WASHINGTON NUCLEAR PROJECT NO. 2 (WNP-2)
Docket No. 50-397

A Summary of Conformance
to Nuclear Regulatory Commission
Regulations of 10 CFR Parts 20, 50 and 100

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

INTRODUCTION

This document has been prepared as a summary of the conformance of Washington Nuclear Project No. 2 (WNP-2) design and operation to the NRC regulations of 10 CFR Parts 20, 50 and 100. Those sections of regulations which specifically impose compliance requirements on licensees are addressed.

10 CFR 20.101 - RADIATION DOSE STANDARDS FOR INDIVIDUALS IN RESTRICTED AREAS

STATEMENT OF SECTION 20.101 - PARAGRAPH (a)

In accordance with the provisions of Section 20.102 - Paragraph (a), and except as provided in Paragraph (b) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of the standards specified in the following table:

		Rems per Calendar Quarter
1.	Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads	1-1/4
2.	Hands and forearms; feet and ankles	18-3/4
3.	Skin of Whole Body.	7-1/2

EVALUATION OF COMPLIANCE

The personnel occupational dose exposure limits at WNP-2 are the same as listed above.

STATEMENT OF SECTION 20.101 - PARAGRAPH (b)

A licensee may permit an individual in a restricted area to receive a total occupational dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) During any calendar quarter the total occupational dose to the whole body shall not exceed 3 rems; and

(2) The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday; and

(3) The licensee has determined the individual's accumulated occupational dose to the whole body on Form NRC-4, or on a clear and legible record containing all the information required in that form; and has otherwise complied with the requirements of Section 20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active blood-forming organs, head and trunk, or lens of eye.

EVALUATION OF COMPLIANCE

The external occupational exposure received by any one individual in a restricted area shall be limited to a whole body, head or trunk, active blood forming organs, gonads or lens of eyes dose of 3 rem per quarter where the lifetime accumulated dose is 5 (N-18), where N is the individual's age and the person's accumulated occupation dose has been determined by the execution of NRC Form 4, WP-186 or equivalent. However, as part of the Supply System commitment to ALARA, reasonable efforts shall be made to maintain plant personnel exposures at less than 5 rem per calendar year.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.101.

10 CFR 20.102 - DETERMINATION OF PRIOR DOSE

STATEMENT OF SECTION 20.102 - PARAGRAPH (a)

Each licensee shall require any individual, prior to first entry of the individual into the licensee's restricted area during each employment or work assignment under such circumstances that the individual will receive or is likely to receive in any period of one calendar quarter an occupational dose in excess of 25 percent of the applicable standards specified in Section 20.101 - Paragraph (a) and Section 20.104 - Paragraph (a), to disclose in a written, signed statement, either (1) that the individual had no prior occupational dose during the current calendar quarter, or (2) the nature and amount of any occupational dose which the individual may have received during that specifically identified current calendar quarter from sources of radiation possessed or controlled by other persons. Each licensee shall maintain records of such statements until the Commission authorizes their disposition.

EVALUATION OF COMPLIANCE

Health Physics Program, FSAR, Section 12.5

Personnel who enter controlled areas of WNP-2 facilities shall be provided, and required to use, appropriate radiation monitoring equipment. Results of personnel monitoring shall be entered into permanent records and made available to monitored individuals, supervisors, and Health Physics/Chemistry personnel.

An NRC form 4 or equivalent shall be on file prior to issue of a TLD. Monthly thermoluminescent dosimeter (TLD) badges shall be issued to personnel whose radiation exposure is expected to be greater than 300 mrem per quarter. Permanently assigned personnel whose radiation exposure is expected to be less than 300 mrem per quarter, may be issued quarterly TLD badges.

Visitors shall be issued TLD's when entering controlled areas.

STATEMENT OF SECTION 20.102 - PARAGRAPH (b)

Before permitting, pursuant to Section 20.101 - Paragraph (b), any individual in a restricted area to receive an occupational radiation dose in excess of the standards specified in Section 20.101 - Paragraph (a), each licensee shall:

(1) Obtain a certificate on Form NRC-4, or on a clear and legible record containing all information required in that form, signed by the individual showing each period of time after the individual attained the age of 18 in which the individual received an occupational dose of radiation; and (2) Calculate on Form NRC-4 in accordance with the instructions appearing therein, or on a clear and legible record containing all the information required in that form, the previously accumulated occupational dose received by the individual and the additional dose allowed for that individual under Section 20.101 - Paragraph (b).

EVALUATION OF COMPLIANCE

Prior to receiving a personnel TLD dosimeter, all individuals will complete a Form NRC-4 or equivalent. A TLD badge will not be issued if this form or equivalent is not completed.

Form NRC-4 will be filled out in accordance with instructions printed on the reverse side of the form, particularly:

a. If an individual has an exposure history, he is to attach it to the form; or

b. If an individual does not have his exposure history, an appropriate request for exposure history must be completed; or

c. If an individual has never been occupationally exposed to radiation, he is to enter NONE.

All Form NRC-4's or equivalents must be signed and dated in order to be valid.

STATEMENT OF SECTION 20.102 - PARAGRAPH (c)

(1) In the preparation of Form NRC-4, or on a clear and legible record containing all the information required in that form, the licensee shall make a reasonable effort to obtain reports of the individual's previously accumulated occupational dose. For each period for which the licensee obtains such reports, the licensee shall use the dose shown in the report in preparing the form. In any case where a licensee is unable to obtain reports of the individual's occupational dose for a previous complete calendar quarter, it shall be assumed that the individual has received the occupational dose specified in whichever of the following columns apply:

Part of Body	Column 1 Assumed exposure in rems for calendar quarters prior to Jan. 1, 1961	Column 2 Assumed exposure in rems for calendar quarters beginning on or after Jan. 1, 1961
Whole body, gonads, active blood forming organs, head and trunk, lens of eye.	3-3/4	1-1/4

(2) The licensee shall retain and preserve records used in preparing Form NRC-4 until the Commission authorizes their disposition.

If calculation of the individual's accumulated occupational dose for all periods prior to January 1, 1961 yields a result higher than the applicable accumulated dose value for the individual as of that date, as specified in paragraph (b) of Section 20.101, the excess may be disregarded.

EVALUATION OF COMPLIANCE

If a person has had previous occupational radiation exposure, a reasonable effort shall be made to obtain such exposure record from prior employment. When necessary, previous exposures shall be calculated in accordance with 10 CFR 20.102 Paragraph (c).

All personnel who have been issued a TLD badge at WNP-2 will have a Form NRC-4 or equivalent on file.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.102.

10 CFR 20.103 - EXPOSURE OF INDIVIDUALS TO CONCENTRATIONS OF RADIOACTIVE MATERIALS IN AIR IN RESTRICTED AREAS

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(1)

No licensee shall possess, use, or transfer licensed material in such a manner as to permit any individual in a restricted area to inhale a quantity of radioactive material in any period of one calendar quarter greater than the quantity which would result from inhalation for 40 hours per week for 13 weeks at uniform concentrations of radioactive material in air specified in Appendix B, Table I, Column 1. If the radioactive material is of such form that intake by absorption through the skin is likely, individual exposures to radioactive material shall be controlled so that the uptake of radioactive material by any organ from either inhalation or absorption or both routes of intake in any calendar quarter does not exceed that which would result from inhaling such radioactive material for 40 hours per week for 13 weeks at uniform concentrations specified in Appendix B, Table I, Column 1.

EVALUATION OF COMPLIANCE

Internal exposure of any individual in a restricted area during a calendar quarter shall be limited to the quantity which would result from inhalation of the occupational concentrations set forth in 10 CFR 20, Appendix B, Table I, Column 1, for 40 hours per week for 13 weeks.

Compliance with 10 CFR 20.103(a)(1) shall be determined by using suitable measurements of concentrations of radioactive materials in the air, and in liquids that may be ingested. In addition, measurements of radioactivity in the body and measurements of radioactivity excreted from the body shall be utilized for timely detection and assessment of individual uptakes of radioactivity by exposed individuals as appropriate.

Exposure shall be limited as far below the 40 MPC-hrs per week as specified in Appendix B, Table I, as is reasonably achievable using all practicable engineering and process controls.

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(2)

No licensee shall possess, use or transfer mixtures of U-234, U-235, and U-238 in soluble form in such a manner as to permit any individual in a restricted area to inhale a quantity of such material in excess of the intake limits specified in Appendix B, Table I, Column 1 of this part. If such soluble uranium is of a form such that absorption through the skin is likely, individual exposures to such material shall be controlled so that the uptake of such material by any organ from either inhalation or absorption or both routes of intake does not exceed that which would result from inhaling such material at the limits specified in Appendix B, Table I, Column 1 and footnote 4 thereto.

EVALUATION OF COMPLIANCE

Mixtures of U-234, U-235, and U-238 in soluble form are not present at WNP-2.

STATEMENT OF SECTION 20.103 - PARAGRAPH (a)(3)

For purposes of determining compliance with the requirements of this section the licensee shall use suitable measurements of concentrations of radioactive materials in air for detecting and evaluating airborne radioactivity in restricted areas and in addition, as appropriate, shall use measurements of radioactivity in the body, measurements of radioactivity excreted from the body, or any combination of such measurements as may be necessary for timely detection and assessment of individual intakes of radioactivity by exposed individuals. It is assumed that an individual inhales radioactive material at the airborne concentration in which he is present unless he uses respiratory protective equipment pursuant to paragraph (c) of this section. When assessment of a particular individual's intake of radioactive material is necessary, intakes less than those which would result from inhalation for 2 hours in any one day or for 10 hours in any one week at uniform concentrations specified in Appendix B, Table I, Column 1 need not be included in such assessment, provided that for any assessment in excess of these amounts the entire amount is included.

EVALUATION OF COMPLIANCE

Bioassay shall be used to aid in determining the extent of an individual's exposure to concentrations of radioactive materials. Measurements of radioactivity in the body (in vivo), and/or radioactivity excreted from the body (in vitro), shall be performed as necessary for timely evaluation of intakes of radioactivity by exposed individuals.

Health Physics personnel at WNP-2 will evaluate the respiratory protection requirements for an area based on air sampling data and/or contamination surveys. Respiratory protection equipment will then be selected to provide a protection factor greater than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in Table I, Column 1 of Appendix B to 10 CFR 20.

Internal exposure records shall include an assessment of inhaled or absorbed radioactivity when such activity, as determined by airborne measurements, exceeds the requirements of 10 CFR 20.103 (a)(3). When the individual exposure is less than the quantities specified in 10 CFR 20.103 (a)(3), no individual assessment is necessary.
Internal exposure records shall also include the internal exposure of individuals to radioactivity as determined by bioassay.

STATEMENT OF SECTION 20.103 - PARAGRAPH (b)(1)

The licensee shall, as a precautionary procedure, use process or other engineering controls, to the extent practicable, to limit concentrations

of radioactive materials in air to levels below those which delimit an airborne radioactivity area as defined in Section 20.203 - Paragraph (d)(1)(ii).

EVALUATION OF COMPLIANCE

The air cleaning systems which utilize special filtration equipment to limit airborne radioactive contaminants are:

1. Standby Gas Treatment Systems (SGTS) - described in 6.5 of FSAR.
2. Control Room Emergency Filtration System - described in 9.4 and 6.4 of FSAR.
3. Reactor Building Sump Vent Exhaust Filter System - described in 9.4 of FSAR
4. Radwaste Building Exhaust Filtration System - described in 9.4 of FSAR.

In addition, small, local absolute particulated filters are used to locally filter the effluent from sample sink hoods and chemical hoods. These small filter units are all described in 9.4 of FSAR.

STATEMENT OF SECTION 20.103 - PARAGRAPH (b)(2)

When it is impracticable to apply process or other engineering controls to limit concentrations of radioactive material in air below those defined in Section 20.203 - Paragraph (d)(1)(ii), other precautionary procedures, such as increased surveillance, limitation of working times, or provisions of respiratory protective equipment, shall be used to maintain intake of radioactive material by any individual within any period of seven consecutive days as far below that intake of radioactive material which would result from inhalation of such material for 40 hours at the uniform concentrations specified in Appendix B, Table I, Column 1 as is reasonably achievable. Whenever the intake of radioactive material by any individual exceeds this 40-hour control measure, the licensee shall make such evaluations and take such actions as are necessary to assure against recurrence. The licensee shall maintain records of such occurrences, evaluations, and actions taken in a clear and readily identifiable form suitable for summary review and evaluation.

EVALUATION OF COMPLIANCE

Evaluation of Compliance to Section 20.103 - Paragraph (b)(2)

Evaluation of airborne activity in restricted areas is discussed in the evaluation of compliance to Section 20.103 - Paragraph (a)(1).

Issuance and Selection of Respiratory Equipment

Health Physics personnel at WNP-2 will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits specified in Appendix B, Table I, Column 1.

Should an individual receive greater than 40 MPC hours in 7 consecutive days, an evaluation will be made to identify the cause and actions will be taken to prevent recurrence. Records will be maintained for each occurrence.

STATEMENT OF SECTION 20.103 - PARAGRAPH (c)

When respiratory protective equipment is used to limit the inhalation of airborne radioactive material pursuant to paragraph (b)(2) of this section, the licensee may make allowance for such use in estimating exposures of individuals to such materials provided that such equipment is used as stipulated in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection.

EVALUATION OF COMPLIANCE

Health Physics personnel at WNP-2 will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits.

The protection factors used at WNP-2 comply with the protection factors permitted under Regulatory Guide 8.15.

STATEMENT OF SECTION 20.103 - PARAGRAPH (e) AND (f)

(e) The licensee shall notify, in writing, the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D at least 30 days before the date that respiratory protective equipment is first used under the provisions of this section.

(f) A licensee who was authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976 shall bring his respiratory protective program into conformance with the requirements of paragraph (c) of this section within one year of that date, and is exempt from the requirement of paragraph (e) of this section.

EVALUATION OF COMPLIANCE

WNP-2 is in compliance with 10 CFR 20.103 - Paragraph (c) and therefore is exempt from the requirements of Paragraph (e).

CONCLUSIONS

WNP-2 is in compliance with 10 CFR 20.103.

10 CFR 20.104 - EXPOSURE OF MINORS

STATEMENT OF SECTION 20.104

(a) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area who is under 18 years of age, to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of 10 percent of the limits specified in the table in paragraph (a) of Section 20.101.

(b) No licensee shall possess, use or transfer licensed material in such a manner as to cause any individual within a restricted area, who is under 18 years of age to be exposed to airborne radioactive material possessed by the licensee in an average concentration in excess of the limits specified in Appendix B, Table II of this part. For purposes of this paragraph, concentrations may be averaged over periods not greater than a week.

(c) The provisions of Section 20.103 - Paragraphs (b)(2) and (c) shall apply to exposures subject to paragraph (b) of this section except that the references in Section 20.103 - Paragraphs (b)(2) and (c) to Appendix B, Table I, Column 1 shall be deemed to be references to Appendix B, Table II, Column 1.

EVALUATION OF COMPLIANCE

Exposure of individuals under the age of 18 years shall be limited to 0.5 rems per year external exposure and one-tenth of 10 CFR 20, Appendix B, Table I, Columns 1 and 2.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.104.

10 CFR 20.105 - PERMISSIBLE LEVELS OF RADIATION IN UNRESTRICTED AREAS

STATEMENT OF SECTION 20.105 - PARAGRAPHS (a) and (b)

(a) There may be included in any application for a license or for amendment of a license proposed limits upon levels of radiation in unrestricted areas resulting from the applicant's possession or use of radioactive material and other sources of radiation. Such applications should include information as to anticipated average radiation levels and anticipated occupancy times for each unrestricted area involved. The Commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem.

(b) Except as authorized by the Commission pursuant to paragraph (a) of this section, no licensee shall possess, use or transfer licensed material in such a manner as to create in any unrestricted area from radioactive material and other sources of radiation in his possession:

(1) Radiation levels which, if any individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour, or

(2) Radiation levels which, if an individual were continously present in the area, could result in his receiving a dose in excess of 100 millirems in any seven consecutive days.

EVALUATION OF COMPLIANCE

The WNP-2 Health Physics Procedures are written to prevent the spread of contamination outside of control areas. The Procedures are also written to prevent the creation of radiation levels in unprotected areas in excess of the limits of 10 CFR 20.105

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.105.

10 CFR 20.106 - RADIOACTIVITY IN EFFLUENTS TO UNRESTRICTED AREAS

STATEMENT OF SECTION 20.106 - PARAGRAPHS (a) AND (d)

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B", Table II of this part, except as authorized pursuant to Section 20.302 of paragraph (b) of this section. For purposes of this section concentrations may be averaged over a period not greater than one year.

(d) For the purposes of this section the concentration limits in Appendix "B", Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe or similar conduit may be determined with respect to the point where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion, or decay between the point of discharge and boundary.

EVALUATION OF COMPLIANCE

WNP-2 Radiological Safety Technical Specifications, Sections 3.11 and 3.12

The WNP-2 Radiological Safety Technical Specifications provide the limits and conditions for discharging radioactive liquids and gases from WNP-2 so that 10 CFR 20 limitations are not exceeded.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.106.

10 CFR 20.202 - PERSONNEL MONITORING

STATEMENT OF SECTION 20.202 - PARAGRAPH (a)

Each licensee shall supply appropriate personnel monitoring equipment to, and shall require the use of such equipment by:

(1) Each individual who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 25 percent of the applicable value specified in paragraph (a) of Section 20.101.

(2) Each individual under 18 years of age who enters a restricted area under such circumstances that he receives, or is likely to receive, a dose in any calendar quarter in excess of 5 percent of the applicable value specified in pargraph (a) of Section 20.101.

(3) Each individual who enters a high radiation area.

EVALUATION OF COMPLIANCE

The requirements of 10 CFR 20.202 are met by WNP-2 Health Physics procedures and the WNP-2 Health Physics Program, FSAR, Section 12.5.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.202.

10 CFR 20.203 - CAUTION SIGNS, LABELS, SIGNALS AND CONTROLS

STATEMENT OF SECTION 20.203 - PARAGRAPH (a)(1)

Except as otherwise authorized by the Commission, symbols prescribed by this section shall use the conventional radiation caution colors (magenta or purple on yellow background). The symbol prescribed by this section is the conventional three-bladed design:

EVALUATION OF COMPLIANCE

All signs designating a radiation zone shall have a yellow background with magenta or purple lettering. At least one conventional magenta-colored, three-bladed radiation symbol must appear on each sign.

STATEMENT OF SECTION 20.203 - PARAGRAPH (a)(2)

In addition to the contents of signs and labels prescribed in this section, licensees may provide on or near such signs and labels any additional information which may be appropriate in aiding individuals to minimize exposure to radiation or to radioactive material.

EVALUATION OF COMPLIANCE

Additional information may be provided on or near such signs in order to aid an individual to minimize his exposure to radiation or radioactive materials.

STATEMENT OF SECTION 20.203 - PARAGRAPH (b)

Radiation Areas. Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION (OR DANGER)
RADIATION AREA

EVALUATION OF COMPLIANCE

Each radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words CAUTION RADIATION AREA.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(1)

High Radiation Areas. Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION (OR DANGER)
HIGH RADIATION AREA

EVALUATION OF COMPLIANCE

Each high radiation area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words DANGER--HIGH RADIATION AREA.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(2)

Each entrance or access point to a high radiation area shall be:

(i) Equipped with a control device which shall cause the level of radiation to be reduced below that at which an individual might receive a dose of 100 millirems in 1 hour upon entry into the area; or

(ii) Equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry; or

(iii) Maintained locked except during periods when access to the area is required, with positive control over each individual entry.

EVALUATION OF COMPLIANCE

The proposed WNP-2 Technical Specifications, Paragraph 6.12, contains the following requirement with regard to access to high radiation areas:

6.12 HIGH RADIATION AREA (OPTIONAL)

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

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*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual, i.e., qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the unit Health Physicist in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the unit Health Physicist. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

Acknowledgement is made that subsequent negotiations on the WNP-2 Technical Specifications could change this proposal.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(3)

The controls required by paragraph (c)(2) of this paragraph shall be established in such a way that no individual will be prevented from leaving a high radiation area.

**Measurement made at 18" from source of radioactivity.

EVALUATION OF COMPLIANCE

Locks and keys are administratively controlled. The locks on high radiation area doors provide one-way control; i.e., they permit free egress from inside the room or enclosure.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(4)

In the case of a high radiation area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted for the controls required by subparagraph (2) of this paragraph.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System (Supply System) presently complies with subparagraph (2) and will comply with subparagraph (4) for temporary high radiation areas as the need arises. The Supply System does acknowledge this requirement and will comply with it if the requirements of (2) cannot be met.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(5)

Any licensee, or applicant for a license, may apply to the Commission for approval of methods not included in subparagraphs (2) and (4) of this paragraph for controlling access to high radiation areas. The Commission will approve the proposed alternatives if the licensee or applicant demonstrates that the alternative methods of control will prevent unauthorized entry into a high radiation area, and that the requirement of subparagraph (3) of this paragraph is met.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System (Supply System) presently complies with subparagraph (2) and will comply with subparagraph (4) for temporary high radiation areas as the need arises. Furthermore, the Supply System will comply to options for controlling access to high radiation areas as delineated in the plant technical specifications.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(6)

Each area in which there may exist radiation levels in excess of 500 rems in one hour at one meter from a sealed radioactive source that is used to irradiate materials shall:

(1) Have each entrance or access point equipped with entry control devices which shall function automatically to prevent any individual from inadvertently entering the area when such radiation levels exist; permit deliberate entry into the area only after a control device is actuated that shall cause the radiation level within the area, from the sealed source, to be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and prevent operation of the source if the source would produce radiation levels

in the area that could result in a dose to an individual in excess of 100 mrem in one hour. The entry control devices required by this paragraph (c)(6) shall be established in such a way that no individual will be prevented from leaving the area.

(ii) Be equipped with additional control devices such that upon failure of the entry control devices to function as required by paragraph (c)(6)(i) of this section the radiation level within the area, from the sealed source, shall be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make an individual attempting to enter the area aware of the hazard and the licensee or at least one other individual, who is familiar with the activity and prepared to render or summon assistance, aware of such failure of the entry control devices.

(iii) Be equipped with control devices such that upon failure or removal of physical radiation barriers other than the source's shielded storage container the radiation level from the source shall be reduced below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour; and visible and audible alarm signals shall be generated to make potentially affected individuals aware of the hazards and the licensee or at least one other individual, who is familiar with the activity and prepared to render or summon assistance, aware of the failure or removal of the physical barrier. When the shield for the stored source is a liquid, means shall be provided to monitor the integrity of the shield and to signal, automatically, loss of adequate shielding. Physical radiation barriers that comprise permanent structural components, such as walls, that have no credible probability of failure or removal in ordinary circumstances need not meet the requirements of this paragraph (c)(6)(iii).

(iv) Be equipped with devices that will automatically generate visible and audible alarm signals to alert personnel in the area before the source can be put into operation and in sufficient time for any individual in the area to operate a clearly identified control device which shall be installed in the area and which can prevent the source from being put into operation.

(v) Be controlled by use of such administrative procedure and such devices as are necessary to assure that the area is cleared of personnel prior to each use of the source preceeding which use it might have been possible for an individual to have entered the area.

(vi) Be checked by a physical radiation measurement to assure that prior to the first individual's entry into the area after any use of the source, the radiation level from the source in the area is below that at which it would be possible for an individual to receive a dose in excess of 100 mrem in one hour.

(vii) Have entry control devices required in paragraph (c)(6)(i) of this section which have been tested for proper functioning prior to initial

operation with such source of radiation on any day that operations are not uninterruptedly continued from the previous day or before resuming operations after any unintended interruption, and for which records are kept of the dates, times, and results of such tests of function. No operations other than those necessary to place the source in safe condition or to effect repairs on controls shall be conducted with such source unless control devices are functioning properly. The licensee shall submit an acceptable schedule for more complete periodic tests of the entry control and warning systems to be established and adhered to as a condition of the license.

(viii) Have those entry and exit portals that are used in transporting materials to and from the irradiation area, and that are not intended for use by individuals, controlled by such devices and administrative procedures as are necessary to physically protect and warn against inadvertent entry by any individual through such portals. Exit portals for processed materials shall be equipped to detect and signal the presence of loose radiation sources that are carried toward such an exit and to automatically prevent such loose sources from being carried out of the area.

EVALUATION OF COMPLIANCE

WNP-2 does not possess or intend to use such a source. Therefore, the requirements of this subparagraph are not applicable.

STATEMENT OF SECTION 20.203 - PARAGRAPH (c)(7)

Licensees with, or applicants for licenses for radiation sources that are within the purview of paragraph (c)(6) of this section, and that must be used in a variety of positions or in peculiar locations, such as open fields or forests, that make it impracticable to comply with certain requirements of paragraph (c)(6) of this section, such as those for the automatic control of radiation levels, may apply to the Director, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, for approval prior to use of safety measures that are alternative to those specified in paragraph (c)(6) of this section, and that will provide at least an equivalent degree of personnel protection in the use of such sources. At least one of the alternative measures must include an entry-preventing interlock control based on a physical measurement of radiation that assures the absence of high radiation levels before an individual can gain access to an area where such sources are used.

EVALUATION OF COMPLIANCE

This requirement is not applicable to WNP-2.

STATEMENT OF SECTION 20.203 - PARAGRAPH (d)

Airborne Radioactivity Areas. (1) As used in the regulations in the Part, "Airborne radioactivity area" means (i) any room, enclosure, or operating area in which airborne radioactive materials, composed wholly

or partly of licensed material, exist in concentrations in excess of the amounts specified in Appendix "B", Table I, Column 1 of this Part; or (ii) any room, enclosure, or operating area in which airborn radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25% of the amounts specified in Appendix "B", Table I, Column 1 of this part.

(2) Each airborne radioactivity area shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION
AIRBORNE RADIOACTIVITY AREA

EVALUATION OF COMPLIANCE

An area in which airborne radioactive materials exist in concentrations which exceeds 25% of the concentrations specified in 10 CFR 20, Appendix B, Table I, shall be posted and include the words "CAUTION, AIRBORNE RADIOACTIVITY AREA".

STATEMENT OF SECTION 20.203 - PARAGRAPH (e)

Additional Requirements: (1) Each area or room in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding 10 times the quantity of such material specified in Appendix "C" of this Part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION
RADIOACTIVE MATERIAL(S)

(2) Each area or room in which natural uranium or thorium is used or stored in an amount exceeding one-hundred times the quantity specified in Appendix "C" of this Part shall be conspicuously posted with a sign or signs bearing the radiation caution symbol and the words:

CAUTION
RADIOACTIVE MATERIAL(S)

EVALUATION OF COMPLIANCE

Any area or room except those areas specifically exempted by 10CFR 20.204 in which licensed material is used or stored and which contains any radioactive material (other than natural uranium or thorium) in an amount exceeding ten times the quantity of such material specified in 10 CFR 20, Appendix C, shall be posted and include the words "CAUTION, RADIOACTIVE MATERIAL."

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(1)

Containers. (1) Except as provided in subparagraph (3) of this paragraph, each container of licensed material shall bear a durable, clearly visible label identifying the radioactive contents.

EVALUATION OF COMPLIANCE

Radiation Protection General Procedures

Containers intended for offsite disposal must be numbered, weighed, surveyed, and labelled when placed in the temporary storage area.

Inventory and Leak Test of Radiation Sources

Each container of nonexempt radioactive sources shall have a durable, clearly visible label identifying the radioactive contents.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(2)

A label required pursuant to subparagraph (1) of this paragraph shall bear the radiation caution symbol and the words "CAUTION, RADIOACTIVE MATERIAL" or "DANGER, RADIOACTIVE MATERIAL". It shall also provide sufficient information to permit individuals handling or using the containers, or working in the vicinity thereof, to take precautions to avoid or minimize exposures.

EVALUATION OF COMPLIANCE

Radioactive material labels shall have a yellow background with magenta or purple lettering and at least one conventional magenta three-bladed radiation symbol on each label.

Space is provided on the label for insertion of an explanatory message.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(3)

Notwithstanding the provisions of subparagraph (1) of this paragraph, labeling is not required.

(i) For containers that do not contain licensed materials in quantities greater than the applicable quantitites listed in Appendix C of this part.

(ii) For containers containing only natural uranium or thorium in quantities no greater than ten times the applicable quantitites listed in Appendix C of this part.

(iii) For containers that do not contain licensed materials in concentrations greater than the applicable concentrations listed in Column 2, Table I, Appendix B of this part.

(iv) For containers when they are attended by an individual who takes the precautions necessary to prevent the exposure of any individual to radiation or radioactive materials in excess of the limits established by the regulations in this part.

(v) For containers when they are in transport and packaged and labeled in accordance with regulations of the Department of Transportation.

(vi) For containers which are accessible only to individuals authorized to handle or use them, or to work in the vicinity thereof, provided that the contents are identified to such individuals by a readily available written record.

(vii) For manufacturing or process equipment, such as nuclear reactors, reactor components, piping, and tanks.

EVALUATION OF COMPLIANCE

WNP-2 recognizes these exceptions to the labelling requirements for containers.

STATEMENT OF SECTION 20.203 - PARAGRAPH (f)(4)

Each licensee shall, prior to disposal of an empty container to unrestricted areas, remove or deface the radioactive material label or otherwise clearly indicate that the container no longer contains radioactive materials.

EVALUTAION OF COMPLIANCE

Signs will warn of an existing radiological hazard and will be promptly removed when no longer required.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.203.

10 CFR 20.205 - PROCEDURES FOR PICKING UP, RECEIVING, AND OPENING PACKAGES

STATEMENT OF SECTION 20.205 - PARAGRAPH (a)(1)

Each licensee who expects to receive a package containing quantitites of radioactive material in excess of the Type A quantitites specified in paragraph (b) of this section shall:

(i) If the package is to be delivered to the licensee's facility by the carrier, make arrangements to receive the package when it is offered for delivery by the carrier; or

(ii) If the package is to be picked up by the licensee at the carrier's terminal, make arrangements to receive notification from the carrier of the arrival of the package, at the time of arrival.

EVALUATION OF COMPLIANCE

Only authorized individuals shall receive, store or ship radioactive materials. These individuals shall be responsible for verification of license limitations and notifying the Health Physics/Chemistry Manager as far in advance as possible of such operations.

STATEMENT OF SECTION 20.205 - PARAGRAPH (a)(2)

Each licensee who picks up a package of radioactive material from a carrier's terminal shall pick up the package expeditiously upon receipt of notification from the carrier of its arrival.

EVALUATION OF COMPLIANCE

A radioactive material shipment to be picked up from a carrier's terminal shall be made as expeditiously as possible upon notification by the carrier.

STATEMENT OF SECTION 20.205 - PARAGRAPH (b)(1) (SUMMARY)

Each licensee, upon receipt of a package of radioactive material, shall monitor the external surfaces of the package for radioactive contamination caused by leakage of the radioactive contents, except as noted in this paragraph.

The monitoring shall be performed as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or eighteen hours if received after normal working hours.

EVALUATION OF COMPLIANCE

Upon arrival of the shipment, Health Physics/Chemistry personnel shall be notified immediately so that surveys can be made of the vehicle transporting the radioactive material and the package containing the material.

STATEMENT OF SECTION 20.205 - PARAGRAPH (b)(2)

If removable radioactive contamination in excess of 0.01 microcuries (22,000 disintegration per minute) per 100 square centimeters of package surface is found on the external surfaces of the package, the licensee shall immediately notify the final delivering carrier and, by telephone and telegraph, mailgram, or facsimile, the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office shown in Appendix D.

EVALUATION OF COMPLIANCE

The shipment shall be surveyed before unloading to assess contamination levels. The survey shall include radiation and contamination surveys as well as checks for irregularities in packaging and labeling. The surveys shall be documented in the Radioactive Materials Receipt and Inventory Log.

All 10 CFR 20 reporting requirements are complied with per WNP-2 procedures.

STATEMENT OF SECTION 20.205 - PARAGRAPH (c)(1)

Each licensee, upon receipt of a package containing quantities of radioactive material in excess of the Type A quantities specified in paragraph (b) of this section, other than those transported by exclusive use vehicle, shall monitor the radiation levels external to the package. The package shall be monitored as soon as practicable after receipt, but no later than three hours after the package is received at the licensee's facility if received during the licensee's normal working hours, or 18 hours if received after normal working hours.

EVALUATION OF COMPLIANCE

Upon arrival of the shipment, Health Physics personnel shall be notified immediately so that surveys can be made of the vehicle transporting the material and the package containing the material.

STATEMENT OF SECTION 20.205 - PARAGRAPH (c)(2)

If radiation levels are found on the external surface of the package in excess of 200 millirem per hour, or at three feet from the external surface of the package in excess of 10 millirem per hour, the licensee shall immediately notify, by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D, and the final delivering carrier.

EVALUATION OF COMPLIANCE

The shipment shall be surveyed before unloading to assess exposure rates. Health Physics Supervisor shall be notified if the exposure rate exceeds 200 millirem per hour at contact with the package and/or 10 millirem per hour 3 feet from the package.

All 10 CFR 20 reporting requirements are complied with per WNP-2 procedures.

STATEMENT OF SECTION 20.205 - PARAGRAPH (d)

Each licensee shall establish and maintain procedures for safely opening packages in which licensed material is received, and shall assure that such procedures are followed and that due consideration is given to special instructions for the type of package being opened.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System (Supply System) presently complies with paragraph (d) and maintains the referenced procedures in the Plant Procedures Manual, Volume 11.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.205.

10 CFR 20.206 - INSTRUCTION OF PERSONNEL

STATEMENT OF SECTION 20.206

Instructions required for individuals working in or frequenting any portion of a restricted area are specified in Section 19.12 of this chapter (which is provided below).

STATEMENT OF SECTION 19.12

Instructions to workers. All individuals working or frequenting any portion of a restricted area shall be kept informed of the storage, transfer, or use of radioactive materials or of radiation in such portions of the restricted area; shall be instructed in the health protection problems associated with exposure to such radioactive materials or radiation, in precautions or procedures to minimize exposure, and in the purposes and functions of protection devices employed; shall be instructed in, and instructed to observe, to the extent within the worker's control, the applicable provisions of Commission regulations and licenses for the protection of personnel from exposure to radiation or radioactive materials occuring in such areas; shall be instructed of their responsibility to report promptly to the licensee any condition which may lead to or cause a violation of Commission regulations and licenses or unnecessary exposure to radiation or to radioactive material; shall be instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radition or radioactive material; and shall be advised as to the radiation exposure reports which workers may request pursuant to Section 19.13. The extent of these instructions shall be commensurate with potential radiological health protection problems in the restricted area.

EVALUATION OF COMPLIANCE

Each individual is trained to minimize his exposure consistent with discharging his duties. Each individual is responsible for observing rules adopted for his safety and that of others.

Health Physics personnel evaluate radiological conditions of operations and establish the procedures to be followed by all personnel. They ensure that all applicable regulations are complied with and that the required radiation protection records are adequately maintained.

All individuals receiving personnel dosimeters must have received radiation training commensurate with the potential radiological health protection problems associated with their work assignment.

All personnel entering a radioactive materials area are required to wear the protective clothing specified by Health Physics personnel. The clothing requirements are established based on evaluation of the radiological conditions of the area.

If it is determined by fixed and/or portable radiation monitoring devices that radiation from, or within, the station is such that permissible exposures in restricted and unrestricted areas will be exceeded if occupancy of these areas is continued, the evacuation alarm is sounded, the unit is shut down, and all personnel not essential to the emergency shutdown procedures immediately assemble at a safe location in accordance with WNP-2 emergency procedures.

The evaluation of 10 CFR 20.409 discusses the notification of individuals pursuant to 10 CFR 19.13.

Guidance for work performed in controlled areas is provided in the WNP-2 Health Physics Procedures.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.206.

10 CFR 20.207 - STORAGE AND CONTROL OF LICENSED MATERIALS IN UNRESTRICTED AREAS

STATEMENT OF SECTION 20.207

(a) Licensed materials stored in an unrestricted area shall be secured from unauthorized removal from the place of storage.

(b) Licensed materials in an unrestricted area and not in storage shall be tended under the constant surveillance and immediate control of the licensee.

EVALUATION OF COMPLIANCE

Licensed materials are not stored outside the restricted area at WNP-2 site.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.207.

10 CFR 20.301 - WASTE DISPOSAL GENERAL REQUIREMENT

STATEMENT OF SECTION 20.301

No licensee shall dispose of licensed material except:

(a) By transfer of an authorized recipient as provided in the regulations in Part 30, 40, or 70 of this Chapter, whichever may be applicable; or

(b) As authorized pursuant to Section 20.302; or

(c) As provided in Section 20.303 or 20.304, applicable respectively to the disposal of licensed material by release into sanitary sewage systems or burial in soil, or in Section 20.106 (Radioactivity in effluents to unrestricted areas).

EVALUATION OF COMPLIANCE

Procedures addressing the "Offsite Shipment of Radioactive Material" require WNP-2 personnel to verify that the consignee is licensed to receive the type and amount of radioactive material being shipped.

Paragraph (b) is not applicable for WNP-2.

Section 20.304 pertains to burial of radioactive waste and, therefore, is not applicable for evaluation. The evaluation of compliance for Sections 20.106 and 20.303 was presented previously.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.301.

10 CFR 20.303 - DISPOSAL BY RELEASE INTO SANITARY SEWERAGE SYSTEMS

STATEMENT OF SECTION 20.303

No licensee shall discharge licensed material into a sanitary sewerage system unless:

(a) It is readily soluble or dispersible in water; and

(b) The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

(1) The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee, will result in an average concentration equal to the limits specified in Appendix "B", Table I, Column 2 of this Part; or

(2) Ten times the quantity of such material specified in Appendix "C" of this Part;

(c) The quantity of any licensed or other radioactive material released in any one month, if diluted by the average monthly quantity of water released by the licensee, will not result in an average concentration exceeding the limits specified in Appendix "B", Table I, Column 2 of this Part; and

(d) The gross quantity of licensed and other radioactive material, excluding hydrogen-3 and carbon-14, released into the sewerage system by the licensee does not exceed one curie per year. The quantities of hydrogen-3 and carbon-14 released into the sanitary sewerage system may not exceed 5 curies per year for carbon-3 and 1 curie per year for carbon-14. Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitation contained in this section.

EVALUATION OF COMPLIANCE

Licensed material discharged to the sanitary sewage system, if done, is done in compliance with this section.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.303.

10 CFR 20.401 - RECORDS OF SURVEYS RADIATION MONITORING, AND DISPOSAL

STATEMENT OF 20.401 - PARAGRAPH (a)

Each licensee shall maintain records showing the radiation exposures of all individuals for whom personnel monitoring is required under paragraph 20.202 of the regulations in this part. Such records shall be kept on Form NRC-5 in accordance with the instructions contained in that form or on clear and legible records containing all the information required by Form NRC-5. The doses entered on the forms or records shall be for periods of time not exceeding one calendar quarter.

EVALUATION OF COMPLIANCE

Records of exposure are maintained for each individual for whom personnel monitoring is required by 10 CFR 20.202. Such a record is kept on a form containing all the information required by NRC Form 5.

STATEMENT OF SECTION 20.401 - PARAGRAPH (b)

Each licensee shall maintain records in the same units used in this part showing the results of surveys required by 20.201(b); monitoring required by 20.205(b) and 20.205(c); and disposals made under 20.302, 20.303, and 20.304.

EVALUATION OF COMPLIANCE

See evaluations of the sections that are referenced in Section 20.401(b).

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(1)

Records of individual exposure to radiation and to radioactive material which must be maintained pursuant to the provisions of paragraph (a) of this section and records of bioassays, including results of whole body counting examinations, made pursuant to 20.108, shall be preserved until the Commission authorizes disposition.

EVALUATION OF COMPLIANCE

Whole body counts are performed on a routine schedule. The results of these counts shall be entered into the individual's permanent record and shall be preserved until the commission authorizes disposition.

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(2)

Records of the results of surveys and monitoring which must be maintained pursuant to paragraph (b) of this section shall be preserved for two years after completion of the survey except that the following records shall be maintained until the Commission authorizes their disposition: (i) records of the results of surveys to determine compliance with 20.103(a); (ii) in the absence of personnel monitoring data, records of the results of

surveys to determine external radiation dose; and (iii) records of the results of surveys used to evaluate the release of radioactive effluents to the environment.

EVALUATION OF COMPLIANCE

Routine survey and monitoring records are kept on file or microfilmed until the Commission authorizes their disposition. In the absence of personnel monitoring data, records of the results of surveys to determine external radiation dose become part of a worker's permanent file.

STATEMENT OF SECTION 20.401 - PARAGRAPH (c)(3)

Records of disposal of licensed material made pursuant to Sections 20.302, 20.303, or 20.304 shall be maintained until the Commission authorizes their disposition.

EVALUATION OF COMPLIANCE

Sources are inventoried and handled according to WNP-2 procedure. Records are either filmed or maintained on hard copy.

STATEMENT OF SECTION 20.401 - PARAGRAPHS (c)(4) and (c)(5)

(4) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations.

(5) If there is a conflict between the Commission's regulations in this part, license conditions, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to 20.501, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

EVALUATION OF COMPLIANCE

Records are maintained in compliance with these regulations.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.401.

10 CFR 20.402 - REPORTS OF THEFT OR LOSS OF LICENSED MATERIAL

STATEMENT OF SECTION 20.402

(a) Each licensee shall report by telephone to the Director of the appropriate Nuclear Regulatory Commission Inspection and Enforcement Regional Office listed in Appendix D, immediately after its occurrence becomes known to the licensee, any loss or theft of licensed material in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas.

(b) Each licensee who is required to make a report pursuant to paragraph (a) of this section shall, within thirty (30) days after he learns of the loss or theft, make a report in writing to the appropriate NRC Regional Office listed in Appendix D with copies to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, setting forth the following information:

(1) A description of the licensed material involved, including kind, quantity, chemical, and physical form;

(2) A description of the circumstances under which the loss or theft occurred;

(3) A statement of disposition or probable disposition of the licensed material involved;

(4) Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas:

(5) Actions which have been taken, or will be taken to recover the material; and

(6) Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

(c) Subsequent to filing the written report, the licensee shall also report any substantive additional information on the loss or theft which becomes available to the licensee, within 30 days after he learns of such information.

(d) Any report filed with the Commission pursuant to this section shall be so prepared that names of individuals who may have received exposure to radiation are stated in a separate part of the report.

EVALUATION OF COMPLIANCE

The disposal and inventory of radiation sources for WNP-2 will be in accordance with the requirements of Section 20.402.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.402.

10 CFR 20.403 - NOTIFICATION OF INCIDENTS

STATEMENT OF SECTION 20.403 - PARAGRAPH (a)

Immediate notification. Each licensee shall immediately notify by telephone or telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving byproduct, source, or special nuclear material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation; or

(2) The release of radioactive material in concentration which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one working week or more of the operation of any facilities affected; or

(4) Damage to property in excess of $200,000.

EVALUATION OF COMPLIANCE

In case of the incidents delineated in Section 20.403 - Paragraphs (a)(1), (a)(2), (a)(3), and (a)(4) the Supply System will provide immediate notification to the NRC.

STATEMENT OF SECTION 20.403 - PARAGRAPH (b), (c) and (d)

Twenty-four hour notification. Each licensee shall within 24 hours notify by telephone and telegraph, mailgram, or facsimile, the Director of the appropriate NRC Regional Office listed in Appendix D of any incident involving licensed material possessed by him and which may have caused or threatens to cause:

(1) Exposure of the whole body of any individual to 5 rems or more radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms to 75 rems or more radiation; or

(2) The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II; or

(3) A loss of one day or more of the operation of any facilities affected; or

(4) Damage to property in excess of $2,000.

(c) Any report filed with the Commission pursuant to this section shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

(d) For nuclear power reactors licensed under Paragraph 50.21 or Paragraph 50.22, the incidents included in paragraph (a) and paragraph (b) in this section shall in addition be reported pursuant to Paragraph 50.72.

EVALUATION OF COMPLIANCE

In case of the incidents specified in Section 20.403 - Paragraphs (b)(1), (b)(2), (b)(3), and (b)(4), WNP-2 will provide twenty-four hour notification to the NRC.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.403.

10 CFR 20.405 - REPORTS OF OVEREXPOSURES AND EXCESSIVE LEVELS AND CONCENTRATIONS

STATEMENT OF SECTION 20.405 - PARAGRAPH (a)

(a) In addition to any notification required by 20.403, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Appendix D with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, of:

(1) each exposure of an individual to radiation in excess of the applicable limits in Sections 20.101 or 20.104(a) or the license; (2) each exposure of an individual to radioactive material in excess of the applicable limits in Sections 20.103(a)(1), 20.103(a)(2), 20.104(b), or the license; (3) levels of radiation or concentrations of radioactive material in a restricted area in excess of any other applicable limit in the license; (4) any incident for which notification is required by Section 20.403; and (5) levels of radiation or concentration of radioactive material (whether or not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any applicable limit set forth in this part or in the license. Each report required under this paragraph shall describe the extent of exposure of individuals to radition or to radioactive material, including estimates of each individual's exposure as required by paragraph (b) of this section; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

EVALUATION OF COMPLIANCE

In the case of an overexposure or excessive levels and concentrations, notification is made in compliance with the specifications. WNP-2 Health Physics Program requires compliance with 10 CFR 20.

STATEMENT OF SECTION 20.405 - PARAGRAPH (b) AND (c)

(b) Any report filed with the Commission pursuant to this section shall include for each individual exposed the name, social security number, and date of birth; and an estimate of the individual's exposure. The report shall be prepared so that this information is stated in a separate part of the report.

(c) In addition to any notification required by Section 20.403, each licensee shall make a report in writing within 30 days to the appropriate NRC Regional Office listed in Appendix D, with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission , Washington, D. C. 20555, of levels of radiation or releases of radioactive material in excess of limits specified by 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations", or in excess of license conditions related to compliance with 40 CFR Part 190. Each report required under this paragraph shall describe the extent of exposure

of individuals to radiation or to radioactive materials; levels of radiation or releases of radioactive material in excess of limits specified by 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," or in excess of license conditions related to compliance with 40 CFR Part 190. Each report required under this paragraph shall describe the extent of exposure of individuals to radiation or to radioactive material; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence, including the schedule for achieving conformance with 40 CFR Part 190 and associated license conditions.

EVALUATION OF COMPLIANCE

Notification will be made in compliance with these specifications.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.405.

10 CFR 20.407 - PERSONNEL MONITORING REPORTS

STATEMENT OF SECTION 20.407

Each person described in Section 20.408 of this part shall, within the first quarter of each calendar year, submit to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years. All other persons specifically licensed by the Commission shall, within the first quarter of calendar years 1979 and 1980, submit to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, the reports specified in paragraphs (a) and (b) of this section covering the preceding calendar years 1978 and 1979.

(a) A report of either (1) the total number of individuals for whom personnel monitoring was required under 20.202(a) or 34.33(a) of this chapter during the calendar year; or (2) the total number of individuals for whom personnel monitoring was provided during the calendar year: Provided, however, that such total includes at least the number of individuals required to be reported under paragraph (a)(1) of this section. The report shall indicate whether it is submitted in accordance with paragraph (a)(1) or (a)(2) of this section. If personnel monitoring was not required to be provided to any individual by the licensee under 20.202(a) or 34.33(a) of this chapter during the calendar year. The licensee shall submit a negative report indicating that such personnel monitoring was not required.

(b) A statistical summary report of the personnel monitoring information recorded by the licensee for individuals for whom personnel monitoring was either required or provided as described in paragraph (a) of this section, indicating the number of individuals whose total whole body exposure recorded during the previous calendar year was in each of the following estimated exposure ranges:

Estimated Whole Body Exposure Range (REMS)*	Number of Individuals in Each Range
No measurable exposure.	. . . .
Measurable exposure less than 0.1	. . . .
0.1 to 0.25	. . . .
0.25 to 0.5	. . . .
0.5 to 0.75	. . . .
0.75 to 1	. . . .
1 to 2.	. . . .
2 to 3.	. . . .
3 to 4.	. . . .
4 to 5.	. . . .
5 to 6.	. . . .
6 to 7.	. . . .
7 to 8.	. . . .
8 to 9.	. . . .
9 to 10	. . . .
10 to 11.	. . . .
11 to 12.	. . . .
12 +.	. . . .

*Individual values exactly equal to the values separating exposure ranges shall be reported in the higher range.

The low exposure range data are required in order to obtain better information about the exposure actually recorded. This section does not require improved measurements.

EVALUATION OF COMPLIANCE

The Supply System provides the reports required for radiation exposure statistics at WNP-2. These statistics are in the prescribed format.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.407.

10 CFR 20.408 - REPORTS OF PERSONNEL MONITORING ON TERMINATION OF EMPLOYMENT

STATEMENT OF SECTION 20.408

(a) This section applies to each person licensed by the Commission to:

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to 50.21(b) or 50.22 of this chapter or a testing facility as defined in 50.2(r) of this chapter.

(2) Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter;

(3) Possess or use at any one time, for purposes of fuel processing, fabrication, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof pursuant to Part 70 of this chapter; or

(4) Possess high-level radioactive waste at a geologic respository operations area pursuant to Part 60 of this chapter.

(5) Possess or use at any one time, for processing or manufacturing for distribution pursuant to Part 30, 32, or 33 of this chapter, byproduct material in quantities exceeding any one of the following quantities.

Radionuclide*	Quantity in Curies
Cesium-137.	1
Cobalt-60	1
Gold-198.	100
Iodine-131.	1
Iridium-192	10
Krypton-85.	1,000
Promethium-147.	10
Technetium-99m.	1,000

*The Commission may require, as a license condition, or by rule, regulation or order pursuant to Section 20.502, reports from licensees who are licensed to use radionuclides not on this list, in quantities sufficient to cause comparable radiation levels.

(b) When an individual terminates employment with a licensee described in paragraph (a) of this section, or an individual assigned to work in

such a licensee's facility but not employed by the licensee, completes the work assignment in the licensee's facility, the licensee shall furnish to the Director of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, C. C. 20555, a report of the individual's exposures to radiation and radioactive material, incurred during the period of employment or work assignment in the licensee's facility, containing information recorded by the licensee pursuant to Sections 20.401(a) and 20.108. Such report shall be furnished within 30 days after the exposure of the individual has been determined by the licensee or 90 days after the date of termination of employment or work assignment, whichever is earlier.

EVALUATION OF COMPLIANCE

A report is generated in accordance with Section 20.408.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.408.

10 CFR 20.409 - NOTIFICATION AND REPORTS TO INDIVIDUALS

STATEMENT OF SECTION 20.409

(a) Requirements for notifications and reports to individuals of exposure to radiation or radioactive material are specified in Section 19.13 of this chapter.

(b) When a licensee is required pursuant to Sections 20.405 or 20.408 to report to the Commission any exposure of an individual to radiation or radioactive material, the licensee shall also notify the individual. Such notice shall be transmitted at a time not later than the transmittal to the Commission, and shall comply with the provisions of Section 19.13(a) of this chapter.

EVALUATION OF COMPLIANCE

A copy of the letter generated in accordance with Section 20.408 is sent to the subject worker at the time of submittal to the NRC.

CONCLUSION

WNP-2 is in compliance with 10 CFR 20.409.

10 CFR 50.34 - CONTENTS OF APPLICATION: TECHNICAL INFORMATION

STATEMENT OF SECTION 50.34 - PARAGRAPH (b)

Final safety analysis report. Each application for a license to operate a facility shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design basis, and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. . . .

EVALUATION OF COMPLIANCE

The WNP-2 Final Safety Analysis Report (FSAR) was initially submitted on March 24, 1977, and was docketed (No. 50-397) on June 22, 1978, pursuant to Section 50.34 of 10 CFR 50. This document and its numerous amendments have been undergoing review by the NRC and its staff.

STATEMENT OF SECTION 50.34 - PARAGRAPH (c)

Physical security plan. Each application for a license to operate a production or utilization facility shall include a physical security plan. The plan shall consist of two parts. Part I shall address vital equipment, vital areas, and isolation zones, and shall demonstrate how the applicant plans to comply with the requirements of Part 73 of this chapter, if applicable, at the proposed facility. Part II shall list tests, inspections, and other means to be used to demonstrate compliance with such requirements, if applicable.

EVALUATION OF COMPLIANCE

The physical security plan for WNP-2 was submitted to the Nuclear Regulatory Commission (NRC) on December 21, 1978. This plan has been accepted and approved by NRC (see NRC internal memo from McCorkle to Schwencer dated November 6, 1981.)

STATEMENT OF SECTION 50.34 - PARAGRAPH (d)

Safeguards contingency plan. Each application for a license to operate a production or utilization facility that shall be subject to Sections 73.50, 73.55, or 73.60 of this chapter shall include a licensee safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR Part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and industrial sabotage, as defined in Part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information, as set forth in Appendix C to 10 CFR Part 73, are Background, Generic

Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval).

EVALUATION OF COMPLIANCE

The safeguards contingency plan for WNP-2 was submitted to the NRC on March 23, 1979. This plan has been accepted and approved by NRC (see NRC internal memo from McCorkle and Schwencer dated November 6, 1981).

CONCLUSION

WNP-2 has addressed and complied with the requirements of 10 CFR 50.34.

10 CFR 50.34a - DESIGN OBJECTIVES FOR EQUIPMENT TO CONTROL RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS - NUCLEAR POWER REACTORS

STATEMENT SECTION 50.34a - PARAGRAPH (a)

An application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable. The term "as low as is reasonably achievable" as used in this part means as low as is reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations and in relation to the utilization of atomic energy in the public interest. The guides set out in Appendix I provide numerical guidance on design objectives for light water-cooled nuclear power reactors to meet the requirement that radioactive material in effluents releases to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit application was filed on August 19, 1971. WNP-2 compliance to Appendix I is documented in the FSAR Chapter 11 and ER Section 5.2.

STATEMENT OF SECTION 50.34 - PARAGRAPH (b)

Each application for a permit to construct a nuclear power reactor shall include:

1. A description of the preliminary design of equipment to be installed pursuant to Paragraph (a) of this section:

2. An estimate of:

   (i) The quantity of each of the principal radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations; and

   (ii) The quantity of each of the principal radionuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal operations.

3. A general description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

EVALUATION OF COMPLIANCE

A description of the design of the subject equipment installed at WNP-2 is included in the FSAR, Sections 11.2, 11.3, and 11.4.

STATEMENT OF SECTION 50.34a - PARAGRAPH (c)

Each application for a license to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, pursuant to paragraph (a) of this section; and (2) a revised estimate of the information required in paragraph (b)(2) of this section if the expected releases and exposures differ significantly from the estimates submitted in the application for a construction permit.

EVALUATION OF COMPLIANCE

The information required in the paragraph concerning the description of equipment is contained in the FSAR as discussed in the Evaluation of Compliance for Paragraphs (a) and (b).

The estimated annual releases of principal radionuclides in gaseous and liquid effluents are presented in the FSAR, Tables 11.2-9 and 11.3-9 .

The procedures for control of gaseous and liquid effluents are contained in the proposed WNP-2 Radiological Effluent Technical Specifications. NRC review of these items is not yet complete.

CONCLUSION

WNP-2 has addressed and complied with the requirements of 10 CFR 50.34a as documented in Chapter 11 of the FSAR, proposed Radiological Effluent Technical Specifications on effluents.

10 CFR 50.36 - TECHNICAL SPECIFICATIONS

STATEMENT OF SECTION 50.36 - PARAGRAPHS (a) AND (b)

(a) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(b) Each license authorizing operation of a production or utilization facility of a type described in Section 50.21 or Section 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to Section 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

EVALUATION OF COMPLIANCE

The WNP-2 applications for operating licenses incorporate the technical specifications requirements of 10 CFR 50.36 - Paragraphs (a) and (b) above. Drafts of these technical specifications were submitted to NRC for review in February 1982.

STATEMENT OF SECTION 50.36 - PARAGRAPH (c)

Technical Specifications will include items in the following categories:

(1) Safety limits, limiting safety system settings, and limiting control settings;

(2) Limiting conditions for operation;

(3) Surveillance requirements;

(4) Design Features; and

(5) Administrative controls.

EVALUATION OF COMPLIANCE

All of the above listed categories are contained in the current draft technical specifications. The technical specifications will be a "living" document. Amendments, therefore, will be necessary to maintain the document current with NRC requirements, as well as with plant modifications for improved operations.

CONCLUSION

The proposed technical specifications for WNP-2 meet the requirements of 10 CFR 50.36.

10 CFR 50.36a - TECHNICAL SPECIFICATIONS ON EFFLUENTS FROM NUCLEAR POWER REACTORS

STATEMENT OF SECTION 50.36a - PARAGRAPHS (a) AND (b)

(a) In order to keep releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable, each licensee authorizing operation of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of Section 20.106 of this chapter, require:

1. The operating procedures developed pursuant to Section 50.34a(c) for the control of effluents be established and followed and that equipment installed in the radioactive waste system pursuant to Section 50.34a(a) be maintained and used.

2. The submission of a report to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter within sixty (60) days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous six (6) months of operation, and such other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases. Copies of such report shall be sent to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. If quantities of radioactive materials released during the reporting period are significantly above design objectives, the report shall cover this specifically. On the basis of such report and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

(b) In establishing and implementing the operating procedures described in paragraph (a) of this section, the licensee shall be guided by the following considerations: Experience with the design, construction and operation of nuclear power reactors indicates that compliance with the technical specifications described in this section will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in Section 20.106 of this chapter and in the operating license. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small percentages, but still within the limits specified in Section 20.106 of this chapter and the operating license. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as is reasonably achievable. The

guides set out in Appendix I provide numerical guidance on limiting conditions for operation for light water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept as low as is reasonably achievable.

EVALUATION OF COMPLIANCE

The operating procedures pursuant to paragraph 50.34a(c) are contained in the proposed WNP-2 Radiological Effluent Technical Specifications.

The WNP-2 requirements for reporting semiannual releases are included in the WNP-2 Radiological Effluent Technical Specifications and are in conformance with the above requirements.

CONCLUSION

The proposed WNP-2 Radiological Effluent Technical Specifications is in compliance with the requirements of Section 50.36a - paragraphs (a) and (b).

10 CFR 50.44 - STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER COOLED POWER REACTORS

STATEMENT OF SECTION 50.44 - PARAGRAPH (a)

Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, shall, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA), by (1) metal-water reaction involving the fuel cladding and the reactor coolant, (2) radiolytic decomposition of the reactor coolant, and (3) corrosion of metals.

EVALUATION OF COMPLIANCE

The following modes of hydrogen gas control are available at WNP-2:

a. Hydrogen mixing system and hydrogen recombiners,

b. Containment inerting system and

c. Containment purge system, if necessary.

STATEMENT OF SECTION 50.44 - PARAGRAPH (b)

Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding shall be provided with the capability for (1) measuring the hydrogen concentration in the containment, (2) insuring a mixed atmosphere in the containment, and (3) controlling combustible gas concentrations in the containment following a postulated LOCA.

EVALUATION OF COMPLIANCE

A hydrogen concentration monitoring system measures the amount of hydrogen in the drywell and suppression chamber atmosphere. A hydrogen mixing system is provided to ensure a well mixed atmosphere in the drywell and suppression chamber. The following methods are available to control combustible gas concentration in containment following a postualted LOCA. These are: use of hydrogen recombiner, use of the containment purge system and the inerting of the primary containment.

STATEMENT OF SECTION 50.44 - PARAGRAPH (c)

For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (1) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (2) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function. If

neither of these conditions can be shown, the containment shall be provided with an inerted atmosphere or an oxygen deficient condition in order to provide protection against hydrogen burning and explosions during this time period.

EVALUATION OF COMPLIANCE

The analysis of hydrogen generation presented in the FSAR Section 6.2.5.3 determined that the uncontrolled hydrogen concentration in the drywell reaches the lower flammability limit of 4% by volume approximately 10 hours after the occurrence of a LOCA if the hydrogen recombiner is not in operation. The Supply System has committed to inert the WNP-2 containment.

The hydrogen recombiner is permanently installed and can be brought online to control hydrogen concentration in the containment below the lower flammability limit.

STATEMENT OF SECTION 50.44 - PARAGRAPH (d)

(1) For facilities that are in compliance with Section 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with Section 50.46 (b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.0023 inch (0.0058mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs. (2) For facilities as to which no evaluation of compliance in accordance with Section 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reactor of 5 percent of the mass metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume.

EVALUATION OF COMPLIANCE

The analysis performed to determine hydrogen generation at WNP-2 was done in accordance with the most restrictive requirements of Section 50.44 - paragraph (d). This analysis assumed that the extent of metal-water reaction (percentage of fuel cladding that reacts with water) was the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel to a depth of .0023 inch.

STATEMENT OF SECTION 50.44 - PARAGRAPH (e)

For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for controlled

purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas control system, such as recombiners, that does not result in a significant release from containment.

EVALUATION OF COMPLIANCE

WNP-2 is equipped with hydrogen recombiners and a containment inerting system as primary means for controlling combustible gases following a LOCA. In addition, a containment purge system for controlled purging of the containment is available as a backup control system.

STATEMENT OF SECTION 50.44 - PARAGRAPH (f)

For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970. . . .

EVALUATION OF COMPLIANCE

Notice of hearing on the application for a construction permit for WNP-2 was published after November 5, 1970.

STATEMENT OF SECTION 50.44 - PARAGRAPH (g)

For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization if a repressurization system is provided) and the postulated LOCA calculated in accordance with Section 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

EVALUATION OF COMPLIANCE

Notice of hearing on the application for a construction permit for WNP-2 was published after December 22, 1968.

CONCLUSION

The combustible gas control systems at WNP-2 meet the requirements of 10 CFR 50.44, however, subparagraph 50.44(g) does not apply because of the later application date.

10 CFR 50.46 - ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS

STATEMENT OF SECTION 50.46

EVALUATION OF COMPLIANCE

Compliance with 10 CFR 50.46 is documented in the FSAR, Section 6.3.3. This analysis shows that WNP-2 meets 10 CFR 50.46 criteria and the ECCS equipment will perform its function in an acceptable manner.

CONCLUSION

WNP-2 meets 10 CFR 50.46 criteria.

10 CFR 50.47 - EMERGENCY PLANS

STATEMENT OF SECTION 50.47

(a) (1) No operating license for a nuclear power reactor will be issued unless a finding is made by NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

(2) The NRC will base its finding on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and capable of being implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on a question of adequacy.

(b) The onsite and offsite emergency response plans for nuclear power reactors must meet the following standards. These standards are addressed by specific criteria in NUREG-0654; FEMA-REP-1 entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants--For Interim Use and Comment" January 1980.

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

(3) Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

(5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

(6) Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

(7) Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contract with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

(9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

(10) A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

(11) Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

(12) Arrangements are made for medical services for contaminated injured individuals.

(13) General plans for recovery and reentry are developed.

(14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilitues, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

(15) Radiological emergency response training is provided to those who may be called on to assist in an emergency.

(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

(c) (1) Failure to meet the standards set forth in paragraph (b) of this subsection may result in the Commission declining to issue an Operating License; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operation.

(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZ's surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZ's also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

EVALUATION OF COMPLIANCE

Refer to evaluation of compliance for Appendix E to 10 CFR 50.

10 CFR 50.48 - FIRE PROTECTION

STATEMENT OF SECTION 50.48

(a) Each operating nuclear power plant shall have a fire protection plan that satisfies Criterion 3 of Appendix A to this part. This fire protection plan shall describe the overall fire protection program for the facility, identify the various positions within the licensee's organization that are responsible for the program, state the authorities that are delegated to each of these positions to implement those responsibilities, and outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan shall also describe specific features necessary to implement the program described above, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities, automatic and manually operated fire detection and suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.

(b) Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate prior to January 1, 1979. Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1 reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position BTP APCSB 9.5-1 was published in August 1976. With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate prior to January 1, 1979 shall satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.

(c) All fire protection modifications require to satisfy the provisions of Appendix R to this part or directly affected by such requirements shall be completed on the following schedule:

(1) Those fire protection features that involve revisions of administrative controls, manpower changes, and training, shall be implemented within 30 days after the effective date of this section and Appendix R to this part.

(2) Those fire protection features that involve installation of modifications that do not require prior NRC approval or plant shutdown shall be implemented within 9 months after the effective date of this section and Appendix R to this part.

(3) Those fire protection features, except for those requiring prior NRC approval by paragraph (c)(5) of this section, that involve installation of modifications that do require plant shutdown, the need for which is justified in the plans and schedules required by the provisions of paragraph (c)(5) of this section, shall be implemented before startup after the earliest of the following events commencing 180 days or more after the effective date of this section and Appendix R to this part:

(i) the first refueling outage;

(ii) another planned outage that lasts for at least 60 days, or

(iii) an unplanned outage that lasts for at least 120 days.

(4) Those fire protection features that require prior NRC approval by paragraph (c)(5) of this section, shall be implemented within the following schedule: Dedicated shutdown systems--30 months after NRC approval; modifications requiring plant shutdown--before startup after the earliest of the events given in paragraph (c)(3) commencing 180 days after NRC approval; modifications not requiring plant shutdown--6 months after NRC approval.

(5) Licensees shall make any modifications necessary to comply with these requirements in accordance with the above schedule without prior review and approval by NRC except for modifications required by Section III.G.3 of Appendix R to this part. Licensees shall submit plans and schedules for meeting the provisions of paragraphs (c)(2), (c)(3), and (c)(4) within 30 days after the effective date of this section and Appendix R to this part. Licensees shall submit design descriptions of modifications needed to satisfy Section III.G.3 of Appendix R to this part within 30 days after the effective date of this section and Appendix R to this part.

(6) In the event that a request for exemption from a requirement to comply with one or more of the provisions of Appendix R filed within 30 days of the effective date of this rule is based on an assertion by the licensee that such required modifications would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety, the schedule requirements of paragraph (c) shall be tolled until final Commission action on the exemption request upon a determination by the Director of Nuclear Reactor Regulation that the licensee has provided a sound technical basis for such assertion that warrants further staff review of the request.

(d) Fire protection features accepted by the NRC staff in Fire Protection Safety Evaluation Reports referred to in paragraph (b) of this section and supplements to such reports, other than features covered by paragraph (c), shall be completed as soon as practicable but no later than the completion date currently specified in license conditions or technical

specifications for such facility, or the date determined by paragraphs (d)(1) through (d)(4) of this section, whichever is sooner, unless the Director of Nuclear Reactor Regulation determines, upon a showing by the licensee, that there is good cause for extending such date and that the public health and safety is not adversely affected by such extension. Extensions of such date shall not exceed the dates determined by paragraphs (c)(1) through (c)(4) of this section.

(1) Those fire protection features that involve revisions of administrative controls, manpower changes and training shall be implemented within 4 months after the date of the NRC staff Fire Protection Evaluation Report accepting or requiring such features.

(2) Those fire protection features involving installation of modifications not requiring prior approval or plant shutdown shall be implemented within 12 months after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features.

(3) Those fire protection features, including alternative shutdown capability, involving installation of modifications requiring plant shutdown shall be implemented before the startup after the earliest of the following events commencing 9 months or more after the date of the NRC staff Fire Protection Safety Evaluation Report accepting or requiring such features:

(i) The first refueling outage;

(ii) Another planned outage that lasts for at least 60 days, or

(iii) An unplanned outage that lasts for at least 120 days

(4) Those fire protection features involving dedicated shutdown capability requiring new buildings and systems shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.

(e) Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of Appendix A to this part in accordance with the provisions of their licenses.

EVALUATION OF COMPLIANCE

Refer to evaluation of compliance for Appendix R to 10 CFR 50.

10 CFR 50.54 - CONDITIONS OF LICENSES

STATEMENT OF SECTION 50.54 - PARAGRAPHS (a) THROUGH (h)

Whether stated therein or not, the following shall be deemed conditions in every license issued:

(a) (Deleted, effective March 9, 1967 (32 F. R. 2562.)

(b) No right to the special nuclear material shall be conferred by the license except as may be defined by the license.

(c) Neither the license, nor any right thereunder, nor any right to utilize or produce special nuclear material shall be transferred, assigned, or disposed of in any manner, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall, after securing full information, find that the transfer is in accorance with the provisions of the Act and give its consent in writing.

(d) The license shall be subject to suspension and to the rights of recapture of the material or control of the facility reserved to the Commission under Section 108 of the Act in a state of war or national emergency declared by Congress.

(e) The license shall be subject to revocation, suspension, modification, or amendment for cause as provided in the Act and regulations, in accordance with the procedures provided by the Act and regulations.

(f) The licensee will at any time before expiration of the license, upon request of the Commission submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not the license shall be modified, suspended or revoked.

(g) The issuance or existence of the license shall not be deemed to waive, or relieve the license from compliance with, the antiturst laws, as specified in subsection 105a of the Act. In the event that the licensee should be found by a court of competent jurisdiction to have violated any provision of such antitrust laws in the conduct of the licensed activity, the Commission may suspend or revoke the license or take such other action with respect to it as shall be deemed necessary.

(h) The license shall be subject to the provisions of the Act now or hereafter in effect and to all rules, regulations, and orders of the Commission. The terms and conditions of the license shall be subject to amendment, revision, or modification, by reason of amendments of the Act or by reason of rules, regulations and orders issued in accordance with the terms of the Act.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System acknowledges the above listed conditions and complies with them.

STATEMENT OF SECTION 50.54 - PARAGRAPH (i)

Except as provided in Section 55.9 of this chapter, the licensee shall not permit the manipulation of the controls or any facility by anyone who is not a licensed operator or senior operator as provided in Part 55 of this chapter.

EVALUATION OF COMPLIANCE

A minimum of one trained licensed Nuclear Station Operator will be assigned responsibility for the operation of WNP-2 Operating reactor each shift. No persons other than licensed operators will be permitted to manipulate the controls of WNP-2 operating reactor.

STATEMENT OF SECTION 50.54 - PARAGRAPH (i-1)

Within three (3) months after issuance of an operating license, the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A of Part 55 of this Chapter. Notwithstanding the provisions of Section 50.59 the licensee shall not, except as specifically authorized by the Commission, make a change in an approved operator qualification program by which the scope, time allotted for the program or frequency in conducting different parts of the program is decreased. Holders of operating licenses in effect on September 17, 1973, shall implement an operator requalification program which, as a minimum, meets the requirements of Appendix A of Part 55 of this chapter which was submitted for approval by the Atomic Energy Commission.

EVALUATION OF COMPLIANCE

Following initial licensing of shift personnel, a retraining program implementing the requirements of 10 CFR 55, Appendix A shall be initiated to maintain the knowledge level and operating proficiency of licensed personnel. The retraining program cycle will be based on a two year period with training distributed fairly evenly over that period.

STATEMENT OF SECTION 50.54 - PARAGRAPHS (j) AND (k)

(j) Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of a reactor, shall be manipulated only with the knowledge and consent of an operator or senior operator licensed pursuant to Part 55 of this chapter present at the controls.

(k) An operator or senior operator licensed pursuant to Part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

EVALUATION OF COMPLIANCE

The minimum shift crew composition is presented in FSAR Table 13.1-2 and complies with the requirement of this paragraph.

STATEMENT OF SECTION 50.54 - PARAGRAPH (l)

The licensee shall designate individuals to be responsible for directing the licensed activities of licensed operators. These individuals shall be licensed as senior operators pursuant to Part 55 of this chapter.

EVALUATION OF COMPLIANCE

The Shift Manager and Control Room Supervisor, who must be licensed Senior Operators, shall be responsible for directing the licensed activities of subordinate licensed operators.

STATEMENT OF SECTION 50.54 - PARAGRAPH (m)

A senior operator licensed pursuant to Part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial startup and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

EVALUATION OF COMPLIANCE

A senior reactor operator is assigned to each shift and is present or available in accordance with the requirements of this section.

STATEMENT OF SECTION 50.54 - PARAGRAPH (n)

The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or construction permit pursuant to Section 50.36.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System acknowledges and complies with this requirement. All proposed changes pursuant to 10 CFR 50.59 are reviewed to verify that such actions do not constitute an unreviewed safety question.

STATEMENT OF SECTION 50.54 - PARAGRAPH (o)

Primary reactor containment for water cooled power reactors shall be subject to the requirements set forth in Appendix J.

EVALUATION OF COMPLIANCE

WNP-2 complies with requirements of 10 CFR 50 Appendix J as discussed in section 6.2.6 of the FSAR.

STATEMENT OF SECTION 50.54 - PARAGRAPH (p)

The licensee shall prepare and maintain safeguards contingency plan Procedures in accordance with Appendix C of 10 CFR Part 73 for effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan. The licensee may make no change which would decrease the effectiveness of a security plan prepared pursuant to Section 50.34(c) or Part 73 of this chapter, or of the first four categories of information (Background, Generic Planning Base, Licensee Planning Base, Responsibility Matrix) contained in a licensee safeguards contingency plan prepared pursuant to Section 50.34(d) or Part 73, as applicable, without prior approval of the Commission. A licensee desiring to make such a change shall submit an application for an amendment to his license pursuant to Section 50.90. The licensee may make changes to the security plan or to the safeguards contingency plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of changes to the plans made without prior Commission approval for a period of two years from the date of the change, and shall furnish to the Director of Nuclear Material Safety and Safeguards (for enrichment and reprocessing facilities) or to the Director of Nuclear Reactor Regulation (for nuclear reactors), U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate NRC Regional Office specified in Appendix A of Part 73 of this chapter, a report containing a description of each change within two months after the change is made. Prior to the safeguards contingency plan being put into effect, the licensee shall have:

(1) All safeguards capabilities specified in the safeguards contingency plan available and functional.

(2) Detailed Procedures developed according to Appendix C to Part 73 available at the licensee's site, and

(3) All appropriate personnel trained to respond to safeguards incidents as outlined in the plan and specified in the detailed Procedures.

The licensee shall provide for the development, revision, implementation, and maintenance of his safeguards contingency plan. To this end, the licensee shall provide for a review at least every 12 months of the safeguards contingency plan by individuals independent of both security program management and personnel who have direct responsibility for implementation of the security program. The review shall include a review and audit of safeguards contingency procedures and practices, an audit of the security system testing and maintenance program, and a test of the safeguards system along with commitments establish for response by local law

enforcement authorities. The results of the review and audit, along with recommendations for improvements, shall be documented, reported to the licensee's corporate and plant management, and kept available at the plant for inspection for a period of two years.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System acknowledges and complies with this requirement. Changes to the security plan are maintained in the Supply System file.

STATEMENT OF SECTION 50.54 - PARAGRAPH (q)

(q) A licensee authorized to possess and/or operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in section 50.47(b) and the requirements in Appendix E of this Part. A licensee authorized to possess and/or operate a research reactor or a fuel facility shall follow and maintain in effect emergency plans which meet the requirements in Appendix E of this Part. The nuclear power reactor licensee may make changes to these plans without Commission approval only if such changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of Section 50.47(b) and the requirements of Appendix E of this Part. The research reactor licensee and/or the fuel facility licensee may make changes to these plans without Commission approval only if such changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the requirements of Appendix E of this Part. Proposed changes that decrease the effectiveness of the approved emergency plans shall not be implemented without application to and approval by the Commission. The licensee shall furnish 3 copies of each proposed change for approval; and/or if a change is made without prior approval, 3 copies shall be submitted within 30 days after the change is made or proposed to the Director of the appropriate NRC regional office specified in Appendix D, 10 CFR Part 20, with 10 copies to the Director of Nuclear Reactor Regulation, or, if appropriate, the Director of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555.

EVALUATION OF COMPLIANCE

Refer to evaluation of compliance for Appendix E to 10 CFR 50.

STATEMENT OF SECTION 50.54 - PARAGRAPH (r)

(r) Each licensee who is authorized to possess and/or operate a research or test reactor facility with an authorized power level greater than or equal to 500 kW thermal, under a license of the type specified in section 50.21(c), shall submit emergency plans complying with 10 CFR Part 50, Appendix E, to the Director of Nuclear Reactor Regulation for approval within one year from the effective date of this rule. Each licensee who is authorized to possess and/or operate a research reactor facility with an authorized power level less than 500 kW thermal, under a

license of the type specified in section 50.21(c), shall submit emergency plans complying with 10 CFR Part 50, Appendix E, to the Director of Nuclear Reactor Regulation for approval within two years from the effective date of this amendment.

EVALUATION OF COMPLIANCE

This paragraph is not applicable to WNP-2.

STATEMENT OF SECTION 50.54 - PARAGRAPH (s)

(s) (1) Each licensee who is authorized to possess and/or operate a nuclear power reactor shall submit to NRC within 60 days of the effective date of this amendment the radiological emergency response plans of State and local governmental entities in the United States that are wholly or partially within a plume exposure pathway EPZ, as well as the plans of State governments wholly or partially within an ingestion pathway EPZ.[1, 2] Ten (10) copies of the above plans shall be forwarded to the Director of Nuclear Reactor Regulation with 3 copies to the Director of the appropriate NRC regional office. Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs for a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway EPZ shall focus on such actions as are appropriate to protect the food ingestion pathway.

(2) For operating power reactors, the licensee, State, and local emergency response plans shall be implemented by April 1, 1981, except as provided in Section IV, D.3 of Appendix E of this Part. If, after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown or other enforcement action is appropriate, the Commission shall take into account, among other factors, whether the licensee can demonstrate to the Commission's satisfaction that the deficiencies in the plan are not significant for the plant in question, or that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons for continued operation.

(3) The NRC will base its finding on a review of the FEMA findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the licensee's emergency plans are adequate and capable of being implemented. Nothing in the paragraph shall be construed as limiting the authority of the Commission to take action under any other regulation or authority of the Commission or at any time other than that specified in this paragraph.

EVALUATION OF COMPLIANCE

Refer to evaluation of compliance for Appendix E to 10 CFR 50.

STATEMENT OF SECTION 50.54 - PARAGRAPH (t)

(t) A nuclear power reactor licensee shall provide for the development, revision, implementation, and maintenance of its emergency preparedness program. To this end, the licensee shall provide for a review of its emergency preparedness program at least every 12 months by persons who have no direct responsibility for implementation of the emergency preparedness program. The review shall include an evaluation for adequacy of interfaces with State and local governments and of licensee drills, exercises, capabilities, and procedures. The results of the review, along with recommendations for improvements, shall be documented, reported to the licensee's corporate and plant management, and retained for a period of five years. The part of the review involving the evaluation for adequacy of interface with State and local governments shall be available to the appropriate State and local governments.

EVALUATION OF COMPLIANCE

Washington Public Power Supply System acknowledges and will comply with this requirement.

STATEMENT OF SECTION 50.54 -PARAGRAPH (u)

(u) Within 60 days after the effective date of this amendment, each nuclear power reactor licensee shall submit to the NRC plans for coping with emergencies that meet standards in section 50.47(b) and the requirements of Appendix E of the Part.

EVALUATION OF COMPLIANCE

Refer to evaluation of compliance for Appendix E to 10 CFR 50.

CONCLUSION

WNP-2 complies with the requirements of 10 CFR 50.54.

SECTION 50.55a - CODES & STANDARDS

STATEMENT OF SECTION 50.55a - PARAGRAPH (a)(1)

Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

EVALUATION OF COMPLIANCE

Structures, systems and components are classified in FSAR Table 3.2-1 using the following Quality Class Designations:

a. Quality Class I - Any nuclear system, structure, subassembly, component or design characteristics that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. All engineered safeguards fall within this category. All Quality Class I items meet the applicable provisions of 10 CFR Part 50 Appendix B.

b. Quality Class II - Any system, structure, subassembly, component or design characteristic which could cause a safety hazard to plant personnel, an extended reduction in unit output, an unscheduled unit trip, or equipment damage. Appropriate quality assurance requirements for these items are assigned in the purchase specifications.

c. Quality Class G - Any non-nuclear system, structure, subassembly, component or design characteristic to which quality assurance requirements are assigned in accordance with the consequences of failure, operating costs or procurement costs.

WNP-2 structures, systems and components are designed to the codes and standards listed in the applicable FSAR Tables.

STATEMENT OF SECTION 50.55a - PARAGRAPH (a)(2)

As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), (f), (g), and (i) of this section shall meet the requirements described in those paragraphs, except that the American Society of Mechanical Engineers (hereinafter referred to as ASME) Code N-symbol need not be applied, and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (h) of this section, except as authorized by the Commission or the Atomic Energy Commission upon demonstration by the applicant for or holder of a construction permit that:

(i) Design, fabrication, installation, testing of the specified system or component, is to the maximum extent practical, in accordance with generally recognized codes and standards, and compliance

with the requirements described in paragraphs (c) through (i) of this section or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety; or

(ii) Proposed alternatives to the described requirements or portions thereof will provide an acceptable level of quality and safety. For example, the use of inspection or survey systems other than those required by the specified ASME Codes and Addenda may be authorized under this subparagraph provided that an acceptable level of quality and safety in design, fabrication, installation, and testing is achieved.

EVALUATION OF COMPLIANCE

An evaluation of compliance is provided for each of the CFR 50.55a paragraphs (c), (d), (e), (f), (h), and (i) referenced in paragraph (a)(2) above.

STATEMENT OF SECTION 50.55a - PARAGRAPH (b)

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1977 Edition and addenda through the Summer 1979 Addenda.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1 and include editions through the 1977 Edition and addenda through the Summer 1979 Addenda, subject to the following limitations and modifications:

(i) Applicability of specific editions and addenda. When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must be used.

(ii) Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB-2500 and IWB-2500-1 and Category B-J). If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB-2500 and Table IWB-2600, Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) Steam generator tubing (modifies Article IWB-2000). If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications.

(iv) Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC-2520 or IWC-2520-1, Category C-F).

(A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, shall be examined. The extent of examination for these systems shall be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code.

(B) For a nuclear power plant whose application for a construction permit is docketed prior to July 1, 1978, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph, IWC-1220, Table IWC-2520 Category C-F and C-G and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

EVALUATION OF COMPLIANCE

WNP-2 was designed and constructed to the ASME Code editions and addenda described in the evaluation of compliance for 10 CFR 50.55a--Paragraphs (c), (d), (e), (f), and (g) below. The WNP-2 construction permit was issued on March 19, 1973. Therefore, the applicable ASME Section XI Code is the 1971 edition and addenda through the Winter 1971. The Supply System has voluntarily upgraded the preservice inspection program to the 1974 edition and addenda through Summer 1975. The currently submitted ISI programs for pressure retaining piping and components meet the requirements of Section XI of the ASME Code, 1974 edition, Summer 1975. Proposed Technical Specifications addresses Inservice Inspection also. ISI for pump and valves meet 1977 edition and addenda through Summer 1978.

WNP-2 will perform inservice inspections and tests to the requirements in the submitted program where approved by NRC and will conform to the requirements in the Technical Specifications when issued in agreement thereto.

STATEMENT OF SECTION 50.55a - PARAGRAPH (c)

Pressure vessels: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operating pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A vessels set forth in Section III of the ASME Boiler and Pressure Vessel Code, applicable Code Cases, and Addenda in effect on the date of order of the vessel. The pressure vessels may meet the requirements set forth in editions of this Code, applicable Code Cases, and Addenda which have been effective after the date of vessel order.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A or Class 1

vessels set forth in Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pressure vessel: Provided, however, that if the pressure vessel is ordered more than 18 months prior to the date of issuance of the construction permit, compliance with the requirements for Class A or Class 1 vessels set forth in editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 18 months prior to the date of issuance of the construction permit is required. The pressure vessels may meet the requirements set forth in editions of this Code and Addenda which have become effective after the date of vessel order or after 18 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code: Provided, that the ASME Code provisions applied to the pressure vessels shall be no earlier than those of the Summer 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

WNP-2 construction permit was issued on March 19, 1973. Therefore, 10 CFR 50.55a, Paragraph (c)(2) applies.

Application for WNP-2 was filed with the Commission in August 1971. At that time a construction permit was expected before the end of 1972, but requests for additional seismic data in August 1972 caused the issuance of the construction permit to go beyond the end of the year to March 19, 1973. As is common practice in the utility industry, the Washington Public Power Supply System proceeded with the engineering, design, and material and components procurement in anticipation of the award of a construction permit to meet construction schedules. Had the construction permit been issued as initially expected, the requirements of 10 CFR 50.55a would have been met to the letter of the law.

In this case the code version applied was one addenda earlier (six months) than the code version required by the rules of 10 CFR 50.55a, paragraph (c)(2). The changes embodied in the later addenda are listed in Tables 5.2-13 and 5.2-14 of FSAR.

These changes have been reviewed with the conclusion that the addenda required by the rules of 10 CFR 50.55a, paragraph (c)(2) affected documentation format, but imposed no new technical requirements or changes in quality control procedures from the code version applied in the procurement of the components. The level of safety and quality provided by conformance to the earlier code edition and addenda applied in procurement is equivalent to that which would be required by strict application of the rules of 10 CFR 50.55a. The effort and expense of recertification of these components, which have all been shipped to the construction site, would not provide a compensating increase in the level of safety and quality.

STATEMENT OF SECTION 50.55a - PARAGRAPH (d)

Piping: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, piping which is part of the reactor coolant pressure boundary shall meet the requirements set forth in:

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases or the Class I Section of the USA Standard Code for Pressure Piping (USAS B31.7) in effect on the date of order of the piping and

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards of Class I piping of the USA Standard Code for Pressure Piping (USAS B31.7) may be applied.

The piping may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0 and USAS B31.7 Addenda and Code Cases which become effective after the date of order of the piping.

(2) For construction permits issued on or after January 1, 1971 but before July 1, 1974, piping which is part of the reactor coolant pressure boundary shall meet the requirements for Class 1 piping set forth in the USA Standard Code for Pressure Piping (USAS B31.7) and Addenda in effect on the date of order of the piping and the requirements applicable to piping of articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the piping, or (ii) the requirements applicable to Class 1 piping of editions of Section III of the ASME Boiler and Pressure Vessel Code Addenda in effect on the date of the order of the piping: Provided, however, that if the piping is ordered more than 6 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I or Class 1 piping set forth in editions of USAS B31.7 or Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit is required. The piping may meet the requirements set forth in editions of these Codes and Addenda which have become effective after the date of piping order or after 6 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, piping which is part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code. Provided, that the ASME Code provisions applied to the piping shall be no earlier than those of Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit was issued on March 19, 1973, therefore, 10 CFR 50.55a paragraph (d)(2) applied. Justification for the use of specific codes other than those issued specifically within 6 months of

the Construction Permit is given in FSAR Section 5.2.1. Table 5.2-5 tabulates the comparison of the purchase specification edition of the code and the 10 CFR 50.55a edition of the code. For the recirculation system piping the differences are trivial.

Recirculation piping and fittings were designed to ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Summer 1971 Addenda, Paragraph N-153.

STATEMENT OF SECTION 50.55a - PARAGRAPH (e)

Pumps: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, pumps which are part of the reactor coolant pressure boundary shall meet--

(i) The requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power Addenda, and Code Cases in effect on the date of order of the pumps, or

(ii) The nondestructive examination and acceptance standards set forth in ASA B31.1 Code Cases N7, N9, and N10, except that the acceptance standards for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps may be applied.

The pumps may meet the requirements set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which became effective after the date of order of the pumps.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, pumps which are part of the reactor coolant pressure boundary shall meet the requirements for Class I pumps set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the pumps and the requirements applicable to pumps set forth in articles 1 and 8 of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps, or (ii) the requirements applicable to Class 1 pumps of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the pumps: Provided, however, that if the pumps are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I pumps set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda and the requirements applicable to pumps set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda, or for Class 1 pumps of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The pumps may meet the requirement set forth in editions of these Codes or Addenda which have become effective after the date of pump order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, pumps which are part of the reactor coolant pressure boundary shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code: Provided, that the ASME Code provisions applied to the pumps shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit was issued on March 19, 1973, therefore, paragraph (e)(2) applies.

The reactor coolant pump casings were designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition Paragraph N-153. The recirculation pumps for WNP-2 were ordered in April 1971 to ASME Section III, 1971 Edition, whereas the 10 CFR 50.55a requirements would include the Summer 1971 Addenda. Table 5.2-5 tabulates the differences in these codes. Section 5.2.1 concludes that the differences are trivial for these pump bodies which make up the pressure boundary. Thus conformance exists.

STATEMENT OF SECTION 50.55a - PARAGRAPH (f)

Valves: (1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in:

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases, or the USA Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases, in effect on the date of order of the valves or the Class I section of the Draft ASME Code for Pumps and Valves for Nuclear Power Addenda, and Code Cases in effect on the date of order of the valves or

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N2, N7, N9, and N10, except that the acceptance standards for Class 1 valves set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the valves may be applied.

The valves may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases which become effective after the date of order of the valves.

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974, valves which are part of the reactor coolant pressure boundary shall meet the requirements for Class I valves set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power

and Addenda in effect on the date of order of the valves and the requirements applicable to valves set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valves, or (ii) the requirements applicable to Class 1 valves of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valve: Provided, however, that if the valves are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class 1 valves set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda and the requirements applicable to valves set forth in articles 1 and 8 of editions of Section III of the ASME Boiler and Pressure Vessel Code and Addenda or for Class 1 valves of Section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The valves may meet the requirements set forth in editions of these codes or Addenda which have become effective after the date of valve order or after 12 months prior to the date of issuance of the construction permit.

(3) For construction permits issued on or after July 1, 1974, valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in Section III of the ASME Boiler and Pressure Vessel Codes, provided, that the ASME Code provisions applied to the valves shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit was issued on March 19, 1973, therefore, paragraph (f)(2) applies. The main steam isolation valves were ordered in April 1971. The recirculation systems suction and discharge valves were ordered in June 1971 as pressure boundary components (Non Class 1E operators). The feedwater valves inside the reactor coolant pressure boundary were ordered in December 1973. These are non-steam Type Class B valves. The Crosby safety-relief valves were ordered in April 1971. They are not a ferritic steel hence need no testing requirements for toughness.

The Main Steam Isolation Valves are designed to the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, Paragraph N-153 and Winter 1972 Addenda. Conformance exists. For the MSIV's the procurement documents required ASME Section III 1971 Edition with Summer 1972 Addenda. For the recirculation system which is not ferritic steel, pressure boundary integrity conforms to ASME Section III Edition 1971 with Winter 1971 Addenda. The pressure boundary of the feedwater valves inside the RCPB was designed to ASME Section III Class 1, 1971 standards. The WNP-2 SRV's were designed to ASME Section III Class 1, Edition 1971 with Summer 1972 Addenda with respect to pressure boundary considerations. Conformance exists.

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(1) THROUGH (g)(3)

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves shall meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i) Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component in accordance with paragraphs (c), (d), (e), or (f) of the section.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component.

(iii) Pumps and valves which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the

construction of the particular pump or valve in accordance with paragraphs (c) and (f) of this section or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

EVALUATION OF COMPLIANCE

The WNP-2 Construction permit was issued on March 19, 1973, therefore, paragraph (g)(2) is applicable. WNP-2 safety-related components have been classified as ASME Code Class 1, 2 or 3 in accordance with the requirements of paragraph (g)(2). Table 5.2-5 tabulates the comparison of the purchase specification edition of code and 10 CFR 50.55a edition of the code. The preservice examination to be performed on Class 1 components and piping pursuant to the requirements of the 1974 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, including Summer 1975 Addenda for both RPV and associated piping, pumps, and valves are detailed in the WNP-2 Preservice Inspection Program Plan which was submitted in March 1979. Inservice Inspection Program Plan for Class 1 pressure retaining components and piping pursuant to the requirements of 10 CFR 50.55a (g)(2) will be submitted in April 1983. The WNP-2 Pump, Valve Inservice Test program addresses the operability testing of active Class 1, 2 and 3 components. This program has already been submitted to NRC.

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(4)

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements except design and access provisions and preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, conducted during the

initial 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the code incorporated by reference in paragraph (b) of this section of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examinations of components, inservice tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, conducted during successive 120-month inspection intervals shall comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) For a facility whose operating license was issued prior to March 1, 1976, the provisions of paragraph (g)(4) of this section are effective after September 1, 1976, at the start of the next one-third of a 120-month inspection interval. During that third of an inspection interval and the remainder of the inspection interval, the inservice examinations of components, tests to verify operational readiness of pumps and valves whose function is required for safety and system pressure tests, for such facilities shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the start of that third of an inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iv) Inservice examinations of components, tests of pumps and valves, and system pressure tests, may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section and subject to Commission approval. Portions of editions of addenda may be used provided that all related requirements of the respective editions or addenda are met.

EVALUATION OF COMPLIANCE

The baseline inspection report and the proposed inservice inspection program for WNP-2 complies with ASME Code Section XI, 1974 Edition, Summer 1975 Addenda, to the extent that the design of the plant, state of nondestructive testing technology and access to areas to be inspected allow. FSAR Section 5.2.4 explains this conformance.

STATEMENT OF SECTION 50.55a - PARAGRAPH (g)(5) AND (g)(6)

(5) (i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility shall be revised by the licensee, as necessary to meet the requirements of paragraph (g)(4) of this section.

(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. This application shall be submitted at least 6 months before the start of the period during which the provisions become applicable as determined by paragraph (g)(4) of this section.

(iii) If a licensee has determined that conformance with certain code requirements is impractical for his facility, the licensee shall notify the Commission and submit information to support his determinations.

(iv) Where an examination or test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination or test is determined to be impractical.

(6) (i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

EVALUATION OF COMPLIANCE

The NRC has not yet approved the proposed WNP-2 Technical Specifications to conform with the submitted ISI program. The WNP-2 inservice inspections and tests will be performed to the requirements in the submitted program when approved by NRC and will make necessary changes to the Technical Specification whenever required to conform to 10 CFR 50.55a(g).

STATEMENT OF SECTION 50.55a - PARAGRAPH (h)

Protection systems: For construction permits issued after January 1, 1971, protection systems shall meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generation Station," (IEEE-279) in effect on the formal docket date of

the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit was issued on March 19, 1973.

All features of the protection systems which actuate reactor trip and engineered safety features action are designed and/or built to conform to the intent of the criteria specified in IEEE-279-1971. Conformance is noted in FSAR Chapter 7.0.

STATEMENT OF SECTION 50.55a - PARAGRAPH (i)

Fracture toughness requirements: Pressure-retaining components of the reactor coolant pressure boundary shall meet the requirements set forth in Appendices G and H to this part.

EVALUATION OF COMPLIANCE

WNP-2 has addressed and complies with the requirements of 10 CFR Part 50, Appendices G and H as discussed in the evaluation of compliance for each of these appendices. FSAR Tables 5.3-1a and 5.3-1b summarize the conformance positions with regard to fracture toughness of RCPB materials.

STATEMENT OF SECTION 50.55a - PARAGRAPH (j)

Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit has been published on or before December 31, 1970, may meet the requirements of paragraphs (c)(1), (d)(1), (e)(1), and (f)(1) of this section instead of paragraph (c)(2), (d)(2), (e)(2), and (f)(2) of this section, respectively.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit was issued on March 19, 1973.

CONCLUSION

WNP-2 has addressed and complies with all applicable aspects of 10 CFR 50.55a.

10 CFR 50.59 - CHANGES, TESTS AND EXPERIMENTS

STATEMENT OF SECTION 50.59 - PARAGRAPHS (a), (b) AND (c)

(a) (1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. (2) A proposed change, test or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

(b) The licensee shall maintain records of changes in the facility and of changes in the procedures made pursuant to this section, to the extent that such changes constitute changes in the facility as described in the safety analysis report or constitute changes in procedures as described in the safety analysis report. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph (a) of this section. These records shall include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The licensee shall furnish to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this chapter with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. Any report submitted by a licensee pursuant to this paragraph will be made part of the public record of the licensing proceeding. In addition to a signed original, 39 copies of each report of changes in the facility of the type described in Section 50.21(b) or Section 50.22 or a testing facility, and 12 copies of each report of changes in any other facility, shall be filed. The records of changes in the facility shall be maintained until the date of termination of the license, and records of changes in procedures and records of tests and experiments shall be maintained for a period of five years.

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involves an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to Section 50.90.

EVALUATION OF COMPLIANCE

The Plant Operations Committee shall review:

a. The safety evaluation for tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59; and

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

Records of changes made to the equipment or review of tests and experiments to comply with 10 CFR 50.59 shall be kept in a manner convenient for review and shall be retained for at least five years.

WNP-2 will comply with the reporting requirements set forth in 10 CFR 50.59.

CONCLUSION

WNP-2 complies with the requirements of 10 CFR 50.59.

10 CFR 50.70 - INSPECTIONS

STATEMENT OF SECTION 50.70

(a) Each licensee and each holder of a construction permit shall permit inspection, by duly authorized representatives of the Commission, of his records, premises, activities, and of licensed materials in possession or use, related to the license or construction permit as may be necessary to effectuate the purposes of the Act, including Section 105 of the Act.

(b) (1) Each licensee and each holder of a construction permit shall upon request by the Director, Office of Inspection and Enforcement, provide rent-free office space for the exclusive use of the Commission inspection personnel. Heat, air conditioning, light, electrical outlets and janitorial services shall be furnished by each licensee and each holder of a construction permit. The office shall be convenient to and have full access to the facility and shall provide the inspector both visual and acoustic privacy.

(2) For a site with single power reactor or fuel facility licensed pursuant to Part 50, the space provided shall be adequate to accommodate a full-time inspector, a part-time secretary and transient NRC personnel and will be generally commensurate with other office facilities at the site. A space of 250 square feet either within the site's office complex or in an office trailer or other on-site space is suggested as a guide. For sites containing multiple power reactor units or fuel facilities, additional space may be requested to accommodate additional full-time inspectors). The office space that is provided shall be subject to the approval of the Director, Office of Inspection and Enforcement. All furniture, supplies and communication equipment will be furnished by the Commission.

(3) The licensee or construction permit holder shall afford any NRC resident inspector assigned to that site, or other NRC inspectors idientified by the Regional Director as likely to inspect the facility, immediate unfettered access, equivalent to access provided regular plant employees, following proper identification and compliance with applicable access control measures for security, radiological protection and personal safety.

EVALUATION OF COMPLIANCE

Plant operating records shall be kept in a manner convenient for review.

Rental-free office space as detailed above is provided at WNP-2 for the NRC Resident Inspector's Office.

Plant access equivalent to that provided regular plant employees is provided to NRC inspectors identified by the Regional Director of Region V as likely to inspect the facility.

CONCLUSION

WNP-2 complies with the requirements of 10 CFR 50.70.

10 CFR 50.71 - MAINTENANCE OF RECORDS, MAKING OF REPORTS

STATEMENT OF SECTION 50.71 - PARAGRAPHS (a) THROUGH (d)

(a) Each licensee and each holder of a construction permit shall maintain such records and make such reports, in connection with the licensed activity, as may be required by the conditions of the license or permit or by the rules, regulations, and orders of the Commission in effectuating the purposes of the Act, including Section 105 of the Act.

(b) With respect to any production or utilization facility of a type described in Section 50.21(b) or Section 50.22, or a testing facility, each licensee and each holder of a construction permit shall, upon each issuance of its annual financial report, including the certified financial statements, file a copy thereof with the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

(c) Records which are required by the regulations in this part, by license condition, or by technical specification, shall be maintained for the period specified by the appropriate regulation, license condition, or technical specification. If a retention period is not otherwise specified, such records shall be maintained until the Commission authorizes their disposition.

(d) (1) Records which must be maintained pursuant to this part may be the original or a reproduced copy or microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of reproducing a clear and legible copy after storage for the period specified by Commission regulations.

(2) If there is a conflict between the Commission's regulations in this part, license condition, or technical specification, or other written Commission approval or authorization pertaining to the retention period for the same type of record, the retention period specified in the regulations in this part for such records shall apply unless the Commission, pursuant to Section 50.12, has granted a specific exemption from the record retention requirements specified in the regulations in this part.

EVALUATION OF COMPLIANCE

The maintenance of these records and the making of required reports is specified in the WNP-2 Technical Specifications which are updated periodically and are subject to NRC approval.

STATEMENT OF SECTION 50.71 - PARAGRAPH (e)

Each person licensed to operate a nuclear power reactor pursuant to the provisions of Section 50.21 or Section 50.22 shall update periodically as provided in paragraphs (e)(3) and (e)(4) of this section, the final

safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.

(1) Revisions containing updated information shall be submitted on a replacement-page basis and shall be accompanied by a list which identifies the current pages of the FSAR following page replacement. One signed original and 12 additional copies of the required information shall be filed with the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

(2) The submittal shall include (i) a certification by a duly authorized officer of the licensee that either the information accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement, or that no such changes were made; and (ii) an identification of changes made under the provisions of Section 50.59 but not previously submitted to the Commission.

(3) (i) A revision of the original FSAR containing those original pages that are still applicable plus new replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

(ii) No less than 15 days before Section 50.71(e) becomes effective, the Director of the Office of Nuclear Reactor Regulation shall notify by letter the licensees of those nuclear power plants initially subject to the NRC's systematic evaluation program that they need not comply with the provisions of this section while the program is being conducted at their plant. The Director of the Office of Nuclear Reactor Regulation will notify by letter the licensee of each nuclear power plant being evaluated when the systematic evaluation program has been completed.

Within 24 months after receipt of this notification, the licensee shall file a complete FSAR which is up to date as of a maximum of 6 months prior to the date of filing the revision.

(4) Subsequent revisions shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing.

(5) Each replacement page shall include both a change indicator for the area changed, e.g., a bold line vertically drawn in the margin adjacent to the portion actually changed, and a page change identification (date of change or change number or both).

EVALUATION OF COMPLIANCE

Washington Public Power Supply System shall submit a revision of the original WNP-2 FSAR within 24 months of the issuance of the operating license and shall submit subsequent revisions to this document as required by this paragraph.

CONCLUSIONS

WNP-2 complies with the requirements of 10 CFR 50.71. An updated FSAR shall be submitted in accordance with the schedule established by the NRC in Section 50.71 - Paragraph (e).

10 CFR 50.72 - NOTIFICATION OF SIGNIFICANT EVENTS

STATEMENT OF SECTION 50.72 - PARAGRAPHS (a) AND (b)

(a) Each licensee of a nuclear power reactor licensed under Section 50.21 or Section 50.22 shall notify the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any of the following significant events and shall identify that event as being reported pursuant to this section:

(1) Any event requiring initiation of the licensee's emergency plan or any section of that plan.

(2) The exceeding of any Technical Specification Safety Limit.

(3) Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shut down.

(4) Any act that threatens the safety of the nuclear power plant or site personnel, or the security of special nuclear material, including instances of sabotage or attempted sabotage.

(5) Any event requiring initiation of shutdown of the nuclear power plant in accordance with Technical Specification Limiting Conditions for Operation.

(6) Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems, and components important to safety that are needed to (i) shut down the reactor safely and maintain it in a safe shutdown condition, or (ii) remove residual heat following reactor shutdown, or (iii) limit the release of radioactive material to acceptable levels or reduce the potential for such release.

(7) Any event resulting in manual or automatic actuation of Engineered Safety Features, including the Reactor Protection System.

(8) Any accidental, unplanned, or uncontrolled radioactive release. (Normal or expected releases from maintenance or other operational activities are not included.)

(9) Any fatality or serious injury occurring on the site and requiring transport to an offsite medical facility for treatment.

(10) Any serious personnel radioactive contamination requiring extensive onsite decontamination or outside assistance.

(11) Any event meeting the criteria of 10 CFR 20.403 for notification.

(12) Strikes of operating employees or security guards, or honoring of picket lines by these employees.

(b) With respect to the events reported under subparagraphs (1),(2), (3), and (4) of paragraph (a) of this section, each licensee, in addition to prompt telephone notification, shall also establish and maintain an open continuous communication channel with the NRC Operations Center, and shall close this channel only when notified by NRC.

EVALUATION OF COMPLIANCE

The Supply System's Corporate Policy and Procedure 4.3.4, "Reporting of Accident/Incidents," is in draft stage which requires each Directorate to have in their procedure the notification of NRC of applicable events within the time frame required by this section.

APPENDIX B TO 10 CFR 50 - QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

The WNP-2 program for quality assurance during the operations phase is described in report number WPPSS-QA-004, Revision 6, "WPPSS Operational Quality Assurance Program Description" which has been reviewed and approved by the NRC. This topical report complies with 10 CFR 50, Appendix B.

APPENDIX C TO 10 CFR 50 - A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR FACILITY CONSTRUCTION PERMITS AND OPERATING LICENSES

Application for Operating License for WNP-2 was tendered on March 24, 1978 which included financial data and related information. This is under NRC review.

APPENDIX E TO 10 CFR 50 - EMERGENCY PLANS FOR PRODUCTION AND UTILIZATION FACILITIES

The Washington Public Power Supply System WNP-1, -2, -4 Hanford Site Emergency Preparedness Plan, Revision 2, was submitted to NRC in December 1981. This plan outlines the emergency actions of the Supply System and offsite emergency response organizations as required by Appendix E to 10 CFR 50. This plan is developed in conjunction with Washington State and Benton and Franklin Counties fixed Nuclear Facility Emergency Response Plans. The Emergency Preparedness Plan is developed using the guidance of NUREG-0654/ FEM-REP-1, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in support of Nuclear Power Plants.

APPENDIX G TO 10 CFR 50 - FRACTURE TOUGHNESS REQUIREMENTS

Compliance with General Design Criteria 14 and 15 is documented in FSAR Section 3.1.2. Compliance with 10 CFR 50, Appendix G, is documented in the WNP-2 FSAR, Table 5.3-1a, and in FSAR responses to Series 121 questions. The technical requirements of Appendix G were evaluated and general compliance was determined to exist. The FSAR responses have provided the test data which confirms the prior analytical conclusions of adequate fracture toughness of ferritic materials in these BWR's. Conformance exists.

SECTION III - FRACTURE TOUGHNESS TESTS

STATEMENT OF SECTION III - PARAGRAPH A

To demonstrate compliance with the minimum fracture toughness requirements of Section IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code Section NB-2300, "Fracture Toughness Requirements for Materials." Both unirradiated and irradiated ferritic materials shall be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials shall be tested by means of the dropweight test specified by paragraph NB-2321.1 of the ASME Code. Provision shall be made for supplemental tests in crucial situations such as that described in Section V - Paragraph C.

EVALUATION OF COMPLIANCE

Reactor coolant pressure boundary materials testing complied with ASME Special Material Testing Requirements current at the time of construction of the WNP-2 components. A discussion of the details of this testing and of compliance with the existing standards based on prescribed operating limits is given in the WNP-2 FSAR, Section 5.3.1.5. Additional test data has been included in FSAR responses to Series 121 and 122 questions. Conformance exists.

STATEMENT OF SECTION III - PARAGRAPH B

Charpy V-notch impact tests and dropweight tests shall be conducted in accordance with the following requirements:

1. Location and orientation of impact test specimens shall comply with the requirements of Paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens shall be representative of the actual materials of the finished component as required by the applicable rules of the construction code under which the component is built pursuant to 10 CFR 50.55a, except that ferritic materials intended for the reactor vessel beltline region shall comply with the additional requirements of Section III - Paragraph C of this appendix.

3. Calibration of temperature instruments and Charpy V-notch impact test machines used in impact testing shall comply with the requirements of Paragraph NB-2360 of the ASME Code.

4. Individuals performing fracture toughness tests shall be qualified by training and experience and shall have demonstrated competency to perform the tests in accordance with written procedures of the component manufacturer.

5. Fracture toughness test results shall be recorded and shall include a certification by the licensee or person performing the tests for the licensee that:

a. the tests have been performed in compliance with the requirements of this appendix,

b. the test data are correctly reported and identified with the material intended for a pressure-retaining component,

c. the tests have been conducted using machines and instrumentation with available records of periodic calibration, and

d. records of the qualifications of the individual performing the tests are available upon request.

EVALUATION OF COMPLIANCE

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the time of construction of the WNP-2 components. A discussion of the details of this testing and of compliance with the existing standards based on prescribed operating limits is given in the WNP-2 FSAR, Section 5.3.1.5. Materials testing data for pressure boundary materials selected by the NRC staff has been reported in FSAR Series 121 and Series 122 Responses. Voluminous fabrication data are available in the design files to substantiate ASME code compliance. Conformance exists.

STATEMENT OF SECTION III - PARAGRAPH C

In addition to the test requirements of Section III - Paragraph A of this appendix, tests on materials of the reactor vessel beltline shall be conducted in accordance with the following minimum requirements:

1. Charpy V-notch (C_V) impact tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_V test curves (including the uppershelf levels) in terms of both fracture energy and lateral expansion of specimens. Location and orientation of impact test specimens shall comply with the requirements of Paragraph NB-2322 of the ASME Code.

2. Materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from excess material and welds in

the vessel shell course(s) following completion of the production longitudinal weld joint, and subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process, in accordance with paragraph NB-2211 of the ASME Code. Where seamless shell forgings are used, or where the same welding process is used for longitudinal and circumferential welds in plants, the test specimens may be taken from a separate weldment provided that such a weldment is prepared using excess material from the shell forging(s) or plates, as applicable, the same heat of filler material, and the same production welding conditions as those used in joining the corresponding shell materials.

EVALUATION OF COMPLIANCE

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the time of construction of the WNP-2 components. Additional sample materials testing was accomplished for beltline plates and weld materials per NRC requirements. A decision of the details of this testing and of compliance with the existing standards based on prescribed operating limits with the existing standards based on prescribed operating limits is given in the WNP-2 FSAR, Section 5.3.1.5 and Series 121 Responses.

SECTION IV - FRACTURE TOUGHNESS REQUIREMENTS

STATEMENT OF SECTION IV - PARAGRAPH A.1

The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials shall meet the following requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences:

1. The materials shall meet the acceptance standards of paragraph NB-2330 of the ASME Code, and the requirements of Section IV.A.2, 3, 4, and IV.B of this appendix.

EVALUATION OF COMPLIANCE

Reactor coolant system boundary materials testing complied with ASME Special Material Testing Requirements current at the time of construction of the WNP-2 components. A discussion of the details of this testing and of compliance with the existing standards based on prescribed operating limits is given in the WNP-2 FSAR, Section 5.3.1.5. Additional test data is reported in FSAR Series 121 Responses, as requested by the NRC staff. These recent data confirm the original test data upon which the conclusions of Section 5.3.1.5 were based.

(Compliance with Sections IV.A.2, 3, and IV.B are listed below. There is no Section IV-Paragraph A.4 as referenced in Paragraph A.1).

STATEMENT OF SECTION IV - PARAGRAPH A.2

For vessels, exclusive of bolting or other fasteners:

a. Calculated stress intensity factors shall be lower than the reference stress intensity factors by the margins specified in the ASME Code Appendix G, "Protection Against Non-Ductile Failure." The calculation procedures shall comply with the procedures specified in the ASME Code Appendix G, but additional and alternative procedures may be used if the Commission determines that they provide equivalent margins of safety against fracture, making appropirate allowance for all uncertainties in the data and analyses.

b. For nozzles, flanges, and shell regions near geometric discontinuities, the data and procedures required in addition to those specified in the ASME Code shall provide margins of safety comparable to those required for shells and heads remote from discontinuities.

c. Whenever the core is critical, the metal temperature of the reactor vessel shall be high enough to provide an adequate margin of protection against fracture, taking into account such factors as the potential for overstress and thermal shock during anticipated operational occurrences in the control of reactivity. In no case when the core is critical (other than for the purpose of low-level physics tests) shall the temperature of the reactor vessel be less than the minimum permissible temperature for the inservice system hydrostatic pressure test nor less than 40°F above that temperature required by Section IV.A.2.a.

d. If there is no fuel in the reactor during the initial preoperational system leakage and hydrostatic pressure tests, the minimum permissible test temperature shall be determined in accordance with paragraph G2410 of the ASME Code except that the factor of safety applied to each term making up the calculated stress intensity factor may be reduced to 1.0. In no case shall the test temperature be less than RT_{NDT}+60°F.

EVALUATION OF COMPLIANCE

a. Evaluation of compliance to 10 CFR 50, Appendix G paragraph IV A.2a is recorded in FSAR Section 5.3.1.5. Additional information is contained in FSAR response to question 121.12.

b. An alternative method was used to show compliance of WNP-2 by adjusting the BWR/6 analyses RT_{NDT} conditions. A 60°F adjustment was added to RT_{NDT} for flanges.

c. An alternative method of compliance was accepted for WNP-2 via GE Topical Report NEDO-21778-A.

d. WNP-2 preoperational testing limitations are in compliance.

STATEMENT OF SECTION IV - PARAGRAPH A.3

Materials for piping, pumps, and valves shall meet the requirements of paragraph NB-2332 of the ASME Code. Materials for bolting and other fasteners shall meet the requirements of paragraph NB-2333 of the ASME Code.

EVALUATION OF COMPLIANCE

Evaluation of compliance to 10 CFR 50, Appendix G Section IV paragraph A.3 is recorded in FSAR Section 5.2.3.3 and Section 5.3.1.5. Additional information is contained in FSAR response to question 121.15.

STATEMENT OF SECTION IV - PARAGRAPH B

Reactor vessel beltline materials shall have minimum upper-shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with paragraph NB-2332.2(a) of the ASME Code of 75 ft. lbs. unless it is demonstrated to the Commission by appropriate data and analysis that the lower values of upper-shelf fracture energy still provide adequate margin for deterioration from irradiation.

EVALUATION OF COMPLIANCE

Evaluation of compliance to 10 CFR 50, Appendix G Section IV papagraph B is recorded in Section 5.3.1.5 of the FSAR. Additional information is contained in FSAR response question 121.13.

STATEMENT OF SECTION IV - PARAGRAPH C

Reactor vessels for which the predicted value of adjusted reference temperature exceeds 200°F shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

EVALUATION OF COMPLIANCE

Annealing of the WNP-2 reactor vessels is not necessary. The requirements of this section are not applicable to WNP-2.

SECTION V - INSERVICE REQUIREMENTS - REACTOR VESSEL BELTLINE MATERIAL

STATEMENT OF SECTION V - PARAGRAPH A

The properties of reactor vessel beltline region materials, including welds, shall be monitored by a material surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements" set forth in Appendix H.

EVALUATION OF COMPLIANCE

Evaluation of compliance to 10 CFR 50, Appendix H is recorded in FSAR Section 5.3.1.6. The WNP-2 materials surveillance program has been upgraded to conform to NRC's new requirements. See the Appendix H evaluation which is discussed in the next major section of this conformance report.

STATEMENT OF SECTION V - PARAGRAPH B

Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV.A.2 are satisfied, using the predicted value of the adjusted reference temperature at the end of the service period to account for the effects of irradiation on the fracture toughness of the beltline materials. The basis for the prediction shall include results from pertinent radiation effects studies in addition to the results of the surveillance program of Section V.A.

EVALUATION OF COMPLIANCE

To meet the requirements of Section IV paragraph A.2 Appendix G, the material information obtained from the surveillance program will be tested in accordance with the requirements of Section III NB 2300 of the ASME Code and Appendix H Section III. This material data will be used to generate new heat up and cool down curves, if necessary, as described in the FSAR Section 5.3.1.5.

STATEMENT OF SECTION V - PARAGRAPH C

In the event that the requirements of Section V.B cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied.

1. An essentially complete volumetric examination of the beltline region of the vessel including 100% of any weldments shall be made in accordance with the requirements of Section XI of the ASME Code.

2. Additional evidence of the changes in fracture toughness of the beltline materials resulting from exposure to neutron irradiation shall be obtained from results of supplemental tests, such as measurements of dynamic fracture toughness of archive material that has been subjected to accelerated irradiation.

3. A fracture analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of adequate margins for continued operation.

EVALUATION OF COMPLIANCE

Compliance with Appendix G Section V paragraph C can be committed to except for item No. 1. At this time the Supply System is not capable of

performing 100% volumetric examination on WNP-2 vessel. Technological developments are currently under way which will allow remote examination in the future.

STATEMENT OF SECTION V - PARAGRAPH D

If the procedures of Section V.C do not indicate the existence of an adequate safety margin, the reactor vessel beltline region shall be subjected to a thermal annealing treatment to effect recovery of material toughness properties. The degree of such recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishment of the adjusted reference temperature after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A.2, using the values of adjusted reference temperature that include the effects of annealing and subsequent irradiation.

EVALUATION OF COMPLIANCE

Compliance with these requirements can be discussed if a future need arises. Standby capsules are provided in the material surveillance programs to measure the degree of recovery achieved by annealing and to monitor the effects of subsequent irradiation in the highly unlikely event that these requirements become applicable to WNP-2. Generally, this vintage of BWR vessels has not shown a dependence of material toughness properties that lie beyond current understanding and present forecast tolerances.

STATEMENT OF SECTION V - PARAGRAPH E

The proposed programs for satisfying the requirements of Section V.C and V.D shall be reported to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B.

EVALUATION OF COMPLIANCE

Should the need arise, Washington Public Power Supply System will alert the NRC in the unlikely event that a future need is identified for new programs to satisfy the requirements of Sections V.C and V.D.

APPENDIX H TO 10 CFR 50 - REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

Compliance with 10 CFR 50, Appendix H, is documented in the WNP-2 FSAR Section 5.3.1.6 and Table 5.3-1b. The sections of Appendix H that contain technical requirements were evaluated and compliance recorded for those requirements.

SECTION II - SURVEILLANCE PROGRAM CRITERIA

STATEMENT OF SECTION II - PARAGRAPH A

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E 1 MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

EVALUATION OF COMPLIANCE

Peak neutron fluence has been conservatively demonstrated using accepted analytical methods to be 1.1×10^{18}n/cm^2. The Supply System has provided for a reactor vessel materials surveillance program.

STATEMENT OF SECTION II - PARAGRAPH B

Reactor vessels constructed of ferritic materials which do not meet the conditions of Section II - Paragraph A shall have their beltline regions monitored by surveillance program complying with the American Society for Testing and Materials (ASTM) Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, ASTM Designation: E-185-73, except as modified by this appendix.

EVALUATION OF COMPLIANCE

WNP-2 has instituted a reactor vessel material surveillance program which will meet the intent of the NRC Reactor Vessel Material Surveillance Programs for Nuclear Power Reactors, 10 CFR 50 Appendix H. The WNP-2 ferritic materials irradiation surveillance program is not strictly in accordance with ASTM-E-185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" in that the surveillance specimens are not from the limiting beltline material. Specimens made from representative materials are used for the program. This resulted from vessel fabrication predating implementation of a formal ferritic materials surveillance program, 10 CFR 50 Appendix H.

STATEMENT OF SECTION II - PARAGRAPH C.1

Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G. The

specimen types shall comply with the requirements of Section III - Paragraph A of Appendix G (except that dropweight specimens are not required).

EVALUATION OF COMPLIANCE

WNP-2 surveillance program is not in compliance with this requirement. Specimens were not made from material adjacent to the fracture toughness specimens required in Section III of Appendix G. Also, transverse CVN specimens were not available. The noncompliance is due to the predating of vessel fabrication to implementation of 10 CFR 50 Appendix H. Material for surveillance specimens was obtained from actual beltline plate. The plate material, although not having the highest predicted EDL RT_{NDT} does have the greatest RT_{NDT} (which is the current basis in ASTM E-185-79 Table 1). Additional information is contained in FSAR response to question 121.14. The longitudinal surveillance specimens are consistent with the specimen orientation used to qualify the actual beltline material and the net effect of specimen orientation on radiation damage has been shown to be insignificant. Additional information is contained in FSAR response to question 121.14.

STATEMENT OF SECTION II - PARAGRAPH C.2

Surveillance specimen capsules shall be located near the inside vessel wall in the beltline region, so that the specimen irradiation history duplicates to the extent practicable, within the physical constraints of the system, the neutron spectrum temperature history and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds shall be done according to the requirements for permanent structural attachments to reactor vessels given in the ASME Code*, Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in Section II Paragraph C.3. (*Defined in Section II - Paragraph A of 10 CFR 50.)

EVALUATION OF COMPLIANCE

WNP-2 surveillance program is in compliance with this paragraph. Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline. The sealed capsules are not attached to the vessel wall. Capsule holder brackets are welded to the vessel inner surface. The welding of the bracket was conducted in accordance with Section III of the ASME Code.

STATEMENT OF SECTION II - PARAGRAPHS C.3.a, C.3.b, and C.3.c

Paragraph C.3.a

For reactor vessels for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steel, making

appropriate allowances for uncertainties in the measurements, the adjusted reference temperature established in accordance with Section III - Paragraph B will not exceed 100°F at the end of the service lifetime of the reactor vessel, at least three surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule - One-fourth service life

Second capsule - Three-fourths service life

Third capsule - Standby

In the event that the surveillance specimens exhibit, at one-quarter of the vessel's life, a shift of the reference temperature greater than originally predicted for similar material as recorded in the applicable technical specification, the remaining withdrawal schedule shall be modified as follows:

Second capsule - One-half service life

Third capsule - Standby

Paragraph C.3.b

For reactor vessels which do not meet the conditions of Section II - Paragraph C.3.a but for which it can be conservatively demonstrated by experimental data and tests performed on comparable vessel steels that the adjusted reference temperature will not exceed 200°F at the end of the service lifetime of the reactor vessel, at least four surveillance capsules shall be provided for the subsequent withdrawal as follows:

First capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50° or at one-fourth service life, whichever is earlier.

Second capsule - At approximately one-half of the time interval between first and third capsule withdrawal.

Third capsule - Three-fourths service life.

Fourth capsule - Standby.

Paragraph C.3.c

For reactor vessels which do not meet the conditions of Section II - Paragraph C.3.b, at least five surveillance capsules shall be provided for subsequent withdrawal as follows:

WITHDRAWAL SCHEDULE

First capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50°F or at one-fourth service life, whichever is earlier.

Second and third capsules - At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.

Fourth capsule - Three-fourths of service life.

Fifth capsule - Standby.

EVALUATION OF COMPLIANCE

WNP-2 is required to conform with Section II - Paragraph C.3.a as the adjusted reference temperature for both reactor vessels will not exceed 100°F. The withdrawal schedule complies with this section.

STATEMENT OF SECTION II - PARAGRAPH C.3.d

Provisions shall also be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation.

EVALUATION OF COMPLIANCE

WNP surveillance program is equipped with a standby surveillance capsule identical to those required in Section II paragraph C.3a Appendix H of the Code. This capsule can be used to monitor pressure vessel response to further irradiation in the unlikely event that an anneal is required.

STATEMENT OF SECTION II - PARAGRAPHS C.3.e, C.3.f, AND C.3.g

Paragraph C.3.e

Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

Paragraph C.3.f

If accelerated irradiation capsules are employed in addition to the minimum required number of surveillance capsules, the withdrawal schedule may be modified, taking into account the test results obtained from testing of the specimens in the accelerated capsules. The proposed modified withdrawal schedule in such cases shall be approved by the Commission on an individual case basis.

Paragraph C.3.g

Proposed withdrawal schedules that differ from those specified in paragraphs a. through f. shall be submitted with a technical justification therefor to the Commission for approval. The proposed schedule shall not be implemented without prior Commission approval.

EVALUATION OF COMPLIANCE

WNP surveillance program does not anticipate the need for accelerated testing due to the relative small shift in RT_{NDT}. Should the conditions dictate as Section I Paragraph C of Appendix G to 10 CFR 50, WNP-2 will obtain commission approval for change in the withdrawal schedule that differs from Section III - Paragraph C.3a.

STATEMENT OF SECTION II - PARAGRAPH C.4

For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

EVALUATION OF COMPLIANCE

Not applicable for WNP-2.

SECTION III - FRACTURE TOUGHNESS TESTS

STATEMENT OF SECTION III - PARAGRAPH A

Fracture toughness testing of the specimens withdrawn from the capsules shall be conducted in accordance with the requirements of Section III of Appendix G, "Fracture Toughness Requirements."

STATEMENT OF SECTION III - PARAGRAPH B

The adjusted reference temperatures for the base metal, heat-affected zone, and weld metal shall be obtained from the test results by adding to the reference temperature the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 foot-pound level or that measured at the 35 mil lateral expansion level, whichever temperature shift is greater. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements of Section V - Paragraph B of Appendix G are satisfied.

EVALUATION OF COMPLIANCE

WNP-2 surveillance program will test surveillance specimens in accordance with ASME Code Section III and ASTM recommended testing practices.

Adjusted reference temperature shall be determined in accordance of Section III, Paragraph B of Appendix H to 10 CFR 50.

SECTION IV - REPORT OF TEST RESULTS

STATEMENT OF SECTION IV - PARAGRAPH A

Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be provided to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. The report shall include a schematic diagram of the capsule locations in the reactor vessel, identification of specimens withdrawn, the test results, and the relationship of the measured results to those predicted for the reactor vessel beltline region.

STATEMENT OF SECTION IV - PARAGRAPH B

The report shall also include the dosimetry measurements performed at each specimen withdrawal, analyses of the results which yield the calculated neutron fluence received by the reactor vessel beltline region through the time of the tests, and comparisons with the originally predicted values of fluence.

STATEMENT OF SECTION IV - PARAGRAPH C

The operating pressure and temperature limitations established for the period of operation of the reactor vessel between any two surveillance specimen withdrawals shall be specified in the report, including any changes made in operational procedures to assure meeting such temperature limitations.

EVALUATION OF COMPLIANCE

A technical summary report including locations, orientations, and traceable identifiers for each capsule and specimen, test results, analysis and comparison to predicted results shall be provided to the Commission Director of Requalification. The Technical Summary Report shall also include dosimetry measurements, calculated peak neutron fluence in the beltline region, and a comparison to predicted values. The operating pressure and temperature limits for the period prior to capsule withdrawal will be reported. Any changes in operating procedures will be documented in accordance with technical specification revision procedures, as required.

APPENDIX I TO 10 CFR 50 - NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS FOR OPERATION TO MEET THE CRITERION " AS LOW AS IS REASONABLY ACHIEVABLE" FOR RADIOACTIVE MATERIAL IN LIGHT-WATER COOLED NUCLEAR POWER REACTOR EFFLUENTS

SECTION II - GUIDE ON DESIGN OBJECTIVES FOR LIGHT-WATER COOLED NUCLEAR POWER REACTORS LICENSED UNDER 10 CFR 50

STATEMENT OF SECTION II - PARAGRAPH A

The calculated annual total quantity of all radioactive material above background* to be released from each light-water cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways or exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

*Here and elsewhere in this Appendix background means radioactive materials in the environment and in the effluents from light-water cooled power reactors not generated in, or attributed to, the reactors of which specific account is required in determining design objectives.

EVALUATION OF COMPLIANCE

The radiological impact of radionuclides released in liquid effluents is discussed in subsection 5.2.4.1 of the WNP-2 ER-OL. Using the source terms and assumptions in Tables 5.2-1, 5.2-5, and 5.2-6 and the models in Appendix II to ER-OL, doses are estimated for individuals living near the plant. Tables 5.2-9 and 5.2-14 summarize the annual radiation doses to an individual which could be attributed to WNP-2. These calculated doses are within 10 CFR 50 Appendix I guidelines.

STATEMENT OF SECTION II - PARAGRAPH B.1

The calculated annual total quantity of all radioactive material above background to be released from each light-water cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

EVALUATION OF COMPLIANCE

The radiological impact of radionuclides in gaseous effluents are discussed in subsection 5.2.4.2 of the WNP-2 ER-OL. The calculated air doses for individuals living near the plant are listed in Tables 5.2-9 and 5.2-14. These doses are within 10 CFR 50 Appendix I guidelines.

STATEMENT OF SECTION II - PARAGRAPH B.2

Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

EVALUATION OF COMPLIANCE

The calculated individual doses due to gaseous and particulate effluents are given in Tables 5.2-9 and 5.2-14 of ER-OL. These doses are within 10 CFR 50 Appendix I guidelines.

STATEMENT OF SECTION II - PARAGRAPH C

The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

EVALUATION OF COMPLIANCE

The calculated annual doses to various organs for gaseous and liquid effluents including iodine are given in Tables 5.2-9 and 5.2-14. These doses are well within Appendix I to 10 CFR 50 guidelines.

STATEMENT OF SECTION II - PARAGRAPH D

In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable costbenefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until

establishment and adoption of better values (or other appropriate criteria), the values \$1000 per total body man-rem and \$1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pages 25-30, reproduced in the Annex to this Appendix I.

EVALUATION OF COMPLIANCE

The WNP-2 application for construction permit was docketed in August 1971. The question of eligibility of WNP-2 to dispense with the Appendix I cost-benefit analysis has been reviewed in the ASLB hearing of November 11, 1975 on the Supply System's Nuclear Project No. 1 (WNP-1) and Nuclear Project No. 4 (WNP-4), Docket Nos. 50-460 and 50-513. The NRC staff testimony concluded that "the aggregate doses associated with WPPSS Nuclear Project No. 1, WPPSS Nuclear Project No. 2, and WPPSS Nuclear Project No. 4 operation meet the RM-50-2 design objectives." These conclusions were ratified by the ASLB in its decision of December 22, 1975 (RAI-75/12 922, 934). Hence, no cost-benefit analysis is required.

SECTION III - IMPLEMENTATION

STATEMENT OF SECTION III - PARAGRAPH A.1

Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimation of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: provided, that if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of Paragraph C of Section II with respect to radioactive iodine if estimations of exposure are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

EVALUATION OF COMPLIANCE

The dispersion of radionuclides through liquid and gaseous pathways has been calculated using conservative data and methodology which are discussed in section 5.2 of the ER-OL.

STATEMENT OF SECTION III - PARAGRAPH A.2

The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

EVALUATION OF COMPLIANCE

Radiological impacts calculations have been performed using conservative data and methodology which are discussed in section 5.2 of the ER-OL. The liquid waste and gaseous waste management system, described in sections 11.2 and 11.3 of FSAR, are designed to keep the radiation dose in unrestricted areas as low as reasonably achievable within the guidelines of Appendix I to 10 CFR 50.

STATEMENT OF SECTION III - PARAGRAPH B

If the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives;

2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and

3. The content of radioactive iodine and foods involved in the changes if and when they occur.

EVALUATION OF COMPLIANCE

The Operational Radiological Environmental Monitoring Program (REMP) will implement the guidance of Regulatory Guide 4.8 and will include effluent

and environmental media monitoring and census program necessary to substantiate compliance with the Appendix I numerical guides and to assess the radiological impact of plant operation.

SECTION IV - GUIDES ON TECHNICAL SPECIFICATION FOR LIMITING CONDITIONS FOR OPERATION OF LIGHT-WATER COOLED NUCLEAR POWER REACTORS LICENSED UNDER 10 CFR 50

STATEMENT OF SECTION IV - PARAGRAPH A (SUMMARY)

If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall investigate, take corrective action, and report actions to the appropriate NRC Regional Office.

STATEMENT OF SECTION IV - PARAGRAPH B (SUMMARY)

The licensee shall establish an appropriate surveillance and monitoring program.

EVALUATION OF COMPLIANCE

The Limiting Conditions for Operation and the Surveillance Requirements of the WNP-2 Environmental Radiological Monitoring Program are stated in the radiological technical specifications for WNP-2. The reporting requirements are also stated in Technical Specifications.

Radiological Technical Specifications will be submitted to the NRC for review in the last quarter of 1982.

SECTION V - EFFECTIVE DATES

STATEMENT OF SECTION V - PARAGRAPH A

The guides for limiting conditions for operation set forth in this Appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water cooled nuclear power reactor.

EVALUATION OF COMPLIANCE

The WNP-2 application was filed on August 19, 1971. Radiological Technical Specification will be submitted to the NRC for review in February 1982.

STATEMENT OF SECTION V - PARAGRAPH B

For each light-water cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the

holder of the permit or a license authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission.

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by Section 50.34(a), (b), and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

EVALUATION OF COMPLIANCE

The WNP-2 construction permit application was filed on August 19, 1971. Therefore, this paragraph is not applicable to WNP-2.

CONCLUSION

Washington Public Power Supply System has addressed and complies with the requirements of Appendix I, 10 CFR 50 as documented in FSAR sections 11.2 and 11.3 and ER-OL section 5.2. The Radiological Technical Specification for WNP-2 will be submitted to the NRC for review in the last quarter of 1982.

APPENDIX J - REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER COOLED POWER REACTORS

SECTION III - LEAKAGE TESTING REQUIREMENTS

SECTION III - PARAGRAPH A - TYPE A TEST

EVALUATION OF COMPLIANCE

The proposed WNP-2 Technical Specifications, Paragraph 4.6.1.2 contains the following requirements with regard to reactor primary containment leak rate testing:

> 4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.5 - 1972:
>
> a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at P_a, 24.7 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant in-service inspection.
>
> b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a, at which time the above test schedule may be resumed.
>
> c. The accuracy of each Type A test shall be verified by a supplemental test which:
>
> 1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.
>
> 2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
>
> 3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 34.7 psig.

It should be noted that subsequent negotiations on the WNP-2 Technical Specifications could change this proposal.

SECTION III - PARAGRAPHS B AND C - TYPE B AND C TESTS

EVALUATION OF COMPLIANCE

The proposed WNP-2 Technical Specification, Paragraph 4.6.1.2 also contains the following requirements with regard to reactor primary containment leak rate tests.

d. Type B and C tests shall be conducted with gas at P_a, 34.7 psig, at intervals no greater than 24 months except for tests involving:

  1. Air locks,

  2. Main steam line isolation valves,

  3. Penetrations using continuous leakage monitoring systems,

  4. Valves pressurized with fluid from a seal system, and

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line valves shall be leak tested at least once per 18 months.

g. Type B periodic tests are not required for penetrations continuously monitored by the Containment Penetration Pressurization System, provided the system is OPERABLE per Specification 3.6.1.9.

h. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the valves are pressurized to at least 1.10 P_a, 38.2 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

j. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 34.7 psig, at intervals no greater than once per 3 years.

k. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurements system.

l. The provisions of Specification 4.0.2 are not applicable.

It should be noted that subsequent negotiations on WNP-2 Technical Specifications could change this proposal.

SECTION III - PARAGRAPH D - PERIODIC RETEST SCHEDULE

EVALUATION OF COMPLIANCE

The retest schedules for Type A, B, and C tests shall be in accordance with Section III, Paragraph D of Appendix J with the exception stated in 4.6.1.3 of the technical specifications.

SECTION IV - SPECIAL TESTING REQUIREMENTS

SECTION IV - PARAGRAPH A - CONTAINMENT MODIFICATION

EVALUATION OF COMPLIANCE

No major containment modification is anticipated at WNP-2.

SECTION IV - PARAGRAPH B - MULTIPLE LEAKAGE BARRIER OR SUBATMOSPHERIC CONTAINMENTS

EVALUATION OF COMPLIANCE

The proposed WNP-2 Technical Specifications, Paragraph 4.6.5.1 contain the following requirements with regard to secondary containment.

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inches of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All secondary containment equipment hatches and blowout panels are closed and sealed.

   2. At least one door in each access to the secondary containment is closed.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 2240 CFM.

It should be noted that subsequent negotiations on the WNP-2 Technical Specifications could change this proposal.

SECTION V - INSPECTION AND REPORTING OF TESTS

SECTION V - PARAGRAPH A - CONTAINMENT INSPECTION AND PARAGRAPH B - REPORT OF TEST RESULTS

EVALUATION OF COMPLIANCE

Inspection and reporting of tests shall be in accordance with Section V of Appendix J. The results of the preoperational containment leak rate test will be submitted to the NRC in 1982.

CONCLUSION

The Supply System has addressed and complies with the requirements of 10 CFR 50, Appendix J, for WNP-2.

APPENDIX K TO 10 CFR 50 - ECCS EVALUATION MODES

Compliance with 10 CFR 50 Appendix K is specifically required by 10 CFR 50.46. WNP-2 compliance is documented in Section 6.3 of the WNP-2 FSAR.

APPENDIX R TO 10 CFR 50 - FIRE PROTECTION PROGRAM FOR NUCLEAR POWER FACILITIES OPERATING PRIOR TO JANUARY 1979

Compliance is documented in Amendment No. 19, Fire Protection Evaluation, to the WNP-2 FSAR which was submitted to the NRC on October 12, 1981.

10 CFR 100.10 - FACTORS TO BE CONSIDERED WHEN EVALUATING SITES

STATEMENT OF SECTION 100.10

Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should ensure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in detemining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

(1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," describes the nature of investigations required to obtain the geologic and seismic data necessary to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes

procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow in to nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

EVALUATION OF COMPLIANCE

The factors required to be considered with regard to both the plant design and the site have been provided in the WNP-2 FSAR and Environmental Report. Site specifics, including seismology, meteorology, geology, and hydrology, are presented in FSAR, Chapter 2. The FSAR also describes the characteristics of reactor design and operation.

CONCLUSION

Conformance with the intent and specific requirements of 10 CFR 100.10 is documented for WNP-2. No site constraints have been identified in the FSAR, Environmental Report, nor during CP deliberations.

10 CFR 100.11 - DETERMINATION OF EXCLUSION AREA, LOW POPULATION ZONE, AND POPULATION CENTER DISTANCE

STATEMENT OF SECTION 100.11

(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not inititate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of containment accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous release. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

EVALUATION OF COMPLIANCE

The exculsion area, low population zone and population center distance are described in the FSAR, Chapter 2. All requirements of this section with regard to these areas and distances are met. The FSAR accident analyses, particularly those in Chapters 6 and 15, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.

CONCLUSION

Conformance with the intent and specific requirements of 10 CFR 100.11 is documented in the WNP-2 FSAR.

APPENDIX A TO 10 CFR 100 - SEISMIC AND GEOLOGIC SITING CRITERIA FOR NUCLEAR POWER PLANTS

STATEMENT OF APPENDIX A TO 10 CFR 100 (SUMMARY)

General Design Criterion 2 of Appendix A to Part 50 of this chapter requires that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. It is the purpose of these criteria to set forth the principal seismic and geologic considerations which guide the Commission in its evaluation of the suitability of proposed sites for nuclear power plants, and the suitability of the plant design bases established in consideration of the seismic and geologic characteristics of the proposed sites.

These criteria are based on the limited geophysical and geological information available to date concerning faults and earthquake occurrence and effect. They will be revised as necessary when more complete information becomes available.

In general, the criteria are set forth with regard to: (1) the investigations required for vibratory ground motion, for surface faulting, for seismically induced floods and water waves; (2) the seismic and geologic design basis for the plan with respect to the results of these investigations; and (3) the application of these bases to the engineering design.

EVALUATION OF COMPLIANCE

Structures and equipment important to plant safety are protected from or designed to withstand all appropriate natural phenomena at the plant site. Design is based on the most severe phenomena probable with special consideration for the uncertainty in prediction. Detailed discussions of the phenomena themselves, and how they are applied to the structures and equipment, are found in the following FSAR sections:

Meteorology, Section 2.3;

Hydrology, Section 2.4;

Geology and Seismology, Section 2.5;

Classification of Structures, Components, and Systems, Section 3.2;

Wind and Tornado Design Criteria, Section 3.3;

Water Level Design Criteria, Section 3.4;

Missile Protection Criteria, Section 3.5;

Seismic Design, Section 3.7;

Design of Seismic Category I Structures, Section 3.8;

Mechanical Systems and Components, Section 3.9;

Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment, Section 3.10; and

Environmental Design of Mechanical and Electrical Equipment, Section 3.11.

CONCLUSION

The Supply System complies with the requirements of Appendix A to 10 CFR 100 for WNP-2. The WNP-2 site and plant environmental design considerations meet the NRC seismic and geologic siting criteria.

PFR NO. / 2408-PFR-F001

REVISION ___________

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

PREPARATION BY GA INITIATOR

AFFECTED ITEMS: Safety Injection Line to Reactor Coolant Loop 1A
Piping Stress Analysis Package PSG-74

REQUIREMENT REFERENCE DOCUMENTS:

Pipe Support Description List Sheets

BASIC REQUIREMENT: The design load should be an absolute summation of the seismic inertia loads and the seismic anchor movement loads combined with an algebraic summation with the dead weight and thermal loads, with the exception of the special case where ZPAs occur, this uses $(DBE^2 + ZPA^2)^{1/2}$

DESCRIPTION OF POTENTIAL FINDING:

The above summation of weight, thermal, seismic inertia and seismic anchor movement loads exceeds the design load of five of the twenty-two supports that have been considered in PSG-74. These supports are PS-12, PS-14, PS-16, PS-17 and PS-22.

Alan C Lewis

PREPARED BY: Alan C. Lewis DATE: 25 Feb, 1982.

REJECTION OF GA TASK LEADER COMMENTS BY: ___________ DATE: ___________

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: ___________ DATE: ___________

B. REVIEW BY GA TASK LEADER COMMENTS

☒ AGREE PF IS VALID BY [illegible signature] DATE 2/25/82

☐ REQUEST RE-REVIEW BY ___________ DATE ___________

☐ DISAGREE BY ___________ DATE ___________

☐ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: ___________ DATE: ___________

8203230674

PFR NO. 2408-PFR-F001

REVISION

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

COMMENTS

☐ AGREE PF IS VALID

☐ DISAGREE

BY: __________ DATE: __________

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" __________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. L. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: G. L. Wessman DATE: 3/18/82

2408 PFR NO. F001

REVISION ________

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION COMMENTS

The correct method for combining pipe support design loads is given in Section 27.6.8 of the SONGS 2 & 3 Pipe Support Group Design Manual. See attached sheets 27, 28 and [illegible] from this manual. Sheet 27 shows that the procedure for combining seismic inertia and seismic anchor movement loads is by the SRSS method. This method is considered conservative due to the extremely low probability of a maximum seismic inertia load occurring at the same instant in time as the maximum load due to seismic anchor movements.

☐ AGREE PF IS VALID

☒ DISAGREE

This combined seismic load is then added algebraically with weight and thermal loads.

BY: Fred B Mord DATE: 3/1/82

I agree with this PFR - being invalid - see attachment to this document. JLP 3/10/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☐ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☐ INVALID
10 CFR 21:	☐ NOT APPLICABLE	☐ APPLICABLE
10 CRF 50.55(e):	☐ NOT APPLICABLE	☐ APPLICABLE
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

Concur. [illegible] 3/10

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: ________ DATE: ________

E. TPT PROJECT MANAGER

☐ ACCEPT

☐ REJECT

BY: ________ DATE: ________

3408-PFR-F001
3/18/8_

The following combination shall be used for NB-3652 equation (9) faulted condition.

$$B_1 \frac{PD_o}{2t} + B_2 \frac{D_o}{2I} M_i \leq 3S_m$$

where

$$M_i = \sqrt{M_x^2 + M_y^2 + M_z^2}$$

For faulted condition, the moments in each individual direction shall be combined as follows:

$$M_j = M_{jwt} \pm \sqrt{M_{j\,\text{DBE INERTIA}}^2 + \left(|M_{j\,\text{JI}}| + |M_{j\,\text{LOCA DISP}}|\right)^2}$$

where

$j = x, y, z$

27.6.7 PIPE BREAK EFFECTS

Stress summaries will be provided to Mechanical Group and the nozzle loads are furnished to CE. Jet impingement loads are described in Section 27.6.6.

27.6.8 PIPE SUPPORT DESIGN LOADS

Definition of Terms

TH = Loads from Thermal Analysis

DW = Dead Weight Loads

EQ = Load Associated with DBE earthquake,

$= \sqrt{SI^2 + SAM^2}$

where

SI = Loads from seismic inertia analysis, DBE

SAM = Loads from Seismic Anchor Movement Analysis, DBE

HYDRO WT. = Dead Weight Loads Associated with Hydro Test Loads

2408-PRK-F001
SDC 5/15/82

FV = Loads from Fast Valve Closure

RVC = Loads from Relief Valve Opening - Closed System (Transient)

RVO = Loads from Relief Valve Opening - Open System (Sustained)

PD = Loads from Untied Expansion Joint at Design Pressure

LOCA = Loads Associated with LOCA Event

= (JI $\pm$ DISP)

where

JI = Loads from Jet Impingement

DISP = Loads Associated with NSSS Vessel During LOCA Condition

DF = Dynamic Events Associated with LOCA (Piping Must Remain Intact)

DU = Dynamic Events Associated with Upset Plant Condition

DE = Dynamic Events Associated With Emergency Plant Condition

ASME Class 2 and 3 Piping Systems

Using the load combinations shown the pipe support design loads are the load combination which gives the largest algebraic value in each direction.

DW. $\pm$ EQ. + PD
TH + DW $\pm$ EQ + RVO + PD
HYDRO WT. + PD
TH + DW + DF + PD
TH + DW + FV + PD
TH + DW + RVC + PD
TH + DW + DU + PD
TH + DW + DE + PD

For most ASME Class 2 and 3 piping the governing load combinations are:

1) HYDRO WT.

2) WT $\pm$ EQ

3) TH + WT $\pm$ EQ

2408-PFR-F001
3/15/82

B31.1 Power Piping Systems, Seismic Category II

Load combinations sets used to obtain maximum pipe support design loads in each direction that the support acts

$$\begin{cases} \text{HYDRO} + \text{PD} \\ \text{TH} + \text{DW} + \text{FV} + \text{PD} \\ \text{TH} + \text{DW} + \text{RVC} + \text{PD} \\ \text{DW} \pm \text{Static Seismic} + \text{PD} \\ \text{TH} \mp \text{DW} \pm \text{Static Seismic} \\ \quad + \text{RVO} + \text{PD} \end{cases}$$

Static seismic per Appendix 4G or upgrade criteria. SAMs are not included in static seismic analysis.

$$\text{State Seismic} = \sqrt{\left(\begin{matrix}\text{N - S}\\ \text{STATIC}\end{matrix}\right)^2 + \left(\begin{matrix}\text{VERT.}\\ \text{STATIC}\end{matrix}\right)^2 + \left(\begin{matrix}\text{E - W}\\ \text{STATIC}\end{matrix}\right)^2}$$

B31.1 Power Piping Systems, Seismic Category I

Load Combinations

$$\begin{cases} \text{HYDRO} + \text{PD} \\ \text{TH} + \text{DW} + \text{PV} + \text{PD} \\ \text{TH} + \text{DW} + \text{RVC} + \text{PD} \\ \text{DW} \pm \text{EQ} + \text{PD} \\ \text{TH} \mp \text{DW} \pm \text{EQ} + \text{PD} + \text{RVO} \end{cases}$$

ASME Class 1 Piping Systems

Loads combinations are the same as for ASME Class 2 and 3 piping systems with the addition of the following LOCA condition:

$$\text{TH} + \text{WT} \pm \sqrt{\text{EQ}^2 + \text{LOCA}^2}$$

$$= \text{TH} + \text{WT} \pm \sqrt{\text{DBE SI}^2 + \text{DBE SAM}^2 + \text{JI} \pm \text{DISP}^2}$$

The above load combination is the governing criteria for design loads on Class 1 pipe supports.

27.6.9 STRESS INTENSIFICATION FACTORS (SIF)

A. SIF for elbows, tees, reinforced, and unreinforced branch connections shall be per the Code.

B. SIF for Sweepolets shall be the larger of the SIF for a welding tee or the SIF as specified by the vendor.

C. SIF for weldolets shall be the larger of the SIF for an unreinforced branch connection or the SIF as specified in the vendor catalog. Deviation from this requires EGS approval.

2408-PFR-F001
3/18/82

Attachment to PFR-2408-PFR-F001 and F002

When values from the dynamic analysis and those from the static differential displacement analysis (SAM) are combined it is NRC's position that the absolute sum method be used.* BPC used the SRSS method in combining the load from seismic inertia and SAMs. However, the SSE is a faulted event which according to the ASME Code does not include consideration of any secondary loading (SAM and THERMAL). Therefore, BPC is conservative in their approach by including the secondary loadings in the faulted event and the method used for summation is not a point of concern.

* Per NRC's review of BPC's BP-TOP-1, "Seismic Analysis of Piping Systems," Rev. 3, 1/76.

Concur with recommendation to invalidate this PFR. Note that A. Lewis, the initiator of this PFR, is offsite. The processing of this PFR has been assigned to L. J. Pickering. A. Lewis' concurrence with the disposition of this PFR was obtained (see attached telephone record).

[signature]
3/18/82

FROM: LARRY PICKERING LOCATION: GA DATE: 3/17/82

TO: FS OPLE LOCATION: GA DATE:

2408-PFR-F001
3/16/81

TELEPHONE COMMUNICATION RECORD

(PLEASE HAND LETTER LEGIBLY IN BLACK OR RED INK)

CALL INITIATED BY: LARRY PICKERING AT GAC ☒ OTHER:

CALL RECEIVED BY: ALLEN LEWIS AT GAC ☐ OTHER: BECHTEL, LA
(213) 862-8631 x 648

OTHER PARTICIPANTS:

DATE	TIME	PROGRAM NAME	PROGRAM NUMBER
3/17/82	MORN	SCE DESIGN REVIEW	2408-400

SUBJECT: PFR's F001 AND F002

SUMMARY: I INFORMED ALLEN THAT THE REPLY TO HIS PFR's WAS THAT BPC IS CONSERVATIVE IN THEIR APPROACH BY INCLUDING THE SECONDARY LOADINGS IN THE FAULTED EVENT AND THE METHOD USED FOR SUMMATION IS NOT A POINT OF CONCERN. ALLEN AGREED WITH THAT POSITION.

ACTION ITEMS:	Date Required	Person Responsible

DISTRIBUTION:

File No.:

PFR NO. 2408-PFR-F002

REVISION

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION.

PREPARATION BY GA INITIATOR

AFFECTED ITEMS: Safety Injection Line to Reactor Coolant Loop 1A
Piping Stress Analysis Package PSG-74

REQUIREMENT REFERENCE DOCUMENTS:

Anchor Load Sheets

BASIC REQUIREMENT: The total net design load should be an absolute summation of the seismic inertia loads and the seismic anchor movement loads combined by an algebraic summation with the dead weight and thermal loads.

DESCRIPTION OF POTENTIAL FINDING:

The above summation of weight, thermal, seismic loads inertia and seismic anchor movement loads from both sides of the anchor using the latest seismic computer run exceeds the ~~total~~ design load of both of the anchors.

AL

PREPARED BY: A. Lewis Alan C Lewis DATE: 25 Feb, 1982.

REJECTION OF GA TASK LEADER COMMENTS BY: DATE:

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: DATE:

B. REVIEW BY GA TASK LEADER

COMMENTS

AL ☒ AGREE PF IS VALID BY [signature] DATE 2/25/82

☐ REQUEST RE-REVIEW BY DATE

☐ DISAGREE BY DATE

☐ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: DATE:

2408 PFR NO. F002

REVISION

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION — COMMENTS

The correct method for combining pipe support design loads is given in Section 27.6.8 of the SONGS 2 & 3 Pipe Support Group Design Manual. See attached sheets 27, 28 and 29 from this manual. Sheet 27 shows that the procedure for combining seismic inertia and seismic anchor movement loads is by the SRSS method. This method is considered conservative due to the extremely low probability of a maximum seismic inertia load occurring at the same instant in time as the maximum load due to seismic anchor movements.

This combined seismic load is then added algebraically with weight and thermal loads.

☐ AGREE PF IS VALID

☒ DISAGREE

BY: Fred R Maul DATE: 3/1/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
~~10 CFR 21:~~	~~☐ NOT APPLICABLE~~	~~☐ APPLICABLE~~
~~10 CFR 50.55(e):~~	~~☐ NOT APPLICABLE~~	~~☐ APPLICABLE~~
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

SLK 3/18/82

I agree with their PFR as being invalid see attachment to their document JJ Pickering 3/18/82

Concur [initials] 3/18

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. L. Koutz DATE: 3/18/82

E. TPT PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: G L Weisman DATE: 3/18/82

2408-PFR-F002
[handwritten initials] 3/18/82

The following combination shall be used for NB-3652 equation (9) faulted condition.

$$B_1 \frac{PD_o}{2t} + B_2 \frac{D_o}{2I} M_i \leq 3S_m$$

where

$$M_i = \sqrt{M_x^2 + M_y^2 + M_z^2}$$

For faulted condition, the moments in each individual direction shall be combined as follows:

$$M_j = M_{jwt} \pm \sqrt{M_{j\,\substack{DBE\\INERTIA}}^2 + \left(|M_{j\,JI}| + |M_{j\,\substack{LOCA\\DISP}}|\right)^2}$$

where

$j = x, y, z$

27.6.7 PIPE BREAK EFFECTS

Stress summaries will be provided to Mechanical Group and the nozzle loads are furnished to CE. Jet impingement loads are described in Section 27.6.6.

27.6.8 PIPE SUPPORT DESIGN LOADS

Definition of Terms

TH = Loads from Thermal Analysis

DW = Dead Weight Loads

EQ = Load Associated with DBE earthquake,

$= \sqrt{SI^2 + SAM^2}$

where

SI = Loads from seismic inertia analysis, DBE

SAM = Loads from Seismic Anchor Movement Analysis, DBE

HYDRO WT. = Dead Weight Loads Associated with Hydro Test Loads

2406-PFR-F002
SPC 3/18/82

FV = Loads from Fast Valve Closure

RVC = Loads from Relief Valve Opening - Closed System (Transient)

RVO = Loads from Relief Valve Opening - Open System (Sustained)

PD = Loads from Untied Expansion Joint at Design Pressure

LOCA = Loads Associated with LOCA Event

= (JI ± DISP)

where

JI = Loads from Jet Impingement

DISP = Loads Associated with NSSS Vessel During LOCA Condition

DF = Dynamic Events Associated with LOCA (Piping Must Remain Intact)

DU = Dynamic Events Associated with Upset Plant Condition

DE = Dynamic Events Associated With Emergency Plant Condition

ASME Class 2 and 3 Piping Systems

Using the load combinations shown the pipe support design loads are the load combination which gives the largest algebraic value in each direction.

DW. ± EQ. + PD
TH + DW ± EQ + RVO + PD
HYDRO WT. + PD
TH + DW + DF + PD
TH + DW + FV + PD
TH + DW + RVC + PD
TH + DW + DU + PD
TH + DW + DE + PD

For most ASME Class 2 and 3 piping the governing load combinations are:

1) HYDRO WT.

2) WT ± EQ

3) TH + WT ± EQ

2408-PFR-F002
JWC 3/18/82

B31.1 Power Piping Systems, Seismic Category II

Load combinations sets used to obtain maximum pipe support design loads in each direction that the support acts	HYDRO + PD TH + DW + FV + PD TH + DW + RVC + PD DW ± Static Seismic + PD TH ∓ DW ± Static Seismic + RVO + PD

Static seismic per Appendix 4G or upgrade criteria. SAMs are not included in static seismic analysis.

$$\text{State Seismic} = \sqrt{\left(\begin{matrix}\text{N - S}\\ \text{STATIC}\end{matrix}\right)^2 + \left(\begin{matrix}\text{VERT.}\\ \text{STATIC}\end{matrix}\right)^2 + \left(\begin{matrix}\text{E - W}\\ \text{STATIC}\end{matrix}\right)^2}$$

B31.1 Power Piping Systems, Seismic Category I

Load Combinations	HYDRO + PD TH + DW + PV + PD TH + DW + RVC + PD DW ± EQ + PD TH ∓ DW ± EQ + PD ∓ RVO

ASME Class 1 Piping Systems

Loads combinations are the same as for ASME Class 2 and 3 piping systems with the addition of the following LOCA condition:

$$TH + WT \pm \sqrt{EQ^2 + LOCA^2}$$

$$= TH + WT \pm \sqrt{DBE\ SI^2 + DBE\ SAM^2 + (JI \pm DISP)^2}$$

The above load combination is the governing criteria for design loads on Class 1 pipe supports.

27.6.9 STRESS INTENSIFICATION FACTORS (SIF)

A. SIF for elbows, tees, reinforced, and unreinforced branch connections shall be per the Code.

B. SIF for Sweepolets shall be the larger of the SIF for a welding tee or the SIF as specified by the vendor.

C. SIF for weldolets shall be the larger of the SIF for an unreinforced branch connection or the SIF as specified in the vendor catalog. Deviation from this requires EGS approval.

2408-PFR-F001
SPC 3/18/82

Attachment to PFR-2408-PFR-F001 and F002

When values from the dynamic analysis and those from the static differential displacement analysis (SAM) are combined it is NRC's position that the absolute sum method be used.* BPC used the SRSS method in combining the load from seismic inertia and SAMs. However, the SSE is a faulted event which according to the ASME Code does not include consideration of any secondary loading (SAM and THERMAL). Therefore, BPC is conservative in their approach by including the secondary loadings in the faulted event and the method used for summation is not a point of concern.

* Per NRC's review of BPC's BP-TOP-1, "Seismic Analysis of Piping Systems," Rev. 3, 1/76.

Concur with recommendation to invalidate this PFR. Note that A. Lewis, the initiator of this PFR, is offsite. The processing of this PFR has been assigned to L. F. Pickering. A. Lewis' concurrence with the disposition of this PFR was obtained (see attached telephone record).

[signature] 3/18/82

FROM: LARRY PICKERING LOCATION: GA DATE: 3/17/82

TO: FS OPLE LOCATION: GA DATE:

2408-PFR-F002

TELEPHONE COMMUNICATION RECORD

FC 3/18/82

(PLEASE HAND LETTER LEGIBLY IN BLACK OR RED INK)

CALL INITIATED BY: LARRY PICKERING AT GAC ☒ OTHER:

CALL RECEIVED BY: ALLEN LEWIS AT GAC ☐ OTHER: BECHTEL, LA

(213) 862-8631 x 648

OTHER PARTICIPANTS:

DATE	TIME	PROGRAM NAME	PROGRAM NUMBER
3/17/82	MORN	SCE DESIGN REVIEW	2408-400

SUBJECT: PFR's F001 AND F002

SUMMARY: I INFORMED ALLEN THAT THE REPLY TO HIS PFR'S WAS THAT BPC IS CONSERVATIVE IN THEIR APPROACH BY INCLUDING THE SECONDARY LOADINGS IN THE FAULTED EVENT AND THE METHOD USED FOR SUMMATION IS NOT A POINT OF CONCERN. ALLEN AGREED WITH THAT POSITION.

ACTION ITEMS:	Date Required	Person Responsible

DISTRIBUTION:

File No.:

2408 PFR NO. F005
REVISION A

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

A. PREPARATION BY GA INITIATOR

AFFECTED ITEMS: CEN-99(S), "Seismic Program for Qualification of NSSS-Supplied Instrumentation Equipment."

REQUIREMENT REFERENCE DOCUMENTS:
QADP 5.8, Rev. 0, "Other Design Documents."

BASIC REQUIREMENT:
Design documents must be reviewed, approved, issued and revisions controlled.

DESCRIPTION OF POTENTIAL FINDING:
CEN-99(S) appears to exist only as a marked-up draft copy. Combustion Engineering furnished this marked-up draft to GA as the only version available. A copy of the Title page is attached as an illustration of the condition of the document. Also attached is a section from the FSAR that references CEN-99(S).

Agree PFR is invalid based on CE's response that specifications, RAR's & Test Reports were used. George Chandler 3/16/82

PREPARED BY: George W. Chandler DATE: 2/24/82

REJECTION OF GA TASK LEADER COMMENTS BY: ______ DATE: ______

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: ______ DATE: ______

B. REVIEW BY GA TASK LEADER — COMMENTS

Agree PFR is invalid per above. SB 3/16/82

☒ AGREE PF IS VALID BY S. Brown DATE 2/24/82

☐ REQUEST RE-REVIEW BY ______ DATE ______

☐ DISAGREE BY ______ DATE ______

☑ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: S. Brown DATE: 3/16/82

2408 PFR NO. F005
REVISION A

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

COMMENTS

While C-E concedes that CEN-99(S) exists with handwritten annotations, it is neither a draft nor a design document. It is a finalized, controlled licensing document that was submitted, on short notice, to the NRC in support of San Onofre's licensing effort as a partial response to Question 032.4 (see attached). CEN-99(S) and CEN-94(S) are plant specific replacements for the two-part, generic topical report, CENPD-182.

As the documents' Abstract says: "CEN-99(S) is a summary of the C-E seismic qualification program utilized to demonstrate the seismic design adequacy of the instrumentation and control equipment used in the C-E supplied NSSS for San Onofre Units 2 & 3."

☐ AGREE PF IS VALID

☒ DISAGREE

BY: [illegible] DATE: 3/12/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ______

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. L. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: [illegible] DATE: 3/18/82

As to the references in the FSAR to CEN-99(S), the early sections of CEN-99(S) merely describe the scope and definitions of the seismic qualification program, sections 3.5 and 3.6 describe the qualification procedures in accordance with IEEE Std's 344-1971 and 344-1975 respectively, and section 4.0 describes the administrative procedures C-E follows for seismic qualification. Within section 4.0 there is a discussion of how C-E sets forth the equipment specification, the vendor submits his design and qualification program to C-E for approval, C-E follows the qualification including the witnessing of tests if necessary, the test facility forwards a test/analysis report to C-E and C-E's acceptance as a qualification report. The documents used in this procedure (C-E Specifications, RAR's (Request for Approval or Review), Test Procedures, Test Plans and Test Reports) have been reviewed by GA's Quality Assurance reviewers on a number of items already. Any other criteria in CEN-99(S) draw from the above mentioned IEEE Std's.

In summation, CEN-99(S) is not a draft copy of a design document, but rather, a finalized, controlled licensing document that gives a summary of C-E's seismic qualification program and contains a series of references for the use of those concerned in the qualification program.

2408 PFR No. F005
JRB for FDC 3/16/82

Question 032.4

Reference is made to Combustion Engineering Topical Report, CENPD-182 for the seismic qualification of the Class IE instrumentation and control equipment in the NSSS scope of supply.

This topical report is presently under review by the NRC and the staff has requested CE to provide additional information with regard to their seismic qualification program.

In order to proceed with the review of San Onofre 2 and 3 in this area, we require a commitment from the applicant to accept the generic resolution achieved on the report between the NRC and Combustion Engineering.

Therefore, provide your commitment in this area.

Response

By the letter referenced below[a] the requested additional information was provided with regard to the seismic qualification program. Should additional concerns arise as a result of the staff's review of the referenced letter[a] or the updated program and test results in CEN-99(S)[c] and CEN-94(S)[b], respectively, the applicants will evaluate the impact on San Onofre Units 2 and 3 to determine if any action is required. 13

Reference

See FSAR section 3.10. No FSAR change was made.

a. C-E letter from A. E. Scherer to K. Kniel (NRC), LD-77-068 June 28, 1977.

b. Seismic Qualification Data for NSSS-Supplied Instrumentation Equipment," Combustion Engineering, Inc., CEN-94(S), July 1978.

c. "Seismic Program for Qualification of NSSS-Supplied Instrumentation Equipment," Combustion Engineering, Inc., CEN-99(S), August 1978. 13

PFR-005

Rakowski's Copy

SAN ONOFRE UNITS 2 AND 3

DOCKETS 50-361 AND 362

Seismic ~~ENVIRONMENTAL~~ Program for QUALIFICATION ~~DATA FOR~~ OF

NSSS-SUPPLIED INSTRUMENTATION EQUIPMENT

CEN-~~95~~ 99(S)

~~JULY~~ August 1978

COMBUSTION ENGINEERING, INC.
POWER SYSTEMS
WINDSOR, CONNECTICUT

1300-9...

p' 2408 PFR - F005 next page

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

The seismic qualification of Category I instrumentation and electrical equipment demonstrates an equipment's ability to perform its required function during and after the postulated design basis earthquake. The demonstration has been accomplished by either of the following two methods (or combinations thereof):

A. Analysis

The equipment performance is predicted by mathematical analysis techniques and accompanied by sufficient followup testing of the equipment to verify the mathematical predictions of the natural frequency and damping.

B. Testing

The equipment performance is determined by testing under simulated seismic conditions as given herein and accompanied by sufficient mathematical analysis to extract the needed information from the test results.

The choice of method was based on the practicality of the method for the type, size, shape, and complexity of the equipment and the reliability of the conclusions.

The Design Basis Earthquake (DBE) and the Operating Basis Earthquake (OBE) horizontal and vertical floor response spectra, reflecting in-structure floor accelerations, were provided to the vendor for a given instrumentation or electrical equipment location. The vendor then determined the appropriate acceleration levels for qualification from these spectra.

In designing the equipment, the vendor combined the effects of gravity loads, normal operating loads, operating temperature loads, other loads that may be included in the specification, and the appropriate DBE and OBE seismic loads.

3.10.1.1 Vendor Documentation

The adequacy of the seismic qualification program is demonstrated in documentation requirements that the vendor fulfills for each equipment type. The documentation demonstrates that the equipment meets its performance requirements when subjected to the loads for which it was qualified. Documentation was required from vendors as described in reference 1 for NSSS equipment and appendix 3.10A for other equipment.

none given, see page 3.10-12

2408-MK-r-003 JLC 3/18/82

3.10.1.2 Analysis Method Requirements

When the analysis method was used, vendors met the requirements of CEN-99(S)(4) for NSSS equipment and appendix 3.10A for other equipment.

3.10.1.3 Test Method Requirements

13 When the test method was used, vendors met the requirements of CEN-99(S)(4) for NSSS equipment and appendix 3.10A for other equipment.

All test data submitted by the vendor to satisfy these requirements was obtained from these test programs which show evidence of performance, supervision, and witnessing of all testing by qualified personnel.

3.10.1.4 Acceptance Evaluation

Upon receiving the equipment supplied by the vendor, selective tests were performed to determine equipment adequacy to meet the specified seismic requirements. The following tests and analyses have been used:

A. Determination of natural frequencies by field testing.

B. Formulation and analysis of a mathematical model.

C. Testing to the stress levels indicated by the analysis of the mathematical model.

13 A list of all Seismic Category I instrumentation, electrical equipment, and supports can be found in table 3.10-1 and CEN-94(S)(3).

Appendix 4F

For further information refer to appendix 3.10A, Criteria for Seismic Qualification of Seismic Category I Equipment, which incorporates the information found in IEEE Standard 344-1971. Appendix 3.10A is the specification provided to vendors, which presents the criteria for seismic qualification of Seismic Category I equipment for San Onofre Units 2 and 3.
13 The criteria used for NSSS equipment is contained in CEN-99(S)(4).

3.10.1.5 Overall Seismic Criteria and Implementation Program

The program for overall seismic adequacy is addressed in San Onofre Nuclear Generating Station, Units 2 and 3 Seismic Category I Criteria and Implementation Program(2). This document describes the scope of the Seismic Category I design program and includes sufficient detail to provide the technical basis for design criteria, analysis methods, and design control implementation.

2408 PFR NO. F049

REVISION - -

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

PREPARATION BY GA INITIATOR

AFFECTED ITEMS: Bechtel Site audit No. 1567, conducted 8/28-29/79.

REQUIREMENT REFERENCE DOCUMENTS:

1) Bechtel QA Standard No. 5.1, Rev. 13, issued 5/10/79, "Project QA Audits", Section 4.2 "auditors"
2) Peabody Testing/X-ray Engineering Co., QA Plan, Rev. C, per audit checklist 5-3.
3) Corrective action statement for characteristics 24 and 25 from the audit check list.

BASIC REQUIREMENT:

SEE ATTACHMENT I

DESCRIPTION OF POTENTIAL FINDING:

Corrective action taken was inappropriate, since it failed to correct the deficient requests for examination.

Per BPC this audit was in error, what the auditor interpreted as requests for examination (after reexamination) were actually the records of notification (scheduling data), the formal requests for examination with proper data, dwg etc. were hand carried to the vendor by a field engineer (Class I+II also QC engineer), therefore this PFR is invalid. Robert A. Sweig 3/18/82

PREPARED BY: Robert A. Sweig DATE: 3/5/82

REJECTION OF GA TASK LEADER COMMENTS BY: ______ DATE: ______

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: ______ DATE: ______

B. **REVIEW BY GA TASK LEADER** COMMENTS

Agree PFR is invalid, per above
SB 3/18/82

☒ AGREE PF IS VALID BY S. Bernal DATE 3/5/82

☐ REQUEST RE-REVIEW BY ______ DATE ______

☐ DISAGREE BY ______ DATE ______

☒ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: S. Bernal DATE: 3/18/82

2408 PFR NO. F049

REVISION ______

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION COMMENTS

SEE ATTACHMENT II

☐ AGREE PF IS VALID

☒ DISAGREE

BY: Fred B Maul DATE: 3/11/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ______

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. L. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: G.K. Wessman DATE: 3/18/82

ATTACHMENT I

BASIC REQUIREMENT:

1) "Review results of the audit, identify individuals responsible for actions to be taken and establish due dates. In the event corrective action cannot be completed within 30 days, the audited organization's response shall include a scheduled date for the corrective action."

2) (From Audit Checklist)
Characteristic 24, procedure paragraph 6.1.1, "Requests for Radiographic Examination. The client will submit a written list of weld joints to be radiographed. The list will indicate the following:

a) Weld joint identification
b) Weld joint size
c) Wall thickness
d) Weld joint location
e) Code or Specification to which radiography is to be performed
f) Other information that may apply to the radiography, i.e., Repair No."

(Auditor's Remark for Characteristic 24)
"OBSERVATION: requests written by D. Bently 8/22/79 and M. Sibley 8/22/79, 8/23/79 do not contain"...subparagraph 6.1.1 e) required data.

(From Audit Checklist)
Characteristic 25, procedure paragraph 6.2.1, "Requests for Other Nondestructive Examinations. Requests will be received in writing or verbally from the client and shall include the following information:

a) Type of nondestructive examination to be performed
b) Identity of item(s)
c) Location of item(s)
d) Weld or material thickness (for MT and UT only)
e) Weld configuration (for UT only)
f) Applicable Code or Specification
g) Any other information that may apply.

(Auditor's Remark for Characteristic 25)
"OBSERVATION: requests for MT or UT examinations do not contain weld or material thicknesses."

3) "LWQCE, D. Martin has instructed the responsible personnel to comply with sub para...."

SUBJECT: BPC Response to GA Potential Finding Report PFR #F-049

Following is BPC response to the finding reported in PFR #F-049:

GA Finding: Corrective action taken was inappropriate, since it failed to correct the deficient requests for examination.

BPC Response: The above finding references characteristic #24 and #25 of Audit Report #1567. The "Requests" for NDE are used merely to notify Peabody (GEO Testing) that the joint or material is ready for specified inspection. The technical data, such as wall thickness, weld joint size, code or specification, etc., are obtained from the welding checklist, which is a "Quality" document and is available at the work location. Further, the above mentioned technical data used by Peabody (GEP Testing) is verified by BPC personnel and ANI.

Accordingly, when the omission of this information from the "Request" forms was noted in the subject audit, corrective action was taken to assure that in the ~~failure~~ this form is properly completed for the appropriate NDE method. Previously completed forms were not revised (backfitted) since it was apparent that the proper data was utilized in the performance of the inspections.

future ?
SB 3/15/82

These "Requests" are non-quality documents and have no impact on quality.

FBM
3-11-82

2408-PFR-F049

Genoral Atomic Company

3/18/82

QUALITY ASSURANCE DEPARTMENT

Record of Long Distance Telephone Call

Party: Called ☒ Calling ☒

Date: 3/16/82 (CONFRENCE CALL)

Time: Completed 1:13 Started 1:10 On-line .3

Name SHELLY Fred (witter) John Shepard (site)

Company BPC

(See F051 for other calls)

Location (SEE ABOVE)

Telephone No: A/C 213 No. 946 1811 ex 273
714 498 1000 ex 250

Discussion

Per John Shepard this was the first audit of a new auditor. When these requests for examination (those reported in error by the audit) were re-examined they were found to be correct. The auditor examined Req notifications that an examination was required and misinterpreted these as the requests for examination. This notification is a note or tel/con to schedule the examination. The request for examination with drawings etc are brought to the vendor before the work starts by the field engineer. For Q class I and II the Quality engineer discusses the QC requirements of the examination with the vendor before work begins.

I said this information was sufficient to close out this PFR and that I was satisfied that it was an erroneous audit and would invalidate the PFR.

Record Made by: Robert H Bewing

Distribution: ≡

PFR NO. 2408-PFR-F059

REVISION ______

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

A. PREPARATION BY GA INITIATOR

AFFECTED ITEMS: Control Room Relay Panels 2L-71 and 3L-71

REQUIREMENT REFERENCE DOCUMENTS: Quality Class II Specification for Quality Class II Panels, Relays and Devices for the Southern California Edison Company, San Onofre Nuclear Generating Station, Units 2 and 3, San Onofre, California. "Specification Number SO23-306-1, SCE Number 3274, July 31, 1945".

BASIC REQUIREMENT: A. Section 3.10A, 3.3.1.1 of Appendix 3.10A of the FSAR defines two methods for test qualification of assemblies, i.e., fully operational assemblies versus cabinet testing with dummy weights. B. Section 3.10A.3.3.2 states that the assembly shall be mounted to the vibration generator in a manner that simulates the intended service mounting.

DESCRIPTION OF POTENTIAL FINDING: A. The method of testing assemblies with dummy weights is to be used if the fully operational test is not practical, however, in this case it is practical. Hence, by providing representative equipment as stated in Paragraph 4.6.3.5 of the specification, with no requirement for the vendor to install dummy weights elsewhere, the vendor could have interpreted the specification to call for a cabinet test with only the representative equipment installed. If this was the case, the test was not valid because the amplification and frequency content of the test configuration may be quite different from the one installed in the plant and equipment operability would not be verified. (See attached sheet).

PREPARED BY: J. Rakowski J Rakowski DATE: 2/25/82

REJECTION OF GA TASK LEADER COMMENTS BY: ______ DATE: ______

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: ______ DATE: ______

B. REVIEW BY GA TASK LEADER

COMMENTS

ar ☒ AGREE PF IS VALID BY [signature] DATE 2/26/82

☐ REQUEST RE-REVIEW BY ______ DATE ______

☐ DISAGREE BY ______ DATE ______

☐ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: ______ DATE: ______

PFR NO. 2408-PFR-F039

REVISION

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

See attached sheet.

COMMENTS

Concur with recommendation to invalidate this PFR. Reviewer's comments are shown on attachments. Note that initiator of PFR (J. Rakowski) is offsite. Processing of this PFR has been assigned to A. Schwartz. FBO 3/18/82

☐ AGREE PF IS VALID

☒ DISAGREE

BY: Fred B Marsh DATE: 3/2/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY: ☒ ADEQUATE ☐ INADEQUATE

VALIDITY: ☐ VALID ☒ INVALID

CLASSIFICATION: ☐ OBSERVATION ☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ____________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. A. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: [signature] DATE: 3/18/82

Attachment PFR No: 2408-PFR-F059

Description of Potential Finding (Continued)

B. Purchaser determination of a specific test panel configuration indicates that either panels 2L-71 and 3L-71, the only panels covered by this specification, are different, or only one dissimilar section of the overall panel need be tested. Paragraph 4.1.2 indicates that dissimilar panel sections may make up 2L/3L-71. There is no requirement that the test configuration determined by the Purchaser is to be representative of all fully assembled panels covered by this specification. If only one panel section is to be tested, there is no requirement that the qualification remain valid when the sections are joined together.

Bechtel Response
to 2408-PFR-F059

Relay panels 2L-71, 3L-71 and their components were seismically qualified by combined analysis and type testing which is acceptable. The design engineer approved the above approach based on detailed reviews and evaluations with the vendor as indicated in the following documents.

a. NMC telexes to Dave Morrow and Perry Kine dated 2/1/77, 2/2/77 and 1/26/77

b. NMC letter to Bechtel dated 10/24/77, 11/8/77, 12/14/77 and 12/1/78

The following are additional documents which substantiate the method chosen and the qualification of the panels.

a. Test procedure for the seismic qualification (BPC Log S023-306-1-8-2).

b. Seismic qualification of relay panels 2L-71 and 3L-71 (BPC Log S023-306-1-37-1).

c. Seismic vibration analysis of control room relay panels 2L-71 and 3L-71 (BPC Log S023-306-1-37-2).

d. Procedure for the seismic qualification of control room panel 2L-71 and 3L-71 (BPC Log S023-306-1-43).

e. Seismic vibration analysis of panel 2L-71 and 3L-71 (BPC Log S023-306-1-44-0).

The above documentation clearly indicate that qualification tests and analyses were completed to include all configurations appropriate. The purchase specification need not include a complete detailed discussion of these considerations so long as the post-procurement documentation properly addresses the design requirements and is reviewed for acceptability by Bechtel.

~~[illegible]~~
3/3/82

Based on the information contained in the test procedure and verbally supplied by the original design organization it was found that.

a) The purchaser did not supply representative equipment

b) The purchaser did not determine which panel section was to be tested

c) Panel sections were qualified by analysis separately and together

Therefore this PFR is considered invalid. For further details see attachment.

[illegible] for J. Kalcowski
3/18/82

Attachment to PFR 2408-PFR-F059

Paragraph 4.6.3.5 of the specification requires qualification by test, and further states that the purchaser will determine the specific panel configuration to be tested and will also supply representative panel mounted equipment to be included in the panel test program. However, paragraph 4.1.3 requires that the vendor shall be responsible for purchase, installation and wiring of all panel mounted equipment. Also, Appendix 4F of the specificatin permits qualification by test, analysis, or a combinatin thereof. The documentation supplied by the original design organization indicated that the panels were qualified by analysis and the equipment was separately qualified by test (on a shaker table) to a peak acceleration ~~euql~~ equal to or greater than the acceleration obtained in the analysis at the mounting location of maximum response for the equipment type.

Regarding potential finding A, the purchaser did not supply representative equipment. Since this potential finding is based on such supply, it is not applicable. The specification adequately covers qualification when all equipment is vendor supplied. The analytical qualificatin procedure describes how vendor supplied equipment is treated.

Regarding potential finding B, the purchaser did not determine which panel section or configuration was to be tested. There are two types of panel sections per panel. The test procedure requires that they were both to be qualified by analysis. Additionally, an analysis was to be performed on the panel sections joined together to determine that the qualification will remain valid under that condition. These analyses were performed as evidenced by the ~~test~~ report.

qualification ok

2408 PFR NO. F079

REVISION --

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

A. PREPARATION BY GA INITIATOR

AFFECTED ITEMS:

Combustion Engineering Component Specifications - see attached

REQUIREMENT REFERENCE DOCUMENTS:

PE-001, Rev. 1, Section 6.1.4.3 (Combustion Engineering Procedure)

BASIC REQUIREMENT:

"The Design Requirements document will specify functions, definitions, performance requirements, compliance with codes, standards, regulations, mechanical and material considerations, interface and testing requirements, and other design basis to the level of detail necessary to permit the design activity to be carried out in a correct manner and to provide a consistent basis for making design decisions, accomplishing design verification measures, and evaluating design changes."

DESCRIPTION OF POTENTIAL FINDING:

(SEE ATTACHMENT I)

PREPARED BY: DATE: 2/26/82

REJECTION OF GA TASK LEADER COMMENTS BY: ______ DATE: ______

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: DATE: 3/17/82

B. REVIEW BY GA TASK LEADER

COMMENTS

Although CE did supply "a design requirement documents" in their response, these do not represent ~~the~~ formal comprehensive + auditable document which we believe would be appropriate to meet the Basic Requirement above.

SB 3/17/82

☒ AGREE PF IS VALID BY DATE 2/26/82

☐ REQUEST RE-REVIEW BY ______ DATE ______

☐ DISAGREE BY ______ DATE ______

☒ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: DATE: 3/17/82

2408 PFR NO. F079

REVISION ---

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

COMMENTS

There appears to be some confusion as to the basic criteria to be used in TASK B - Design Procedure Implementation Review. It is C-E's understanding that the design work was to be evaluated against the procedures in effect when the design work was done. Nowhere in the Project 2408 Program Plan is PE-001, for instance, specified as the basic criteria to be used. This confusion has led to the generation of several PFR's that appear to require the documents be "back-fitted" to comply with procedures that become effective several years after the document was originally produced. The PFR is such a case.

☐ AGREE PF IS VALID

☒ DISAGREE

Nevertheless, the design input discussed in the PFR was contained in the Specifications for the equipment listed. C-E produced these specifications to be used for the actual design work on the equipment mentioned in 2408-PFR-F079 were seen by GA's Quality Assurance

BY: F.D. Bennett for VC Hill DATE: 3/17/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY: ☒ ADEQUATE ☐ INADEQUATE

VALIDITY: ☒ VALID ☐ INVALID

CLASSIFICATION: ☒ OBSERVATION ☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ________

COMMENT ON "OBSERVATION" CLASSIFICATION

Design Requirements Documents required by procedure were not produced. However, for equipment items, design input is contained in the equipment Specifications. Although beyond the scope of this PFR, the committee is concerned about the control of systems related design input information.

BY: S. L. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: G.H. Wessman DATE: 3/18/82

team during their review at Windsor during the week of 2/15/82.

Having stated the above, C-E still wishes to address any concerns raised by this PFR. With that in mind, attached are a sample of the memos and letters that deal with design input for the equipment in question. Included also are the title pages from the Plant Engineering Design Quality Assurance procedures in effect in 1970, 1971 and 1973.

Memos attached:

S-CE-187	Transmitted Pressurizer Seismic Accelerations (Project Manager to Chattanooga, 10/8/70)
S-PM-142	Transmitted Preliminary Loading Date for Bottom of Support Skirt (Project Manager to Chattanooga, 4/30/71)
IOC	W. Stolecki to G. Huba/E. A. Steen, 4/29/71 (transmitted data used in S-PM-142)
S-CE-1027	Transmitted Preliminary Pressurizer Support Load Table (Project Manager to Bechtel, 2/22/74)
IOC	Transmitted Max. Accelerations for Pressurizer CG (R. Kassawara to T. Ianuzzi, 2/14/75)
S-PSE-079	Transmitted Seismic Loads for SONGS Pressurizer Support (D. Satter to R. D. Haun, 4/4/75)
S-PM-15	Transmitted Specification for Seismic Design Criteria (Asst. Project Manager to Engineering, 2/24/70)
IOC	Transmitted SCE Response Spectra - Containment Bldg. (A. Swan to P. W. Weilhouwer, et. al., 7/12/74)
BC-1268	Transmitted Final Response Spectra for Aux. Bldg., Containment Bldg., etc. (Bechtel to Project Manager, 10/14/77; sent to I&CE and Plant Engineering by routing stamp
BC-?	Transmitted Response Spectra for Aux. Bldg. (Bechtel to Project Manager, 7/24/73; sent to I&CE and Plant Engineering by routing stamp)
PSE-77-025	Transmitted Proposed Rev. to R. V. Assembly Specification (R. P. Kassawara to W. E. Stolecki, 3/4/77)
S-PM-1262	Discussion of R. V. Column Supports Seismic Loads (Project Manager to D. A. Peck, 11/18/76)
S-PCE-334	Transmitted Seismic Requirements for Valves (A. J. Tillman to Engineering, 4/22/75)
S-PCE-263	Discussion of Design Changes due to 33H Requirements (P. R. Wade to Project Manager, 2/26/75)

S-CE-3364	Discussion of Pressurizer Branch Line Seismic Spectra (Project Manager to Bechtel, 10/22/76)
BC-PKGE#574	Transmitted Safeguard System Pumps Spec. Comments (Bechtel to Project Manager, 8/9/74; sent to I&CE and Plant Engineering by routing stamp)
S-PSE-039	Transmitted SCE R. V. Support Loads (W. E. Stolecki to E. E. Magette, 10/10/74)

2408 PFR No. F079

ATTACHMENT I

DESCRIPTION OF POTENTIAL FINDING:

Design input was not available as required in PE-001. It is recognized that certain of the attached items may have been prepared prior to the issuance of PE-001. However, PE-001 is being used as the basic criteria as specified in the Project 2408 Program Plan, and no evidence could be found that the basic requirement was met in any manner.

The point of concern is the control of design input in the 1969-1975 period. Design input by today's standards must be available when design work commences, must be reviewed and approved and must be controlled as changes occur.

Components examined which had design work in the 1969-1975 period were:

Safety Injection Tank T008
LPSI Pump P016
Valves 2FV-0306 Spec. 1370-PE-704 (SI 306)
2HV-9342 Spec. 1370-PE-704 (SI 611)
2HV-9341 Spec. 1370-PE-704 (SI 618)
2HV-9322 Spec. 1370-PE-705 (SI 635)
Reactor Coolant Pump
Reactor Vessel Supports
Pressurizer Unit 2
Pressurizer Unit 3
Containment Spray System Pump
Boric Acid Make-up Tank - Unit 3

IMPACT ASSESSMENT

2408 PFR NO. F079

AFFECTED ITEM: Combustion Engineering Component Specification

1. IS THERE THE POTENTIAL FOR REDUCING DESIGN MARGINS TO THE EXTENT DESIGN ALLOWABLES ARE EXCEEDED OR DESIGN REQUIREMENTS ARE NOT MET?

   N/A

2. IS THERE THE POTENTIAL THAT THE ITEM MIGHT FAIL OR ENDANGER OTHER ITEMS DURING AN SSE?

   N/A

3. COULD THE FAILURE OF THIS ITEM DURING AN SSE CREATE A SUBSTANTIAL SAFETY HAZARD?

   N/A

4. COULD THE PROCEDURAL VIOLATION CREATE A SUBSTANTIAL SAFETY HAZARD?

   The lack of rigid control of design inputs such as by a "Design Requirements Document" could have caused designs not being responsive to needed inputs.

5. ARE OTHER SIMILAR DEVIATIONS LIKELY TO EXIST?

   N/A

6. OTHER COMMENTS:

   This PFR is of a procedural nature - no actual design deficiencies were noted in this part of the study.

PREPARED BY: George Chanelle DATE: 3/17/82

COMMENTS: None

BY: S Brown DATE: 3/17/82

PFR 2408 + 077

General Atomic Company

QUALITY ASSURANCE DEPARTMENT

Record of Long Distance Telephone Call

Party: Called ☑ Date: 3/17/82
Calling ☐ Time: Completed 9:45
Started 9:30
On-line :15

Name Jake Westhaven

Company Combustion Engineering

Location Windsor, Conn.

Telephone No: A/C 203 No. 688 1911 X 4114

Discussion

Subj. PFR F079 & F081

The status of these PFR's was discussed - They're at The stage where I'm asked to accept or reject The original design organization response.

I told Jake I was going To Reject their response because they did not have "Design Requirements Documents" and because memos and routing copies of the Contract, in my opinion, were not good methods of providing design bases.

Jake pointed out that The General Spec's were design requirement-containing documents and That he Thought the contract was a good thing to work from.

Record Made by [signature]

Distribution: S. Bresnick, PFR 081 File, PFR 079 File, Reg. 2408 File

2408-PFR/F079
Sce 3/18/82

T W X

DATE RECEIVED
OCT 14 1970

OCTOBER 8, 1970

TO: W. W. ROBERTS - CHATTANOOGA

SUBJECT: SAN ONOFRE PRESSURIZER

CONFIRMING TELECON, G. J. HUBA TO W. W. ROBERTS, OCTOBER 1, 1970, THE FOLLOWING ARE THE SEISMIC ACCELERATIONS (g's) FOR THE SAN ONOFRE PRESSURIZER. THESE WILL APPEAR IN 4.6.8 AND 4.6.9 OF THE SPECIFICATION. "DESIGN" IN 4.6.8 WILL BE CHANGED TO "OPERATIONAL BASIS EARTHQUAKE" AND "MAXIMUM" IN 4.6.9 WILL BE CHANGED TO "DESIGN BASIS EARTHQUAKE".

OPERATIONAL BASIS EARTHQUAKE		DESIGN BASIS EARTHQUAKE	
HORIZONTAL	VERTICAL	HORIZONTAL	VERTICAL
.8	.525	1.6	1.05

VERY TRULY YOURS,

COMBUSTION ENGINEERING, INC.

V C Hall / [illegible]

V. C. HALL

VCH/GJH/JOL:ljs
S-PA-097
S-CE-187

	ACT	INF
EJP		
EAS	E. A. S.	OCT 14 1970
EEM		
WOW		
FILE		✓

6

SCE SAN ONOFRE 2 & 3	A	I	ENCL.
PROJ MGR		✓	
ASST MGR		✓	
PROJ ADM			
Core Dsgn			
Plt Eng		✓	
Huba		✓	
Safety			
Prod. Plan.			
Purch.			
P EUXD		✓	
Contracts		✓	
Field Off.			
Rdg. File			
Hold File			
Chrono. File		✓	
Q. C.			

SCE PRESSR. SUM

INTEROFFICE CORRESPONDENCE

2406-PFR-F079
[initials] 3/18/82

COMBUSTION DIVISION

To: W. W. Roberts (3)

cc: G. J. Huba ✓
D. R. Wade
W. E. Stolecki
File

Southern California Pressurizer

V. C. Hall

S-PM-142

April 30, 1971

Enclosure: (1) Loading at Bottom of Support Skirt-Preliminary-dated April 29, 1971.

Enclosure (1) is forwarded for your use in sizing the support skirt and ordering material.

V C Hall

V. C. Hall, Jr.
Project Manager

VCH/GJH/EAS:iy

S-PA-383

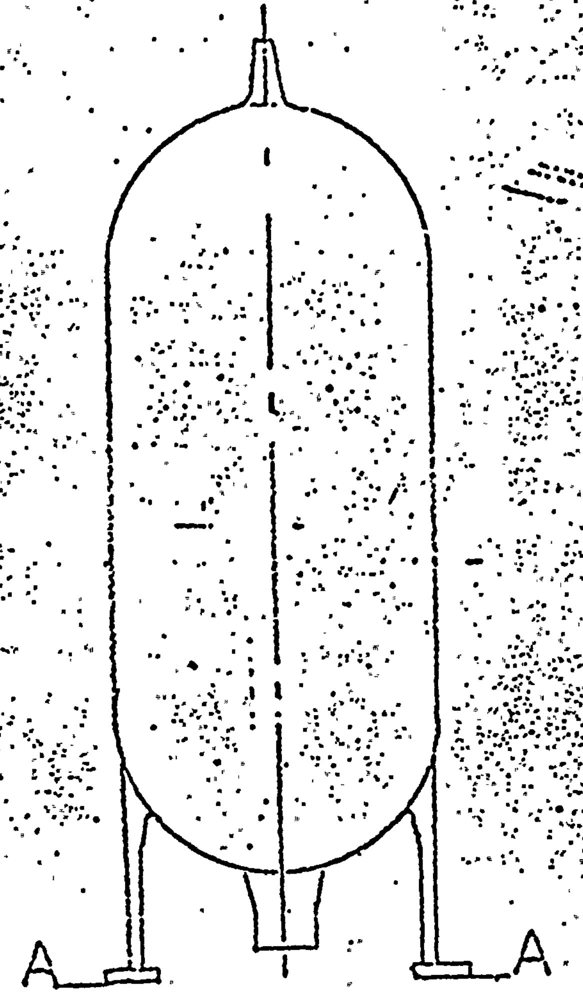
DATE RECEIVED
MAY 5 1971

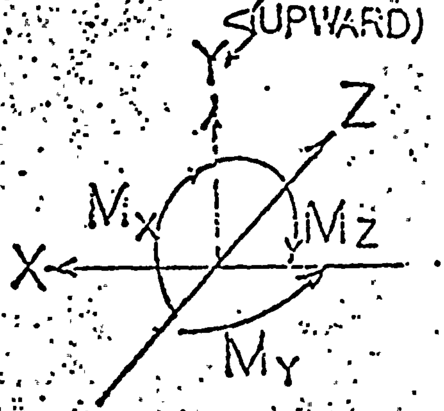
[initial]	ACT	INF
EJP		
EAS		E.A.S. MAY - 6 1971
EEM		
WOW		
FILE		✓

2408 PFR-F079

Encl. (1) to S-PM-383

S-PM-142





PRELIMINARY

CONDITION		FORCES KIPS Fx	FORCES KIPS Fy	FORCES KIPS Fz	MOMENTS IN-KIP Mx	MOMENTS IN-KIP My	MOMENTS IN-KIP Mz
THERMAL EXPAN.		−5.2	−2.2	+20.3	−325	−1533	−192
DEAD WEIGHT		−.2	+306.6	−.2	−247	−13	+262
COMBINED		−5.4	+304.6	+20.1	−572	−1546	+70
OPERAT BASIS EARTHQ (NOTE 1)	± X	±259	0	±9.6	±221	±326	±53,400.
	± Y	0	±256	0	±930	±99.4	±905
	± Z	±6.4	1.0	±269	±53,600.	±246	±228
PIPE	CASE 1	F_V= 402.5		F_H= 0	M_B= 29,000.		M_T= 0
RUPT	CASE 2	F_V= 0		F_H= 51.5	M_B= 23,242.		M_T= 0
NOTE 2	CASE 3	F_V= 0		F_H= 402.5	M_B= 0		M_T= 19,100.

NOTES: (1) DESIGN BASIS EARTHQUAKE = 2 X OPERATIONAL BASIS EARTHQUAKE

(2) SUBSCRIPTS DENOTE V = VERTICAL, H = HORIZONTAL, B = BENDING, T = TORSION & ACT IN THE DIRECTIONS WHICH GIVE THE WORST LOADING COMBINATIONS DOES NOT INCLUDE NORMAL OPERATING AND SEISMIC LOADS

(3) FORCES ACT ON SUPPORT SKIRT

dsb 4/29/71

LOADING AT BOTTOM OF SUPPORT SKIRT

CONNECTICUT BUSINESS FORMS, INC NORWALK, CONN.

2406-PFR-F079
3/18/82

COVER ONE SUBJECT ONLY IN EACH LETTER

3730

E 3637 (4/67)

INTER OFFICE CORRESPONDENCE

TO: HUBA / E.A. STEEN

DEPT.-LOCATION

FROM: W. STOLECKI

DEPT.-LOCATION

SUBJECT: SCE #2 - PRESSURIZER DESIGN LOADS

DATE: 4/29/71

MESSAGE:

ATTACHED HEREWITH ARE THE SUPPORT SKIRT LOADS FOR THE SUBJECT PRESSURIZER. THESE LOADS ARE PRELIMINARY AND CAN BE USED FOR SIZING AND DESIGN CALCULATIONS. THEY SHOULD NOT BE USED FOR A FINAL STRESS REPORT.

ORIGINATOR - DO NOT WRITE BELOW THIS LINE

SIGNED W. STOLECKI

REPLY:

E.A.S. APR 29 1971

To NCD:
S-PA-383
S-PM-142

4/30/71

DEPT.-LOCATION

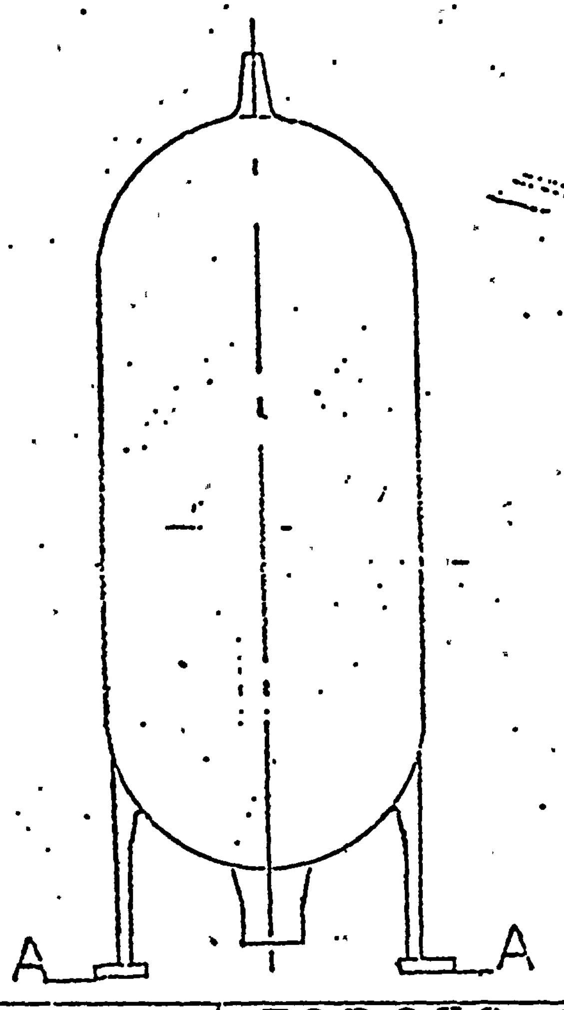
SIGNED

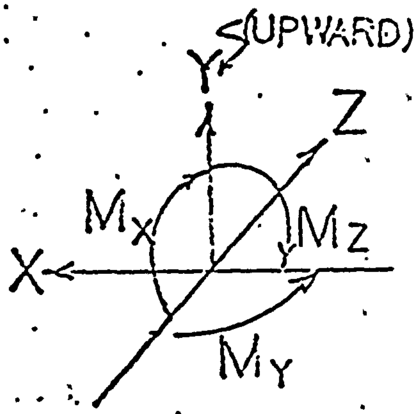
DATE / /

SEND WHITE AND PINK COPIES WITH CARBON INTACT—PINK WILL BE RETURNED WITH REPLY.

2408-PFR-F079-

Add. (1) to S-PA-
S-PM- 3/15/82





PRELIMINARY

CONDITION		FORCES KIPS Fx	Fy	Fz	MOMENTS IN-KIP Mx	My	Mz
THERMAL EXPAN.		-5.2	-2.2	+20.3	-325	-1533	-192
DEAD WEIGHT		-.2	+306.6	-.2	-247	-13	+262
COMBINED		-5.4	+304.6	+20.1	-572	-1546	+70
OPERAT BASIS EARTHQ (NOTE 1)	± X	±259	0	±9.6	±221	±326	±53,400.
	± Y	0	±256	0	±930	±99.4	±905
	± Z	±6.4	±0	±269	±53,600.	±246	±228
PIPE	CASE 1	F_V=402.5		F_H= 0	M_B=29,000.		M_T= 0
RUPT	CASE 2	F_V= 0		F_H= 51.5	M_B=23,242.		M_T= 0
NOTE 2	CASE 3	F_V= 0		F_H= 402.5	M_B= 0		M_T=19,100.

NOTES: (1) DESIGN BASIS EARTHQUAKE = 2 X OPERATIONAL BASIS EARTHQUAKE

(2) SUBSCRIPTS DENOTE V = VERTICAL, H = HORIZONTAL, B = BENDING, T = TORSION & ACT IN THE DIRECTIONS WHICH GIVE THE WORST LOADING COMBINATIONS DOES NOT INCLUDE NORMAL OPERATING AND SEISMIC LOADS

(3) FORCES ACT ON SUPPORT SKIRT

dated 4/29/71

LOADING AT BOTTOM OF SUPPORT SKIRT

Specification No. 1370 -PE-130	Revision	Figure 16 Sheet —

2408-PER-F079
3/18/82

DATE RECEIVED
FEB 26 1974

COMBUSTION DIVISION, COMBUSTION ENGINEERING, INC.
WINDSOR, CONN 06095
203-688-1911 CABLE: COMBENG

COMBUSTION DIVISION

February 22, 1974

S-CE-1027
S-PA-776

Southern California Edison Company
San Onofre Units 2 and 3
SCE Order No. N1800001
Bechtel Job No. 1304-606
C-E Contracts 1370 and 1470

E.A.S. FEB 28 1974

T.W. FEB 28 1974

Mr. J. D. Houchen
Bechtel Corporation
P. O. Box 60860
Terminal Annex
Los Angeles, California 90060

Subject: Pressurizer Supports

References: (A) BC-223, J. D. Houchen to R. W. DeVane, August 15, 1973, Pressurizer Support Spectra
(B) BC-234, J. D. Houchen to R. W. DeVane, September 18, 1973, Pressurizer Support Locations

See PR Sum.

Attachment: (L) Preliminary Support Load Table

Dear Mr. Houchen:

Review of the design data transmitted by Reference (A) indicates that the pressurizer support skirt is capable of withstanding the resulting seismic loadings without requiring the upper horizontal support keys discussed in Reference (B). It is proposed to delete the upper horizontal keys from the pressurizer.

Concurrence on your part should include consideration of the support loads shown in Attachment (1) and consideration of seismic motions for branch lines at the top of the unkeyed pressurizer. The maximum horizontal acceleration and displacement at the top of the unkeyed pressurizer are 1.67g's and ±.050 inches respectively for OBE, and DBE equal to 1.67 times OBE.

1408-PFR-F074.
JDC 3/18/82

February 22, 1974
S-CE-1027

In order to maintain our pressurizer fabrication schedule we request your concurrence with this change no later than March 22, 1974.

Very truly yours,

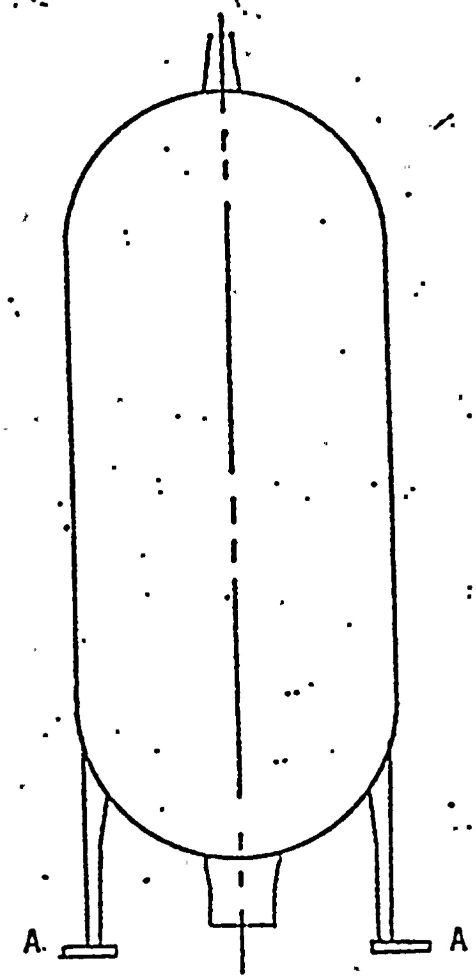
R.W. DeVane /

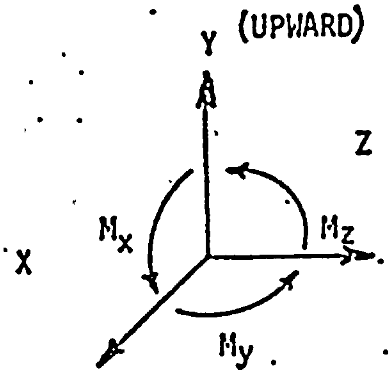
R. W. DeVane, Jr.
Project Manager

RWD/DDM:jj

cc: L. D. Hamlin (SCE)
R. G. Lacy (San Diego G&E Co.)
W. L. MacDonald (CE-Orange, Calif.)

2408-PFR-F074
JAC 3/18/82





PRELIMINARY

CONDITION		FORCES KIPS Fx	Fy	Fz	MOMENTS IN-KIP Mx	My	Mz
THERMAL EXPAN.		-5.2	-2.2	+20.3	-325	-1533	-192
DEAD WEIGHT		-.2	+306.6	-.2	-247	-13	+262
COMBINED		-5.4	+304.6	+20.1	-572	-1546	+70
OPER. BASIS EARTH-QUAKE	± X	150	0	0	0	0	33,380
	± Y	0	100	0	0	0	0
	± Z	0	0	150	33,380	0	0
PIPE	CASE 1	FV = 402.5		FH = 0	MB = 29,000		MT = 0
RUPT.	CASE 2	FV = 0		FH = 51.5	MB = 23,242		MT = 0
NOTE 2	CASE 3	FV = 0		FH = 402.5	MB = 0		MT = 19,100

NOTES: (1) DESIGN BASIS EARTHQUAKE = 1.67 OPERATIONAL BASIS EARTHQUAKE

(2) SUBSCRIPTS DENOTE V = VERTICAL, H = HORIZONTAL, B = BENDING, T = TORSION & ACT IN THE DIRECTIONS WHICH GIVE THE WORST LOADING COMBINATIONS

(3) FORCES ACT ON SUPPORT SKIRT

Title: LOADING AT BOTTOM OF SUPPORT SKIRT

Figure 9 Sheet

GUSTAVE FISCHER COMPANY
HARTFORD, CONNECTICUT 06130

2405 PE RE 079

COVER ONE SUBJECT ONLY IN EACH LETTER

INTER OFFICE CORRESPONDENCE

TO T. Januzzi	DEPT.-LOCATION 487-4
FROM R. Kassawara	DEPT.-LOCATION 487-4
SUBJECT SCE Pressurizer g-loads	DATE 2/14/75

MESSAGE:

The following are the maximum accelerations to be applied to the pressurizer CG for seismic loading

ØBE { 1.65 g in any horizontal direction
{ .30 g in the vertical direction

DBE { 1.75 g in any horizontal direction
{ .50 g in any vertical direction

These are to be considered final loads.

ORIGINATOR - DO NOT WRITE BELOW THIS LINE SIGNED R. P. Kassawara

REPLY:

FILE SCE PRESS SUMMARY

DEPT.-LOCATION	SIGNED	DATE / /

SEND WHITE AND PINK COPIES WITH CARBON INTACT—PINK WILL BE RETURNED WITH REPLY.

INTEROFFICE CORRESPONDENC

2408·PER-F079

COMBUSTION DIVISION

TO: R. D. Haun

SCE
San Onofre Units 2 & 3
Pressurizer Support
Seismic Loading

D. Sattar

S-PSE-079

April 4, 1975

Enclosure: Seismic Loads for San Onofre Pressurizer Support

Subject enclosure supersedes what is currently in the specification. Please note that the format has not changed, only the magnitude of numbers.

D. Sattar

D. Sattar

W. Stoleki

DS:dmk
encl.

LE SCE PRESS

2408-PFR-F079

COMBUSTION ENGINEERING, INC.
WINDSOR, CONN.

CUSTOMER SOUTHERN CALIFORNIA EDISON CONT. No. 1370 MADE BY D.S. DATE 4-4
LOCATION SAN ONOFRE 2 AND 3 DWG. No. ____ CHK'D BY ____ DATE ____

SEISMIC LOADS FOR PRESSURIZER SUPPORT

CONDITION			FORCE - KIPS			MOMENTS - IN. KIPS			KIPS
			Fx	Fy	Fz	Mx	My	Mz	Fy
SEISMIC	OBE	X±Y	120	180	10	2610	260	7565	75
		Z±Y	20	200	150	3790	1310	5440	100
	DBE	X±Y	165	240	10	3650	315	11440	85
		Z±Y	30	340	150	4330	1600	9490	115

SHEET 1 OF 1

2408-PFR-F079

COMBUSTION ENGINEERING, INC.

ADDRESSEE	SUBJECT	FROM — DATE
Engrg E. P. Flynn J. E. Myers	Transmit SCE Specifications	D. F. Streinz February 24, 1970 S-PM-15

PROJ OFF

cc: B. J. Cochran (w/o encl.)
J. D. Crawford (w/o encl.)
W. W. Albert (w/o encl.)

Chrons

Enclosures:
(1) Specification for Seismic Design Criteria San Onofre Nuclear Generating Station Units 2 and 3
(2) Southern California Edison Company Specification No. A64-1967, 13,200 Volt A-C Induction Motors for Generating Stations
(3) Southern California Edison Company Specification No. A54-1969, 4160 Volt A-C Induction Motors for Generating Stations
(4) Southern California Edison Company Specification No. A65-1969, 460 Volt A-C Induction Motors for Generating Stations
(5) SCE Questions - Nos. 1, 13 and 14 and related answers

The enclosed documents are forwarded for your information and use during the design of San Onofre Units 2 and 3. Enclosure (1) establishes the SCE seismic design criteria which was agreed to by CE except as amended by the answers to questions Nos. 1 and 13. Enclosures (2), (3) and (4) are SCE induction motor specifications and establish the motor standards required for Units 2 and 3. These standards were accepted by CE as explained in the answer to Question No. 14.

D F Streinz

D. F. Streinz

DFS:jmt

SIT

2408-PFR-F079

COVER ONE SUBJECT ONLY IN EACH LETTER

Plant Apparatus JUL 17 1974

INTER OFFICE CORRESPONDENCE

TO: P. Wittlemann / A. Tessor / J.A. Peck — DEPT.-LOCATION:

FROM: A Scurra — DEPT.-LOCATION:

SUBJECT: SCE Response spectra, Containment Bldg — DATE: 7/12/74

MESSAGE:

[illegible] 5023-SE-627 [illegible] 670

for your info
formal Transmittal to follow, by Bechtel

A Scurra

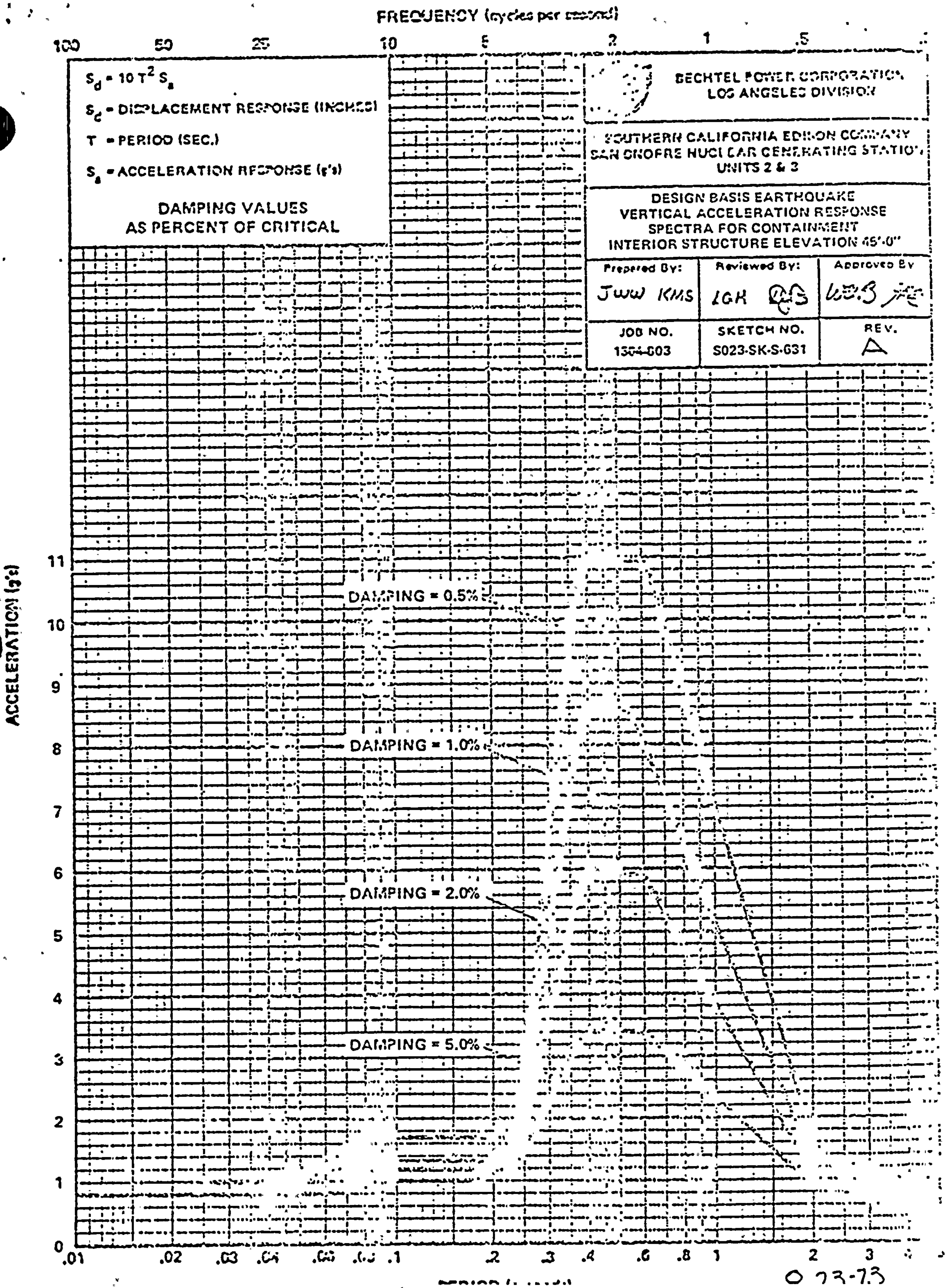
ORIGINATOR • DO NOT WRITE BELOW THIS LINE — SIGNED

REPLY:

DEPT.-LOCATION — SIGNED — DATE / /

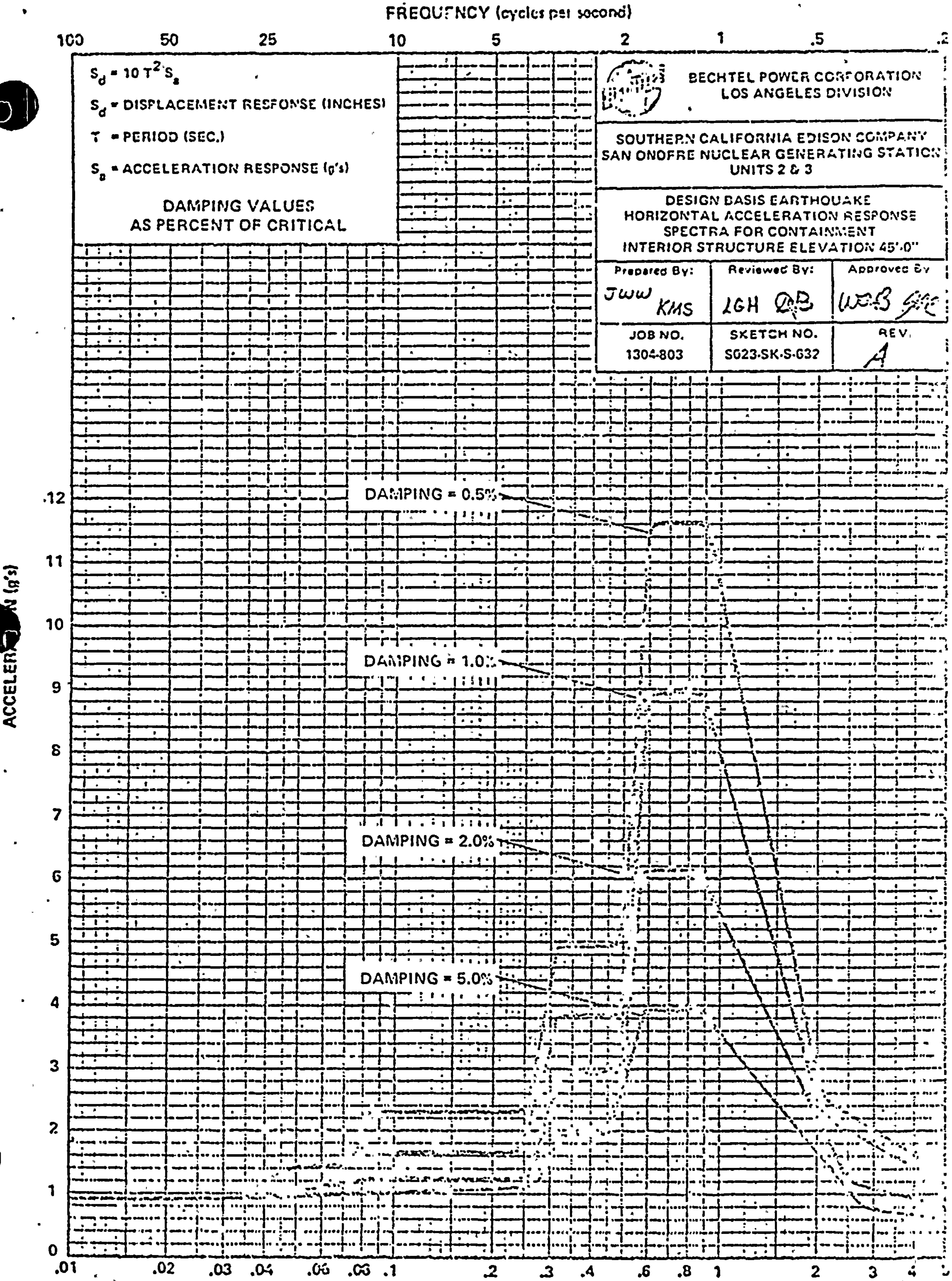
SEND WHITE AND PINK COPIES WITH CARBON INTACT—PINK WILL BE RETURNED WITH REPLY.

2408-PFR-F079

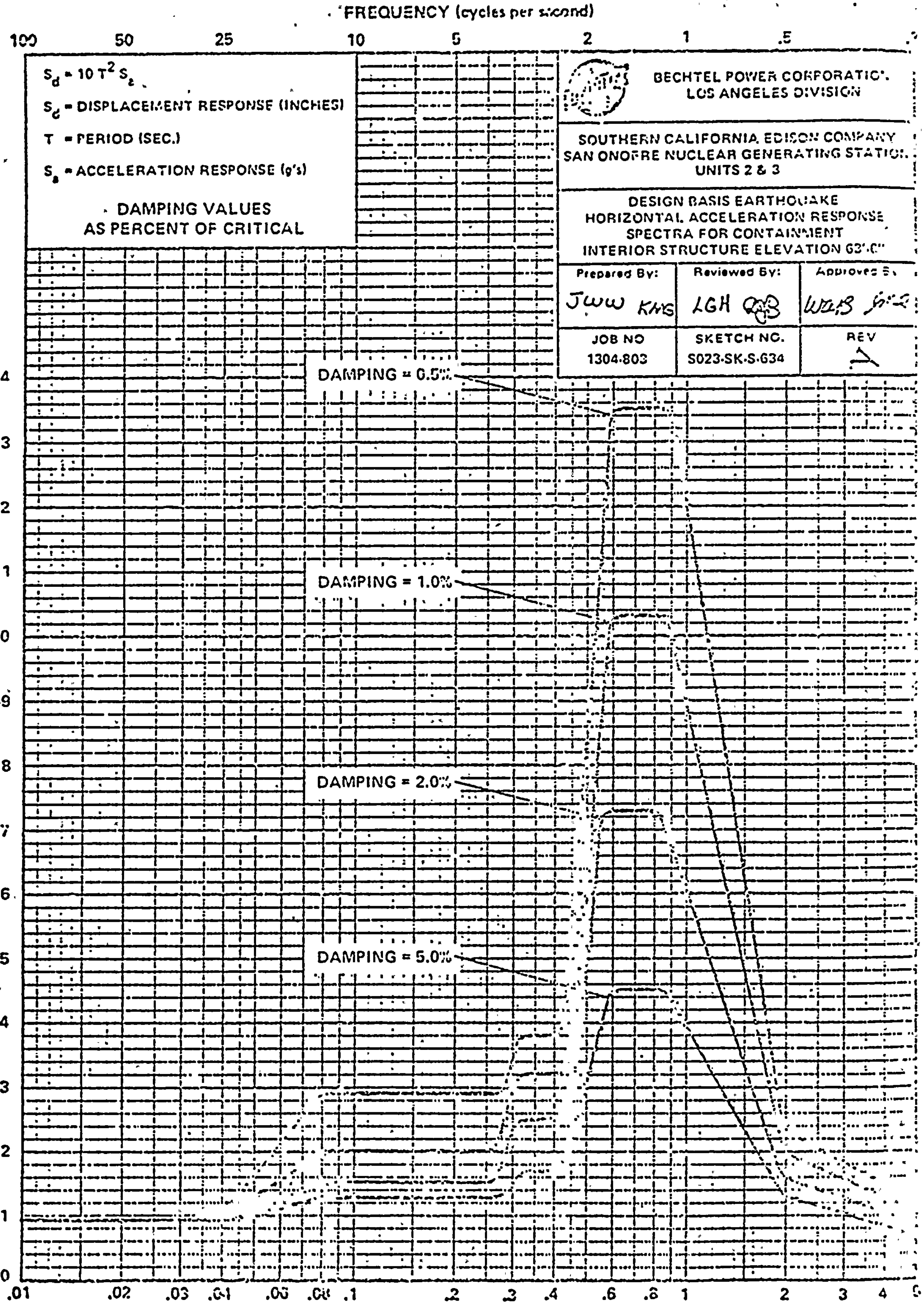


0 23-73

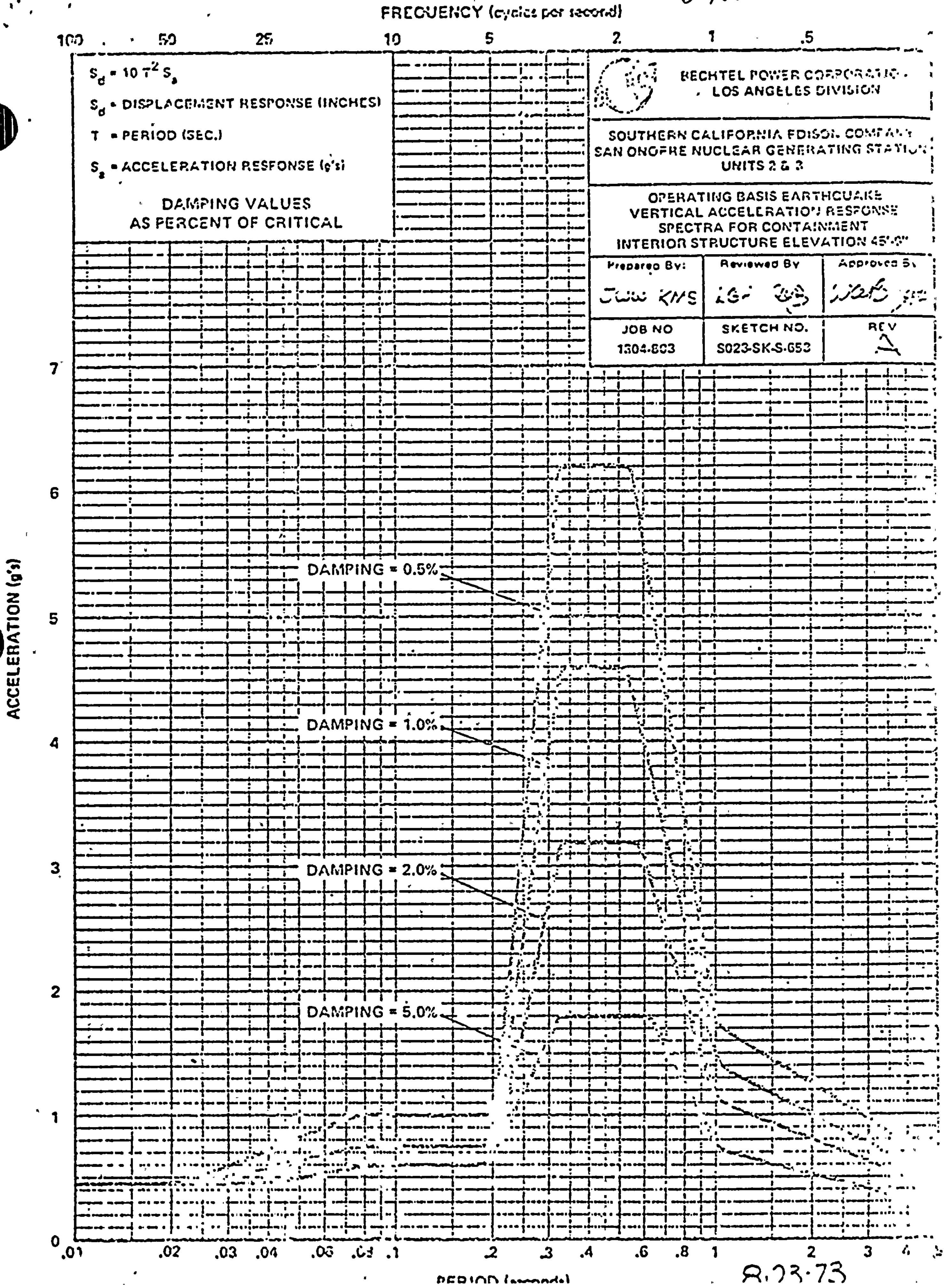
2408-PFR-F079.



2408-PFR-F079-

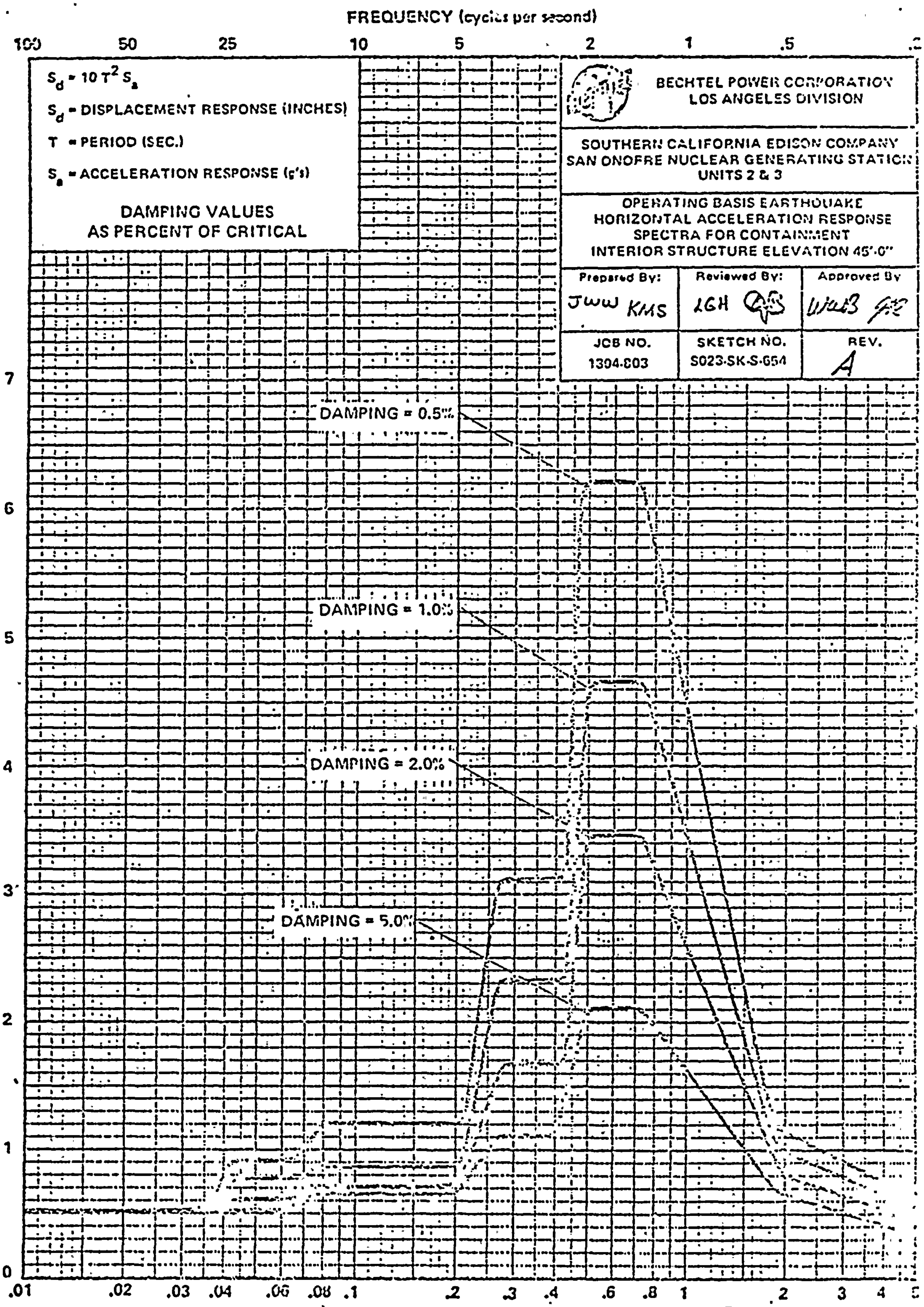


2408-PFR-F079

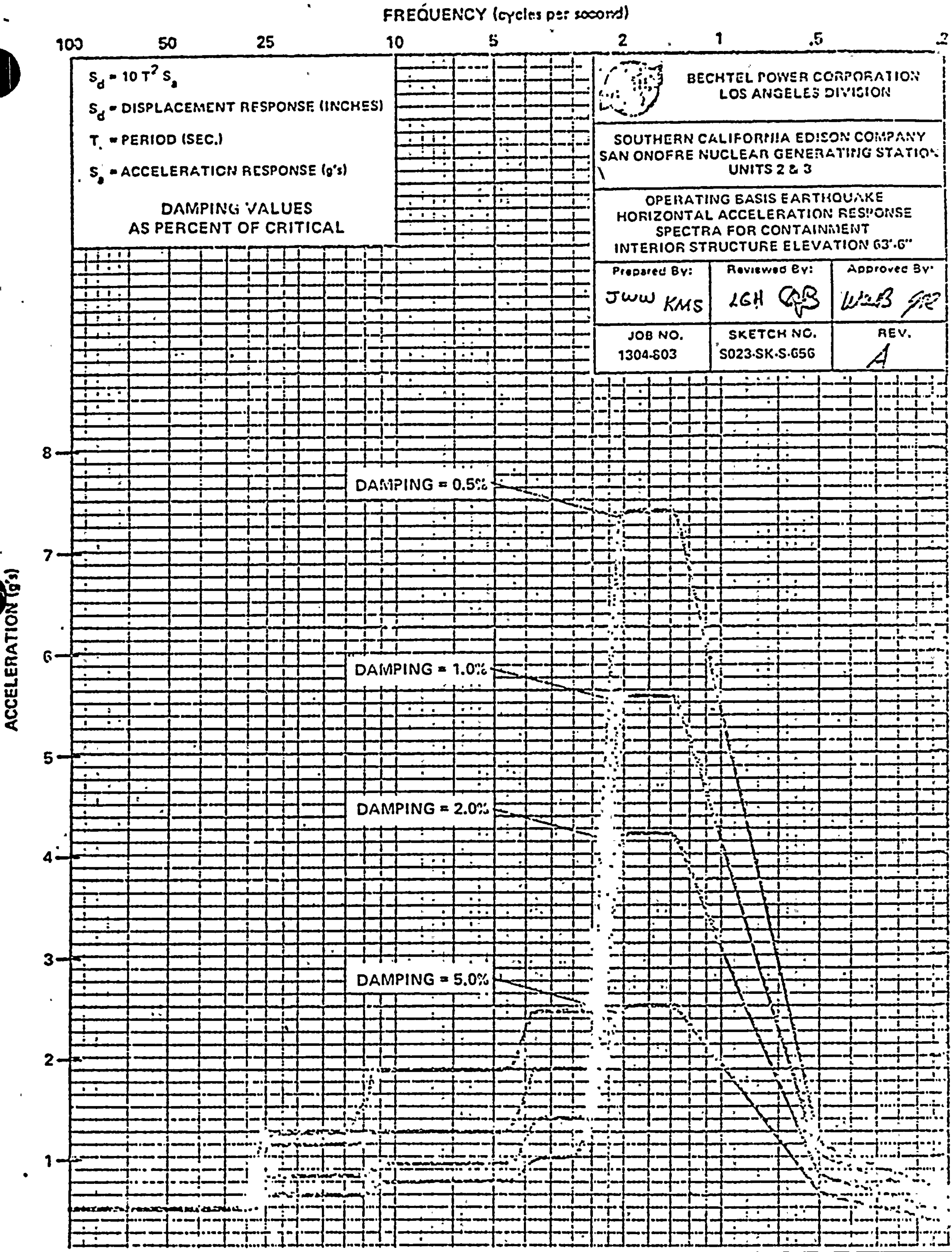


8-23-73

2408-PFR-F079



2408-PFR-F079



R.W.D. OCT 17 1977
2408-PFR-F079

A.J.R. OCT 18 1977

CC: T.E. Burke
R. L. Kassawara
INFO

Bechtel Power Corporation
Engineers - Constructors

12400 East Imperial Highway
Norwalk, California 90650
MAIL ADDRESS
P.O BOX 60860 - TERMINAL ANNEX, LOS ANGELES, CALIFORNIA
TELEPHONE:(213) 864-6011

SCE SAN ONOFRE 2 & 3			
ROUTE/COPY	A	I	ENCL.
R. W. D.		✓	
R. C. C.		✓	
A. H. S.		✓	
A. B. S.			
REAC. ENG.			
	✓		
ICE.	✓		
CONTRACTS			
SITE MGR.			
RES. ENG.			
		✓	

October 14, 1977
BC-1268

Combustion Engineering Co., Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Attention: Mr. R. W. DeVane, Jr., Project Manager

Subject: Southern California Edison Company
San Onofre Nuclear Generating Station, Units 2 & 3
Bechtel Job 10079
Seismic Response Spectra INDEF
File: S023-250B

Reference: (A) CE letter S-CE-4213 dated September 29, 1977.

Gentlemen:

In response to Reference (A), the following Bechtel transmittals represent final spectra:

SIT Containment Building, BC-205 dated June 28, 1973 ✓
Safety Equipment Building, BC-911 dated January 15, 1976 ✓
Fuel Handling Building, BC-529 dated January 8, 1975 ✓
BAMT Auxiliary Building, BC-213 dated July 24, 1973. ✓

Very truly yours,

BECHTEL POWER CORPORATION

J. D. Houchen
Project Engineer
Los Angeles Power Division

KL:lp

cc: Mr. S. V. Tashjian, SCE
Mr. W. MacDonald, CE-Orange, Calif.

SAN ONOFRE UNITS 2 & 3 PROJECT
SO

BAMT

Info. copy to D. Peck
2408-PFR-F079
X 3

SCE SAN ONOFRE 2 & 3

PROJ MGR	ASST MGR						Prod. Plan	Purch.	Proj. Serv.	Contract	Field	Calc.	Field	Construc.	Q.C.

Bechtel Power Corporation

Engineers – Constructors

12400 East Imperial Highway
Norwalk, California 90650
MAIL ADDRESS
P.O. BOX 60860 - TERMINAL ANNEX, LOS ANGELES, CALIFORNIA 90060
TELEPHONE: (213) 864-6011

July 24, 1973

R.W.D. JUL 30 1973

Combustion Engineering Co , Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

D.A.P. APR 9 1974

Attention: Mr. R. W. DeVane, Jr., Project Manager

Subject: Southern California Edison Company
San Onofre Nuclear Generating Station, Units 2 & 3
Bechtel Job 1304-803
In-Structure Response Spectra for Auxiliary Building
File: S023-258-A, Log BC-213

Enclosures: Three (3) copies each of:

(1) In-Structure Response Spectra for the Auxiliary Building, Design Basis and Operating Basis Earthquakes

Sketch Numbers S023-SK-S-689 through 740, Revision A

	Drawing Number	Title
(2)	C-25318-A	Auxiliary Building, Seismic Analysis, Lumped Parameter Model
(3)	C-25300-A	Auxiliary Building, Seismic Analysis, Plans and Sections

Gentlemen:

Enclosed are the in-structure response spectra for the Auxiliary Building for both horizontal and vertical response. Analysis were performed and response spectra developed for both Operating Basis and the Design Basis Earthquakes.

2408-PFR-F079

Bechtel Power Corporation

Mr. R. W. DeVane, Jr., Project Manager
Combustion Engineering Co., Inc.
Page Two - Log BC-213

July 24, 1973

The response spectra presented (See Enclosure (1)) are referenced by direction of response, node number and elevation within the Auxiliary Building. Enclosure (2) presents the definition of the nodal points, including their coordinate location in three-dimensional space and an outline of the portions of the building associated with each node. Enclosure (3) is for general information and offers a schematic representation of the Auxiliary Building.

If we can be of any further assistance please let us know.

Very truly yours,

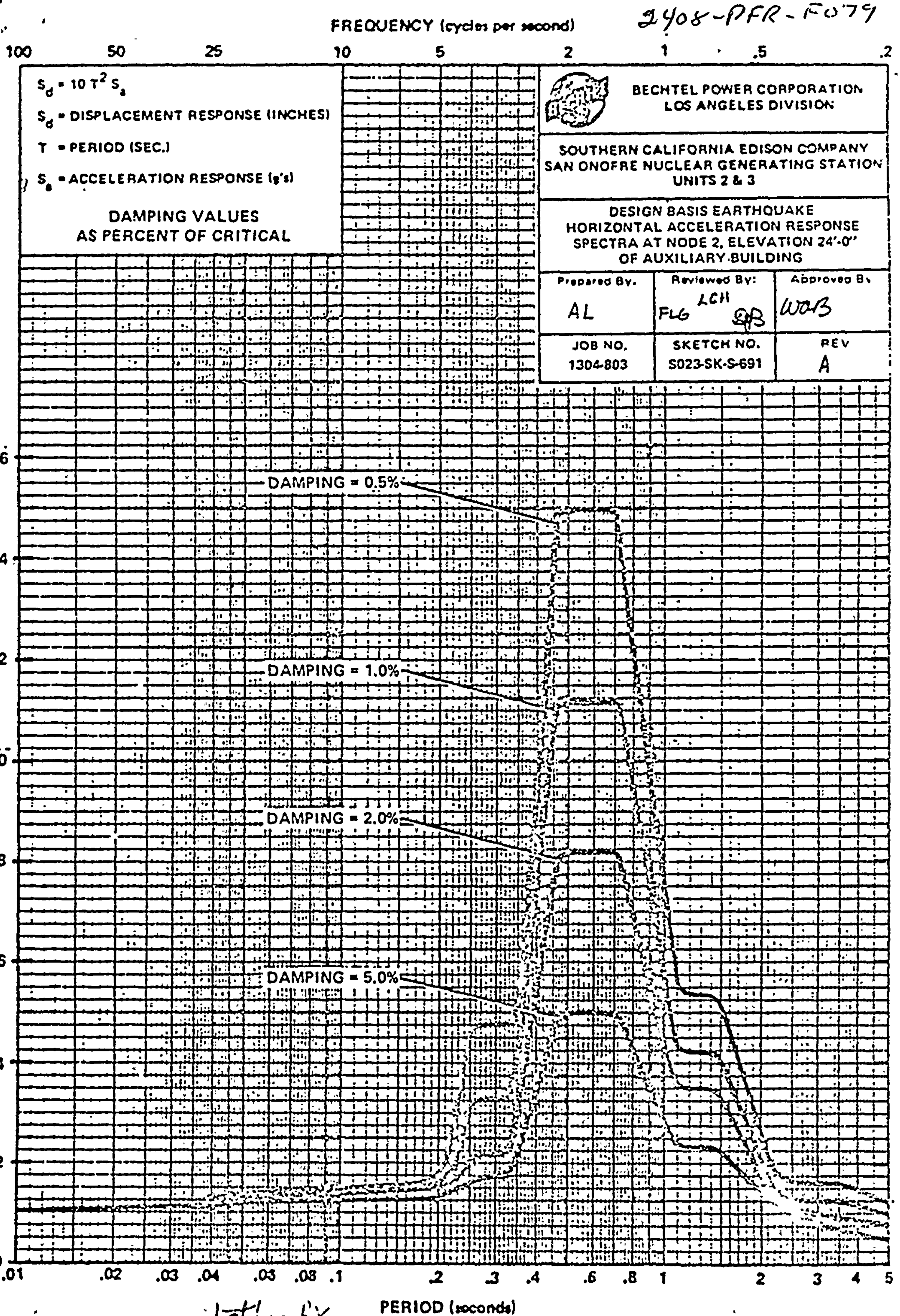
BECHTEL POWER CORPORATION

J. D. Houchen

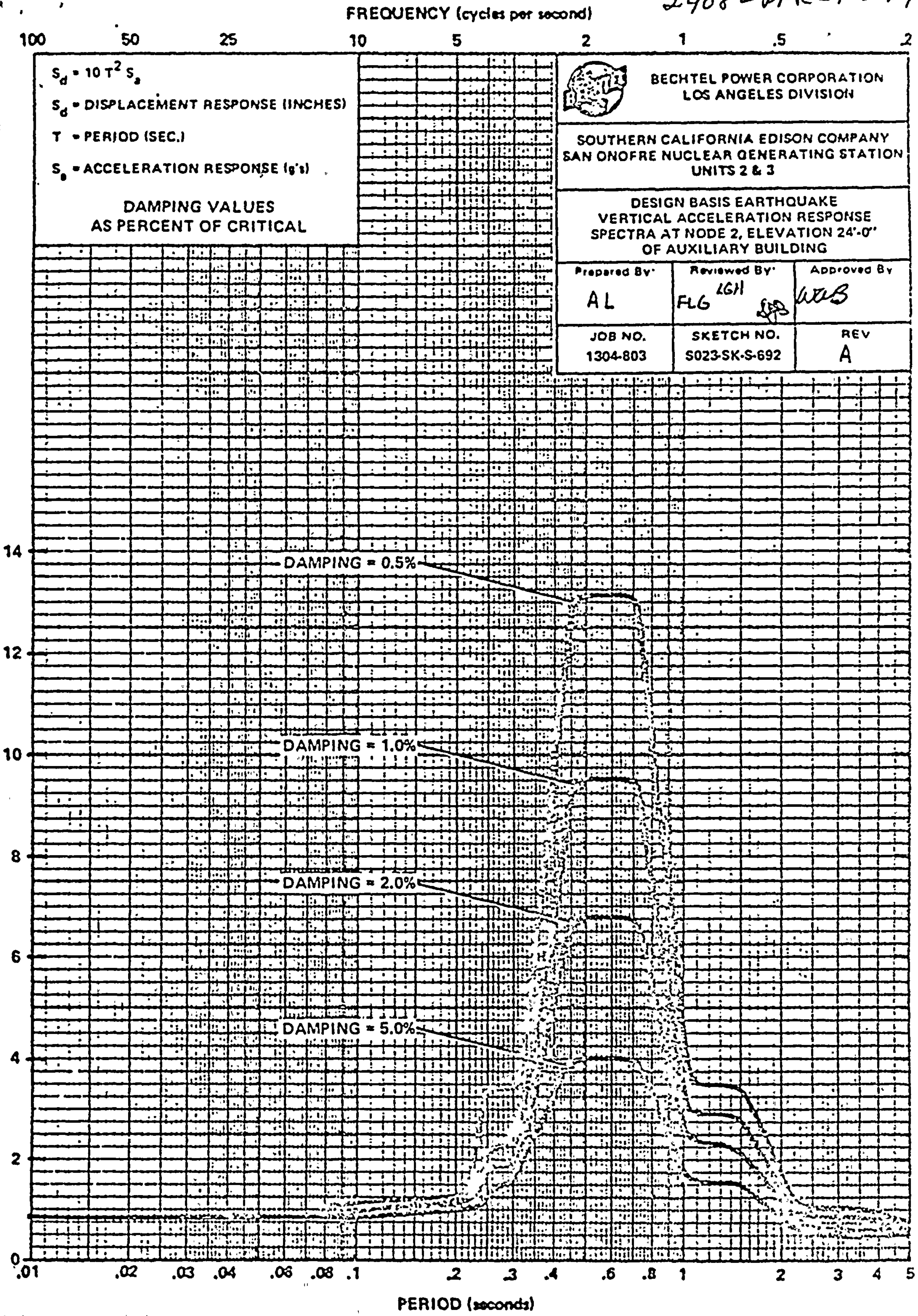
J. D. Houchen
Project Engineer
Los Angeles Division

LGH:dsr
Enclosures
cc: Mr. L. D. Hamlin, SCE
Mr. D. F. Martin, SCE
Mr. W. McDonald, CE-Orange, Calif.

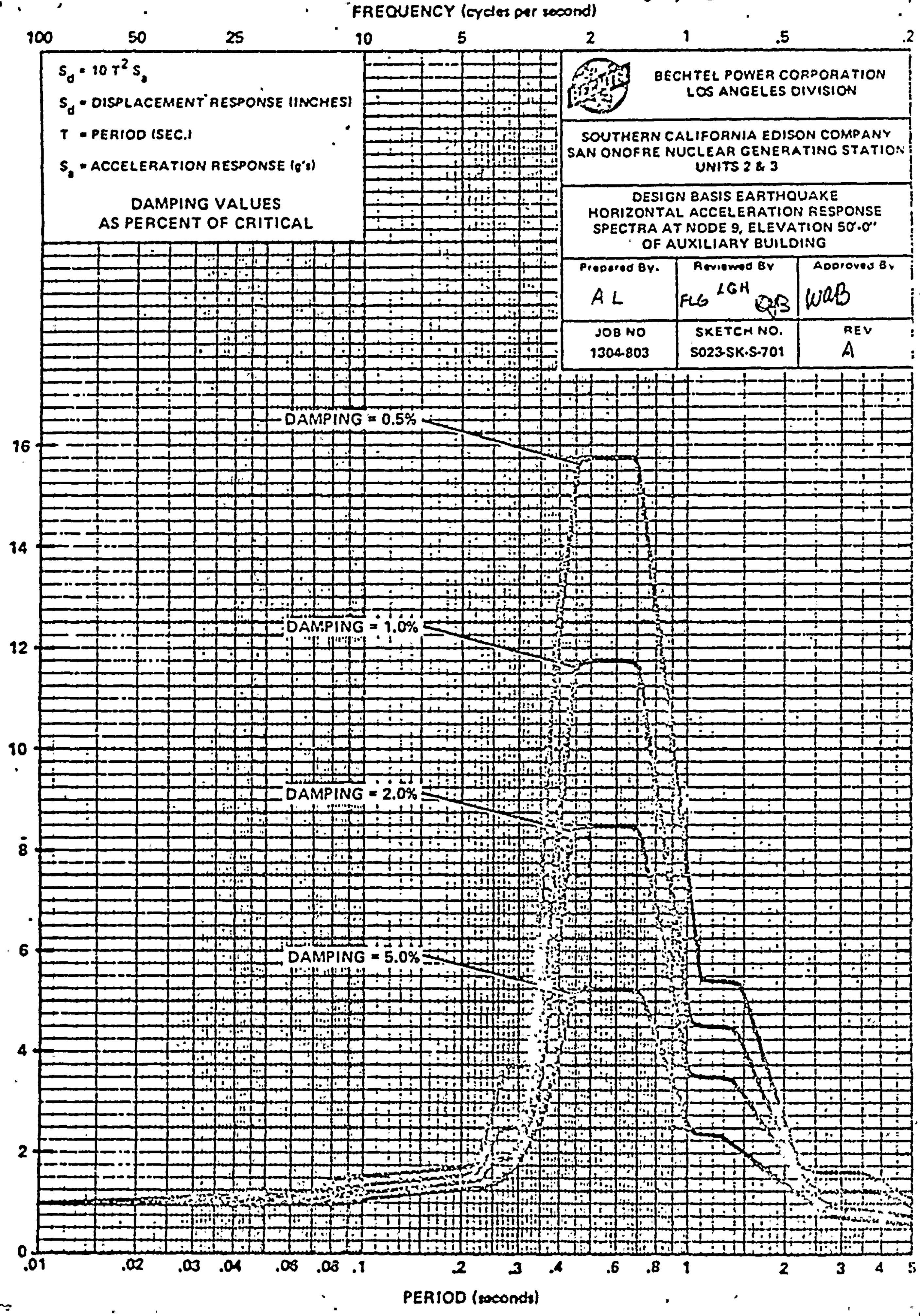
2408-PFR-F079



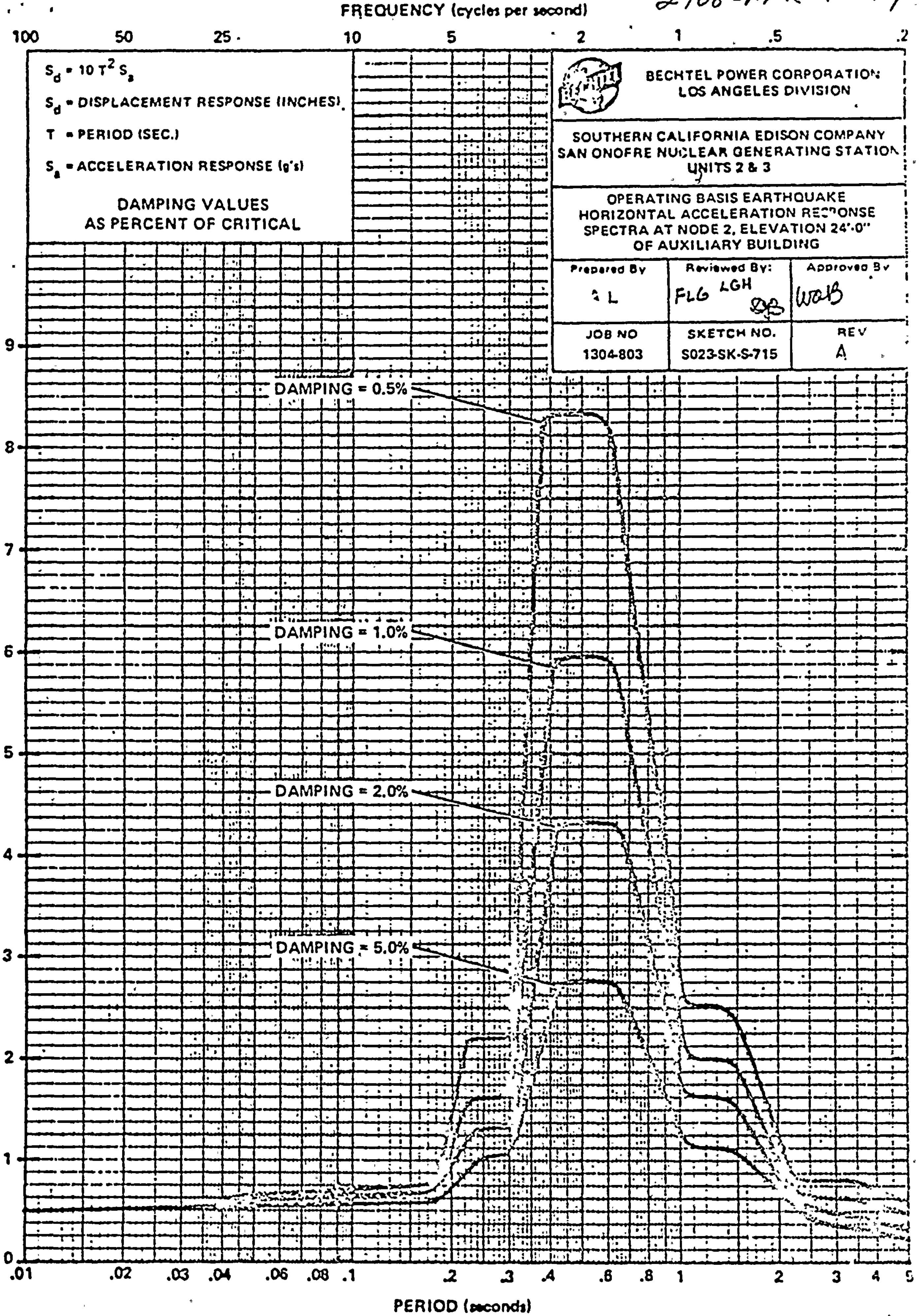
2408-PFR-F079



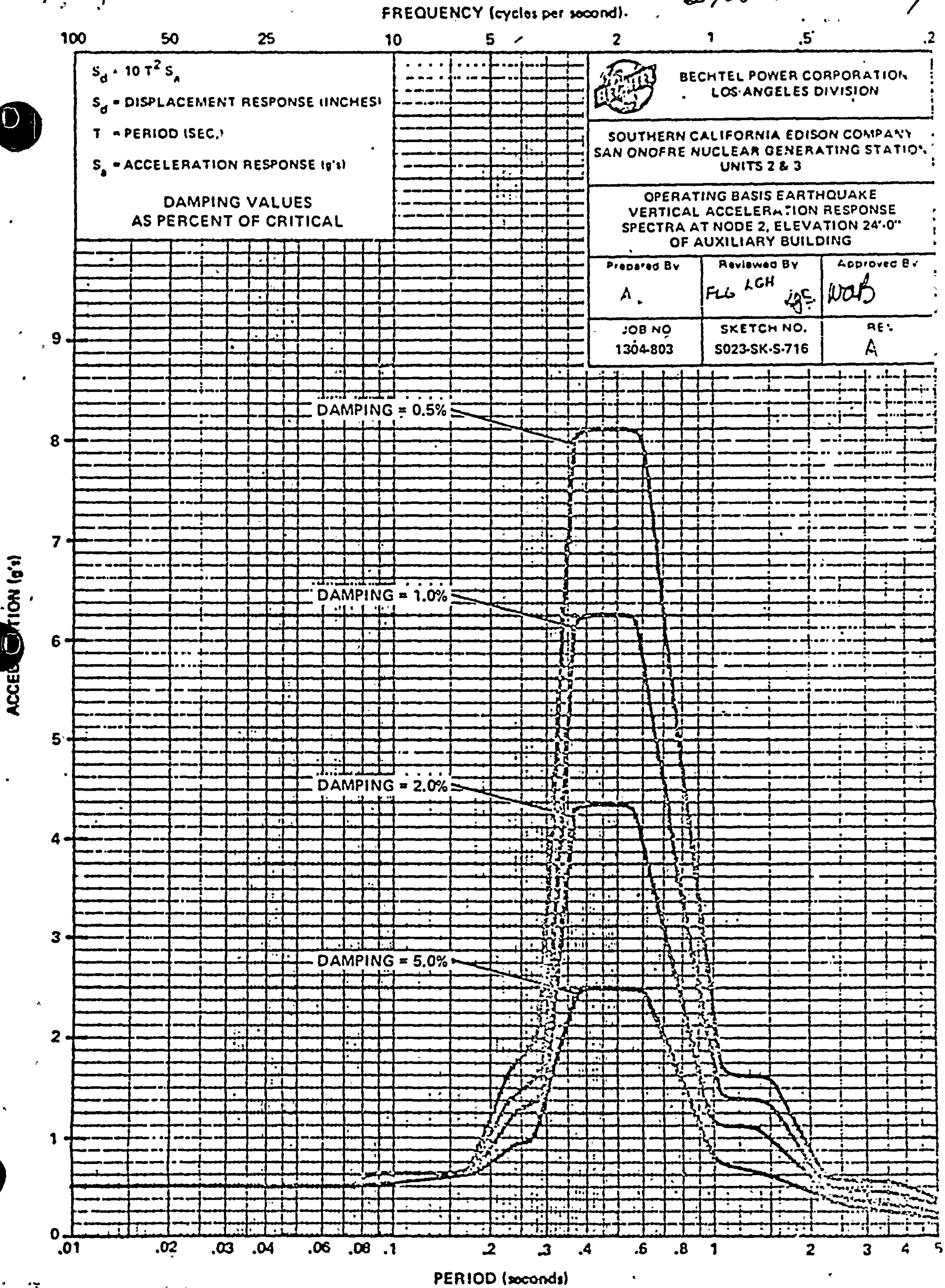
2408-PFR-F079



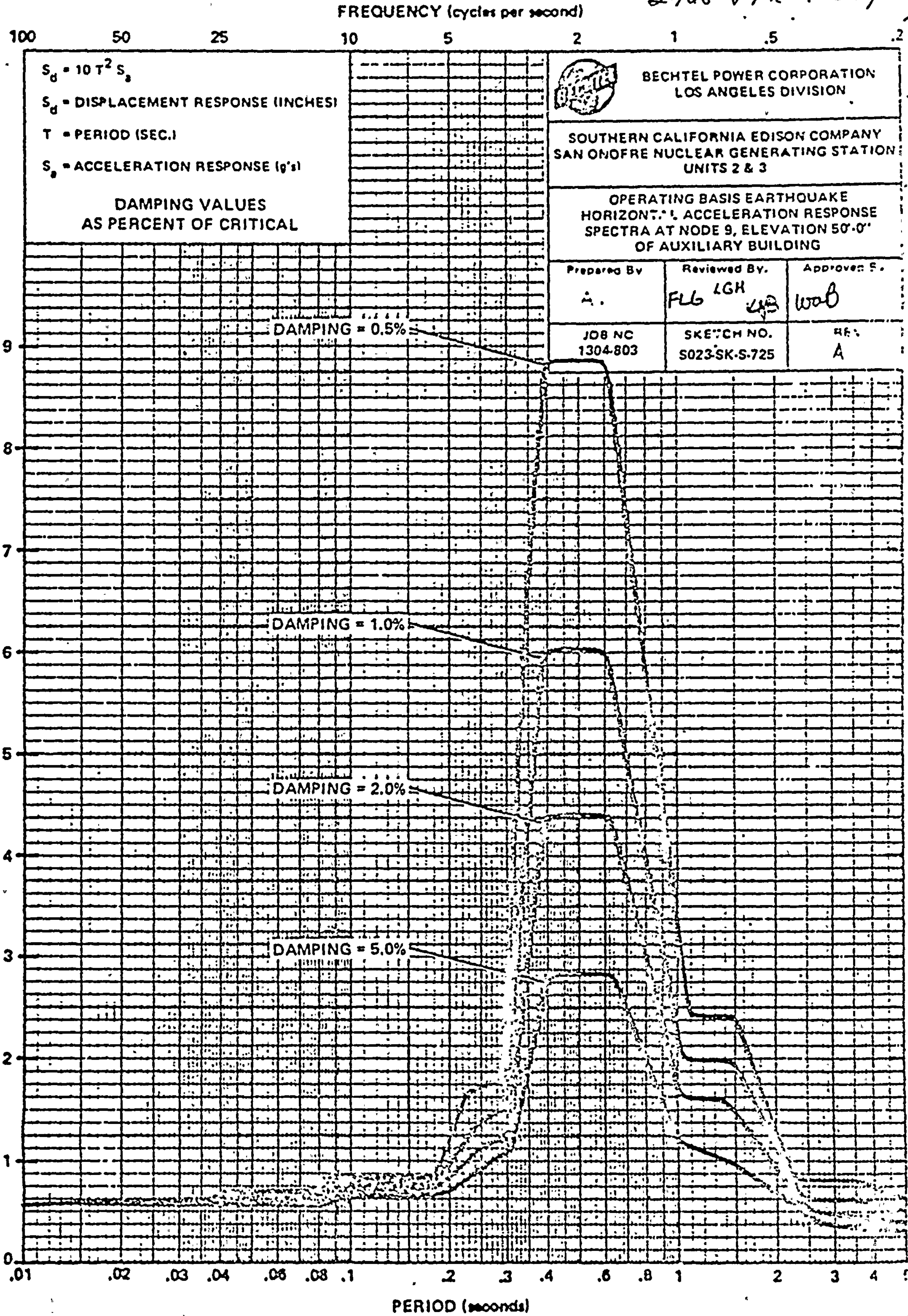
2408-PFR-F079



2408-PFR-F079



2408-PFR-F079



Interoffice Correspondence

2408-PFR-F079

file

POWER SYSTEMS

Proposed Revision to SONGS 2 & 3 Reactor Vessel Assembly Specification

To: W. E. Stolecki

cc: T. E. Natan
D. A. Peck

R. P. Kassawara

PSE-77-025

March 4, 1977

Reference: (1) Specification Number 01370-PE-110, Rev. 4, Project Specification for A Reactor Vessel Assembly for San Onofre Units 2 & 3

(2) S-CE-3646, Reactor Vessel Support LOCA Loads, R. W. DeVane Jr. to J. D. Houchen, dated February 9, 1977

This memo transmits to you proposed changes to the referenced specification (Ref 1). The changes given in the attachments are a result of the "North Anna" analysis of the reactor vessel supports. The changes are based on results which were transmitted to Bechtel in Reference (2) as "High Confidence Level". QA will be completed by April 15, 1977. Also included are specification changes due to the seismic reanalysis.

R. P. Kassawara

R. P. Kassawara

RPK:car

L.R.Y. NOV 19 1976

Interoffice Correspondence

2408-PFR-F079

POWER SYSTEMS

Dan,
Pls. prepare unplanned WK. statement & then we will send Dick a Budget Change Request. TRY 11/19

D. A. Peck

XC: T. R. Young
D. B. Grogan

Reactor Vessel Column Supports Seismic Loads

R. W. DeVane, Jr.
November 18, 1976
S-PM-1262

As a result of the coupled seismic analysis that you performed on the San Onofre Reactor Coolant System you identified two seismic loads that increased over the present values in the specification.

At that time you suggested that your groups could perform a review of the reactor vessel supports stress analysis to demonstrate that the equipment could take the increased loads. I was not in agreement with you at that time and in fact I thought that the specification loads should be revised. Since that time I have reviewed this situation and I agree with your original proposal.

Please proceed with a review of the vessel supports stress analysis in order to demonstrate that the increased loads are acceptable. The review should be documented and placed in the files for later use if necessary.

R. W. DeVane

R. W. DeVane, Jr.
Project Manager

RWD:AHS:jwb

INTEROFFICE CORRESPONDENCE 2408-PFR-F979

I. R. J.

COMBUSTION DIVISION

R.W.D. APR 23 1975

To: W. W. Albert
W. G. Blowers
R. W. DeVane
S. E. Gilley
T. L. Kettles
A. N. Major
R. L. Moscardini
D. A. Stanton

Implementation of
Southern California Edison
Seismic Requirements
for Valves

A. J. Tillman

S-PCE-334

April 22, 1975

RWD
HVT
AHS
TRY
SO

This correspondence replaces S-PCE-276, dated March 10, 1975.

1. Add the following to Specifications 1370-PE-704 and 1370-PE-705.

4.2.5.1 For "active" Seismic Category I valves, the demonstration shall be documented by a report showing that the minimum natural frequency exceeds 33 hertz and that the valve is capable of operating and remaining intact during and after exposure to the faulted condition of Paragraph 4.2.5.3. The report shall include a description of tests, test results and ~~all~~ calculations used for the demonstration. When the demonstration is by analysis, the analytic method shall be confirmed by tests on at least one valve manufactured by the Supplier. These tests must confirm the natural frequency calculation accuracy and confirm the analytic method used to demonstrate operability. The Purchaser shall approve the selection of valves tested to verify analysis.

4.2.5.3 The faulted condition is the following concurrent loading:

a. Loads due to 3g static acceleration acting on the extended structure in the most severe direction from the standpoint of valve deformation.

b. Loads due to the design pressure.

c. End loads due to valve weight and due to moment imposed by attached pipe stressed to 6000 psi.

2. Add the following to Specifications 1370-PE-704, 1370-PE-705, and 1370-PE-716.

4.2.5.2 For Seismic Category I valves that are not designated "active", the demonstration shall be documented by a report showing that the minimum natural frequency exceeds 33 hertz and that the valve is capable of remaining intact during and after concurrent exposure to design pressure and to a static force due to a 3g acceleration directed through the center of gravity of the extended structure in the weakest direction of the valve assembly. The report shall include a description of tests, test results and calculations used for the demonstration.

SAN ONOFRE UNITS 2 & 3 PROJECT

2408-PFR-F079

3. Add the following paragraph to Specification 1370-PE-704.

4.2.7 Operating pressures for "active" valves will be limited to P_r. Operating pressures of Class 1 Seismic Category I valves that are not designated "active" will be limited to 1.1 P_r during upset, 1.2 P_r during emergency and 1.5 P_r during plant faulted conditions. Operating pressures of Class 2 and 3 Seismic Category I valves that are not designated "active" will be limited to 1.1 P_r during upset and emergency conditions and 1.5 P_r during faulted conditions. P_r is defined as the pressure corresponding to the maximum temperature associated with each plant condition as taken from (for standard valves) or interpolated from (for non-standard valves) the pressure-temperature table(s) imposed by Reference 3.1.1 for the valve's primary pressure rating.

4. Add the following paragraph to Specification 1370-PE-705.

4.2.7 Operating pressures of "active" valves will be limited to P_r. Operating pressures of valves that are not designated "active" will be limited to 1.1 P_r during upset and emergency conditions and to 1.2 P_r during plant faulted conditions. P_r is defined as the pressure corresponding to the maximum temperature associated with each plant condition as taken from (for standard valves) or interpolated from (non-standard valves) the pressure-temperature table(s) imposed by Reference 3.1.1 for the valve's primary pressure rating.

5. Add the following paragraph to Specification 1370-PE-716.

4.2.7 Operating pressures will be limited to 1.1 P_r during upset, 1.2 P_r during emergency and 1.5 P_r during plant faulted conditions. P_r is defined as the pressure corresponding to the maximum temperature associated with each plant condition as taken from (for standard valves) or interpolated from (for non-standard valves) the pressure-temperature table(s) imposed by Reference 3.1.1 for the valve's primary pressure rating.

Costs due to these changes must be accrued for a contract change estimate to obtain reimbursement from the customer.

Al Tillman

A. J. Tillman

AJT:ddc

2408-PFR-F079

Interoffice Correspondence

X 3

ER
EMS

R.W.D. FEB 27 1975

RWD.
HVT.
AHS.
50

SEISMIC

W. DeVane

San Onofre 2 and 3
Design Changes Due to
33Hz Requirements

D. R. Wade

W. Albert
N. Major
L. Kettles
H. Swan

S-PCE-6263

February 26, 1975

sign changes resulting from a minimum of 33Hz natural frequency requirement e as follows for the equipment indicated:

. Valves - They are specified to withstand a seismic acceleration of 3.0g and to have a fn > 33Hz. The architect/engineer is required to examine the exact location of the valve and the response spectra to determine the adequacy of the specified 3.0g - 33Hz requirement. No equipment changes have been made to date. In the event specific valve accelerations exceed 3.0g there will be vendor analysis costs, potential hardware costs, and Plant Engineering follow-costs.

Heat Exchangers - Previous analyses for the Shutdown, Regenerative and Letdown Heat Exchangers were performed with g loadings of 2.0 and 2.8 acting on the center of gravity and included a factor for nozzle loads. Current requirements specify maximum nozzle forces and moments to he evaluated as acting upon the rigid heat exchanger. It is not expected these analyses will require equipment changes.

Pumps - Analyses for the safeguards pumps and charging pumps will be conducted using g accelerations of 1.5 horizontal and 1.0 vertical. In addition the charging pumps have been specified to have a natural frequency of 35 cps or higher. Results of the analyses are not yet available and potential equipment changes are likely for the HPSI pumps. The order for the Boric Acid Makeup Pump has yet to be placed.

ummary of potential costs is as follows:

EM	Hardware Change	Vendor Analysis	Plant Engineering Follow
ves	[illegible]	[illegible]	[illegible]
down t Exchanger	[illegible]	[illegible]	
enerative t Exchanger	[illegible]	[illegible]	[illegible]

Cost Info. Deleted
3/11/82

SAN ONOFRE UNITS 2 & 3 PROJECT

2408-PFR-F079.

ARY (Continued)

	Hardware Change	Vendor Analysis	Plant Engineering Follow
HPSI Pumps	[illegible]	[illegible]	[illegible]
LPSI Pumps	[illegible]	[illegible]	[illegible]
tainment ay Pumps	[illegible]	[illegible]	[illegible]
rging Pumps	[illegible]	[illegible]	[illegible]
ric Acid keup Pumps	[illegible]	[illegible]	[illegible]

Cost INFO. Deleted
3/11/82

D R Wade
D. R. Wade

W:lsf

2408-PFR- F079

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 9-9297

R.W.D. OCT 2? 197?

POWER SYSTEMS

October 22, 1976

S-CE-3364
S-PSE-202

Southern California Edison Company
San Onofre Units 2 and 3
SCE Order No. N1800001
Bechtel Job No. 1304-606
CE Contracts 1370 and 1470

SCE SAN ONOFRE 2 & 3	A	I	INCL
R. W. D.		✓	
K. V. T.			
A. H. S.		✓	
A. S.			
R. C. G.		✓	
STAC. ENG			
FLT. ENG.		✓	
KCL			
CONTRACTS			
D. PECK		✓	
CHRONO		✓	✓

Mr. J. D. Houchen
Bechtel Corporation
P. O. Box 60360
Terminal Annex
Los Angeles, California 90650

Subject: Pressurizer Branch Line Seismic Spectra

Reference: (A) S-CE-3122, R. W. DeVane to J. D. Houchen, July 30, 1976

(B) S-CE-1353, R. W. DeVane to J. D. Houchen, October 4, 1976

Enclosures: 15 Sheets of Spectra

Dear Mr. Houchen:

In discussing the seismic reanalysis of the RCS main loop at the Engineering Interface Meeting held in Windsor on June 10-11, 1976, it was identified that the Pressurizer was not included in the scope of that effort. It was agreed at that meeting to review the results of the main loop analysis before recommending what to do about the high frequency peaks on the Pressurizer branch line seismic response spectra.

Having now completed the main loop reanalysis, we have compared the results with the original main loop branch line spectra, and have prepared a set of OBE spectra for the Surge and Spray Lines at their respective pressurizer nozzles, enclosed for your consideration.

These results were obtained by applying the trends observed in the data supplied by Reference (A), to the original pressurizer spectra of Reference (B). It is our opinion that the trends observed in the calculated data of Reference (A) are sufficiently general and clear to permit this present data to be considered a conservative estimate of the results that would be expected if a detailed reanalysis of the pressurizer were to be performed.

SAN ONOFRE UNITS 2 & 3 PROJECT

2408-PFR-F079

The following observed trends were applied to the spectra of Reference (B) to arrive at this present data:

(1) The response spectrum is unchanged in the frequency range below 20 cps.

(2) The zero period acceleration (ZPA) is less than or equal to the magnitude of the minimum value in the range between 2 and 20 cps.

(3) The ratio of the amplitude of the peaks to the ZPA at frequencies above the fundamental natural frequency of the pressurizer is no greater than the original results.

(4) The vertical response spectra due to vertical excitation are unchanged.

(5) Spectra for responses in directions other than the direction of excitation (such as Z response due to X excitation) are no greater than the original results.

Very truly yours,

R. W. DeVane, Jr.
Project Manager

RWD/DAP:dmh
enclosure

cc: S. V. Tashjian (SCE) w/encl.
W. D. Griffith (SDG&E) w/o encl.
W. L. MacDonald (C-E Orange) w/o encl.

Plant Apparatus

VENDOR DATA TRANSMITTAL FORM

R.W.D. AUG 14 1974

Bechtel
P.O. Box 60860, Terminal Annex
Los Angeles, California 90060
Telephone: (213) 864-6011

TO

Combustion Engineering Inc.
1000 Prospect Hill Road
Windsor, Conn. 06095
ATTN: R.W. DeVane, Jr.

DATE 8-9-74 PACKAGE NO. 574

Status

1 Approved – Manufacturer may proceed.
2 Approved – Submit final drawing – Manufacturer may proceed.
3 Approved except as noted – Make changes and submit final drawing. Manufacturing may proceed, as approved.
4 Not approved – Correct and resubmit.
5 Approval not required. – Manufacturing may proceed.

Subject: Safeguard System Pumps SPEC COMMENTS
Client: Southern California Edison
Client Job: San Onofre Units 2 & 3
Location: San Onofre, California
Client No.: Bechtel Job: SO23
Account No.: File No.: 933
Vendor No.: S-CE-1102 P.O. No.: 933

REQUIRED ACTION BY VENDOR

a. Resubmit by (date) 9-21-74
b. SHOW LOG NUMBER on reproducible and transmittal.
c. Return blue copy, Print Control, Floor No. 45
d. Address correspondence to: ATTN: Vendor Print Control. Also send two copies of letter to ATTN: J.D. Houchen PROJ. ENG.

Enclosed is (1) One copy of the following Drawings:

	LOG NO.	REV. NO.	VENDOR NO.	REV. NO.	TITLE	STATUS
1	SO23-933-I	4	1370-PE-410	05	Project Specification For Safeguards	3
2					Pumps	
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						

SCE SAN ONOFRE 2 & 3			
R. W. D.		✓	
H. V. T.		✓	✓
D. C. M.			
REAC. ENG			
PLT. ENG.			
QC			
CONTRACTS		✓	
D. ALBERT	✓		✓
C. BLANCHARD		✓	✓
AHS		✓	✓
CHECKED		✓	✓

COMMENTS:

cc: D.F.Martin, SCE
Dwg. Control, SCE (2)
W.L. MacDonald, CE-Orange, Calif.

Note: The word "drawing" herein shall mean drawings, data, specifications, and other vendor documents in any form.

IMPORTANT:
Vendor's drawings will be reviewed and approved only as to arrangements and conformance to the specifications. Approval shall not relieve the Vendor's responsibility of adequacy and suitability of materials and/or equipment represented thereon for the intended function.

SIGNED

ENGINEER [signature] DATE 8/9/74

COMMENTS:

PROJECT SPECIFICATION FOR SAFEGUARDS PUMPS (1370-PE-410)

General Comments

1. There are comments outstanding on Revision 4 of the Project and General safeguard pump specifications which were transmitted CE as VDTF Package No. 354, 4-22-74. Approval of the subject specification will be given upon resolution of the Revision 4 comments and the following.

Item	Specific Comment
1.1	The appropriate revision number of specification No. 00000-PE-411 should be included.
2.2	Should not the heat exchanger, blower and piping, etc. for motor cooling be included here as it is not listed under 2.3?
4.2.7	The seismic requirements specified herein are inadequate. The acceleration values must be taken from the instructure response spectra for elevation (-) 15'-6" of the safety injection building. As stated in VDTF 354, this information is on sketches SO23-SK-S-677, -680, -683 and 686. These were transmitted by Bechtel letter log BC-208, 7-17-73. The pumps are Seismic Class I and must follow that design criteria for both the D.B.E. and O.B.E. For the LPSI and Spray Pumps, which will be mounted on supports supplied by Bechtel, an interface must be worked out between Bechtel and the Seller to insure an acceptable stiffness is obtained in the support systems.
4.2.15	Acceptable piping loads are not specified. The connected piping will be routed to limit pipe stresses due to thermal expansion to 3500 psi, and to limit pipe stresses due to the combination of design loads and operating basis earthquake loads to 6000 psi.
4.3.4	Attachment (1) should be deleted and SCE Motor Specification GS-0355 Should be used. Attachment (1) conflicts with the SCE specification.
6.1 and 6.2	Cleaning and painting should conform to the SCE approved specification standard Appendix 4c.

1.0 % in [illegible]
9-6-74

DE

2408-[illegible]

[illegible] DIVISION

TO: E. E. Magette

Southern California Edison
Reactor Vessel
Support Loads

W. E. Stolecki

S-PSE-039

October 10, 1974

Put in
Spec

E.E.M. OCT 10 1974

The following are forwarded	Herewith	Under Separate Cover
	xxx	

Item	Quantity Prts.	Repros	Other	Drawing No. or Title
a				Figure 1, Sheet 1 of 6 R.V. Nozzle & Support Loads
				Figure 1, Sheet 2 of 6 Typical Load Coordinates and Direction [illegible]
				Figure 1, Sheet 3 of 6 Cosines Typical Load Coordinates and Direction Cosines
d				Figure 1, Sheet 4 of 6 Nozzle and Support Loads
e				Figure 1, Sheet 5 of 6 Nozzle and Support Loads
f				Figure 1, Sheet 6 of 6 Nozzle and Support Loads

Submitted For	Approval	Comment	Info.	Other	Approval/Comments Requested By	
			xxx			

Remarks: See Attached Sheet.

W. E. Stolecki

W. E. Stolecki

WES:dmk
Attachment

CE 0090059 Imp (3/71)

2408-PFR-1079

References:

(A) NPD letter S-PSP-0133, dated September 18, 1974.

(B) Telecon between J.T. Wren of NCE and W. E. Stolecki of NPD on September 30, 1974..

Item (a) through (F) present design loads for the subject reactor vessel supports. Nozzle loads will be submitted at a later date. These loads supersede those of Reference(A), and should be included in the next issue of the project specification.

The format in which these loads are presented is based on discussions with NCE, recorded in Reference(B). This format treats a support as a straight column, not as a beam column with an elastic deflection curve. This is done for the following reasons:

1. The effect of column deflections, on the moment distribution and stresses throughout a support, is minor. The effect creates a stress that is approximately 5% of the maximum allowable per Section III.

2. The design loads presented herewith were obtained by increasing actual loads by 15%. With this margin, structural adequacy can be established without considering column deflections.

3. Rigorous means for treating beam column effects are not presently available.

4. Calculations using a straight column are much simpler.

Please note that the loads given exceed those previously furnished and should be checked to see if the materials ordered to previous loads and to NPD specified dimensions are still adequate. NPD would have checked the effect of higher loads if NCE sizing calculations had been available. NCE should complete this check as soon as possible, and inform NPD of any ensuring problem areas that need to be resolved.

2408-MRR-F017

Component	Load	Load Location Co-ord in. Ft. * X	Y	Z	Direction Cosine* Cos x	Cos y	Cos z
Typical Inlet Nozzle & Support (See Fig. 1b Sht 1)	Fa2	-4.975	0	+8.617	- .5	0	+ [illegible]
	Fb2				0	+1	0
	Fc2				- .866	0	- .5
	Ma2				- .5	0	+ .866
	Mb2				0	+1	0
	Mc2				- .866	0	- .5
	Fc3	-4.746	-3.085	+8.219	- .866	0	- .5
	Fa4		-3.26		- .5	0	+ [illegible]
	Fb4				0	+1	0
	Fc4				- .866	0	-.5
	Ma4				- .5	0	+.866
	Mb4				0	+1	0
	Mc4				- .866	0	-.5
	Fa5				- .5	0	+[illegible]
	Fb5				0	+1	0
	Fc5				- .866	0	-.5
	Ma5				- .5	0	+.866
	Mb5				0	+1	0
	Mc5				- .866	0	-.5
	Fc6	-3.933	-23.917	+6.811	- .866	0	-.5
	Fa7	-4.746	-24.417	+8.219	- .5	0	+.866
	Fb7				0	+1	0
	Fc7				- .866	0	-.5
	Ma7				- .5	0	+.866
	Mb7				0	+1	0
	Mc7				- .866	0	-.5

*Right hand system as defined in "R.V. Plan" Fig. 1a Sht 1.

Typical Load Coordinates and Direction Cosines

Specification NO. Rev. Fig. 1 Sht 2 of 6

2408-PMK-?-0??

Component	Load	Load Location Co-ord In. Ft.*			Load Direction Cosines*		
		X	Y	Z	Cos_x	Cos_y	Cos_z
[Typ]ical Outlet [No]zzle (See Fig 1c Sht. 1)	Fa_1	+10.219	0	0	+1.0	0	0
	Fb1				0	+1	0
	Fc_1				0	0	+1.0
	Ma_1				+1.0	0	0
	Mb_1				0	+1	0
	Mc_1				0	0	+1.0

* Right hand system as defined in "R.V. Plan" Fig. 1a Sht 1.

Typical Load Coordinates and Direction Cosines

Specification No. Rev. Fig. 1 Sht 5 of 6

2408-PFR-F079

Normal Operating Loads on Typical RV Nozzles & Supports

Condition	Location	N*	Fan*	Fbn*	Fcn*	Man*	Mbn*	Mcn*
Dead Weight	Outlet Noz	1						
	Inlet Noz	2						
	Hor. Supt	3	0	0	2	0	0	0
	Flange	4	0	470	0	0	0	0
	Column	5	0	-470	0	0	0	0
	Key	6	0	0	0	0	0	0
	Pad	7	0	470	0	1	0	1
Normal Operation (Thermal and Weight)	Outlet Noz	1						
	Inlet Noz	2						
	Hor. Supt	3	0	0	-82	0	0	0
	Flange	4	-38	+1046	2	-22	0	-394
	Column	5	+38	-1046	-2	+22	0	+394
	Key	6	0	0	-2	0	0	0
	Pad	7	-38	1046	4	22	3	409

Units: KIPS and ft.-KIPS

* The numerical value of N defines forces and moments as shown in Fig. 1 Sht 1 and 2.

Title: Nozzle and Support Loads

Specification No. Rev. Fig. 1 Sht 4 of 6

2408-VRC-RO17

SSE LOADS ON TYPICAL R.V. NOZZLES & SUPPORTS

Condition	Location	N*	Fan*	Fbn*	Fcn*	Man*	Mbn*	Mcn*
SSE **	Outlet Noz	1						
	Inlet Noz	2						
	Hor. Supt	3	0	0	±717	0	0	0
	Flange ***	4						
	Column ***	5						
	Key	6	0	0	±401	0	0	0
	Pad	7	±6.6	±628.4	±408.6	± 362	±666	±65

Units: KIPS and ft-KIPS

* The numerical value for N defines forces and moments as shown in Fig. 1 Shts 1 and 2

** Tabulated values are for Safe Shutdown Earthquake. For Operational Basis Earthquake use one half the values shown.

Tabulated forces and moments can be either positive or negative, signs should be chosen to give the most severe loading combination

*** Loads can be determined by transposing pad and key loads with worst case signs on the load components.

Title: Nozzle and Support Loads

Specification No. Rev. Fig. 1 Sht 5 of 6

2408-PFR-F079

ACCIDENT LOADS ON TYPICAL RV NOZZLES & SUPPORTS
(Loads do not include Normal Operation & SSE Loads)

[illegible]n	Location	N*	Fan*	Fbn*	Fcn*	Man*	Mbn*	Mcn*
I Ruptured Line	Outlet Nozzle	1						
	Inlet Nozzle	2						
	Hor. Supt	3			1504.2			
	Flange	4	-47.5	-2914.9	14.9	-255	12	-666
	Column	5	47.5	2914.9	-14.9	255	-12	666
	Key	6			46.6			
	Pad	7	-47.5	-2914.9	-31.8	33	-63	340
II Ruptured Line	Outlet Nozzle	1						
	Inlet Nozzle	2						
	Hor. Supt	3			3852.5			
	Flange	4	34.5	1023.5	-5.7	211	159	348
	Column	5	-34.5	-1023.5	5.7	-211	-159	-348
	Key	6			-115			
	Pad	7	34.5	1023.5	109.3	147	345	-345

Units KIPS and ft.-KIPS

* The numerical value of N defines forces and moments as shown in Fig. 1 Shts 1 and 2

Title: Nozzle and Supports Loads

Specification No. Rev. Fig. 1 Sht 6 of 6

2708-KKK-R017

COMBUSTION ENGINEERING, INC.

ADDRESSEE	SUBJECT	FROM – DATE
TO: W. E. Abbott	PLANT ENGINEERING DESIGN QUALITY ASSURANCE	W. K. Wilhelm PE-70-315 October 30, 1970

cc: L. E. Anderson
J. D. Crawford
E. P. Flynn
R. F. von Hollen
Project Engineers
Project Managers
D. A. Miller
R. D. Haun
W. W. Albert
G. J. Huba
I. Bernstein
R. S. Daleas
H. B. Smith

Reference: (A) SA-69-244, H. von Steiger to Distribution, dated September 18, 1969.

The appended procedure is in response to the referenced letter and describes the procedural manner in which Plant Engineering ~~[illegible] meet the [illegible] of Section III of Appendix B to [illegible]~~

W K Wilhelm
W. K. Wilhelm

WKW/EPF/MWS:bl

XTRA 2408-RFR-F079

TO: W. L. Abbott

Plant Engineering
Design Quality Assurance

W. K. [illegible]

PE-71-345

July 23, 1971

cc. W. W. Albert
L. E. Anderson
I. Bernstein
J. D. Crawford
R. S. Daleas
E. P. Flynn
R. D. Haum
G. J. Huba
Project Engineers
Project Managers
D. A. Miller
H. E. Smith
D. P. Stercinz
M. F. Valerino
E. von Steiger

Reference: QR-71-016, E. von Steiger to W. L. Abbott,
March 23, 1971

Attached is Revision 1 of the subject procedure which reflects modifications which were made to effect alignment with Revision 1 of NPI-18. Specific comments which were a product of the referenced audit of Plant Engineering are disposed of as below:

1. A standard calculation list now appears as Exhibit A.

2. a. Flow of design work is shown in Exhibit B.
   b. Definitions of qualifications needed for reviewing or checking engineer have been incorporated in the procedure.
   c. The procedure has been rewritten in part in order to lend congruence in wording.
   d. Positive identification of data sheets has been added as a requirement for all calculations.

2408-PFR-F079

e. Logic networks for Plant Engineering's tasks will be used as an outline of required work through the evolution of design to system completion. Definitions of qualifications for reviewing and approving engineers and their responsibilities have been added to the procedure.

f. The flow chart, Exhibit B, shows interface requirements and words have been added to the procedure to describe the purpose of interface design concurrence.

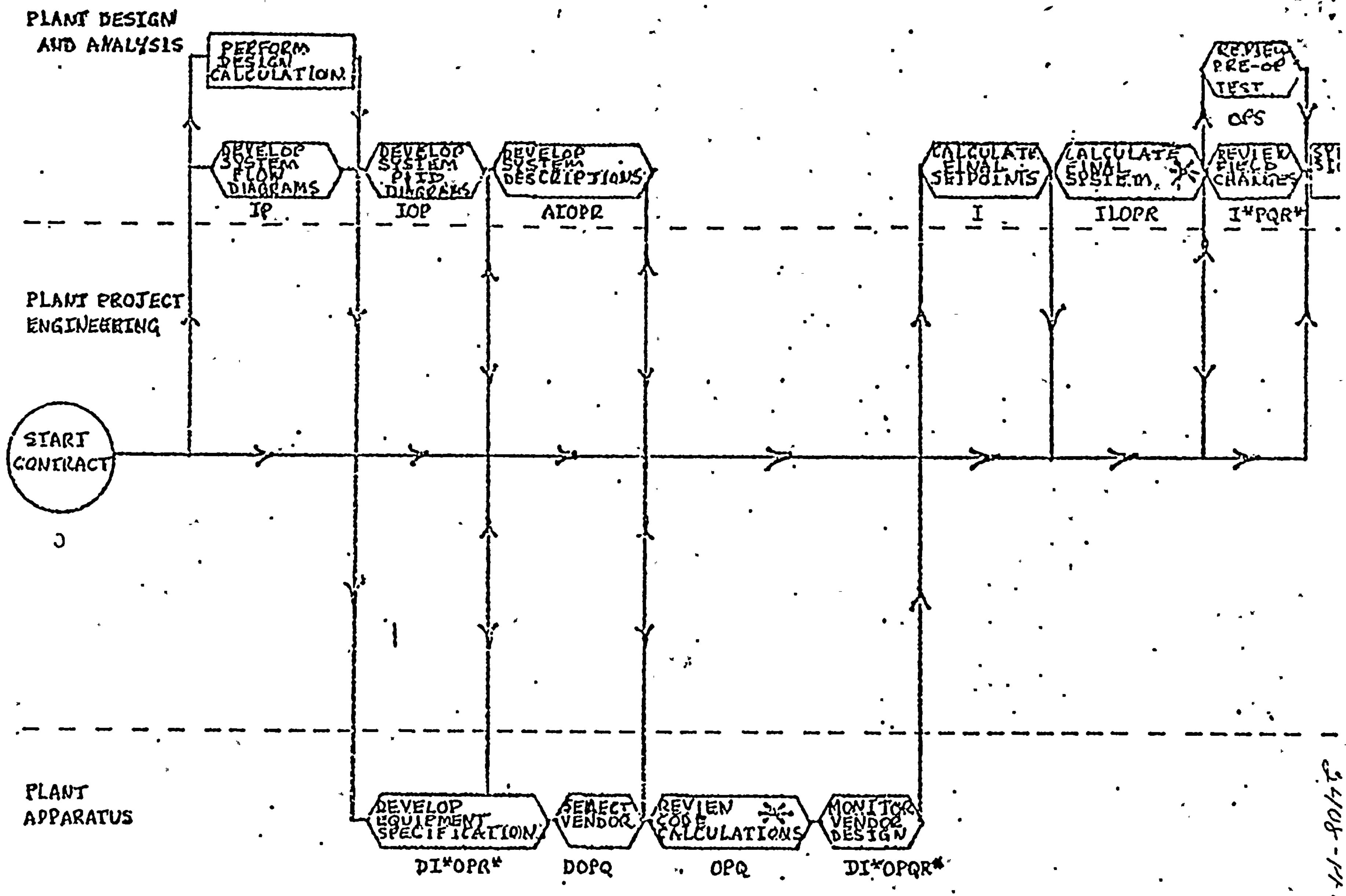
g. Words have been added to the procedure to cover verification of design by test.

3. A review of calculations presently in Plant Engineering files will be conducted to assess and correct non-conformance with the procedure.

W. K. Wilhelm

W. K. Wilhelm

WKW/MWS:jdt



SYMBOLS

▭ INTERNAL REVIEW AND APPROVAL

⬭ EXTERNAL REVIEW AND APPROVAL ALSO REQUIRED [APPROPRIATE GROUPS INDICATED BELOW]

)(AS REQUIRED

A – PLANT APPARATUS, D – PLANT DESIGN & ANALYSIS, I – ICE, L – SAFETY & LICENSING, O – PROJECT OFFICE, [illegible] – [illegible] ENGINEERING, Q – QUALITY CONTROL, R – REACTOR ENGINEERING, S – PROJECT SERVICES

INTEROFFICE CORRESPONDENCE

2408- PFR-F079

COMBUSTION DIVISION

To: V. C. Hall

Plant Engineering
Design Quality Assurance

W. K. Wilhelm

PE-73-107

March 19, 1973

cc: W. W. Albert
L. E. Anderson
I. Bernstein
G. F. Caruthers
J. D. Crawford
R. S. Daleas
E. P. Flynn
R. D. Haun
G. J. Huba
T. E. Natan
D. A. Peck
Project Engineers
Project Managers
G. Requa
H. B. Smith
F. Z. Stiteler
D. F. Streinz

Reference: QR-73-019 dated February 16, 1973, G. Requa to D. R. Wade

Attached is Revision 2 of the subject procedure. [redacted] However, the comments of the referenced memo plus comments resulting from an internal review have also been incorporated. In general, the impact of this document on our modus operandi is that we now must take an extra step or steps. There are no major reversals required in our procedures, but the area of computer code certification might prove tedious.

W K Wilhelm

W. K. Wilhelm

WKW/DRW:sl

PFR NO. 2408-PFR-F 107

REVISION

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

A. PREPARATION BY GA INITIATOR

AFFECTED ITEMS: OCWS Auxiliary Intake Structure, Section F

REQUIREMENT REFERENCE DOCUMENTS:

DC-339, San Onofre 2 and 3 FSAR, NUREG/CR-0098

BASIC REQUIREMENT: Combination of seismic load components should be in accordance with FSAR procedures or it should be justified if other method is used.

DESCRIPTION OF POTENTIAL FINDING:
There is no justification for computing the vertical seismic load as shown in pages F06 - F08 of DC399. Horizontal and vertical load components should be combined in accordance with FSAR, Section 3.7.2. If the approach used in pages F09 to F14 is used, 100% of the effects of the motion in the principal horizontal direction should be combined with 40% of the effects in the perpendicular horizontal direction and 40% of the effects in the vertical direction, as described in NUREG CR-0098.

PREPARED BY: M. Koploy M. Koploy DATE: 3-8-82

REJECTION OF GA TASK LEADER COMMENTS BY: DATE:

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: DATE:

B. REVIEW BY GA TASK LEADER

COMMENTS

☑ AGREE PF IS VALID BY [signature] DATE 3/8/82

☐ REQUEST RE-REVIEW BY DATE

☐ DISAGREE BY DATE

☐ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: DATE:

PFR NO. 2408-PFR-F 107

REVISION

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

COMMENTS

I agree, see attachment for explanation. PFR is invalid
M. Kopling

☐ AGREE PF IS VALID

☒ DISAGREE Comments attached

Concur with reviewer's recommendation to invalidate this PFR. FSO 3/17

RB BY: R. L. Richter DATE: 3/11/82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. L. Koutz DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: [signature] DATE: 3/18/82

2408-PFR-F107
3/18/82

PFR NO. F107

The methodology shown in pages F06-F08 was developed by SCE to apply SRSS techniques to stability analyses. The load factor method was not widely publicized at that time. Standard SRSS techniques are valid for combination of structure stresses but are not directly applicable to stability evaluations. This methodology recognized that seismic forces in the vertical direction were not acting at maximum magnitude at the same time as horizontal forces.

With the acceptance of the load factor method in NUREG/CR-0098, SCE performed a new calculation for DBE stability using this accepted method in lieu of the procedure previously used. The previous calculations were retained since they establish that DBE is clearly the controlling case for stability. A third stability calculation was also performed by the Project Soils Consultant at the same time as the original calculation and this calculation produced comparable safety factors.

SCE's intent in producing these calculations was to establish a clearly conservative calculation basis for this structure. Although 95% of this structure is buried in a competent soil medium, stability was checked using the same amplified static loadings for oscillating elements which were used for determination of structural stresses. In fact, it is anticipated that such a structure, which has similar mass and inertial properties as the displaced soil, would not significantly amplify the seismic motions and would be grossly stable.

In the further interest of this conservatism, it was also determined to analyze stability for each horizontal direction separately combining only vertical seismic load with the individual horizontal component. We have attached 2 calculation sheets in which the calculational technique suggested by TPT was performed for DBE sliding stability. It is noted, because resistance to sliding and overturning is increased by partial mobilization of passive pressure and skin friction in the other horizontal direction, the safety factor is increased from 1.15 to 1.30. The SCE determination is thus more conservative and is the correct calculation basis, since the second horizontal component may have a value less than 40 per cent of maximum.

Prepared by: J. K. Yann 3/11/82

Approved by: H. L. Richter
H. L. Richter 3/11/82

JKY:npy

ENGINEERING DEPARTMENT
CALCULATION SHEET

SUBJECT: Auxiliary Intake Structure DESIGN CALCULATION NO. DC PFR F107 - sh 1

J.O. NO. 9663 MADE BY JKlann DATE 3/10/82 CHK. BY ______ DATE ______

At the suggestion of TPT, the stability calculation for DBE sliding will be checked using the load factor method. Lateral seismic load will be assumed at 100% of the originally calculated value and vertical and longitudinal seismic load will be assumed at 40%. If this calculation demonstrates a need, overturning will also be checked although the overturning safety factor calculated initially was higher than for sliding.

Per original calculations

	Lateral	Longitudinal	Vertical
Structure seismic	749^k (F06, F09)	$.4 \times 531^k$ (F06, F09) = 212	$.4 \times 419^k$ (F06, F09) = 168^k
Active Earth Pressure	255^k (F06, F09)	190.2^k (F14)	NA
	1004	402.2	
Passive Earth Pressure	923 (F06, F13, WCC)	700.9 (F14)	NA
Friction (other than base friction)	52 (F06, F13)	160 (F07, F14)	N.A.
	975^k	861	

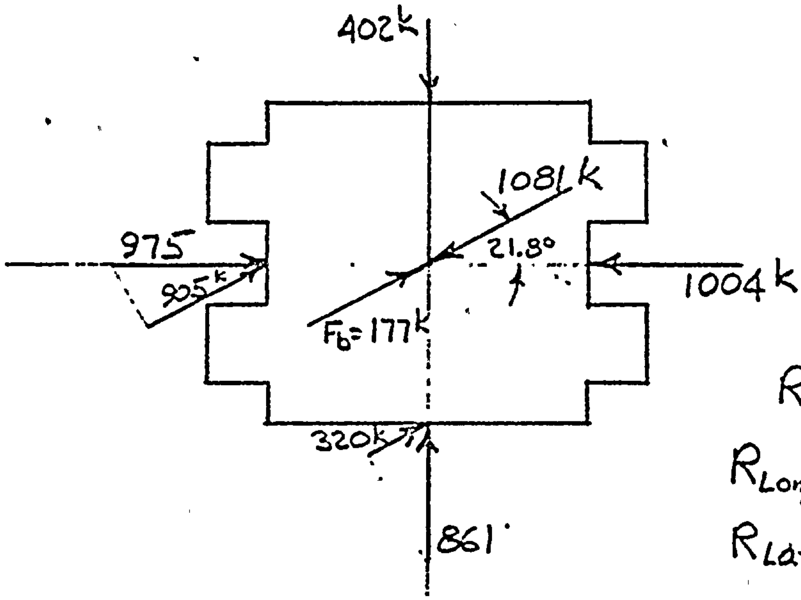
Base friction is based on a vertical reaction resulting from the overturning analysis for lateral load only. Since this reaction will increase only slightly as a result of longitudinal overturning combined with lateral overturning, this same value will be used.

$$F_b = .52 \cdot 340.8 = 177.2\ k$$

DWG. NO.

SUBJECT: Auxiliary Intake Structure DESIGN CALCULATION NO. DC PFR-F107 - sh.2

J.O. NO. ____ MADE BY ____ DATE ____ CHK. BY ____ DATE ____



Resultant sliding force

$$= \sqrt{1004^2 + 402^2} = 1081^k$$

$$\theta = \tan^{-1} \frac{402}{1004} = 21.8°$$

Resisting forces

$$R_{Long} = 861 \sin\theta = 320^k$$

$$R_{Lat} = 975 \cos\theta = 905^k$$

$$\text{total } R = 177 + 320 + 905 = 1402^k$$

$$S.F. = \frac{1402}{1081} = 1.30 > 1.15 \text{ (min S.F for lat. alone)}$$

Since S.F. is higher, it is not necessary to check overturning case, which is more complex.

DWG. NO.

Attachment to PFR-2408-F-107

SCE response to this PFR answers the concerns I had concerning the combination of seismic load components used in Section F, DC-339.

The method used in pages F06 - F08 was developed by SCE to apply SRSS techniques to stability analysis. Yet, the results from this method are not used, instead, the stability analysis is repeated using the accepted load factor method described in NUREG/CR-0098.

In an effort to be more conservative, SCE used the load factor method combining separately each horizontal direction with .40 of the vertical seismic load. SCE shows by performing the calculations that this approach is indeed more conservative than combining 100% of the effects in one horizontal direction plus .40% in the perpendicular horizontal direction plus .40% in the vertical direction.

Therefore I agree PFR is invalid. M.K.

M. A. Koploy M.K

Concur. FSO 3/17/82

2408 PFR NO. F111

REVISION --

POTENTIAL FINDING REPORT
SONGS 2&3 SEISMIC DESIGN VERIFICATION

A. PREPARATION BY GA INITIATOR

AFFECTED ITEMS:

SCE Trend Reports of Conditions adverse to Quality.

REQUIREMENT REFERENCE DOCUMENTS:

PSAR - Attachment 1, SCE Quality Assurance Program Plan
Amendment 20, Sec. 16, "Corrective Action", Para. 16.2.7

BASIC REQUIREMENT:

"A cognizant quality assurance engineer shall determine the existence of significant adverse trends in nonconformances and report them to the SCE chief quality assurance engineer. The chief quality assurance engineer may refer these trends to the Engineering Review Process to determine the need for corrective actions."

DESCRIPTION OF POTENTIAL FINDING:

No evidence of reporting of significant adverse trend which existed in the area of Document Control for 1972-1980 (see PFR F0054).

3/18/82

Originator agrees that PFR is invalid. JE Larsen 3/18/82

PREPARED BY: JE Larsen DATE: 3/8/82

REJECTION OF GA TASK LEADER COMMENTS BY: ______ DATE: ______

REJECTION OF ORIGINAL DESIGN ORG. COMMENTS BY: ______ DATE: ______

B. REVIEW BY GA TASK LEADER

COMMENTS

Agree PFR is invalid.
SA 3/18/82

☒ AGREE PF IS VALID BY [signature] DATE 3/8/82

☐ REQUEST RE-REVIEW BY ______ DATE ______

☐ DISAGREE BY ______ DATE ______

☒ REVIEW OF ORIGINAL DESIGN ORGS. COMMENTS BY: [signature] DATE: 3/18/82

2408 PFR NO. F111

REVISION --

C. REVIEW BY ORIGINAL DESIGN ORGANIZATION

COMMENTS

QAP N2.07 "Reporting of Quality Trends" addresses the SCE Quality Assurance program for analysis and reporting of quality trends. This procedure became effective July 1, 1976. Trend reports have been prepared which cover the period from January 1976 to date. These trend reports are on file with SCE Quality Assurance and establish objective evidence of implementation of PSAR Appendix A, Attachment 1, Paragraph 16.2.7 requirements. Trends regarding the document control function were identified by these trend reports along with appropriate recommendations to SCE Quality Assurance Management regarding required corrective action measures.

☐ AGREE PF IS VALID

☒ DISAGREE

Continued on Attachment 1

BY: JM Curran / ll DATE: 3-10-82

D. RECOMMENDATION BY FINDINGS REVIEW COMMITTEE

DEFINITION ADEQUACY:	☒ ADEQUATE	☐ INADEQUATE
VALIDITY:	☐ VALID	☒ INVALID
CLASSIFICATION:	☐ OBSERVATION	☐ FINDING

JUSTIFICATION:

CLASSIFICATION CRITERION NO. RESULTING IN "FINDING" ____________

COMMENT ON "OBSERVATION" CLASSIFICATION

BY: S. A. Kouty DATE: 3/18/82

E. GA PROJECT MANAGER

☒ ACCEPT

☐ REJECT

BY: AhWeinman DATE: 3/18/82

ATTACHMENT 1

F111

Potential Finding Report 2408 PFR

SONGS 2 & 3 SEISMIC DESIGN VERIFICATION

REVIEW BY RESPONSIBLE DESIGN ORGANIZATION (cont.)

The period 1972-1976, represents an evolutionary process in trending analysis. The frequency of audits and reaudits to verify implementation of proposed corrective actions illustrates the type of activity which constituted the majority of the thrust of SCE QA efforts toward an effective program. The gains made in the formation of an effective quality assurance program in that time span are evident in the change in numbers and character of the Corrective Action Requests and Nonconformance Reports written. During the period, the PSAR Commitment on trend analysis was expressed as QA program policy in the Project QA Manual, Chapter 16, paragraph 16.1.6, "Trending studies are requested by the Vice President, Advanced Engineering or Manager, Quality Assurance." Not until the end of the 1972-1976 period was formal trending analysis requested. The requests resulted in the issue of QAPN2.07. From that point on the trend reports and Quarterly Reports to Management illustrate QAO actions in addressing generic program or hardware problems.