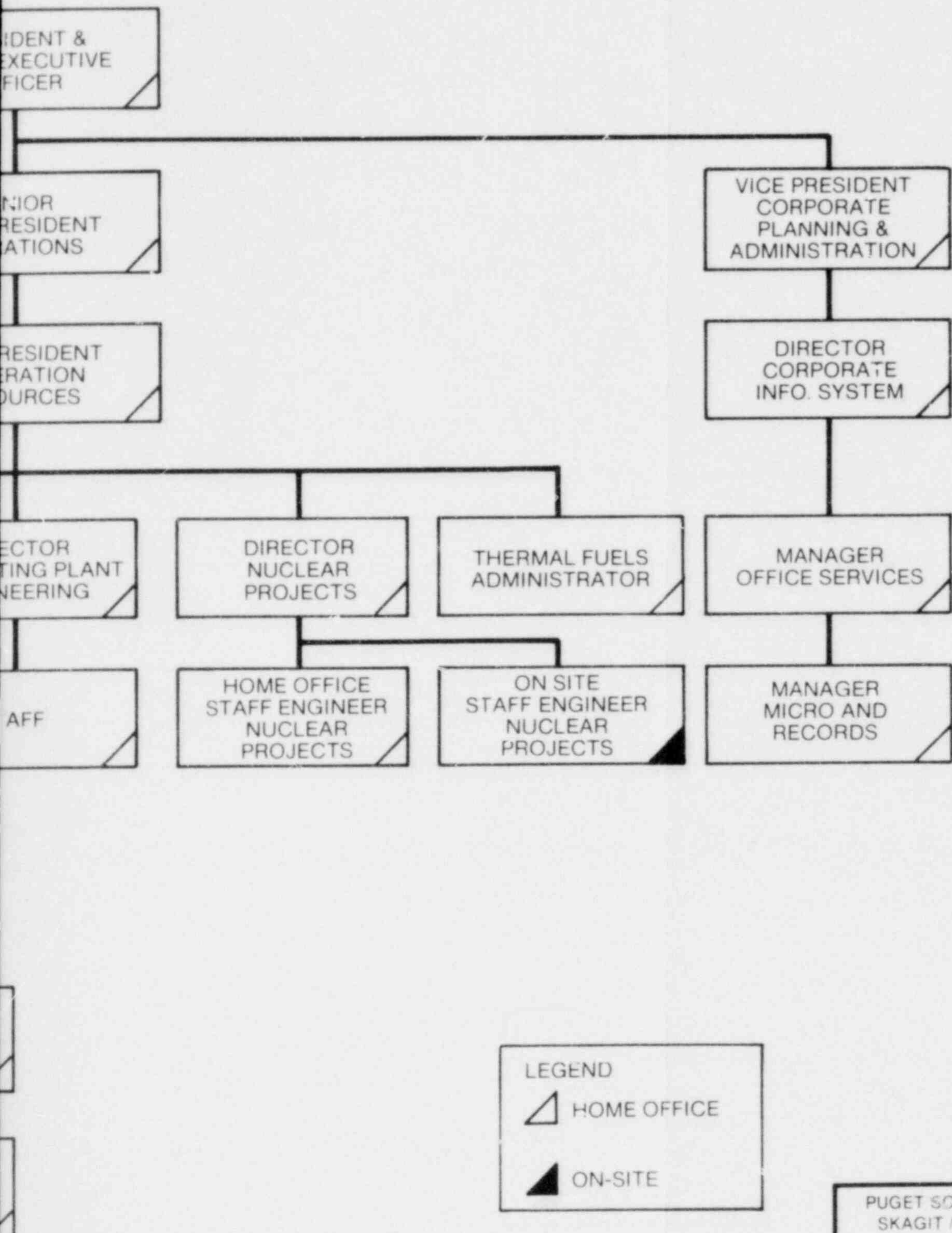
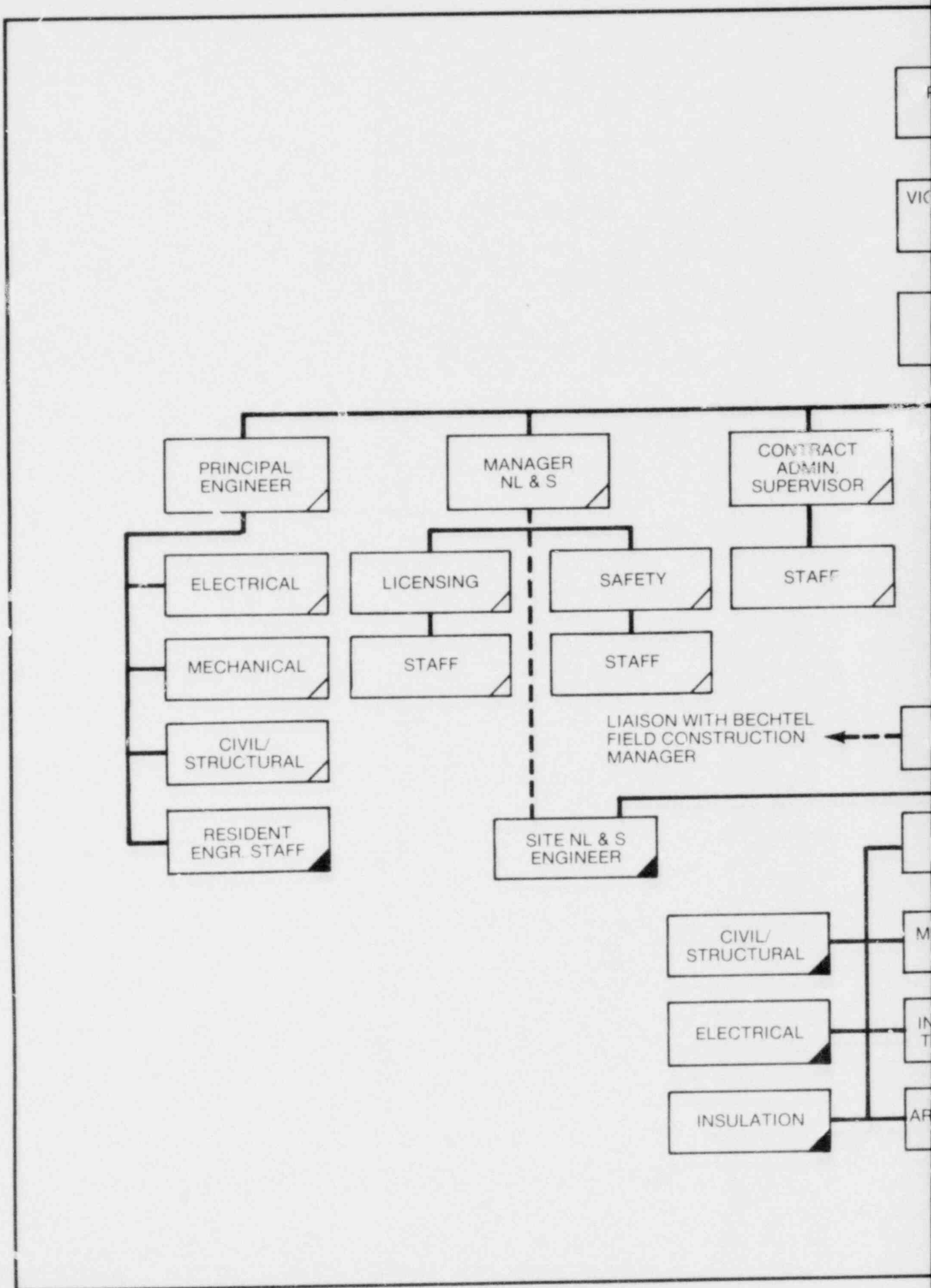
SENIOR STAFF ENGINEER OPERATIONS
DIR. LICENSING & ENVIRONMENTAL COMPLIANCE
HOME OFFICE STAFF
SITE QA ENGINEER
PLANT SUPERINTENDENT (FUTURE)
MANAGER LICENSING & REGULATIONS
MANAGER ENVIRONMENTAL SCIENCES
STAFF
STAFF

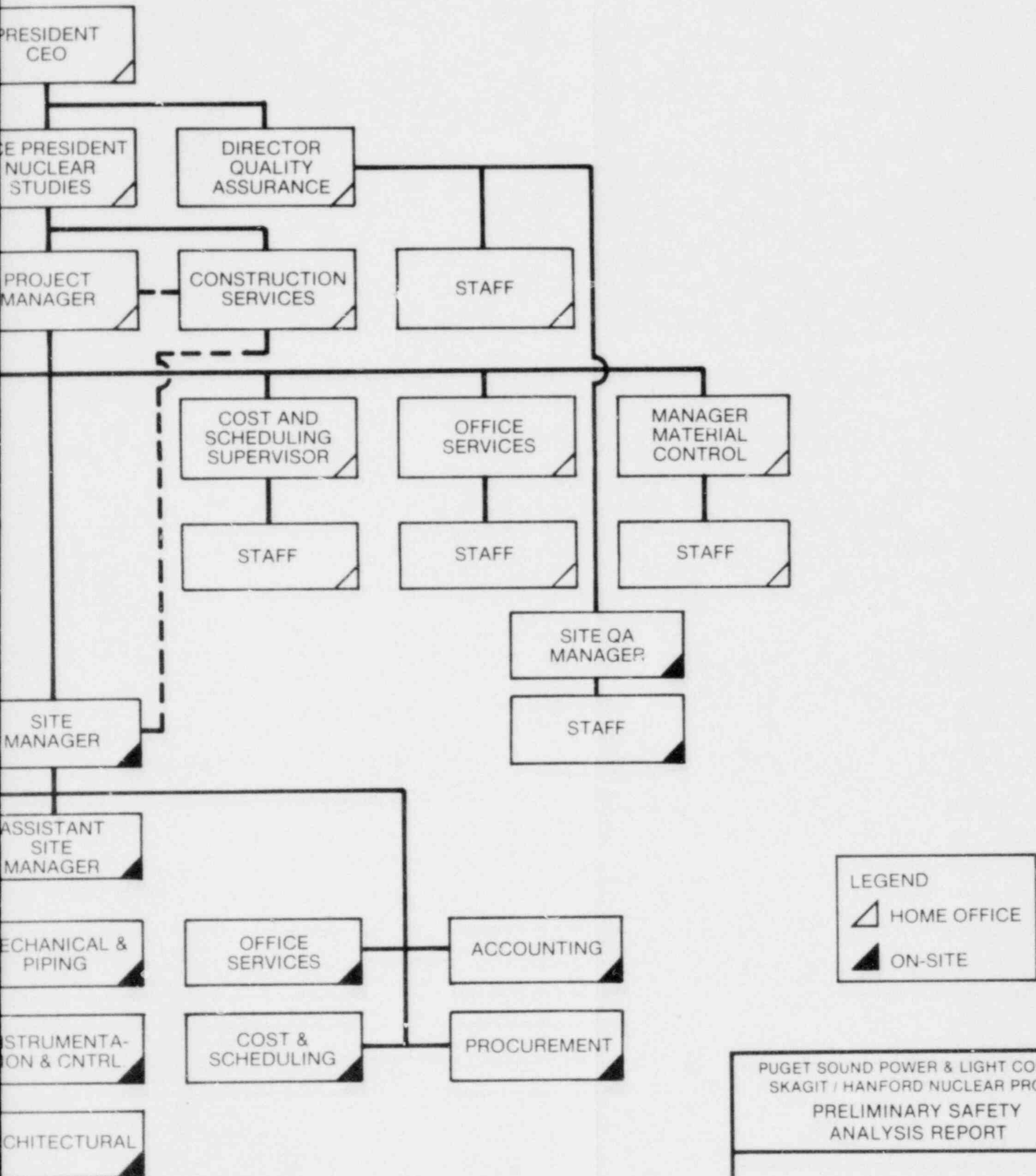


PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

PUGET SOUND POWER AND
LIGHT COMPANY
SKAGIT/HANFORD NUCLEAR
PROJECT ORGANIZATION

FIGURE 13.1-2

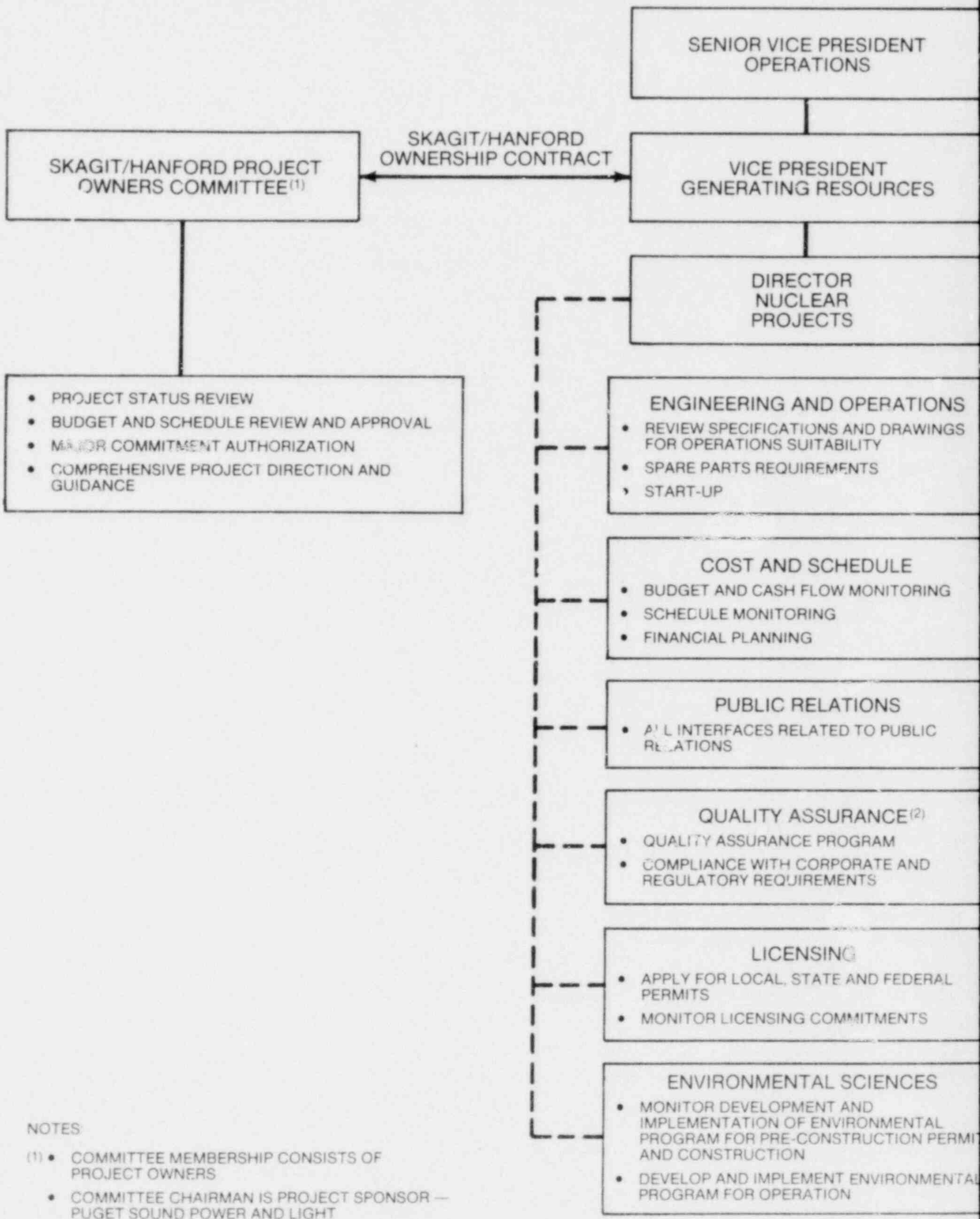

PRINCIPAL ENGINEER
MANAGER NL & S
CONTRACT ADMIN. SUPERVISOR
ELECTRICAL
MECHANICAL
CIVIL/ STRUCTURAL
RESIDENT ENGR. STAFF
LICENSING
STAFF
SAFETY
STAFF
STAFF
LIAISON WITH BECHTEL FIELD CONSTRUCTION MANAGER
SITE NL & S ENGINEER
CIVIL/ STRUCTURAL
ELECTRICAL
INSULATION



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

NESCO
S/HNP ORGANIZATION

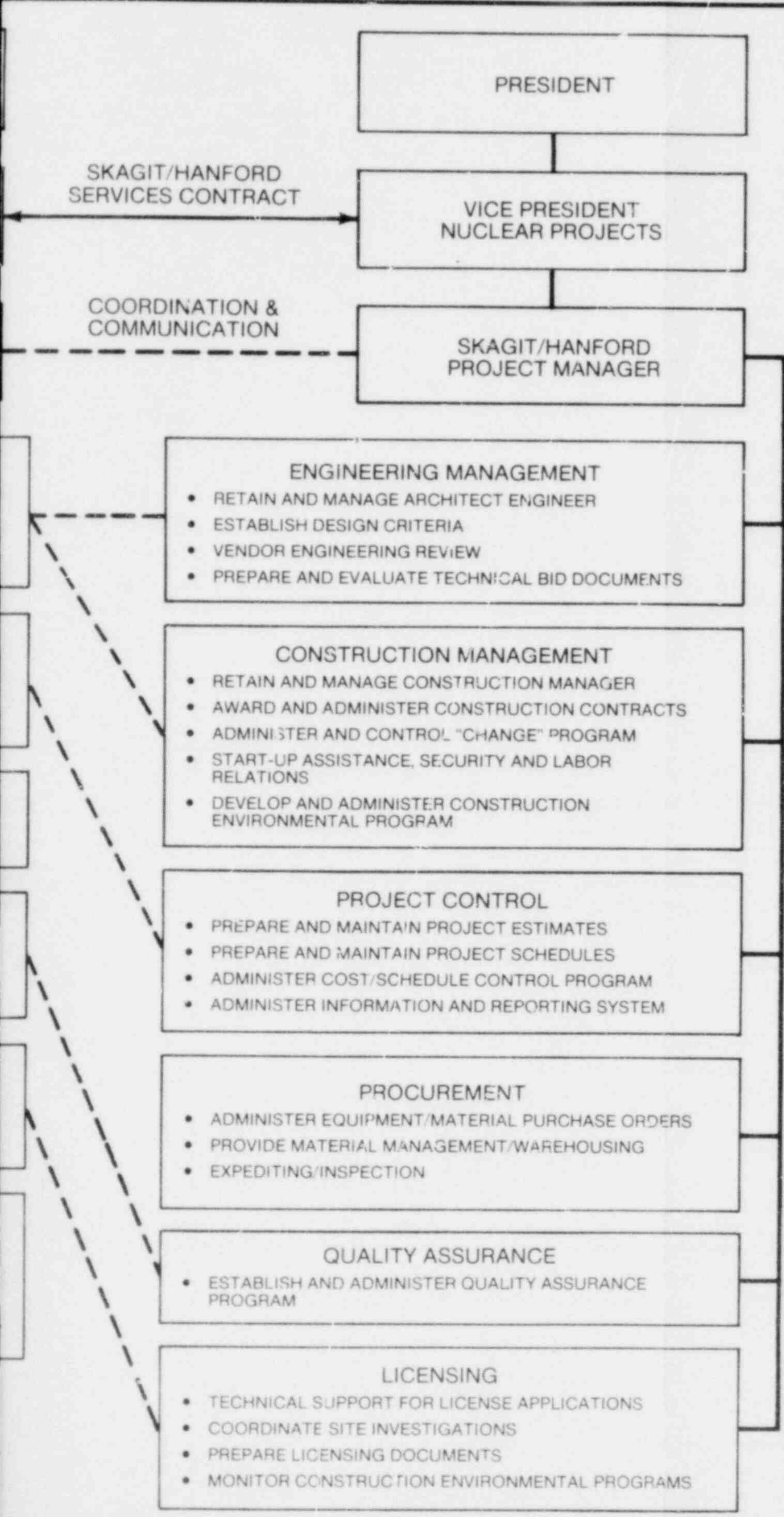
FIGURE 13.1-3



NOTES:

(1) • COMMITTEE MEMBERSHIP CONSISTS OF PROJECT OWNERS
• COMMITTEE CHAIRMAN IS PROJECT SPONSOR — PUGET SOUND POWER AND LIGHT

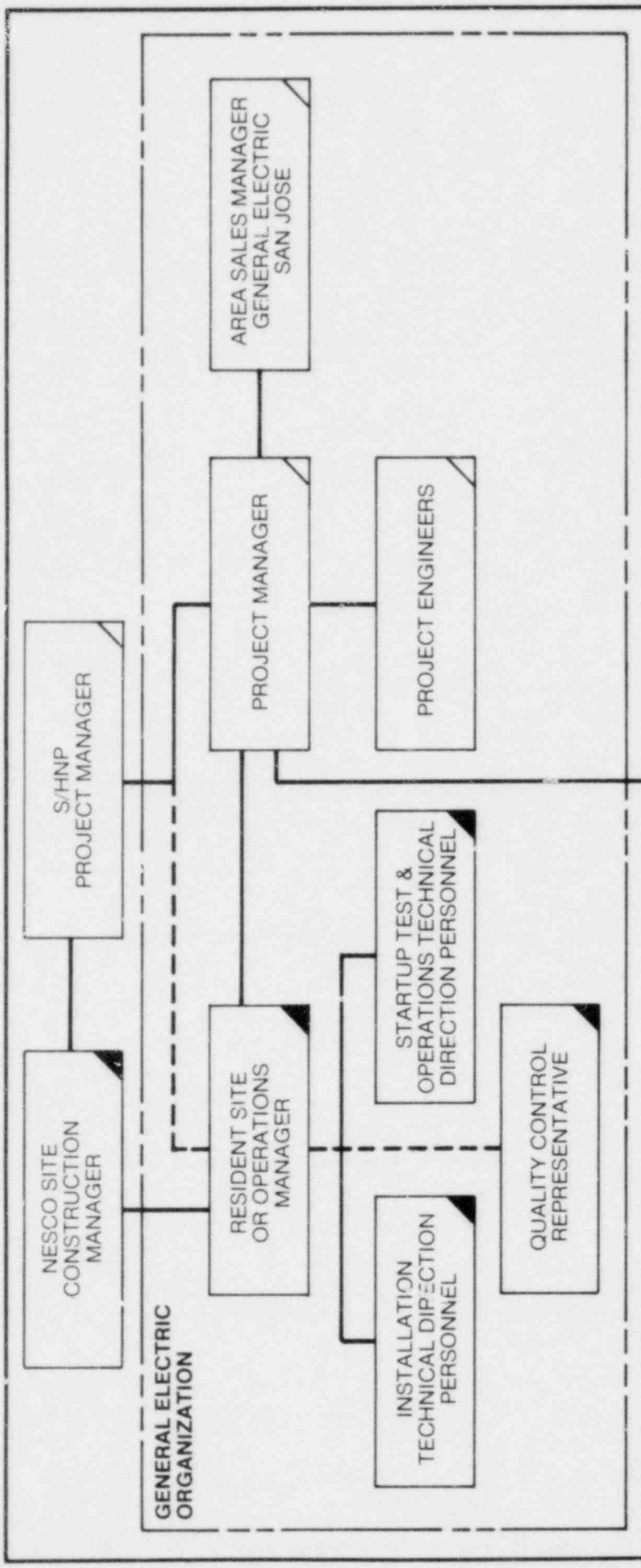
(2) • REPORTS TO VICE PRESIDENT, GENERATION RESOURCES



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

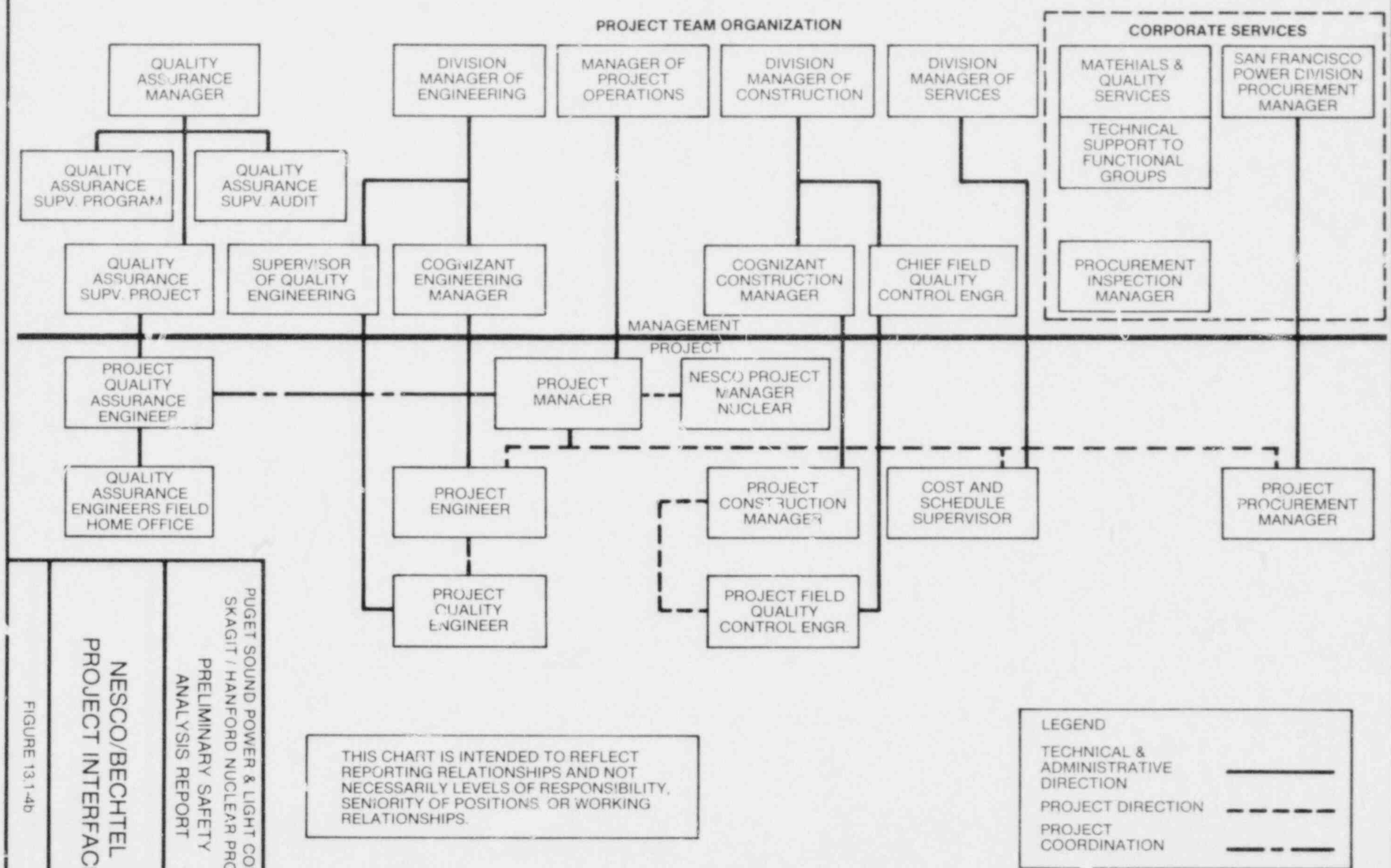
FIGURE 13.1-4



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

NESCO GENERAL ELECTRIC PROJECT INTERFACE

FIGURE 13.1-4a



PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

NESCO/BECHTEL PROJECT INTERFACE

FIGURE 13.1-4b

CHAPTER 17.0

QUALITY ASSURANCE

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QUALITY ASSURANCE

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17.0 QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION PUGET SOUND POWER AND LIGHT COMPANY

Puget Sound Power & Light Company (Puget) will implement an overall Quality Assurance Program (QA Program) for the design, procurement, construction and operation of the S/HNP in accordance with the requirements of 10 CFR 50 Appendix B. As the Applicant, Puget is responsible for the Project and will take appropriate actions to assure that it is designed, constructed, tested and operated in accordance with sound engineering and management principles and practices. Systems, components and structures that are safety-related will be designed, specified, fabricated, installed, tested and operated in accordance with applicable codes, standards, specifications and procedures, and will be within the scope of the Puget QA Program. Puget will use, to the greatest practical extent, the QA experience of the major contractors employed by Puget. However, Puget retains responsibility for institution and implementation of the QA Program. The S/HNP QA Program and related procedures are developed to serve as policy for various quality-related activities and to provide requirements that will assure that quality has been achieved.

17.1.1 ORGANIZATION

17.1.1.1 Puget

The President and Chief Executive Officer of Puget provides
a Statement of Policy in the Quality Assurance Manual
concerning quality as related to the S/HNP. Corrective 21
action items that are not resolved at a lower level within 411.1
a specified time may be brought to the attention of the
Senior Vice President, Operations, and/or the President and 21
Chief Executive Officer for resolution.

Puget is totally responsible for the QA Program on the
S/HNP. The Generation Resources Department of Puget has 21
the responsibility for the overall Project. The department
is under the direction of the Vice President, Generation 16
Resources. The Vice President, Generation Resources,
reports to the Senior Vice President, Operations, and is 21
responsible for:

a. Bringing together in an integrated program the functions of design, engineering, procurement and construction, as necessary, to construct Puget's expanding thermal power generating requirements.

b. Administering the Puget Nuclear Quality Assurance Program, by delegation to the Manager, Quality Assurance, to ensure compliance with NRC regulations and associated guides.

c. Identifying qualified individual(s) or organizational element(s) within the organization as responsible for the quality of delegated work 21
prior to initiation of activities.

d. Assuring that all QA activities are effectively integrated between Generation Resources and other involved departments and contractors.

e. Periodically assessing the effectiveness of the QA Program through evaluation of program audits.

f. Resolving audit recommendations that remain unresolved at the end of the specified period. 21

g. Reviewing, on a quarterly basis, selected organiza- 411.1
tions participating in the QA Program to evaluate the 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, routinely re- 21
ceives QA monthly reports and all Puget issued nonconformance, corrective action and audit reports to aid him in assessing the effectiveness of the QA Program and routinely 411.2
keep him informed of happenings and events related to quality activities.

Puget requires that the authority and duties of persons and organizations performing quality assurance functions be clearly established and delineated in writing. 16

The Puget QA organization that will be located on-Site and 21
off-Site is presented in Figure 17.1-1. Puget/Contractor interfaces are presented in Figure 17.1-2. Puget organ- 21
izations that will be located on-Site and off-Site are 411.4
presented in Chapter 13, Figure 13.1-2. The responsibilities of the organizations performing activities affecting quality, excluding the QA organization, are as 16
contained in Chapter 13. 21

The Puget QA organization is responsible for surveillance and enforcement of all quality-related activities and is 21 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 miti- 16 gate the basic responsibilities of the QA organization for ensuring that the overall intent of the QA Program is carried out and maintained.

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 Assur-
ance, 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. 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 assur-
ance, preferably nuclear equipment manufacture or
nuclear plant construction or construction supervisory
experience. At least one year of this six years' 21
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 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. QA staffing will be based on a long-range projection work schedule and will be periodically re-evaluated and adjusted as necessary. Currently, the QA organization consists of a Manager, Quality Assurance, who reports to the Vice President, Generation Resources.

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, 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

21

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.

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 Director, Quality Assurance, reports directly to the President and is responsible for the NESCO QA Program. 21
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.

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. 21

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 N45.2.12-1974. 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.

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 N45.2.12-1974. QA staffing will be based on a long-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 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.

21

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. They will be involved in day-to-day Plant activities important to safety and will routinely attend

and participate in daily Plant work schedules and status meetings to assure adequate QA coverage. During construction, nonconformance reports will normally 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.

e. 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. 21

Bechtel's QA Program for safety-related activities and services is described in Bechtel's Topical Report BQ-TOP-1, Rev. 1A and Section 17.2.

Construction of the S/HNP will be under the management direction of Bechtel as the Construction Management Contractor. Bechtel will manage and coordinate the activities of constructors responsible for completing the contracts. These contracts will be awarded to constructors during a period of time ranging from approximately one year before the construction permit is issued to several years after the construction permit is issued, depending on construction schedule requirements and the Plant design schedule. 21 411.22

Typical contracts which will be awarded are:

Structures
Containment Liner
Nuclear Equipment Installation
Mechanical Installation
Electrical Installation

The detailed scope of work for each of the constructors will be defined in each contract. 411.22 411.57

As a minimum, the QA Program of the construction contractors and subtier contractors must meet the applicable provisions of 10 CFR 50, Appendix B, and ANSI N45.2-1971. 21

17.1.1.4 General Electric Company

GE has been contracted to provide the Nuclear Steam Supply System which includes the responsibility for the QA/QC activities associated with their scope of supply. The QA Program for safety-related activities and services performed by GE is described in GE's Nuclear Energy Business Group BWR Quality Assurance Program Description, NEDO-11209 04A, Rev. 2. (See Section 17.3.) 21

17.1.1.5 Interfaces

The interface within NESCO during the design and construction phase of the Project is through the Project Manager, Nuclear. NESCO's S/HNP Project Manager is also the primary interface between Puget's Director, Nuclear Projects, Bechtel Project Manager and GE's Project Manager as shown in Figure 17.1-2. During construction, an interface also exists at the Site between Bechtel's Field Construction Manager and NESCO's Site Construction Manager, similar to the interface for engineering and design between Bechtel's Project Engineer and NESCO's Principal Engineer. All interfaces with the constructors will be through Bechtel's Field Construction Manager. 21 411.9

Puget's QA Program encompasses the QA Program of NESCO, Bechtel, GE and major contractors. The relationships between these organizations are shown in Figure 17.1-2. The Puget-Bechtel contact with regard to the Quality Assurance Program is Puget's Manager, Quality Assurance, via Bechtel's Project QA Engineer. The Puget/NESCO contact with regard to QA is Puget's Manager, Quality Assurance, via NESCO's Director, Quality Assurance. The Puget/GE contact with regard to the QA Program is Puget's Manager, Quality Assurance, via [illegible]'s Project Manager. 21

Upon completion of construction and construction testing, systems will be turned over to Puget's Plant staff for operation as described in Section 14.1.1.1. The primary interface during this phase of the Project is between Puget's Plant Superintendent and Bechtel's Field Construction Manager. Subsequent to the acceptance of a system, any further construction work involving that system will be coordinated with the Plant Superintendent. Examples of such work might include the completion of [illegible] outstanding items, modifications, correction of significant deficiencies discovered during preoperational testing, etc. 21 411.9 21

Preoperational testing and startup testing will be performed by Puget's personnel. When the preoperational test phase is completed such that the Plant is ready for fuel loading, Puget will assume full responsibility for its operation and maintenance. All design changes and system modifications after this time will require the approval of the Plant Superintendent. Procedures for the review and approval of modifications and the interface with Puget's Director, Nuclear Projects, and Manager, Quality Assurance, during this period will be described in the FSAR. 21 21

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, Gener- 21
ation Resources, are required on the Quality Assurance
Manual. He reviews, on a quarterly basis, selected organ-
izations participating in the QA Program to evalute the
status and adequacy of their part in the program. The 411.5
review is designed to include all applicable organizations
at least once a year.

The Vice President, Generation Resources, resolves cor- 21
rective action items involving quality arising from a
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 organiza-
tions 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'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 appro-
priate 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 dele-
gated work. Puget's primary tool for ensuring that
delegated portions of the QA Program are being effectively
carried out is a well-planned system of auditing and
surveillance. 21

Puget's QA Program requires that verification of con-
formance 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
with appropriate equipment under suitable environmental 21
conditions. 411.8

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
NESCO. It is also the responsibility of Puget's QA organ- 21
ization to conduct (or have conducted) audits of the major
contractors' QA program activities.

A QA Program will be submitted by the major construction 411.3
contractors for Bechtel's evaluation and concurrence and
Puget's and NESCO's information prior to the start of 21
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, 21
"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 indoc-
trination 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.

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
QA including a review of 10 CFR 50, Appendix B, (18 cri- 21
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 21
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:

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)

NESCO, as Puget's agent, is responsible for preparation of the PSAR and coordinating its review.

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.

Puget, as the applicant, approves and submits the PSAR to the NRC. 21

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 organization designates another responsible organ- 21
ization.

j. Preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design.

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 Principal 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

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 checks for suitability of material, parts, equipment, and processes and that valid industry standards and specifications, 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. 21 411.13

Errors and deficiencies found during the review process that adversely affect safety-related structures, systems, and components are documented by NESCO and appropriate corrective action is taken. 21 411.12

Details of NESCO's review of designated design and design criteria documents are specified in NESCO's Project Procedures Manual. 21

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 specified in Nuclear Licensing and Safety procedures. In addition, NESCO may initiate design changes by letter. 21

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 for any revisions of prior approved design documents or design characteristics changes whose limits are controlled by: 21

a. Design Criteria

b. Safety Analysis Report

c. State and national codes and standards

Changes which are within the specified limits of the above control documents need not have NESCO's release, but are documented by Bechtel and copies transmitted to NESCO for information.

NESCO reviews revisions to prior approved design documents. Such review is accomplished in the same manner as the original document.

Details of how the above design and design change control commitments are implemented by Bechtel and GE are described in Section 17.2 and 17.3, repectively.

Details of how design and design change control is implemented by NESCO, Bechtel and GE are described in their procedures.

NESCO QA personnel audit Bechtel and GE to verify conformance to their design and design change control procedures. 21

17.1.4 PROCUREMENT DOCUMENT CONTROL

The applicable requirements of ANSI N45.2-1971, Section 5 are applied to procurement document control.

Procurement documents are controlled to assure they contain relevant regulatory, code, design basis, QA and commercial requirements necessary to assure the quality, integrity and reliability of procured material, equipment and services.

Procurement and procurement document control has been delegated to Bechtel and GE.

Puget requires that Bechtel and GE provide measures which assure that the following requirements are included in procurement documents:

a. Applicable regulatory, code and design requirements.

b. QA Program requirements.

c. Requirements for supplier documents such as instructions, procedures, drawings, specifications, inspection and test records and supplier QA records to be prepared, submitted, or made available for purchaser review or approval.

d. Requirements for the retention, control and maintenance of supplier QA records.

e. Provision for purchaser's right of access to supplier's facilities and work documents for inspection and audit.

f. Provision for supplier reporting and disposition of nonconformance from procurement requirements.

g. Provision which assures that quality requirements in procurement documents are correctly stated, inspectable and controllable. 21 411.14

h. For commercial "off-the-shelf" items where specific QA controls appropriate for nuclear applications cannot be imposed in a practical manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser. 21

Puget requires that Bechtel's and GE's QA programs include the following requirements:

a. Review and approval of procurement documents are documented and available for verification. 21 411.15

b. Procurement documents, including Puget's procurement documents, identify those records which shall be retained, controlled, maintained or delivered to the purchaser prior to use or installation of the item. 21 411.15

c. Changes or revisions to procurement documents receive the same review and approval requirements as the original documents.

d. Spare or replacement items will be subject to controls at least equivalent to those used for the original equipment. 21

e. Construction contractor's procurement documents, together with any additional QA documentation, are

filed by the contractor in an appropriate QA documentation file and at the conclusion of the contract are turned over to Bechtel.

f. Control responsibilities and action sequence to be taken by competent personnel in the preparation, review, approval and issuance of procurement documents shall be clearly delineated.

g. Procurement documents require suppliers to have and implement an acceptable documented QA Program.

h. Suppliers are evaluated before the award of the procurement order or contract to assure that the supplier can meet procurement requirements.

Bechtel is responsible for the following:

a. Preparing bidders list.

b. Sending bidders list to NESCO for approval.

c. Preparing and issuing bid packages to prospective 21
bidders.

d. Evaluating bid responses and making recommendations to NESCO.

e. Committing awards after Puget's approval.

NESCO, as Puget's agent, is responsible to provide an independent review of procurement documents. NESCO's review does not replace the procurement document control implemented by Bechtel to meet the requirement of 10 CFR 50, Appendix B.

NESCO is responsible for the following:

a. Reviewing and approving bid packages.

b. Reviewing bid evaluations and Bechtel's recommendations.

c. Selecting and recommending the successful bidder to Puget.

NESCO's Director, Quality Assurance, reviews and approves bid packages and bid awards and reviews bid evaluation and recommendation to verify that Bechtel has evaluated the acceptability of the bidder's QA Program and his ability to satisfactorily carry out the program.

Puget's Director, Nuclear Projects, reviews and approves bid packages and bid award, and reviews bid evaluations and recommendations to assure that:

a. Bidders are qualified.

b. Bids are responsive to requirements.

c. Suppliers are evaluated before the award of the procurement order or contract to assure that the contractor and supplier can meet procurement requirements.

NESCO QA personnel audit and perform site surveillance of procurement document control activities to assure compliance with QA Program requirements.

Details of how procurement document control activities are implemented by NESCO, Bechtel and GE are discussed in their procedures.

17.1.5 INSTRUCTIONS, PROCEDURES AND DRAWINGS

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The applicable requirements of ANSI N45.2-1971, Section 6, are applied to instructions, procedures and drawings.

Puget requires that activities affecting quality are prescribed by appropriate documented instructions, procedures or drawings and that these activities are conducted in accordance with these documented instructions, procedures or drawings.

Puget requires that documented instructions, procedures and drawings include appropriate quantitative and qualitative acceptance and rejection criteria for determining that prescribed activities have been satisfactorily accomplished.

It is the responsibility of each Puget and NESCO department associated with the Project to write detailed instructions outlining how they plan to perform their duties affecting quality. These instructions are in addition to instructions specified in the Quality Assurance Manual.

The QA organization reviews and documents concurrence with quality-related procedures. The review is to assure that the procedures are consistent with QA Program commitments and corporate policies and are properly documented and controlled.

Documents are prepared, revised and issued in a controlled manner with a copy sent to Records.

NESCO audits and performs Site 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. Instructions, procedures and drawings include:

a. Design specifications.

b. Design, manufacturing, construction and installation drawings.

c. Procurement documents. 21 411.13

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.

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. 21

Puget requires that Bechtel provide, as appropriate:

a. Evaluation and selection of sources of supply.

b. Surveillance at the supplier's facility in accordance with written procedures during manufacture, inspection and testing of the procured item or service to verify compliance with quality requirements. 21 4

c. Source and/or receipt inspection, in accordance with written procedures and acceptance criteria, of procured items furnished by supplier. 21 4

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 Site prior to installation or use of the procured 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 requirements which have not been met are identified together with a description of those nonconformances dispositioned "accept as is" or "repair". The certification program is administered by Bechtel. Puget exercises its responsibilities through a system of periodic audits by NESCO to insure the PSAR and Puget's QA requirements are not violated. Procured items received at the Plant Site without either complete 21 411.19 21

documented 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, installation or use of the item. 21 411.19

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 requirements are included that assure that purchased material, equipment and services conform to procurement document requirements. 21

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, 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 by the QA organization, as required, throughout fabrication, erection, installation and use of the item, and that the method and location of the identification does not affect the function or quality of the item being identified. 21

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- 21 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.

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.

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: 21

a. Inspection personnel are adequately qualified and qualifications or certifications are kept current and documented.

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. 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. 21 411.21,56

c. Detailed documented inspection instructions and procedures are provided to keep inspectors adequately informed. These procedures require determining the accuracy requirements of inspection equipment and when inspections are required or how and when inspections are performed. The QA organization shall participate in the above function. 21

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

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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.

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. 21

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.

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.

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 contractors are required to perform tests in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design and procurement documents. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation and equipment are available and used, and that the test is performed under suitable environmental conditions 21
and with adequate test methods. 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 of required tests, inspections and other operations important 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 installation is complete and is identified in the construction contract documents. Construction testing involves such testing as hydro and piping leak tests; instrument, control, and stroke valve tests; control and power connections 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.

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.

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 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. 21

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-Site calibration and shall meet the above requirements. 21 411.23

Puget requires that its contractors have procedures for control, calibration and adjustment of measuring and test equipment which enact the above requirements and include: 21

a. Identification of measuring and test equipment.

b. How measuring equipment is routinely monitored and the action required when limits are exceeded.

c. Establishment of calibration frequencies. 21 411.24

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 and major contractors to assure control of measuring and test equipment. 21

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, 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.

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.

NESCO provides surveillance of the transitory storage facility warehousing services contractor to assure conformance to specified requirements. 21

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.

17.1.15 NONCONFORMING MATERIALS, PARTS OR COMPONENTS 21

The applicable requirements of ANSI N45.2-1971, Section 16, are applied to nonconforming materials, parts or components.

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:

(1) Disposition has been satisfactorily completed and closed out.

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

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

j. QA personnel analyze nonconformance reports for trends and chronic problems and, as a minimum, 21
report findings to management on a monthly basis.

k. The Vice President, Generation Resources; Director, Nuclear Projects; Manager, Quality Assurance; and Director, Licensing and Environmental Compliance, evaluate and determine if nonconformances pertain to the conditions outlined in 10 CFR 50.55 (e). The Manager, Licensing and Regulation, assures the timely issuance of any 10 CFR 50.55 (e).

l. The Director, Nuclear Projects, and the Manager, Quality Assurance, are responsible for assuring that work associated with items or activities identified as nonconforming is stopped or conditionally stopped, if the condition that caused the nonconformance has not been satisfactorily corrected.

m. All affected organizations receive copies of nonconformance reports.

Puget requires that its contractors have procedures and a system for controlling nonconforming items. NESCO reviews for information, nonconformance reports as soon as they are issued and suppliers' deviation reports that are dispositioned ("repair" or "use as is"). Such reports become part 21
of the documentation required at the Plant Site. Affected 411.25
organizations are notified by Bechtel or by the responsible supplier issuing the nonconformance report. Nonconformance reports will be reviewed and analyzed by NESCO's QA organization and the results routinely reported to Puget.

Puget holds Bechtel responsible for a final accounting of all nonconformance reports upon completion of each contractor's work for which Bechtel has construction management responsibilities.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to assure that an effective nonconforming material control program has been implemented.

Details of how NESCO, Bechtel and GE assure proper handling of nonconformances are described in their procedures.

17.1.16 CORRECTIVE ACTION

The applicable requirements of ANSI N45.2-1971, Section 17, are applied to corrective action.

Puget's QA Program requires that significant conditions adverse to quality are promptly identified and corrected. The program also requires that the cause of significant conditions adverse to quality be determined, corrective action taken to preclude repetition and the corrective action documented and reported to appropriate levels of management.

Where corrective action is necessary to correct a serious quality problem, a Correction Action Request (CAR) is initiated. The CAR is used only after the more common means of communication have failed to achieve the necessary corrective action. 21

The following example includes situations which may warrant a CAR:

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.

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.

QA is responsible for administering the CAR Program.

Anyone in Puget may initiate a CAR. The CAR initiator completes the applicable portions of the CAR (description

and recommended action) and sends it to the Manager, Quality Assurance.

The Manager, Quality Assurance, is responsible for reviewing and issuing CARs and sending them to applicable management in Puget, NESCO, Bechtel or GE for cause and corrective action.

The action addressee shall note on the CAR the cause and action necessary to remedy the condition and preclude recurrence and return the CAR to the Manager, Quality Assurance.

The Manager, Quality Assurance, reviews the corrective action response for adequacy and, if not satisfactory, returns the CAR unsigned to the action addressee with reasons for rejection.

When a satisfactory corrective action statement is pro- 21
vided, the Manager, Quality Assurance, provides follow-up action and, when the corrective action is implemented, signs the "verified" block on the CAR and sends the CAR to distribution.

The action addressee must respond to the CAR within 30 days after receipt of the CAR unless otherwise stated on the CAR transmittal.

Corrective action items remaining unresolved at the end of 90 days (or any shorter period if the situation is considered urgent) may be brought to the attention of the President and Chief Executive Officer for resolution.

QA personnel are responsible for assigning a number to each CAR. The number and applicable information shall be entered into a CAR Log by QA personnel.

NESCO audits and performs Site surveillance of Bechtel, GE and major contractors to ensure prompt corrective action is being taken when conditions adverse to quality are detected.

Details of how NESCO, Bechtel and GE assure corrective action is taken are described in their procedures.

17.1.17 QUALITY ASSURANCE RECORDS

The applicable requirements of ANSI N45.2-1971, Section 18, and ANSI N45.2.9-1974 are applied to quality assurance records.

Puget's QA Program requires that sufficient records are
maintained to adequately document activities affecting 21
quality and that the record storage facility protects
records from destruction by fire, flooding, tornadoes,
insects and rodents and from deterioration by extremes in
temperature and humidity. Records include:

a. Documentation such as test logs, results of reviews, inspections, tests, audits, monitoring work performance and materials analysis, nonconformance and corrective action reports, procedures, personnel qualifications and drawings.

b. Sufficient information to permit identification of the record with the item or activity to which it applies.

c. Type of operation, the inspector or data recorder, the results, the acceptability and action taken in connection with any deficiencies noted.

Puget has written procedures for controlling QA records. These records are identifiable and retrievable. The procedures require as a minimum:

a. Assignment of responsibility.

b. Establishing a records index indicating:

    (1) Record retention times

    (2) Where records are stored

    (3) Location of records within storage area

c. Establishment of procedures for the distribution and handling of records.

d. Establishment of identification between the record and the item or activity.

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, and ANSI N45.2.12-1974, Draft 3, Rev. 4, are applied to audits.

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.

c. Requests for proposals and evaluation of bids. 21

d. Indoctrination and training programs. 411.27

Puget's external audits include audits of NESCO. Puget also accompanies NESCO as an observer on a sufficent number of NESCO audits to determine their effectiveness. Puget issues an independent report on audit observations. 21

The following organizations within Puget that are participating in the S/HNP are routinely audited by QA:

a. <u>Off-Site</u>

Generation Resources 21
Micrographics and Records 411.26

b. <u>On-Site</u>

Nuclear Projects

Puget's audits are the responsibility of the Manager, Quality Assurance, and are performed as follows:

An audit schedule is prepared and issued by the Manager, Quality Assurance. The scope and who shall participate in audits are determined and specified. 16 Participants are selected based on their knowledge of the activities to be audited and their independence from direct responsibility. Auditors are appropriately trained and qualified. The Manager, Quality Assurance, acts as, or appoints, the audit team leader.

The audit team leader develops an agenda (checklist) for each audit, which forms the basis for the auditor's actions during the audit. Audit agendas are approved by the Manager, Quality Assurance.

The audit team leader reviews with audit team members the procedures governing activities and/or items to be audited. Assignments are made to audit team members by the audit team leader.

Checklists based on the audit agenda are used by the audit team to ensure depth and uniformity of the audit.

Puget QA notifies the organization to be audited and defines general areas, systems or components to be audited.

Applicable elements of the QA Program are audited at least annually or at least once within the life of the activity, whichever is shorter.

The audit team leader prepares and issues an audit report following the conclusion of each audit. The audit report shall be approved by the Manager, Quality Assurance. 21

The Manager, Quality Assurance, forwards audit reports to the audited group with copies as a minimum to:

a. Vice President, Generation Resources.

b. Director, Nuclear Projects.

Responses to formal audit reports are evaluated by the audit team leader who is responsible for preparing the reply and closing the audit. 16

The Manager, Quality Assurance, schedules follow-up audits to verify implementation and effect of corrective action. The follow-up auditor or audit team issues a follow-up audit report.

It is the responsibility of Puget's Vice President, Generation Resources, to periodically assess the effectiveness of the program through evaluation of program audits and other measures.

Any recommendations remaining unresolved at the end of 90 days (or any shorter period, if the situation is considered urgent) may be brought to the attention of the Vice President, Generation Resources, for resolution with the audited organization. 21

QA reports the status of internal and external audits monthly with copies going to the President and Chief Executive Officer and the Vice President, Generation Resources, and his staff.

Documentation pertaining to audits becomes part of the objective evidence of quality and is filed in the Project Records File.

Details of how audits are performed by NESCO, Bechtel, GE and major contractors are described in their procedures.

TABLE 17.1-1
10 CFR 50, APPENDIX B, CRITERIA
PUGET'S MANUALS, PROCEDURES OR INSTRUCTION
QA PROGRAM SYNOPSIS MATRIX

10 CFR 50 Appendix B Criteria	Manual, Procedures or Instructions	QA Program Procedures Synopsis
I Organization	Quality Assurance Procedure #1 (Organization) GPE Procedures Manual Project Procedures Manual	To describe the organizational structure of the S/HNP and the responsibilities of each section in relation to carrying out the overall Quality Assurance Program.
II Quality Assurance	Quality Assurance Procedure #2 and #19 (Quality Assurance Program and Indoctrination and Training) Quality Assurance Manual, Appendix #1 (Quality Assurance for Radioactive Waste Management Systems) Quality Assurance Manual, Appendix #2 (Quality Assurance Concerning Fire Prevention in Safety-Related Areas) Project Procedures Manual Puget S/HNP Licensing Procedures Manual	To describe Puget's Quality Assurance Program for the S/HNP.
III Design Control	Quality Assurance Procedure #3 (Design and Design Change Control) QA Instruction #1 (Design and Procurement Document Review) GPE Procedures Manual Project Procedures Manual Puget S/HNP Licensing Procedures Manual	To establish a system of independent review of design documents for compliance to applicable regulatory requirements, design bases, quality standards, codes and good engineering practices, and to establish a system for control of design changes.
IV Procurement Document Control	Quality Assurance Procedure #4 (Procurement Document Control) QA Insruction #1 (Design and Procurement Document Review) GPE Procedures Manual Project Procedures Manual	To establish a system to assure that procurement documents contain relevant regulatory, code, design bases, quality assurance and commercial requirements necessary to assure the quality, integrity and reliability of procured material, equipment and services.
V Instructions, Procedures and Drawings	Quality Assurance Procedure #5 (Instructions, Procedures and Drawings) GPE Procedures Manual Records and Information Manual Project Procedures Manual	To establish a system to assure that activities affecting quality are prescribed by documented instructions, procedures, and drawings that clearly specify how such activities will be accomplished.
VI Document Control	Quality Assurance Procedure #6 (Document Control) Records and Information Manual GPE Procedures Manual Project Procedures Manual	To establish a system to control the issuance of documents, such as instructions, procedures, and drawings, including changes, which prescribe activities affecting quality.
VII Control of Purchased Material, Equipment and Services	Quality Assurance Procedure #7 (Control of Purchased Material, Equipment and Service) Project Procedures Manual	To establish a system to assure that purchased material, equipment and services conform to the requirements specified in procurement documents.

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10 CFR 50 Appendix B Criteria		Manual, Procedures or Instructions	QA Program Procedures Synopsis
VIII	Identification and Control of Materials, Parts and Components	Quality Assurance Procedure #8 (Identification and Control of Material, Parts and Components) Project Procedures Manual	To establish a system for the identification and control of material, parts, and components, including partially fabricated assemblies to preclude use of incorrect or defective parts, material or components.
IX	Control of Special Processes	Quality Assurance Procedure #9 (Control of Special Processes)	To establish a system to control special processes to assure that they are accomplished.
X	Inspection	Quality Assurance Procedure #10 (Inspection)	To establish a system to assure that inspection activities are performed in accordance with requirements predetermined and delineated in written instructions in a planned and systematic manner.
XI	Test Control	Quality Assurance Procedure #11 (Test Control)	To establish a system of control to assure that construction testing activities are performed in accordance with approved written test procedures which incorporate the requirements and acceptance limits of the design documents.
XII	Control of Measuring and Test Equipment	Quality Assurance Procedure #12 (Control of Measuring and Test Equipment)	To establish a system for the control, calibration and adjustment of tools, gauges, instruments and other inspection, measuring, testing and maintenance devices.
XIII	Handling, Storage and Shipping	Quality Assurance Procedure #13 (Handling, Storage and Shipping) Project Procedures Manual	To establish a system to control handling, storage, shipping, cleaning and preservation of material, parts, components and equipment to prevent damage or deterioration.
XIV	Inspection, Test and Operating Status	Quality Assurance Procedure #14 (Inspection, Test and Operating Status)	To establish a system to indicate the inspection, test and operating status for structures, systems or components to preclude the inadvertent bypassing of their inspection and test requirements, and to prevent their inadvertent use.
XV	Nonconforming Material, Parts or Components	Quality Assurance Procedure #15 (Nonconforming Material, Parts or Components) Quality Assurance Manual, Appendix #3 (Reporting of Significant Deficiencies) Quality Assurance Manual, Appendix #4 (Reportng of Defects and Noncompliances) Project Procedures Manual	To establish a system to assure that nonconforming material, parts, or components are identified, documented, segregated, dispositioned, prevented from being used or installed, and that notification of these actions are transmitted to affected organizations.
XVI	Corrective Action	Quality Assurance Procedure #16 (Corrective Action) Project Procedures Manual	To specify requirements for assuring that prompt, effective corrective action is taken when significant conditions occur that are adverse to quality.
XVII	Quality Assurance Records	Quality Assurance Procedure #17 (Quality Assurance Records) Records and Information Manual Project Procedures Manual	To establish a system for control and maintenance of all records sufficient and necessary to provide objective evidence of activities affecting quality.

411.7
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TABLE 17.1-1
10 CFR 50, APPENDIX B, CRITERIA
PUGET'S MANUALS, PROCEDURES OR INSTRUCTION
QA PROGRAM SYNOPSIS MATRIX

10 CFR 50 Appendix B Criteria	Manual, Procedures or Instructions	QA Program Procedures Synopsis
XVIII Audits	Quality Assurance Procedure #18 (Audits) QA Instruction #2 (Systems and Procedures Audits) QA Instruction #3 (Audit Observer Instructions)	To establish a comprehensive system of planned and periodic audits to verify compliance with all aspects of Puget's Quality Assurance Program and to determine the effectiveness of the program.

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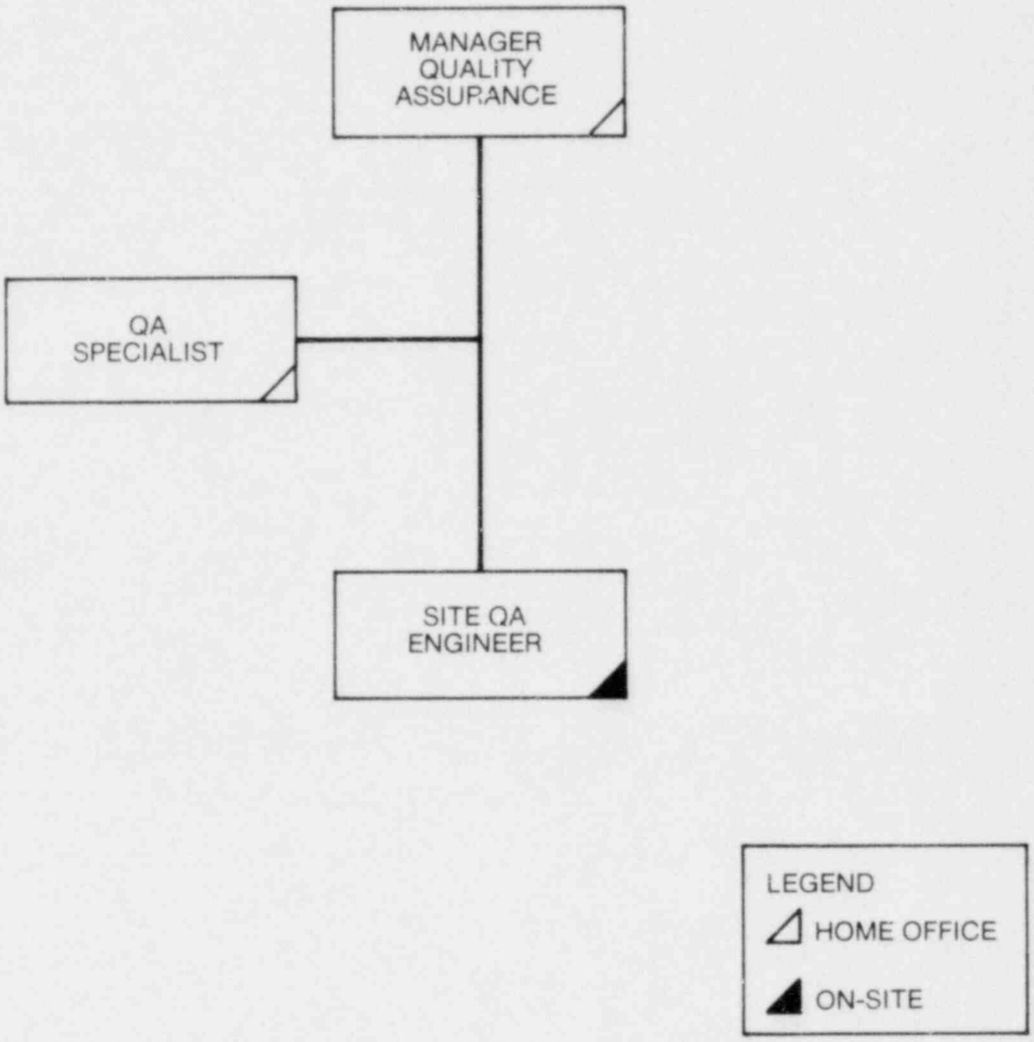
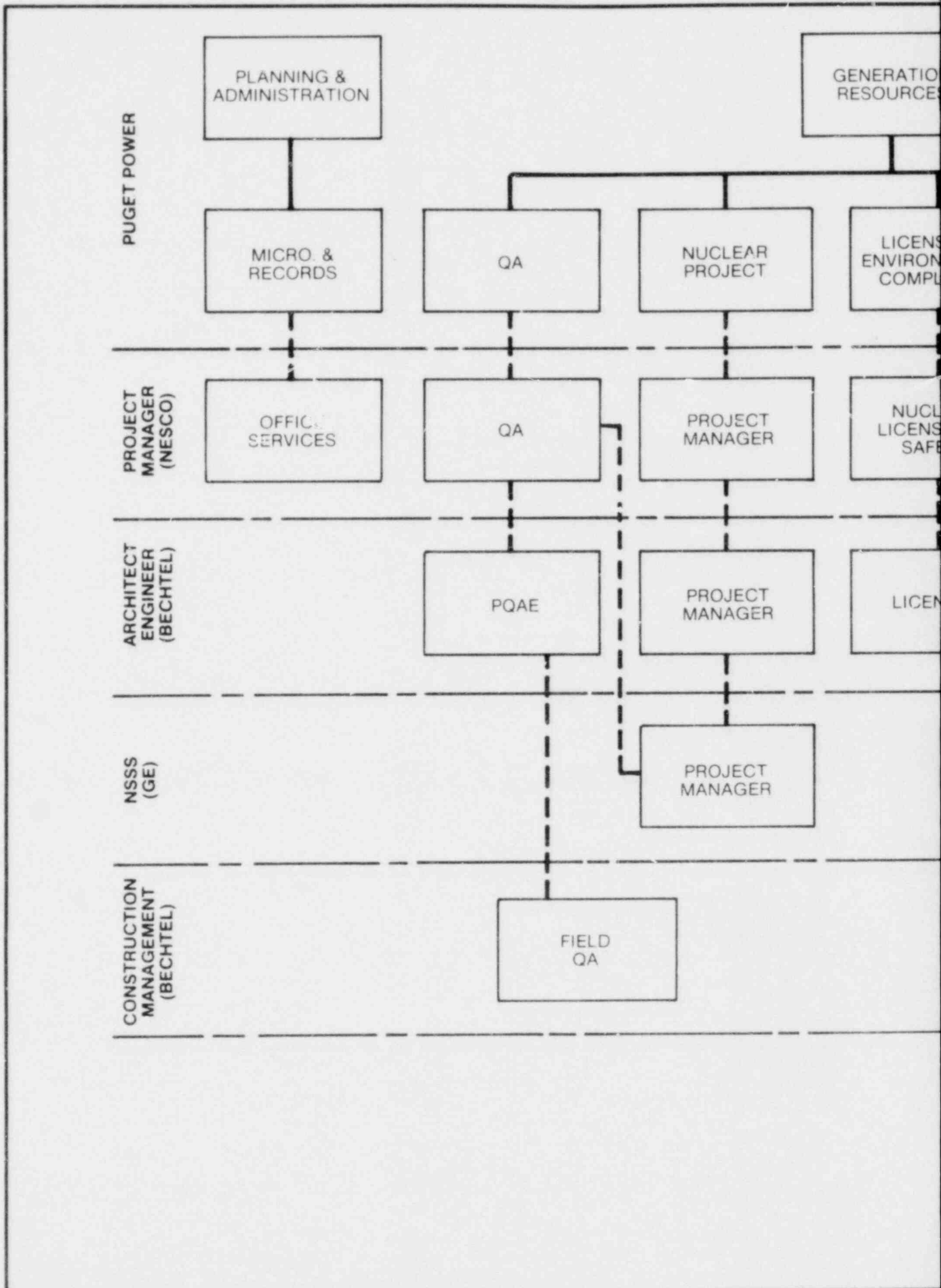
411.7



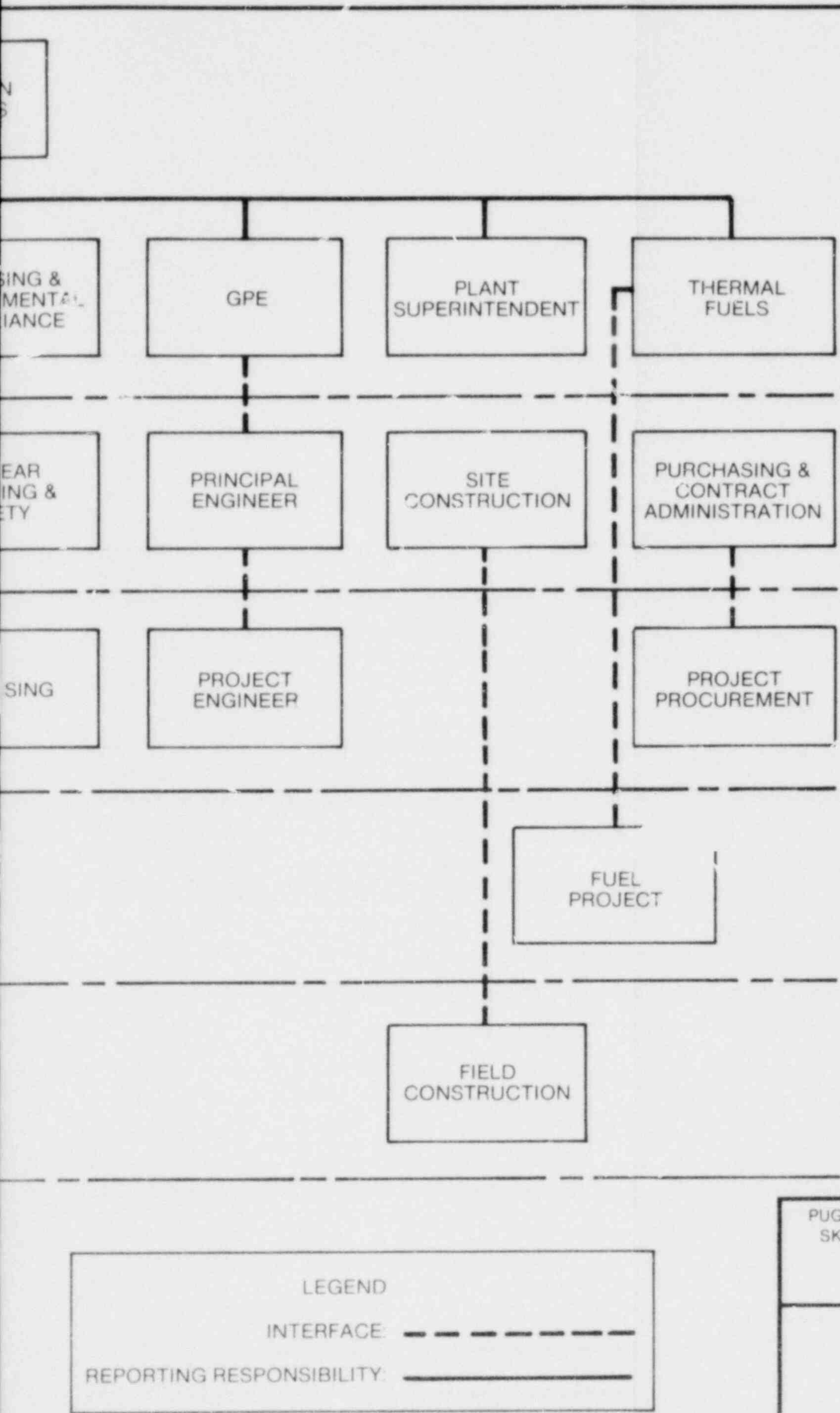
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT / HANFORD NUCLEAR PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT

PUGET
QUALITY ASSURANCE
ORGANIZATION

FIGURE 17.1-1



PUGET POWER
PROJECT MANAGER (NESCO)
ARCHITECT ENGINEER (BECHTEL)
NSSS (GE)
CONSTRUCTION MANAGEMENT (BECHTEL)
PLANNING & ADMINISTRATION
GENERATIO
RESOURCE
MICRO. & RECORDS
QA
NUCLEAR PROJECT
LICENS
ENVIRON
COMPL
OFFIC
SERVICES
QA
PROJECT MANAGER
NUCL
LICENS
SAFE
PQAE
PROJECT MANAGER
LICEN
PROJECT MANAGER
FIELD QA



PUGET SOUND POWER & LIGHT COMPANY
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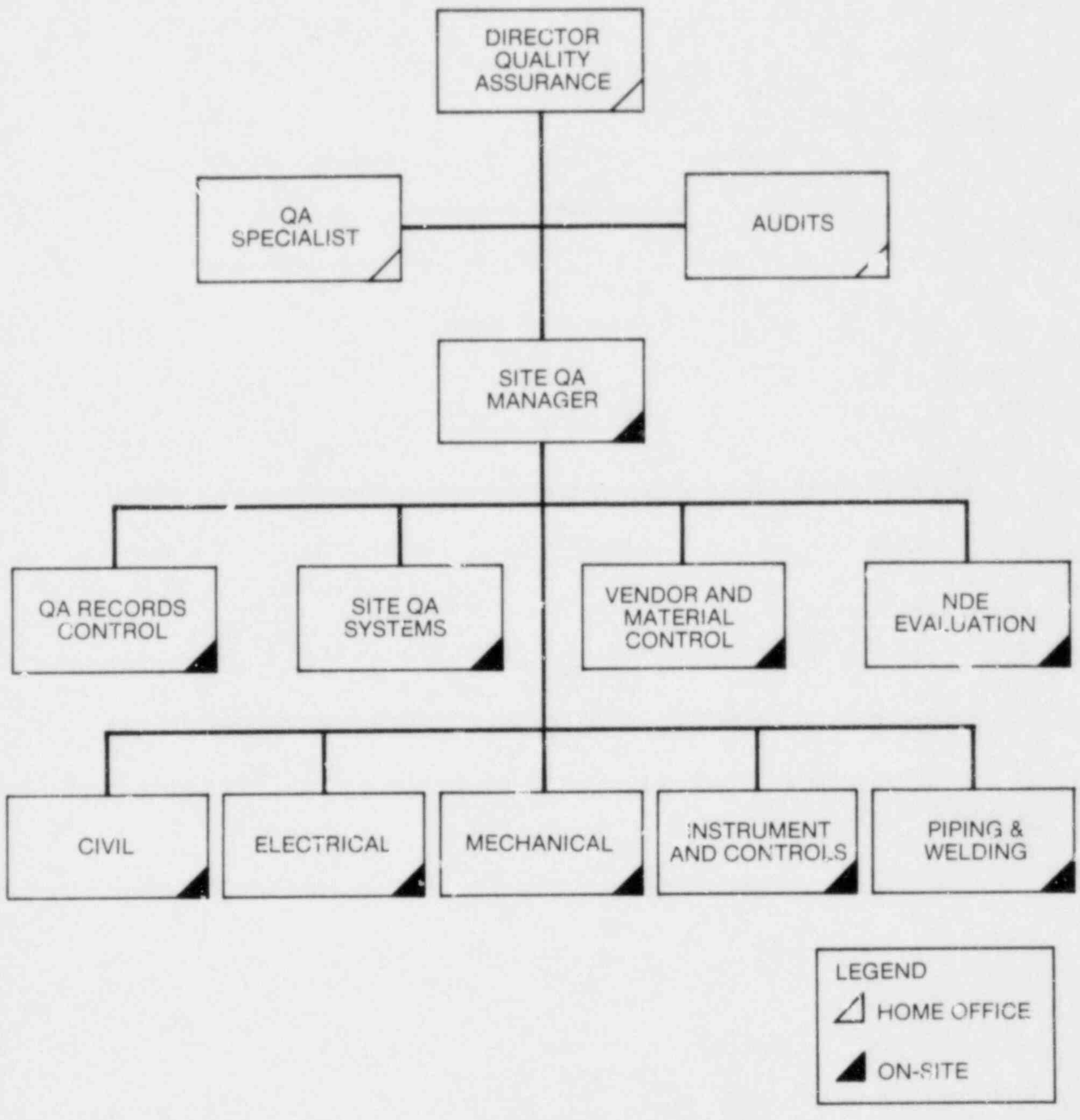
PRELIMINARY SAFETY ANALYSIS REPORT

PUGET/CONTRACTOR INTERFACE

FIGURE 17.1-2



PUGET SOUND POWER & LIGHT COMPANY
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PRELIMINARY SAFETY ANALYSIS REPORT

NESCO
QUALITY ASSURANCE
ORGANIZATION

FIGURE 17.1-3

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, 'Requirements for Auditing of QA Programs' (Draft 3, Rev. 4 - February, 1974)"

"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..." and replace with the word "(1978)."

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."

Page 22:

Replace the word "1973" with "1978" after the words "... ANSI N45.2.6" in subparagraph (2) of the first paragraph.

Add the following to the first paragraph, subparagraph (3), after the words "ANSI N45.2.12":

> ", Draft 3, Rev. 4 - February, 1974."

Add the following subparagraph (5) to the first paragraph:

> "Auditor Qualifications - Personnel performing audits will be qualified in accordance with the appropriate requirements of ANSI N45.2.23 - 1978. (Regulatory Guide 1.146)." 21

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 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 assignments are conducted at least once during the life of the contract. The 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." 21

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.