Portland General Electric Company

James E. Cross
Vice President and Chief Nuclear Officer

February 28, 1994

VPN-013-94
Trojan Nuclear Plant
Docket 50-344
License NPF-1

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

Dear Sirs:

Annual Report of the Trojan Nuclear Plant

Enclosed is one copy of Portland General Electric Company's "Annual Report of the Trojan Nuclear Plant" for the calendar year 1993.

Sincerely,

S. M. Quennoz

S. M. Quennoz
J. E. Cross

Enclosure

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Trojan Nuclear Plant

ANNUAL REPORT OF TROJAN NUCLEAR PLANT FOR 1993

PORTLAND GENERAL ELECTRIC COMPANY

PGE-1015-93

Trojan Nuclear Plant

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ANNUAL REPORT

of

TROJAN NUCLEAR PLANT

for 1993

Docket 50-344
License NPF-1

PORTLAND GENERAL ELECTRIC COMPANY
121 S. W. Salmon Street
Portland, Oregon 97204

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INTRODUCTION

The Annual Report of the Trojan Nuclear Plant for 1993 is submitted in accordance with the requirements of Federal Regulations and Facility Operating License NPF-1. Other required reports are included for ease of reference and completeness.

SUMMARY OF OPERATING EXPERIENCE IN 1993

The year began with the plant in mode 5, making preparations for restart. On January 4, Portland General Electric (PGE) Company's board of directors voted to discontinue plant operation. All fuel was removed from the reactor by January 27. Since then, the reactor has been defueled and will remain as such

PGE received a Possession Only License for the Trojan Nuclear Plant on May 5, 1993. This license allows PGE to possess, use, but not operate the facility. Thus, providing the regulatory basis for activities peformed in the defueled condition.

1. ANNUAL PERSONNEL EXPOSURE AND MONITORING REPORT

Requirement

Trojan Facility Operating License NPF-1, Appendix A, Technical Specification 6.9.1.5 states:

"Reports required on an annual basis shall include:

"a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assessment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions."

Report

Table 1 lists the number of workers receiving exposures greater than 100 mrem per year and the total exposures by work and job function for the year 1993.

Requirement

10 CFR 20.407 requires:

"(a) A report of...(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....

"(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 various exposure ranges)."

Report

Table 2 is the statistical report of radiation exposure required by 10 CFR 20.407(a)(2) and 10 CFR 20.407(b) for the year 1993. Since the revised 10 CFR Part 20 was not effective until January 1, 1994, this information is being provided in the format under the regulation in effect during the reporting period (i.e., 10 CFR 20.407 vice 10 CFR 20.2206).

TABLE 1

REPORT ON NUMBER OF PERSONNEL AND MAN-REM
BY WORK AND JOB FUNCTION
1993

	No. of Personnel (>100 mrem)			Total Man-Rem		
Work and Job Function	Station Employees	Utility Employees	Contract Workers	Station Employees	Utility Employees	Contract Workers
REACTOR OPERATIONS & SURVEILLANCE						
Maintenance Personnel	0	0	0	0.13	0.00	0.01
Operating Personnel	2	0	0	1.30	0.00	0.00
Chemistry & Radiation Control Personnel	4	0	0	1.73	0.00	0.13
Supervisory Personnel	0	0	0	0.24	0.00	0.48
Engineering Personnel	0	0	0	0.17	0.00	0.01
ROUTINE MAINTENANCE & INSERVICE INSPECTION						
Maintenance Personnel	2	0	0	1.69	0.01	0.28
Operating Personnel	0	0	0	0.03	0.00	0.00
Chemistry & Radiation Control Personnel	0	0	0	0.33	0.00	0.04
Supervisory Personnel	0	0	0	0.03	0.00	0.07
Engineering Personnel	0	0	0	0.04	0.00	0.01
SPECIAL MAINTENANCE						
Maintenance Personnel	0	0	0	0.04	0.00	0.16
Operating Personnel	0	0	0	0.01	0.00	0.00
Chemistry & Radiation Control Personnel	0	0	0	0.19	0.00	0.03
Supervisory Personnel	0	0	0	0.00	0.00	0.00
Engineering Personnel	0	0	0	0.00	0.00	0.00
WASTE PROCESSING						
Maintenance Personnel	4	0	0	0.76	0.00	0.01
Operating Personnel	0	0	0	0.34	0.00	0.00
Chemistry & Radiation Control Personnel	0	0	1	0.67	0.00	0.35
Supervisory Personnel	0	0	0	0.03	0.00	0.00
Engineering Personnel	0	0	0	0.01	0.00	0.00

TABLE 2

Work and Job Function	No. of Personnel (>100 mrem) Station Employees	Utility Employees	Contract Workers	Total Man-Rem Station Employees	Utility Employees	Contract Workers
REFUELING						
Maintenance Personnel	7	0	0	1.68	0.00	0.12
Operating Personnel	1	0	0	0.18	0.00	0.00
Chemistry & Radiation Control Personnel	3	0	7	0.76	0.00	1.86
Supervisory Personnel	0	0	0	0.17	0.00	0.01
Engineering Personnel	0	0	26	0.18	0.00	12.15
TOTAL						
Maintenance Personnel	6	0	0	3.07	0.00	0.51
Operating Personnel	1	0	0	1.09	0.00	0.00
Chemistry & Radiation Control Personnel	9	0	6	2.67	0.01	2.06
Supervisory Personnel	0	0	0	0.22	0.00	0.45
Engineering Personnel	0	0	26	0.20	0.00	10.71
GRAND TOTAL	16	0	32	7.25	0.01	13.73

TABLE 2

TROJAN PLANT WHOLE BODY EXPOSURE (REM)
1993

Number of persons monitored = 1311

No detectable exposure; Number of People = 1037

Exposure of at Least 0.001 and Less Than 0.099 Number of People =	220
Exposure of at Least 0.100 and Less Than 0.249 Number of People =	25
Exposure of at Least 0.250 and Less Than 0.499 Number of People =	26
Exposure of at Least 0.500 and Less Than 0.749 Number of People =	3
Exposure of at Least 0.750 and Less Than 0.999 Number of People =	0
Exposure of at Least 1.000 and Less Than 1.999 Number of People =	0
Exposure of at Least 2.000 and Less Than 2.999 Number of People =	0
Exposure of at Least 3.000 and Less Than 3.999 Number of People =	0
Exposure of at Least 4.000 and Less Than 4.999 Number of People =	0
Exposure of at Least 5.000 and Less Than 5.999 Number of People =	0
Exposure of at Least 6.000 and Less Than 6.999 Number of People =	0
Exposure of at Least 7.000 and Less Than 7.999 Number of People =	0
Exposure of at Least 8.000 and Less Than 8.999 Number of People =	0
Exposure of at Least 9.000 and Less Than 9.999 Number of People =	0
Exposure of at Least 10.000 and Less Than 10.999 Number of People =	0
Exposure of at Least 11.000 and Less Than 11.999 Number of People =	0
Exposure of at Least 12.000 and Less Than 100.000 Number of People =	0

Total Plant Exposure = 20.99 man-rem

2. STEAM GENERATOR TUBE INSPECTIONS AND MAINTENANCE

Requirement

Trojan Facility Operating License NPF-1, Appendix A, Technical Specification 6.9.1.5, "Annual Reports", states:

> "Reports required on an annual basis shall include...The completed results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b)."

Technical Specification 4.4.5.5.b states:

> "The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which the inspection was completed. This report shall include:
>
> 1. Number and extent of tube inspected.
> 2. Location and percent of wall-thickness penetration for each indication of an imperfection.
> 3. Identification of tubes plugged or sleeved."

Report

No steam generator tube inspections were performed in 1993.

3. RELIEF VALVE CHALLENGES

Requirement

Trojan Facility Operating License NPF-1, Appendix A, Technical Specification 6.9.1.5.c, "Annual Reports", requires:

> Annual reports shall include..."Documentation of all challenges to the pressurizer power operated relief valves (PORVs) or safety valves."

Report

There were no challenges to the pressurizer power operated relief valves or safety valves in 1993.

4. REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY

Requirement

Trojan Facility Operating License NPF-1, Appendix A, Technical Specification 6.9.1.5.d, "Annual Reports", states:

"6.9.1.5 Reports required on an annual basis shall include:

"d. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit."

Technical Specification 3.4.8, "Reactor Coolant System Specific Activity Limiting Condition for Operation", requires:

"The specific activity of the primary coolant shall be limited to:

"a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

"b. $\leq 100/\bar{E}$ μCi/gram."

Report

During 1993, the Reactor Coolant System specific activity did not exceed the limits of Specification 3.4.8.

5. CHANGES, TESTS, AND EXPERIMENTS

Requirement

Federal Regulation 10 CFR 50.59 and the Trojan Operating License NPF-1 require:

"(a)(1) The holder of a license...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 unreviewed safety question.

"(b)(2) The licensee shall submit...a report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each...annually..."

Report

Section 5 of the Annual Report provides a description of changes, tests, and experiments completed in 1993 in accordance with 10 CFR 50.59.

Safety Evaluation Number 91-030

Subject
Temporary Modification- Backup Diesel Air Compressor

Summary
This Temporary Modification will install a rental diesel driven air compressor and after cooler to provide a backup air supply for instrument Air Loads. If the "B" Joy does not have the capacity required, fire water from a "Y" connection will provide cooling to the rental after cooler. A connection will be installed downstream of a shutoff valve on the "C" Joy receiver. This connection will be the connection point for the rental compressor supply. The use of a backup air compressor is not discussed in FSAR Section 9.3.1. No changes are required to be made to the FSAR or Technical Specifications as this is a temporary modification. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 91-232

Subject
PGE-1049, Amendment 4, "Inservice Inspection Program for the Second Ten-Year Interval"

Summary
Code cases N-356 and N-446 are being adopted to align PGE recertification period for Level III examiners with later approved Code editions and addenda. Since this change is administrative, no unreviewed safety question is involved.

Safety Evaluation Number 91-323

Subject
PMR 91-154

Summary
The purpose of this change is to replace thermal overloads and circuit breakers, which supply overcurrent and short circuit protection to safety-related 480 Vac powered motor-operated valves, to the values calculated in calculation TE-132. The replacement components meet or exceed the design, material, and construction standards for equipment installed in the safety-related 480 Vac load centers B21 through B26 which supply power to the MOVs. The components are sized to meet the sizing criteria of Regulatory Guide 1.106 as referenced in NUREG-1296. The sizing criteria is to select thermal overloads and circuit breakers for safety-related MOVs on the basis of protecting the motor without interfering with the safety function of the valve. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 91-346

Subject
PMR 91-147

Summary
This change involves installing temporary ductwork while performing work on a fire barrier penetration seal and revising the Fire Protection Plan to delete the penetration seal details. The penetration seal installation meets the design, material, and construction standards applicable to proper seal design. The seal installation does not affect the overall performance of any systems which penetrate the fire barrier and does not cause systems to be used out of their design or tested limits. The removal of the penetration details from the Fire Protection Plan has no impact on FSAR evaluated accidents. The ducting which will be temporarily reworked is not used to control the spread of radioactivity and has no safety function. A determination was made that no unreviewed safety function exists.

Safety Evaluation Number 91-405

Subject
CAR C91-0576

Summary
PGE addresses in significant detail the basis for meeting the serviceability requirement of the technical specifications for ODSCC/IGA tube degradation at tube support plate intersections for Operating Cycle 14. Included is extensive analysis regarding the effect of ODSCC/IGA at tube support plate intersections on steam generator tube integrity and the adequacy of steam generator tube inspections and repairs. Other operational enhancements are being implemented prior to and during Operating Cycle 14. It is concluded from this basis that there is not an unreviewed safety question.

Safety Evaluation Number 91-437

Subject
PMR 91-077 (DCP 5), Revisions 3 and 4

Summary
This modification will change the power supply to the heat traced instruments PT-3044B, PT-3046B, PT-3072B and PT-3083B. These instruments provide indication of suction and discharge pressure for the diesel driven auxiliary feed pump. The power supply will be connected from the existing 480V at Box Q73 and step it down to 120/240V via a 1 1/2 kVA transformer for heat trace utilization. All connections are inside the AFP room. The existing 480V power supply have sufficient capacity to accept the additional load. Fuses will be installed on the line side feeding the transformer to isolate it from the existing load. The cable used in this installation is quality-related and the power supply is diesel-backed. The change in power supply will eliminate unnecessary conduit and cable run and fire barrier penetrations. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 92-001

Subject
PMR 90-050, DCP 1

Summary
This modification installs the power supply circuits and distribution panels for the plant process computer equipment and peripherals. The new plant process computer replacement for the existing plant computer and the safety parameters display system computer will not be supplied with 120 Volt AC electrical power from uninterruptible power supply system UPS Y30. Power for the existing plant process computer is supplied from UPS Y30. The load imposed is 5.5 kW. Upon loss of preferred power. Tables 8.3-1 and 8.3-2 from FSAR Section 8, this load would be imposed on the EDGs. The "Max. Output Required from Each Diesel Generator (kW)" column on Tables 8.3-1 and 8.3-2 must be adjusted to show a reduction of 5.5 kW in load after the plant process computer is placed in operation after the 1992 outage.

The load on the EDGs is being decreased. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-037

Subject
PMR 90-068

Summary

The modification is a change to the facility as described in the safety analysis report, including also the Fire Protection Plan, the Environmental Qualification Program, and the Accident Monitoring Program in that it deviates from power supply trains currently treated in these documents. System operation is unaffected, yet reliability is enhanced through increased redundancy; therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report are not increased and the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report is not created. Increased redundancy of accident monitoring instrumentation will enhance rather than reduce the margin of safety as defined in the basis for any technical specification. It is concluded the change does not involve an unreviewed safety question.

Safety Evaluation Number 92-056

Subject

PMR 91-069

Summary

No changes to plant hardware were made as a result of the drawing change. This change only affected a plant drawing regarding the location of a disconnect switch. The disconnect switch is installed to allow controller circuit breaker maintenance. Since no plant changes were made and relocation of the disconnect switch did not affect the function of the circuit, the probability of occurrence or the consequences of a previously analyzed accident or equipment malfunction was not affected, nor was the possibility of a new kind of accident or malfunction introduced. Additionally, the margin to safety, as defined in the basis of the Technical Specifications, was not reduced.

Safety Evaluation Number 92-065

Subject

PMR 92-032

Summary

This PMR moves TE-4936 from its original location immediately downstream of SGBD Hx E-318 to a new location further downstream. The change is necessary since the original location resulted in nonrepresentative temperatures due to thermal stratification. Because of this problem, the configuration is presently altered per Temporary Modification (TM) 88-091. This TM moved TE-4936 to a thermowell in the suction line for SGBD Pump P-335; however, this location is not suitable for a permanent installation since pump bypass flow is not monitored. The function of the TE is to isolate blowdown flow on high temperature to protect the ion exchange resins. This function is not changed by the PMR; in fact, the original cabling and circuitry will be reused due to the short distance the TE is relocated.

The new thermowell detail matches the original. No changes are made to piping material or steam generator chemistry. The function is unchanged and the new location is expected to provide more accurate readings. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-077

Subject

Change to the auxiliary feedwater system

Summary

The changes in description of the auxiliary feedwater system in the FSAR did not affect system configuration. It was concluded that there was not an unreviewed safety question.

Safety Evaluation Number 92-085

Subject
PGE-1043, Revision 2 (LDCR 92-31), "Accident Monitoring Instrumentation Review Plan"

Summary
This revision describes various changes to accident monitoring instrumentation. New or replacement equipment in this revision meet the committed qualification requirements for Regulatory Guide 1.91. The qualification level for radiation effluent monitoring has been increased. The commitment date for improvements to the RCS wide-range temperature and steam generator wide-range level indication has been revised. Also equipment has been added to the scope of the review plan. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-092

Subject
PMR 90-060 (Inverter for Preferred Instrumentation)

Summary
This change will modify all four Preferred Instrument Inverters so that each will have an alarm seal-in capability with manual reset, and the ability to measure voltage and frequency at three different locations inside the Inverter instead of just one.

The existing alarm and lamp control circuit boards will be replaced with boards that incorporate the additional circuit components to provide the alarm seal-in and manual reset. These boards are mounted on the inside of the inverter front door.

The existing input to the voltage and frequency meters will be modified such tat a three position selector switch will choose the source of the input from either the inverter output, the bypass output, or the static switch output.

The design of the alarm seal-in feature uses the same materials and standards as the original design of the Elgar inverters. Construction standards are equal to or better than Elgar's. This circuit modification does not change the safety-related function of the Preferred Instrument Inverters. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-110

Subject
PMR 92-057 - Bearing Cooling Water Filter S/U # 14

Summary
This modification is to install a filtration system in the bearing cooling water (BCW) system downstream of the BCW heat exchangers. The filtration system will interface with an existing bypass assembly previously installed for this purpose. The filtration unit will continuously filter the system. Its purpose is to remove iron oxide and other solid particles.

The BCW system is not safety-related and is not relied upon during an accident. Addition of a filter will have no affect on any evaluated accident. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-131

Subject
PMR 92-059

Summary
The Intake Structure is located on the west bank of the Columbia River and Supplies suction water to the Service Water System and the fire protection water pumps. The bottom of the Intake Structure is lower than the normal bottom of the Columbia River. This causes mud and silt to build up in the Intake Structure that must be dredged twice per year. This modification will install silt retaining walls along the bottom east side to the Intake Structure trash rack, and over both fish escape holes to prevent mud and silt from entering the Intake Structure. Bechtel Doc No. T035518 evaluated the head loss from the addition of the silt retaining wall and concluded the additional head losses were insignificant. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-195

Subject
High Energy Line Break Analysis

Summary
The change incorporates a revised scope of main steam piping welds in the break exclusion zone based on a new definition for the break exclusion boundary. Changes under this revision increase the scope of nondestructive examination to additional mainstream piping welds. This ensures component integrity and is preventive in nature. No unreviewed safety question is involved.

Safety Evaluation Number 92-202

Subject
PGE-1050, Revision 2 (Topical Report)

Summary
This revision to the snubber surveillance program is administrative in nature and removes redundant criteria in the document which are provided in the governing procedure. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-214

Subject
Deletion of PGE-8005, "Radiation Protection Program"

Summary
The Radiation Protection Program document is replaced with procedures. This change does not affect the existing safety-related equipment or operating procedures whatsoever. The "activities" described include only general areas of responsibility for positions and groups within the Radiation Protection organization, and for external positions and groups as they relate to general radiation protection activities. Since this change is administrative in nature, no unreviewed safety question is involved.

Safety Evaluation Number 92-231

Subject
LDCR for PGE-1043, "Accident Monitoring Instrumentation Review Plan"

Summary
The position indicating lights for the control room isolation dampers are modified to have a safety-related power supply instead of a nonsafety lighting circuit. This change allows the damper position lights to function during any event. Since the power supply will allow the operators to more easily check the damper status during an accident, the control room ventilation system is not part of an initiating event described in the SAR, and the

control circuit for the dampers includes a redundant shutdown feature which is not changed, no unreviewed safety question is involved.

Safety Evaluation Number 92-233

Subject
PMR 92-094

Summary
This modification installs two smoke detectors. These smoke detectors will be located on the west side of the diesel driven auxiliary feedwater pump room in the vicinity of the thermo-lagged penetrations into the room. These detectors will be installed to mitigate the effects of fire. NRC Bulletin 92-01 states that Thermo-Lag is not an effective fire barrier. Thus, these smoke detectors are to be installed to provide annunciation in the event of fire. These detectors will be tied into the K-50 computer which will provide annunciation in the Control Room. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 92-239

Subject
PGE 1012, Section 3.7

Summary
This PCC 89-512 involves adding a fire alarm (fire brigade muster) signal to the plant paging system for remote operation from the control room. This is accomplished by installing a selector switch and wiring to CO4, adding new cable and conduit between the sound rack and CO4 and by adding relay to the sound rack.

PGE-1012, Section 3.7 describes the existing evacuation alarm system. An update is required since this PCC 89-512 added a new fire alarm signal to the plant paging system. This modification met the current design, materials, and construction standards and will enhance the ability of plant personnel to respond to different emergency situations. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-266

Subject
TPP 20-2

Summary
This change creates TPP 20-2, "Radiation Protection Program" which replaces the Radiation Protection Manual and PGE-8005, "Radiation Protection Program" which are mentioned in the FSAR.

Since this change is administrative no unreviewed safety question is involved.

Safety Evaluation Number 92-267

Subject
Deletion of FSAR References to ODCM and RPM

Summary
This change deletes references to PGE-8005, "Radiation Protection Program" and POM Volume 10, "Radiation Protection Manual" from the FSAR. These documents have been deleted and replaced by TPP 20-2, "Radiation Protection Program", TPP 20-3, "Conduct of Radiation Protection", and TPD 20-1, "Radiation Protection Program Policies". This change is administrative in nature and involves the transfer of referenced material from one set of FSAR-referenced documents to another (the TPDs and TPPs). No references need to be added to the FSAR

because the TPPs and TPDs are already discussed in general terms in Section 13.5. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 92-271

Subject
PMR 92-107

Summary
This modification involves installation of a tube stabilizer in steam generator B. The acceptability of this stabilizer is discussed in Babcock & Wilcox Nuclear document number 51-1218816-00, "50.59 Input for Trojan B RSG Tube R25C17 Stabilization". The stabilizer is a conservative measure to ensure the defective tube does not come loose and damage and adjacent tube. In any case, failure of an adjacent tube is bounded by existing accident analysis. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 92-274

Subject
PGE 1012, LDCR 92-197

Summary
PGE-1012 Section 4.2.5.5 contains a sentence that does not properly describe the safe shutdown capability in fire area A5. As a result, corrective action 5 of CAR C92-0352 has been issued to NPE Fire Protection to clarify the safe shutdown capability in FA A5. The current sentence states that "adequate separation exists for redundant safe shutdown cables". The E-FM-100 cable database does not support this statement; in that E-FM-100 indicates that there are no safe shutdown cables in this area. The proposed change to PGE-1012 will state that "Area A5 does not contain any safe shutdown equipment or redundant safe shutdown cables".

Corrective action 3 of CAR C92-0224 has been issued to NPE Fire Protection to clarify the A4/A4a fire barrier configuration in PGE-1012. The proposed change to Section 4.2.4.2 of PGE-1012 discusses the lack of a fire damper for the ventilation penetration in the A4/A4a fire barrier and references the evaluation (M-FP-1.1.9) that establishes the technical basis for the non-rated configuration.

During the recent annual fire protection audit conducted by the QA department, an observation noted that the PGE-1012 definition (i.e., the RCS T-avg range) for Hot Shutdown conflicted with the Trojan Technical Specifications. Accordingly, it is proposed that the Hot Shutdown definition in PGE-1012 should be revised and the TTS RCS T-avg range for Hot Standby and Cold Shutdown should be added to PGE-1012 for consistency and completeness. Additional clarifications are proposed to properly reflect the Hot Standby and Hot Shutdown phases of safe shutdown. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 92-285

Subject
PMR 92-503

Summary
This modification will install an additional power supply transformer for the B train heat tracing in the MSSS. The existing installation has a 240-120 V, 3 kVA transformer feeding all of the loads in BTB021. The heat tracing was modified previously and when it was installed, more heat tracing was actually installed than the design anticipated. This led to a condition where the transformer is loaded to its maximum capabilities and the full load current through the fuses is very close to the fuse rating and on several occasions the fuses have blown. This modification will add a new 3 kVA transformer in parallel with the existing transformer and will separate the

existing heat tracing loads such that the transformers will be within their design limits and the normal running current will not cause the fuses to blow.

The addition of this transformer does not alter the heat tracing other than to improve reliability. The new transformer will be installed in parallel with the existing transformer so that the source of power for the heat trace loads is unchanged. There are no previously evaluated accidents that could be impacted by this modification. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-306

Subject
PMR 92-104

Summary
The minor modification replaces the rubber hose installed on drain valve PB-4000 with stainless steel tubing. This tubing allows process steam condensate returning from the Auxiliary Building to be routed to the oily waste system (a monitored discharge system). The FSAR states that this condensate is routed to the condenser. Due to chemistry concerns with this stream, it is no longer acceptable to recycle this flow.

The discharge stream from the process stream system is not safety-related and does not interface with any safety-related systems or any accident analyzed systems. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 92-308

Subject
LDCR PGE-1048, Amendment 7

Summary
This change adds pumps to the program which are already being tested per Technical Specifications. Since this testing is nonintrusive and is already being performed, no unresolved safety issue is involved. More information will be recorded during various tests. These tests are already being performed and do not involve and unreviewed safety issue. Also, an allowance is made to not complete all cold shutdown testing during forced outages. Since this continues to meet ASME requirements an unreviewed safety question is not involved.

Safety Evaluation Number 92-309

Subject
PGE-8010, Revision 14, "Nuclear Quality Assurance Program"

Summary
This change incorporates various organization changes, and clarifies the evaluation and reporting process for Licensee Event Reports and 10 CFR Part 21 reports. Since these changes are organizational and editorial, no unreviewed safety question is involved.

Safety Evaluation Number 93-003

Subject
PCC 87-550 FCN 6

Summary
These startup strainers in the radwaste system do not affect the ability of the plant to safely shut down the reactor. They do not provide any safety-related function, nor does their operation interface with any other safety-related function. Therefore, the change does not involve an unreviewed safety system.

Safety Evaluation Number 93-005

Subject
TM 89-009 for LDCR 93-002

Summary
This temporary modification bypasses a mode selector switch for two three-way solenoid valves on the heater drain tank (HDT) level control system. The only accident that can be initiated by the heater drain system is a loss of feedwater flow. This change will stabilize the operation of the control system, decreasing the probability of this accident. HDT level has no affect on the accident analysis. The change to the level control system will not increase the potential for failure of the pressure boundary, so none of the other safety-related systems in the building will be affected. Therefore, no unreviewed safety question is involved.

Safety Evaluation Number 93-006

Subject
Certified Fuel Handler Program, PGE-1057

Summary
Those accidents which are of concern in a defueled plant have already been analyzed for an operating plant. Training in accident response for these accidents is based on the former licensed operator training. No unreviewed safety question is involved.

Safety Evaluation Number 93-007

Subject
OI 1-3, Revision 14

Summary
This change will allow the used of either the B or C taps for the Startup Transformers during plant shutdown. This will increase operational flexibility so that under-voltage and over-voltage conditions can be avoided. This will improve the reliability of the refueling equipment in that the voltage on the 4160 volt and 480 volt buses will not become degraded, causing and undesired automatic action to clear the buses. Additionally, this will also reduce the likelihood of equipment damage and failure from too high a voltage. As a result, there is no unreviewed safety question involved.

Safety Evaluation Number 93-008

Subject
PGE-8010 - QA portion of Decommissioning Plan

Summary
The change does not change the accident analyses and does not affect the design, operation, or testing of equipment important to safety. No unreviewed safety question is involved.

Safety Evaluation Number 93-009

Subject
Procedure TPP 10-14 (replace AO 1-11)

Summary
This procedure identifies organization and responsibilities for the Plant Modifications department. Since this procedure defines organizational responsibilities only, no change to design, material, construction standards, or system performance are required. Therefore, there is no unreviewed safety question.

Safety Evaluation Number 93-010

Subject
Emergency Plan Amendment (PGE-1008)

Summary
This LDCR is a rewrite of the Emergency Response Plan to reflect the permanent shutdown of the Plant. The emergency plan is implemented after an accident and so could not create an accident. The plan does not direct the operation of equipment important to safety and so could not increase the consequences of an accident. No unreviewed safety question is involved.

Safety Evaluation Number 93-011

Subject
SPEER 92-254 - pH Meter and Probe Replacement for Discharge & Dilution Structure

Summary
SPEER 92-254 will replace the existing nonworking Discharge and Dilution Structure pH probe and meter. Also the replacement of the amp/preamp will also no longer be required and hence deleted. The pH meter and probe are for monitoring discharges to the river for the NPDES permit. Since the equipment is for monitoring only and is not safety-related or relied upon for mitigating accidents, no unreviewed safety question is involved.

Safety Evaluation Number 93-012

Subject
OI 10-3, Revision 31

Summary
This change sets up the Containment HVAC systems for the extended defueled condition. The normal condition of the Reactor Auxiliary Building Chill Water system is projected to be drained, so sections are included in the procedure to fill and start the system, shift chillers and shutdown and drain the system. Steps for venting hydrogen from Containment after a Design Basis Accident have been eliminated from the procedure for the operation of the Hydrogen Vent System and several changes were made to the procedure for venting Containment during normal operations. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 93-014

Subject
Removing Appendix R Requirements from PGE-1012

Summary
Various sections of the Fire Protection Plan as described in Topical Report, PGE-1012, contain the bases, assumptions, performance objectives and results of the Appendix R safe shutdown analysis. The proposed changes

to PGE-1012, which remove the requirements of Appendix R to 10 CFR 50 are required to reflect the reduced regulatory requirements of a Possession Only License (POL) facility. The proposed changes are not be incorporated in PGE-1012 until the POL is obtained.

With the reactor permanently shutdown and defueled with all fuel stored in the Spent Fuel Pool, the requirements of Appendix R to 10 CFR Part 50, which demonstrate the capability to achieve and maintain safe shutdown for the postulated worst case fire scenario(s) are no longer applicable.

Plant fires are not considered or classified as a design basis accident, and a design basis accident is not postulated to occur coincident with a design basis event, such as a fire. However, if a postulated fire event has the potential to provide a mechanism for initiating a design basis accident due to fire-induced damage to Plant equipment, then that increased probability of occurrence of an accident is required to be evaluated. A determination was made that no unreviewed safety question is involved.

<u>Safety Evaluation Number</u> 93-015

<u>Subject</u>
Removing Appendix A to BTP-9.5-1 (from PGE-1012), MPM 30-12, Rev.0, & MPM 30-13 - Door Gate and Hatch Inspections, PSP 22-5, Revision 0: Fire Brigade, Fire Watch and Hot Work Training

<u>Summary</u>
The purpose of the fire protection plan is to prevent the effects of fires from endangering the health and safety of the public and plant property protection. At an operating nuclear power plant, the goal is to safely shut down the plant during postulated fire scenarios. In addition, there is also emphasis to isolate and protect safety-related equipment. At a POL facility the plant is in an inherently safe condition with all of the fuel in the Spent Fuel Pool (SFP). The loss of SFP cooling accident does not present a potential hazard until after an extended period of time. The extended time to correct this condition creates an increase in the margin of safety. With the cessation of power operation, the quantity of activities and personnel on site reduces significantly. The reduced work activity and the deactivation of many of the plant systems reduces the probability of a fire starting in the plant. The changes to PGE 1012 reflect the change to personnel safety and property protection per NFPA Codes and American Nuclear Insurers Fire/All Risk Guidelines.

<u>Safety Evaluation Number</u> 93-017

<u>Subject</u>
AO 1-4, Revision 18 - Procedure Implementation as a Result of LCA 230 (LCR 93-02)

<u>Summary</u>
AO 1-4 Operations Department Responsibilities, Revision 18 proposes to change the responsibilities of the Shift Manager as described in FSAR Section 13.1.2.2 by deleting the words operation of the Reactor and Turbine Generator and adding responsible for operation of the Spent Fuel Pool and all Auxiliary equipment required to support the Spent Fuel Pool and its cooling requirements. This change is being implemented due to the decision to keep the plant in a De-fueled condition permanently. Therefore, responsibility for operation of the reactor and turbine generator are no longer required or needed. No unreviewed safety question is involved.

<u>Safety Evaluation Number</u> 93-018

<u>Subject</u>
Fire Brigade Composition Change Per LCA 230, TPP 13-7

<u>Summary</u>
The composition of the fire brigade is being revised to preclude the use of the new certified fuel handler on shift rather than "members of the minimum shift crew necessary for safe shutdown of the unit." Thus, the shift

supervisor and licensed operator requirements are being changed to "operator" for fire brigade leader, or "security brigade" trained as a leader. TPP 13-7 is being revised to reflect this change. Revision must be accomplished within 30 days of LCA approval. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-019

Subject
Revision of ODCM (PGE-1021) to Address NRC Concerns (Inspection Report 92-26)

Summary
Revision of Offsite Dose Calculation Manual (PGE-1021) to address NRC inspection concerns. This change (LDCR 93-012) to the ODCM incorporates several clarifications to the ODCM and makes specific reference in the ODCM to the computer code models used to calculate offsite doses, thus satisfying the requirement of T.S. 6.9.1.5.4 which requires that "the assessment of offsite doses be performed in accordance with the ODCM."

Safety Evaluation Number 93-020

Subject
CAR C90-3012-01-03; Non-Q MRs for Radwaste Systems

Summary
The corrective action for CAR C90-3012-01-03 required a review of all Maintenance Requests written for the radwaste systems at Trojan that were performed non-quality. Those MRs that replaced or installed equipment/material within Regulatory Guide 1.143 boundaries were identified and evaluated. This safety evaluation documents the acceptance of equipment/material installed under non-quality MRs.

The failure of this equipment would result in a leak or failure of the radioactive liquid waste system. This failure would not induce a failure on plant safety-related systems. Analysis of a leak or failure of a radioactive waste system was not required as a part of the Trojan Nuclear Plant design basis. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 93-031

Subject
Temporary Modification - 230 kV Main Transformer to the 85 Tie Wire Disconnect

Summary
This change affects AC power supply to the plant. It does not increase probability of a loss of power as discussed in Chapter 15 of the FSAR. The probability of losing offsite power is no greater than if the plant were being fed by one offsite source per mode 5 or 6 or defueled technical specification requirements. Procedures are already in place to deal with loss of offsite power. No unreviewed safety question is involved.

Safety Evaluation Number 93-033

Subject
FSAR Section 9.2.2.1, Heat Exchanger Outlet Temperature for CCW, C92-0498-01-01

Summary
The probability of an occurrence of an accident previously evaluated in the FSAR is not increased because Bechtel Calculation 21-11 has determined the maximum CCW Hx outlet temperature is 133.6 degrees F. The limiting safety-related component served by CCW is the SI Pump seal coolers. CCW letter to Bechtel POR-1507, 11/01/72, states that cooling water temperatures as high as 135 degrees F for approximately one day after a DBA are

acceptable. CCW Design Basis Document DBD-16, Section 3.2, discusses the temperature margins for applicable components cooled by CCW with respect to the maximum temperature of 133.6 degrees F. No unreviewed safety question is involved.

Safety Evaluation Number 93-034

Subject
OI-T-72 REV. 0

Summary
OI-T-72 provides instructions to transfer resin from the A CWRT to SRST. Prior to actually transferring the resin, PMW is backflushed into the A CWRT to ensure the tank discharge line is free of blockage. A PMW source is connected to the CWRT pump vent to provide PMW flow back through the pump to the CWRT. The installation of this jumper hose is the "change to the facility" being evaluated by this safety evaluation. These hoses are installed and removed within the procedure, and thus do not represent a permanent change to the Trojan facility. No licensing documents will be changed as a result of this temporary procedure. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-035

Subject
LDCR 93-014

Summary
LDCR 93-014 to PGE-8010, procedure hold status related to biennial review of procedures. A paragraph is being added to PGE-8010 to clarify the PGE position on biennial reviews of procedures.

Safety Evaluation Number 93-036

Subject
LDCR 93-015; add Rad Pro TTS into ODCM

Summary
The Radiological Effluent Technical Specifications (RETS) which place limits on the release of radioactive effluents to the environment and require an environmental monitoring program are being relocated to the Offsite Dose Calculation Manual (ODCM) in accordance with Generic Letter 89-01. This safety evaluation is to support the revision to the Offsite Dose Calculation Manual to accomodate the Radiological Effluent Technical Specifications removed from the Trojan Technical Specifications.

This change is administrative and merely relocates the RETS to the ODCM. No unreviewed safety question is involved.

Safety Evaluation Number 93-037

Subject
TPP 20-2, Radiation Protection Program Procedure

Summary
Revises TPP 20-2, "Radiation Protection Program" to provide clarification, correct errors, and bring into agreement with lower tier procedures. This safety evaluation is required since the Radiation Protection Manual and PGE-8005, "Radiation Protection Program" are mentioned in the FSAR and are being replaced by this procedure.

Since this change is administrative no unreviewed safety question is involved.

<u>Safety Evaluation Number</u> 93-038

<u>Subject</u>
FSAR change to Sec. 9.1 and Chapter 15

<u>Summary</u>
This change revises FSAR Chapter 15 to reflect the permanently defueled condition. The purpose of this proposed FSAR change is to eliminate unnecessary parts of Chapter 15 accident analysis that can no longer occur. the reason these events are described in FSAR Chapter 15 are no longer credible is because power operation with fuel in the reactor vessel is no longer contemplated. Review of the safety status of the defueled plant indicates that there is no need for automatic actions because the time frame to respond to postulated events is much longer than before, and the scope of necessary remedial actions is significantly reduced. Furthermore, as time passes the decay heat power decreases and consequently the need for and timing of remedial actions becomes less stringent.

The proposed change to FSAR Chapter 15 involves revision of the fuel handling accident consequences and the addition of a loss of spent fuel pool cooling description and analysis section. The remaining parts of Chapter 15 that do not support the above two sections can then be eliminated. This evaluation is based on no fuel the reactor vessel; all fuel is in the Spent Fuel Pool.

The Spent Fuel Pool and cooling systems are not changed as a result of the FSAR change. Thus the probability of accidents associated with the Spent Fuel Pool are not changed. A determination was made that no unreviewed safety question is involved.

<u>Safety Evaluation Number</u> 93-040

<u>Subject</u>
Deactivation of Main Transformers, Aux Transformer and Isophase Bus, System 87

<u>Summary</u>
This modification (deactivation) removes disconnect links shown on FSAR dwg. 8.3-1A. As such it results in a change to the Plant design as described in the FSAR. This modification has no effect on Plant safety, the modification affects only Non Q equipment and has no implication which could be outside the accidents summarized in the "<u>Summary</u> of the Permanently Defueled Safety Analysis" in TPP 18-1 Attachment 9. This attachment (TPP 18-1, A-9) applies to all the questions in this Safety Eval and is referred to as Ref. #1.

<u>Safety Evaluation Number</u> 93-041

<u>Subject</u>
System 14, Bearing Cooling Water, Deactivation Plan

<u>Summary</u>
None of the equipment to be deactivated is safety related, none is required in the permanently defueled mode and deactivation will therefore not create any probability of accident occurrence. The purpose of this safety evaluation is to show that no unreviewed safety question exits with respect to deactivation of selected portion of the BCW system.

<u>Safety Evaluation Number</u> 93-042

<u>Subject</u>
Deactivation of Circulating Water, Turbine Building Cooling Water and Cooling Tower Systems, Systems 15, 41, and 42

Summary
None of the equipment to be deactivated is safety related, none is required in the permanently defueled mode and deactivation will therefore not create any probability of accident occurrence. The purpose of this safety evaluation is to show that no unreviewed safety question exits with respect to deactivation of the TBCW, CW & CT (Systems 15 and 42).

Safety Evaluation Number 93-043

Subject
Containment Air Cooler Deactivation, System 60

Summary
The deactivation of the CACS does not result in an Unreviewed Safety Question. This evaluation is conditional in that it assumes that the reactor remains defueled, with all spent fuel stored outside Containment. The remaining credible Chapter 15 accident analyses involve radioactive releases in the Auxiliary or Fuel Buildings. Deactivation of the CACS does not affect these accidents in any manner. Deactivation of the CACS will not affect functioning of any "important to safety" equipment required to function in the defueled condition. The margin of safety as defined in the basis of any Technical Specification is not reduced since Tech Spec related to the operation of the CACS are not applicable in the defueled condition.

Safety Evaluation Number 93-044

Subject
AFW System Deactivation Plan , TPP 30-2, System 45

Summary
The AFW system will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment from service (i.e. valve position changes, fuse removal, breaker rack outs, etc.) will be accomplished using existing plant procedures and processes. The plan will be implemented in accordance with requirements of TPP 30-2. An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-045

Subject
Main Steam Supply System Deactivation, TPP 30-2, System 83

Summary
The Main Steam system will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment from service (i.e. valve position changes, fuse removal, breaker rack outs, etc.) will be accomplished using existing plant procedures and processes. The plan will be implemented in accordance with the requirements of TPP 30-2.

Safety Evaluation Number 93-046

Subject
System 64A (Reactor Coolant Pumps) Deactivation Plan, TPP 30-2

Summary
The accidents considered in this evaluation are 1) Section 15.7, Radioactive Release from a Subsystem or Components, 2) Section 15.7.4, Design Basis Fuel Handling Accidents and 3) Section 15.11 (proposed), Spent Fuel Pool Accidents. The RCPs, Lift Pumps and associated instrumentation are not associated with accident initiation or mitigation systems.

Safety Evaluation Number 93-047

Subject
System 64C (Pressurizer Auxiliaries and Controls) Deactivation Plan, TPP 30-2

Summary
In accordance with TPP 18-1, the only Chapter 15 accidents left to evaluate the proposed activity against are 1) accidents resulting in radioactive release - liquid and gaseous radwaste release, 2) design basis fuel handling accident, and 3) loss of spent fuel cooling accident. Of these accidents, the equipment important to safety is the spent fuel pool structure and liner, the service water Cat. 1 makeup to the spent fuel pool. Potentially other equipment that could be considered would be the control room ventilation system, the spent fuel pool cooling system including CCW and service water the EDGs. The methodology of this safety evaluation is that the pressurizer and associated heaters and instrumentation do not interface or relate to the aforementioned accidents or equipment.

Safety Evaluation Number 93-048

Subject
Reheat and Moisture Separation System Deactivation Plan, System 84

Summary
The MSR System will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment will be accomplished using existing plant procedures and processes. The plan will be implemented in accordance with requirements of TPP 30-2. An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-049

Subject
System Deactivation Plan for Containment Spray System, System 61

Summary
A safety evaluation of this activity is required based on a "Yes" answer to questions 3.a and 3.b of the screening criteria checklist. This evaluation will focus on why the proposed deactivation plan does not involve an unreviewed safety question in the defueled condition.

The applicable accidents in the defueled condition include waste decay tank ruptures, fuel handling accidents in the fuel building, and spent fuel pool cooling accidents. Since the containment spray system has no interaction with any of the systems relating to the above accidents, deactivating the system cannot effect the probability or consequences of the evaluated accidents.

Safety Evaluation Number 93-050

Subject
Deletion of TPP 12-13

Summary
TPP 12-13 deals with the generation of emergency procedures which deal with response to accidents which would occur at power. No unreviewed safety question is involved.

Safety Evaluation Number 93-051

Subject
Deletion of Independent Safety Review Group

Summary
This change is an administrative change which moves responsibility within the ISRG to other organizations. These changes do not result in an unreviewed safety question. Provisions of the Security Plan and QA program are not affected.

Safety Evaluation Number 93-052

Subject
System 66, Hydrogen Recombiner Deactivation Plan

Summary
Because the permanently shutdown and defueled condition, the new safety analysis, TPP 18-1, Rev. 2, states that the accidents which could have occurred during power operation or with fuel in the reactor are no longer applicable at the Trojan Nuclear Plant. The three classes of potential accidents which could result in a radiological release: loss of spent fuel pool cooling or makeup, radioactive waste decay tank ruptures, and fuel handling accidents. These accidents are not related to the hydrogen recombiner system function.

Therefore, the hydrogen recombiner deactivation plan does not adversely affect the health and safety of the public.

Safety Evaluation Number 93-053

Subject
Extraction Steam System Deactivation Plan, TPP 30-2, System 46

Summary
The Extraction Steam system will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment from service (i.e. valve position changes, fuse removal, etc.) will be implemented in accordance with requirements of TPP 30-2. An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-054

Subject
Phase 3 Deactivation of System 86, ATWS Mitigation and Actuation Circuit (AMSAC)

Summary
The Anticipated transient without scram (ATWS) Mitigating System Actuating Circuitry (AMSAC) provides an alternative and backup to the Reactor Protection System (RPS) to trip the main turbine and actuate the Auxiliary Feedwater (AFW) System in case of the possible occurrence of a common mode failure of the RPS and loss of normal feedwater transient. AMSAC was designed to meet the requirements of WCAP-10858-A, Rev. 1, AMSAC Generic Design Package, and satisfies the requirements of 10 CFR 50.62(c)(1) by actuating a turbine trip and AFW flow upon sensing low-low steam generator levels with turbine load greater than 40% power and is not discussed in the Trojan Technical Specifications. Therefore, in modes below mode 1, including the defueled condition, AMSAC is not required and may be deactivated.

Safety Evaluation Number 93-055

Subject
Phase 3, Deactivation Plan for Safety Injection System, Used for Screening for Deletion of OI 5-2, Systems 52 and 67D

Summary
The deactivation plan for the Safety Injection System (SIS) including the Accumulators and Refueling Water Storage Tank (RWST) is to place these systems in a deactivated state in the permanently defueled mode. The deactivated state means de-energizing all power supplies and closing all the appropriate system boundary valves. When the plan is implemented per the approved Trojan Quality Assurance procedures, the deactivated state will change the following in the FSAR:
1. FSAR figures showing the systems,
2. FSAR description of testing will no longer be conducted and
3. FSAR system operation description will change.

The ECCS is designed to cool the reactor and provide additional shut down margin following a LOCA. The primary element for a LOCA is fuel in the reactor vessel. Since the fuel has been permanently removed from the reactor vessel, the function of the ECCS is no longer required. Therefore placing the SIS, Accumulators, and RWST in a deactivated state does not decrease the margin of safety to the public or to the plant. This evaluation will show that deactivating the SIS, Accumulators, and RWST will not decrease the margin of safety.

Safety Evaluation Number 93-056

Subject
Nuclear Instrumentation System Deactivation Plan per TPP 30-2, Phase 3, System 78

Summary
The Nuclear Instrumentation system will be deactivated by de-energizing the electrical power supplies to the various instrument cabinets. As necessary, annunciator input switches (FTAs) will be repositioned to disable the alarms. In cases where some existing components are required for continued plant monitoring, fuses will be removed to de-energize the unnecessary equipment. An LDCR will be initiated in accordance with TPP 30-2 to cover all portions of the deactivation process.

The design modification will be needed to disconnect the automatic containment evacuation alarm. The source range nuclear instruments provide an automatic evacuation alarm based on high neutron flux. This signal can be blocked by an SSPS auxiliary relay. The design modification will lift leads in the auxiliary relay cabinet, thus providing a permanent "Blocked" status. Control room operators will still be able to manually initiate an evacuation alarm using the switch on C04.

Safety Evaluation Number 93-057

Subject
Deactivation of the Containment Hydrogen Sampling System, System 76D

Summary
The 'A' and 'B' trains of containment hydrogen analysis are being deactivated by shutting isolation valves and de-energizing electrical components. The deactivation of the CHAS Systems has no impact on plant safety, or the health and safety of the general public, because the accidents which the CHAS Systems were designed to monitor are no longer possible with the core permanently defueled.

Safety Evaluation Number 93-058

Subject
Deactivation of Post Accident Sampling Systems, System 76

Summary
The RCS and Containment PASS Systems are being deactivated due to the permanent shutdown and defueling of the Trojan Plant. The deactivation of the PASS Systems has no impact on plant safety, or the health and safety of the general public, because the accidents which the PASS Systems were intended to monitor are no longer possible with the Trojan reactor core permanently defueled.

Safety Evaluation Number 93-059

Subject
AO 13-4, Revision 5 (Operator Requalification Program)

Summary
This deletes titles of Control Room Operation, Assistant Control Room Operator, and Shift Supervisor. It deletes requirements for SROs and ROs and related training. Duties of Operations personnel modified due to the permanently defueled plant condition. Qualification of STAs per a repetition of TS 6.3.1 is deleted. Deletion of requirement for an SRO to approve temporary plant procedural changes. This last item replaced with approval by a certified fuel handler. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-060

Subject
PGE-1012 changes removing Fire Protection from TTS, and PSP 22-5, Revision 0: Fire Brigade, Fire Watch and Hot Work Training

Summary
This evaluation was done to support relocation of the Fire Protection Technical Specifications into the PGE-1012, Fire Protection Plan. A determination was made that no unreviewed safety question was involved.

Safety Evaluation Number 93-061

Subject
Stator Cooling System Deactivation Plan, TPP 30-2, System 97

Summary
The system will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment will be accomplished using existing plant procedures and processes. The plan will be implemented in accordance with requirements of TPP 30-2.

An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-062

Subject
Deactivation Plan for Containment Building, System 58

Summary
A safety evaluation of this activity is required based on a "Yes" answer to questions 3.a and 3.b of the screening criteria checklist. This evaluation will focus on why the proposed deactivation plan does not involve an unreviewed safety question in the defueled condition.

The applicable accidents in the defueled condition include waste decay tank ruptures, fuel handling accidents in the fuel building, and spent fuel pool cooling accidents. Since the EPA nitrogen purge system has no interaction with any of the systems relating to the above accidents, deactivating the system cannot effect the probability or consequences of the evaluated accidents.

Documents reviewed include FSAR section 3.8.1, 9.3.5, LDCR transmitted by DAD-0004-92M regarding PMR 91-093, FSAR Chapter 15, and TPP 18-1, Attachment 9.

Safety Evaluation Number 93-063

Subject
System Deactivation Plan for Process Sampling System and Steam Generator Sampling System, Systems 76A and 76E.

Summary
Per FSAR 9.3.2.3, the primary and secondary sampling systems have no safety function. The system is operated on an intermittent basis, other than steam generator blowdown. This deactivation does not affect steam generator blowdown sample lines. The FSAR further states that sample lines are therefore normally closed with no flow. This deactivation plan leaves all sample lines which will not be required in the permanently defueled condition permanently closed.

The only equipment important to safety included by the primary or secondary sampling systems is the sample line containment isolation valves. Containment isolation and integrity are not required in the permanently defueled condition. The secondary sampling system and the portions of the primary sampling system being deactivated do not support any equipment important to safety in the permanently defueled condition. The deactivation plan does not increase the probability or consequences of a malfunction of equipment important to safety.

The sampling system is not involved and does not affect the probability or consequences of any FSAR evaluated accidents or non-FSAR evaluated accidents applicable to the permanently defueled condition.

Safety Evaluation Number 93-064

Subject
System 68, Solid Radwaste System Deactivation Plan

Summary
Per FSAR 11.4.2.2.1.6, the portion of the Solid Radwaste System being deactivated is obsolete. Per FSAR 11.4.2.2, the entire SRWS is designed to Seismic Catego y II standards and has no accident prevention or mitigation functions. SRWS operation is not assumed in any accident evaluated in the FSAR for the plant in its permanently shutdown and defueled condition.

The purpose of this safety evaluation is to show that no unreviewed safety questions exists with respect to deactivation of the SRWS (System 68).

Safety Evaluation Number 93-066

Subject
Auxiliary and Fuel Building Ventilation System Deactivation Plan, System 32

Summary
Because of the permanently shutdown and defueled condition , the new safety evaluation procedure TPP 18-1, Rev. 2, states that the accidents which could have occurred during power operation or with fuel in the reactor are no longer applicable at the Trojan Nuclear Plant. The three classes of potential accidents which could result in a radiological release: loss of spent fuel pool cooling or makeup, radioactive waste decay tank ruptures, and fuel handling accidents. These accidents are not related to the AB/FB Ventilation System function. The AB/FB Ventilation System Deactivation Plan provides the option of maintaining the building occupation for the worker during the daily activities at Trojan. FSAR Section 15 does not take credit for AB Ventilation System during the DBA analysis. However, TPP 18-1,. Rev. 2, Attachment 9 does not credit any SFPEV system in the accident analysis. Therefore, the AB/FB Ventilation Deactivation Plan does not adversely affect the health and safety of the public.

Safety Evaluation Number 93-067

Subject
Control Rod Drive (CRD) Deactivation, TPP 30-2, System 55

Summary
The Control Rod Drive (CRD) System will be deactivated by de-energizing the electrical power supplies to the various system cabinets. The power will be removed by opening breakers, racking out breakers and removing control power fuses. As necessary, annunciator input switches (FTAs) will be repositioned to disable the alarms. The deactivation plan will be implemented when the plant is defueled and the CRD system is not required to be operable (TTS required modes 1 and 2).

Safety Evaluation Number 93-068

Subject
Residual Heat Removal (RHR) System Phase 3 Deactivation, System 49

Summary
The deactivation plan for the RHR System is to place the system in a deactivated state in the permanently defueled mode. The deactivated state means de-energizing all power supplies and closing all the appropriate system boundary valves. When the plan is implemented per the approved Trojan Quality Related procedures the deactivation state will change the following in the FSAR: 1. FSAR figures showing system configuration, 2. FSAR description of testing will no longer be conducted and 3. FSAR system operation description will change.

The RHR system is designed to cool the reactor during normal and emergency conditions and to transfer refueling water between the RWST and the refueling cavity during before and after refueling operations. Since the reactor core has been permanently removed from the reactor vessel, the function of the RHR system for different core cooling conditions and refueling operations is no longer required.

Additionally, RHR cooling of the Spent Fuel Pool (SFP) when a full core has been discharged into the SFP at 150 hours after shutdown (44,300,00 BTU/hr at 150 hours, condition 2 of Section 9.1.3.3 of the FSAR) is no longer required. The decay heat rate of the SFP as of April 7, 1993 (21 weeks since the core was loaded into the SFP) is 6,760,000 BTU/hr per calculation TC-719. This conservative calculation demonstrates that the present decay heat rate is less than the heat load of 20,100,000 BTU.hr stated in Section 9.1.3.3 for condition 3. Condition 3 does not take credit for RHR cooling but relies on additional makeup water and natural circulation cooling.

Therefore, placing the RHR system in a deactivated state does not reduce the margin of safety to the public or to the plant nor does it propose an unreviewed safety question. This evaluation will demonstrate that deactivating the RHR system will not reduce the margin of safety to the plant or purpose an unreviewed safety question.

Safety Evaluation Number 93-069

Subject
Deactivation Plan for Condensate Demineralizer System, System 39

Summary
Per FSAR 10.4.7, the entire Condensate Demineralizer System is designed to Seismic Category II standards and has no accident prevention or mitigation functions. The Demin system is a non-safety and non-quality related system. Operation is not assumed in any accident evaluated in the FSAR for the plant in its permanently shutdown and defueled condition.
The purpose of this safety evaluation is to show that no unreviewed safety question exits with respect to deactivation of the Condensate Demin System (System 39).

Safety Evaluation Number 93-070

Subject
Deactivation Plan for Condenser and Air Removal, System 43

Summary
Per FSAR 10.4.1, 10.4.2, and 11.3.2.3.3, the entire Condenser and Air Removal System is designed to Seismic Category II standards and has no accident prevention or mitigation functions. The system is a non-safety related system. Operation is not assumed in any accident evaluated in the FSAR for the plant in its permanently shutdown and defueled condition.
The purpose of this safety evaluation is to show that no unreviewed safety question exists with respect to deactivation of the Condenser and Air Removal System (System 43).

Safety Evaluation Number 93-071

Subject
Deactivation of Condensate System, SG Feed Pump & Auxiliaries, Feedwater Flow and Level Controls, Feed Line Isolation Actuation, and SG Feedwater Pump Turbine Drivers, Systems 44, 45B, 45E, 45F, and 48

Summary
These systems will be deactivated by de-energizing electrical components and by repositioning certain system valves. Removal of equipment from service (I.e. valve position changes, fuse removal, breaker rack outs, etc.) will be implemented in accordance with requirements of TPP 30-2. An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-072

Subject
Revise OI 9-1 to Allow Running the Main Lube Oil System without the Main Generator Seal Oil System

Summary
Changes to the operation of the main lube oil system will not involve and unreviewed safety question.

Safety Evaluation Number 93-074

Subject
Deactivation Plan for Vibration & Loose Parts Monitoring,, System 75

Summary
A safety evaluation of this activity is required based on a "Yes" answer to question 3.a of the screening criteria checklist. This evaluation will focus on why the proposed deactivation plan does not involve an unreviewed safety question in the defueled condition.

The applicable accidents in the defueled condition include waste decay tank ruptures, fuel handling accidents in the fuel building, and spent fuel pool cooling accidents. Since the VLPMS has no interaction with any of the systems relating to the above accidents, deactivating the system cannot effect the probability or consequences of the evaluated accidents.

Documents reviewed include FSAR section 4.4.6.4, Chapter 15, and TPP 18-1 Attachment 9.

Safety Evaluation Number 93-075

Subject
Miscellaneous Gas Supply (System 74) Deactivation Plan

Summary
The system will be deactivated by de-energizing electrical components, installing blank flanges, and repositioning various system valves. Removal of equipment from service will be accomplished using existing plant procedures and processes. The plan will be implemented in accordance with TPP 30-2. An LDCR will be initiated in accordance with TPP 30-2.

Safety Evaluation Number 93-076

Subject
Steam Generators (System 63A) Deactivation Plan

Summary
Isolating and placing the steam generators in cold wet layup has no impact on safe plant operation, or the health and safety of the general public, because the reactor core has been permanently defueled. The FSAR analyzed accidents involving the steam generators are no longer possible.

Safety Evaluation Number 93-077

Subject
Temporary Modification 93-006 Revision 1

Summary
Temporary Modification 93-006 authorized removal of sprinkler system AL, with system isolation at PIV-21 and a freeze protection gap specified at a blind flange connection in an outside pit. due to leakage past PIV-21, an alternate freeze protection strategy of sealing the blind flange and burying it in the pit is authorized by Revision 1 to the TM. To provide adequate depth of cover, valve FP-253 is removed from the pit. The remaining flange connection will be blanked off, coated, and buried. This is the only change from Revision 0.

The function of the system was strictly to provide fire protection to the radcon access trailer; with the removal of the trailer, the need for the system is eliminated. The system is not relied upon for protection of safe shutdown equipment. Removal of this branch has no detrimental effect on the fire protection system as a whole. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-078

Subject
Remote Shutdown Station Deactivation Plan, TPP 30-2, System 0

Summary
When all the associated systems as described in the Phase III deactivation plan are deactivated, or not required, the RSS will be deenergized by opening breakers and shutting off power to the complete RSS.

Safety Evaluation Number 93-080

Subject
Deactivation Plan for Reactor Non-Nuclear Instrumentation, TPP 30-2, System 80

Summary
The Post-Accident instruments will be deactivated by de-energizing the electrical power supplies to the various instrument cabinets. As necessary, annunciator input switches (FTAs) will be repositioned to disable the alarms. In cases where some existing components are required for continued plant monitoring, fuses [illegible] be removed to de-energize the unnecessary equipment. An LDCR will be initiated in accordance with TP[illegible] to cover all portions of the deactivation process.

Safety Evaluation Number 93-081

Subject
Feedwater Heaters, Vents & Drains System (System 47), Deactivation Plan

Summary
The Heater Drains and Vents System is not required to operate in the current plant condition. It is not used to mitigate the effects of the remaining credible FSAR analyzed accidents. De-energization of electrical equipment and valve position changes will be performed in accordance with existing plant procedures. Therefore, deactivation of the system does not create an unreviewed safety question.

Safety Evaluation Number 93-082

Subject
Containment HVAC System (System 60), Deactivation Plan

Summary
Because the permanently shutdown and defueled condition, the new safety analysis, TPP 18-1, Rev. 2 states that the accidents which could have occurred during power operation or with fuel in the reactor are no longer applicable at the Trojan Nuclear Plant. The three classes of potential accidents which could result in a radiological release: loss of spent fuel pool cooling or makeup, radioactive waste decay tank ruptures, and fuel handling accidents. These accidents are not related to the CS HVAC System function.

Therefore, the CS Ventilation Deactivation Plan does not adversely affect the health and safety of the public.

Safety Evaluation Number 93-083

Subject
Deletion of All EOPs (to include EIs, ESs, ECAs and FRs)

Summary
All Emergency Operating Procedures (EOP's) are written assuming that there is fuel in the Reactor Vessel and that the Reactor Coolant System will be at an elevated temperature prior to the event that initiates the use of the EOP's.

Safety Evaluation Number 93-084

Subject
Deactivation Plan for Domestic Water and Water Pretreatment Systems, Systems 8 and 21

Summary
Per FSAR 9.2.4, the entire Domestic Water/Water Pretreatment System is designed to Seismic Category II standards and has no accident prevention or mitigation functions. The system is a non-safety related system. Operation is not assumed in any accident evaluated in the FSAR for the plant in its permanently shutdown and defueled condition.

The purpose of this safety evaluation is to show that no unreviewed safety question exists with respect to deactivation of the Water Pretreatment System and portions of the Domestic water System (System 8 and 21).

Safety Evaluation Number 93-085

Subject
Engineered Safeguards Actuating System (ESF) (System 53), Deactivation Plan

Summary
The Engineered Safety Features Actuation System will be deactivated by de-energizing the electrical power supplies to the various system cabinets and components. The power will be removed by opening breakers, removing fuses and lifting leads. As necessary, annunciator input switches (FTAs) will be repositioned to disable the alarms. The deactivation plan will be implemented when the plant is defueled and the ESFAS system is not required to be operable (TTS required modes 1 through 6). A PMR is required to make permanent the lifted leads. This deactivation plan also removes the Containment Evacuation Alarm.

Safety Evaluation Number 93-086

Subject
Deactivation Plan for Makeup Demineralizer Water and Condensate Water Storage and Transfer Systems, Systems 22 and 37

Summary
None of the equipment to be deactivated is safety related, none is required in the permanently defueled mode and deactivation will therefore not create any probability of accident occurrence. The purpose of this safety evaluation is to show that no unreviewed safety question exists with respect to partial deactivation of the Systems (Systems 22 and 37)

Safety Evaluation Number 93-087

Subject
Deactivation Plan for Pressurizer Relief System, System 64D

Summary
In accordance with TPP 18-1, the only Chapter 15 accidents left to evaluate the proposed activity against are 1) accidents resulting in radioactive release - liquid and gaseous radwaste release, 2) design basis fuel handling accident, and 3) loss of spent fuel cooling accident. Of these accidents the equipment important to safety is the spent fuel pool structure and liner and the service water SC I makeup to the spent fuel pool. Potentially other

equipment that could be considered would be the control room ventilation system, the spent fuel pool cooling system including CCW and service water and the EDGs. The methodology of this safety evaluation is that the pressurizer relief system and associated instrumentation do not interface or relate to the aforementioned accidents or equipment.

Other equipment that could be considered important to safety would be the pressurizer PORVs for use during a Low Temperature Overpressure (LTOPS) event. The RCS is being maintained vented by a removed pressurizer safety valve. The vent size is greater than the size of an open PORV.

__Safety Evaluation Number__ 93-088

__Subject__
Deactivation Plan for Reactor Coolant System, System 64B

__Summary__
In accordance with TPP 18-1, Attachment 9, the only accidents pertinent to the defueled plant condition are, 1) accidents resulting in radioactive release (liquid and gaseous radwaste tanks), 2) design basis fuel handling accidents, and 3) loss of spent fuel pool cooling. The equipment important to safety for the accidents would involve the spent fuel pool liner and structure and equipment related to the SCI spent fuel pool makeup capability. Other equipment that could be considered important to safety would involve the spent fuel pool cooling system including the SW and CCW system and the EDGs. The methodology of this safety evaluation is that the RCS and related instrumentation do not relate to or interface with the aforementioned accidents or equipment.

__Safety Evaluation Number__ 93-089

__Subject__
PGE-1058, Security Plan, Revision 0

__Summary__
Since this is a change to the Security Plan, the evaluation is considered safeguards information.

__Safety Evaluation Number__ 93-090

__Subject__
Regulatory Guide 1.58 (NDE Qualifications) Revised Position

__Summary__
PGE's letter to the NRC dated March 15, 1993 stated that, upon issuance of the Possession Only License, 10 CFR 50.55a(f) and (g) would no longer apply to Trojan. PGE-1049 and Code Case N-356 therein modify SNT-TC-1A-1980 to establish the NDE personnel qualification system to meet 10 CFR 50.55a. The State of Oregon endorses SNT-TC-1A-1984 for work not governed by ASME Section XI. This modifies our position to delete use of Section XI for NDE Qualifications.

Memorandum HKC-056-93M dated April 13, 1993 states that after receipt of the Possession Only License, Licensing will set aside Topical Reports PGE-1048, 1049, 1050, and 1051. This LDCR evaluates deleting PGE-1049 and PGE-1051 as part of changing NDE qualification requirements.

__Safety Evaluation Number__ 93-092

__Subject__
Deactivation Plan for the Chlorination System, System 10

Summary
None of the equipment to be deactivated is safety related, none is required in the permanently defueled mode and deactivation will therefore not create any probability of accident occurrence. The purpose of this safety evaluation is to show that no unreviewed safety question exists with respect to deactivation of selected portion of Hypochlorite injection system.

Safety Evaluation Number 93-093

Subject
Gaseous Radwaste Deactivation Plan, System 72

Summary
The Gas Collection portion of the GRWS is no longer required to process gas. Deactivation of this portion of the GRWS does not represent an unreviewed safety question since the design objectives of the system are based on an operating plant condition. The containment isolation valves are maintained in the closed condition which is inherently safer than having the valves open.

FSAR Section 15.7 describes a rupture of a Waste Gas Decay Tank. This analysis assumes a large amount of radioactive gas exists in the tank to be ruptured. Prior to the implementation of this plan, all of the Waste Gas Decay Tanks will be completely discharged so that no further stored waste gas exists. Thus, the deactivated condition of the WGDTs is completely bounded by the accident scenario analysis for the operating WGDTs. The deactivated condition is significantly safer than the operating condition due to the lack of waste gas in the tanks.

Safety Evaluation Number 93-094

Subject
System 50, Deactivation Plan for the Chemical & Volume Control System

Summary
Possible FSAR evaluated accidents applicable to the permanently defueled condition are radioactive gas/liquid waste systems leak, or failure, liquid tank failure radioactive releases, and fuel handling accidents. Any changes to the CVCS system would not affect the probability of radwaste system accidents or fuel handling accidents in any way, as CVCS is not involved in these accidents and does not support any systems that are.

Deactivation does not increase the probability of a liquid tank failure radioactive release. Deactivation of CVCS will include draining of all tanks. Due to elimination of pressure in the tanks, any possibility of tank rupture will be reduced. Draining the radioactive liquids from the tanks will reduce radioactive releases which result from a tank rupture.

CVCS does not contain or support any equipment important to safety in the permanently defueled mode.

CVCS does not provide any safety function in the permanently defueled condition. No CVCS components or subsystems support any system which provides for the safety and integrity of radioactive waste tanks or the spent fuel pool.

No margin of safety defined in the Technical Specification Bases applicable in the permanently defueled condition is affected by this activity.

Safety Evaluation Number 93-096

Subject
Deactivation Plan for Generator Hydrogen and Seal Oil System, System 95

Summary
The system will be deactivated by de-energizing electrical components, and repositioning various system valves. Removal of equipment from service will be accomplished using existing plant procedures and processes. This plan will be implemented in accordance with TPP 30-2.

Safety Evaluation Number 93-097

Subject
Deletion of Surveillance and Test Engineering Procedures

Summary
This evaluation shows that there will be no impact to health and safety of the public by deleting the subject procedures and FSAR Ref. Exemption from the requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J was received from the NRC in a letter dated April 12, 1993. PGE Licensing position on ASME Section XI requirements is found in H. K. Chernoff to M. H. Schwartz Memo HKC-056-93M dated April 13, 1993. This memo affirms the STE position that upon receipt of the POL all ASME Section XI testing and inspections are no longer required and may cease. Since the testing and inspections are no longer required or are exempted by the NRC the need for the procedures to administer them are also no longer required and can be deleted. Subject FSAR references and Topicals can similarly be deleted.

LDCR 93-017 proposes changes to Chapter 15 of the FSAR. This LDCR is based on there being very limited credible accidents for the permanently defueled condition. It concerns itself with Fuel Handling and Spent Fuel Pool Cooling accidents. The Fuel Handling equipment and the Spent Fuel Pool Cooling equipment are not ASME Class 1, 2 or 3 components except for those parts that are a part of containment penetrations (already exempted from testing - see above). Waste Gas system components have ASME Class components however they do no meet the scope statement of ASME Section XI for inclusion in IST. Since ASME Section XI only requires IST for those components within the Scope, there are no components involved in credible accidents.

Safety Evaluation Number 93-098

Subject
Deactivation of the Reactor Protection System, System 56

Summary
The Reactor Protection and Control System will be deactivated by de-energizing the electrical power supplies to the Solid State Protection System cabinets, the Hagan racks and the Instrumentation and control loops that appear on the M1T(6) - Series drawings. The power will be removed by opening breakers and removing fuses. The deactivation plan will be implemented when the plant is defueled and the RPS system is not required to be operable (TTS required modes 1 through 6). A PMR is required to reconfigure the power to the several DC power supplies.

Safety Evaluation Number 93-099

Subject
System 33, Deactivation Plan for the HVAC System - Turbine & Turbine Aux Bldg

Summary
Because the permanently shutdown and defueled condition, the new safety analysis, TPP 18-1, Rev. 2 states that the accidents which could have occurred during power operation or with fuel in the reactor are no longer applicable at the Trojan Nuclear Plant. The three classes of potential accidents which could result in a radiological release: loss of spent fuel pool cooling or makeup, radioactive waste decay tank ruptures and fuel handling accidents. These accidents are not related to the deactivated TB HVAC System function. Therefore, the TB Ventilation Deactivation Plan does not adversely affect the health and safety of the public.

Safety Evaluation Number 93-100

Subject
Dirty Radwaste System Deactivation Plan, System 71

Summary
Deactivation of this portion of the DRWS does not represent an unreviewed safety question.

Safety Evaluation Number 93-101

Subject
Clean Radwaste System Deactivation Plan, System 69

Summary
Deactivation of this portion of the CRWS does not represent an unreviewed safety question.

Safety Evaluation Number 93-103

Subject
Deactivation Plan for Main Generator and Excitation System (System 98)

Summary
The system will be deactivated by de-energizing electrical components and/or removing them. Removal of equipment from service will be accomplished using existing plant procedures and processes. This plan will be implemented in accordance with TPP 30-2.

Safety Evaluation Number 93-104

Subject
Deactivation Plan for Turbine Steam Seal and Sealing Drain System (System 92)

Summary
The system will be deactivated by de-energizing electrical components and isolating air from other components. Removal of equipment from service will be accomplished using existing plant procedures and processes. This plan will be implemented in accordance with TPP 30-2.

Safety Evaluation Number 93-107

Subject
Deleting AO-1-15, Planning & Control organization & responsibilities

Summary
AO 1-15, Planning & Control Department Responsibilities, is being deleted. Planning & Control is being disbanded due to Plant closure. Planning & Control functions are, however, being retained and are being transferred to other departments. Therefore, probabilities, consequences, etc. of accidents & malfunctions are not being increased and no reduction in commitment to quality will occur.

Safety Evaluation Number 93-108

Subject
Process and Auxiliary Steam Deactivation Plan, System 28

Summary
The Process and Auxiliary Steam System is not required to operate in the current plant condition. It is not used to mitigate the effects of the remaining credible FSAR analyzed accidents. De-energization of electrical equipment and valve position changes will be performed in accordance with existing plant procedures. Therefore, deactivation of the system does not create an unreviewed safety question.

Safety Evaluation Number 93-109

Subject
As-Built Package 632-91 - Radwaste Piping Code

Summary
As-built package (ABP) 632-91 changes the design code on radwaste piping isometrics from ANSI B31.7 to ANSI B31.1, consistent with Reg. Guide 1.143 and FSAR Table 3.2-3. When the piping was originally downgraded, the isometric drawings were not changed.

While processing the ABP, certain other changes were made to the associated P&IDs. These changes fall under the category of drawing maintenance, and include the following: iso numbers, breaks shown; drains, vents, test connections shown; valve tag numbers corrected; piping tie-in sequences corrected. No equipment important to safety is affected.

Safety Evaluation Number 93-111

Subject
Deletion of Emergency Fire Procedures

Summary
This change deletes procedures which are listed in the FSAR, Section 13.5.8.

Safety Evaluation Number 93-113

Subject
System 79, Radiation Monitoring System Deactivation Plan

Summary
Deactivation of these non-required RMS channels does not represent an unreviewed safety question. Adequate monitoring channels remain to provide dose assessment in the event of the limiting fuel handling accident. PERMs 1B and 1E are not required since no radioactive source exists in Containment that could require their use. PERM 2B is not required since no significant iodine source exists in the Aux/Fuel Buildings (Iodine from SFP is negligible). PERMs 6A, 6B, and 6C and 16A, 16B, 16C, and 16D are not required since there is no radioactive leakage source to the Steam Generators, and the Main Steam System is not in use. PERM 13 is not required since there is no potential for fuel failure. PERMs 26A, 26B, 26C, and 26D are not required since there is no longer any power operation to produce Nitrogen-16.

Safety Evaluation Number 93-114

Subject
QA Reg Guides

Summary
The listing of Regulatory Guides regarding quality assurance to which PGE is committed are being deleted from the Quality Assurance Program Manual and are being placed into the Defueled Safety Analysis Report (DSAR).

These commitments have been evaluated to no longer apply and have been deleted or transferred to the DSAR. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-118

Subject
Deactivation Plan for Main Turbine and Turbine Controls System, System 93

Summary
The system will be deactivated by repositioning various valves, de-energizing electrical components, and isolating air to components. Removal of equipment from service will be accomplished using existing plant procedures and processes. This plant will be implemented in accordance with TPP 30-2.

Safety Evaluation Number 93-121

Subject
System 72A, Deactivation Plan for Gaseous Radwaste System

Summary
Deactivation of the AGA does not represent an unreviewed safety question. The only potential AGA sample input component that will remain in service is the Spent Resin Storage Tank, which will not require hydrogen/oxygen sampling with the plant in the permanently defueled and shutdown condition.

Safety Evaluation Number 93-122

Subject
Revision to PGE-1052, Appendix A, Technical Interpretations

Summary
On January 4, 1993, Portland General Electric (PGE) announced that Trojan would not be restarted. On May 5, 1993, PGE was granted a Possession Only License (POL). The POL ensures that the nuclear fuel will never again be loaded in the reactor.

On March 17, 1993, Trojan Plant Procedure TPP 30-1, Revision 1, Nuclear Division Defueled Requirements and Defueled Systems List, was issued. Nuclear Division Defueled Requirements and Defueled Systems List, was issued. Attachment 1 to this TPP is a list of applicable technical specifications while defueled. Attachment 2 to this TPP is a list of all systems at Trojan and a designator for the required operational status of the system - Operable, Available, Prudent to Operate, or Not Required. Based on this procedure and its attachments, systems are being deactivated at Trojan.

PGE Topical Report 1052 needs to be revised so that systems that are no longer required to be operable, per Trojan's POL and TPP 30-1, are not treated as quality-related. Minimizing the number of quality-related systems at Trojan saves money by simplifying maintenance, drawing updates, etc.

Currently, Section 4.0 of PGE-1052, Quality Classification Evaluation, states that "The criteria in the following subsections form the basis for Trojan Nuclear Plant quality classification evaluation of structures, systems, and components. These criteria are based on Trojan Licensing commitments and are supplemented by the technical interpretations in Appendix A." Appendix A says "to be supplied later".

This LDCR adds the technical interpretation to Appendix A to reflect Trojan's POL and TPP 30-1. System and Component classifications will continue to be performed in accordance with currently approved instructions in PGE-1052 and administrative procedures.

Safety Evaluation Number 93-123

Subject
CAR C92-0040-01-02 LDCR

Summary
Implementation of this LDCR provides consistency between the actual plant configuration and the FSAR description. No safety concerns exist with implementation of this LDCR.

Safety Evaluation Number 93-125

Subject
PSC 93-001, 480 V Load Center Tap Change

Summary
Due to plant closure, loads on the various buses have decreased. The decreased loading has resulted in a rise in 480 Volt bus voltages. In order to decrease this voltage, the 480 Volt load center transformer TAPs will be changed by 2.5% to decrease bus voltage. This change in no way causes any systems to operate outside their design minimum voltages per the FSAR. This change has no effect on the design procedures or operation of the 480 Volt system. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-128

Subject
FSAR change to sections 13.1.2.2 & 13.1.2.6

Summary
The Assistant Operation Manager, Shift Supervisor and Assistant Control Operator positions and functions are described in FSAR Section 13.1.2..2. FSAR Section 13.1.2.6 includes the Assistant Operation Manager in the line of succession of authority. These positions no longer exist as they have been deleted from the Plant Staff due to manning reductions caused by plant shutdown. These positions need to be deleted from the FSAR.

Safety Evaluation Number 93-131

Subject
OI 7-2 Revision 21: Demineralized Water and Primary Makeup Water Operation

Summary
The Makeup Water Treatment System as described in the FSAR has been abandoned. The polisher demineralizer system has also been abandoned. Since the plant is permanently shutdown, the only safety related requirement has been to maintain Spent Fuel Pool temperature and inventory. With no current method for making demineralized water, the plant will deplete its supply in a short time. Demineralized water is the preferred source of makeup to the Spent Fuel Pool. This change will provide makeup to the Makeup Water Treatment System and ensure the preferred source will be available when called upon.

Safety Evaluation Number 93-132

Subject
Reclassify ESF Equipment as not safety-related.

Summary
On January 4, 1993, Portland General Electric (PGE) announced that Trojan would not be restarted. On May 5, 1993, PGE was granted a Possession Only License (POL). The POL ensures that the nuclear fuel will not be

loaded in the reactor. Safety-related, or "Basic Components" are described in 10 CFR 50.2 as any plant structure, system or component (SSC) necessary to assure 1) The integrity of the reactor coolant system, 2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or 3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to those referred in 10 CFR 100. The basis for the SSCs required to meet the three criteria listed above is as described in Trojan's FSAR Chapter 15 Accident Analyses and as events described in FSAR Section 9.1 - Fuel Storage Handling Systems. In March of 1993, Attachment 9 was added to TPP 18-1. This attachment provides a discussion of the effects on the FSAR Chapter 15 Accident Analysis from placing Trojan in a permanently defueled condition. This attachment concludes that the only remaining accidents are an accident resulting in a radioactive release, a fuel handling accident, and a loss of cooling to the spent fuel pool. The above listed ESF systems were designed to mitigate the consequences of a design basis accident, for an operating nuclear power plant, that occur within the Containment Building. Since the reactor has been permanently defueled the possibility of a design basis accident occurring inside the Containment Building no longer exists.

__Safety Evaluation Number__ 93-133

Subject
Downgrading RCS Equipment from Safety-Related

Summary
On January 4, 1993, Portland General Electric (PGE) announced that Trojan would not be restarted. On May 5, 1993, PGE was granted a Possession Only License (POL). The POL ensures that the nuclear fuel will not be loaded in the reactor. Safety-related, or "Basic Components" are described in 10 CFR 50.2 as any plant structure, system or component (SSC) necessary to assure 1) The integrity of the reactor coolant system, 2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or 3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to those referred in 10 CFR 100. The basis for the SSCs required to meet the three criteria listed above is as described in Trojan's FSAR Chapter 15 Accident Analyses and as events described in FSAR Section 9.1 - Fuel Storage Handling Systems.

In March of 1993, Attachment 9 was added to TPP 18-1. This attachment provides a discussion of the effects on the FSAR Chapter 15 Accident Analysis from placing Trojan in a permanently defueled condition. This attachment concludes that the only remaining accidents are an accident resulting in a radioactive release, a fuel handling accident, and a loss of cooling to the spent fuel pool.

The RCS provides a barrier against leakage and the release of radioactivity generated within the reactor. Since the reactor has been permanently defueled and the RCS is not used to mitigate the consequences of the remaining accidents; the RCS, and its components, can be reclassified as nonsafety-related.

__Safety Evaluation Number__ 93-134

Subject
Downgrading CCW System Equipment from Safety-Related

Summary
On January 4, 1993, Portland General Electric (PGE) announced that Trojan would not be restarted. On May 5, 1993, PGE was granted a Possession Only License (POL). The POL ensures that the nuclear fuel will not be loaded in the reactor. Safety-related, or "Basic Components" are described in 10 CFR 50.2 as any plant structure, system or component (SSC) necessary to assure 1) The integrity of the reactor coolant system, 2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or 3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to those referred in 10 CFR 100. The bases for the SSCs required to meet the three criteria listed above is as described in Trojan's FSAR Chapter 15 Accident Analyses and as events described in Trojan's Fuel Storage Handling Systems.

As described in TPP 18-1, Attachment 9 SUMMARY OF THE PERMANENTLY DEFUELED SAFETY ANALYSIS, the only remaining accidents that could have significant radiological consequences are a fuel handling accident and a loss of cooling to the spent fuel pool.

Per Chapter 9.2 of the FSAR, the CCW system's only required functions during accident conditions were to provide cooling flow for the removal of heat from the Reactor Coolant System via the Residual Heat Removal (RHR) heat exchangers, from the Containment atmosphere via the Containment Air Coolers, and from the seals of the RHR pumps, the Containment Spray pumps and Safety Injection pumps. Since the reactor is permanently defueled and there is no fuel in the reactor vessel, power operation can no longer occur. There is no longer a concern for maintaining the reactor coolant pressure boundary, and SSCs associated with safe shutdown of the reactor are no longer applicable. Therefore, the CCW equipment is no longer required to meet the requirements of 10 CFR 50.2, or Chapter 9 and 15 of the FSAR.

Safety Evaluation Number 93-135

Subject
LDCR 93-064, Changing of definition of Quality-Related in the QA Manual with respect to Trojan Technical Specification Monitoring Equipment

Summary
The applicability of each TS has been evaluated to ensure that it supports the FSAR accident analyses. Compliance with applicable TSs ensures that the facility is operated in a safe and responsible manner. Trojan TS currently defueled and is no longer authorized to enter any other defined operating mode making the applicable TS a small subset of the original TS. Reducing the quality status of instruments that are no longer required to ensure compliance with the applicable TS will not adversely affect the safe operation of the facility.

Safety Evaluation Number 93-136

Subject
Reclassify CVCS to nonsafety-related

Summary
On January 4, 1993, Portland General Electric (PGE) announced that Trojan would not be restarted. On May 5, 1993, PGE was granted a Possession Only License (POL). The POL ensures that the nuclear fuel will not be loaded in the reactor. Safety-related, or "Basic Components" are described in 10 CFR 50.2 as any plant structure, system or component (SSC) necessary to assure 1) The integrity of the reactor coolant system, 2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or 3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to those referred in 10 CFR 100. The basis for the SSCs required to meet the three criteria listed above is as described in Trojan's FSAR Chapter 15 Accident Analyses and as events described in FSAR Section 9.1 - Fuel Storage Handling Systems.

In March of 1993, Attachment 9 was added to TPP 18-1. This attachment provides a discussion of the effects on the FSAR Chapter 15 Accident Analysis from placing Trojan in a permanently defueled condition. This attachment concludes that the only remaining accidents are an accident resulting in a radioactive release, a fuel handling accident, and a loss of cooling to the spent fuel pool.

The CVCS was designed to provide support functions for the Reactor Coolant System (RCS). The CVCS maintains water level in the pressurizer, provides seal water to the Reactor Coolant Pumps, controls water chemistry in the RCS, and processes effluent from the RCS. Since the reactor has been permanently defueled, the CVCS is no longer required. In addition, the CVCS is not required to mitigate the consequences from any of the remaining accidents. Therefore, the CVCS is no longer required to be classified as safety-related, and the CVCS components shown as safety-related in the Safety Related Database on the LAN can be reclassified as nonsafety-related.

Safety Evaluation Number 93-137

Subject
Chapter 2 of the DSAR, "Site Characteristics"

Summary
Site Characteristics have been updated to reflect the defueled condition of the plant. Major revisions include: Deletion of discussion for Visitor Center, Ability to use areas in exclusion are for commercial purposes, Correct text on amount of diesel fuel storage buried on site, deleted reference to toxic chemicals no longer stored on site, deleted discussion of control room ventilation response to toxic gas, deleted discussion of cross connecting CCW with CWS, reduced discussion of meteorological monitoring, deleted discussion of flood protection, Add discussion of Seismic Margin Earthquake.

Safety Evaluation Number 93-138

Subject
Chapter 3 of the DSAR, "Design Criteria"

Summary
Chapter 3 describes the facility design features pertinent to the Plant after issuance of the Possession Only License and after 40 weeks since the Plant was shutdown. The limited scope of Chapter 3 now covers general topics such as the remaining regulatory requirements, classification of structures, systems, and components and natural phenomenon (i.e., wind, tornado, flood, seismic, etc.) as well as components such as Fuel Building and spent fuel storage, auxiliary systems, and electric power. The only remaining safety-related components are the Spent Fuel Pool, liner, and racks with all other previously safety-related components now either removed from service or reclassified and maintained as sources of long term makeup water for the Spent Fuel Pool. With the long time available before makeup water is needed (over 4 days) the sources of makeup water can be temporarily out of service and are not required to be safety-related due to the lack of credible significant offsite releases.

Safety Evaluation Number 93-139

Subject
Chapter 7 of the DSAR, "Conduct of Operations

Summary
Discussion of INPO accreditation and systems approach to training is being deleted from the FSAR for Electrician Training, Instrument and Control Technician Training, Chemistry Training Program, Radiation Protection Technician Training, and Technical Staff/Technical Manager Training. With Plant defueling, Trojan will no longer participate in INPO programs. Reduction in training requirements for maintenance personnel are being implemented as a result of the reduction inoperation activities and systems requiring maintenance.

Safety Evaluation Number 93-140

Subject
Chlorine Gas Event Accident Scenarios

Summary
With the permanently defueled condition of Trojan and the issuance of a Possession Only License (POL) a significant reduction in credible accidents has resulted. Only three (3) accidents remain credible.

1. Radioactive Release from a Subcomponent or System.

As a result of plant shutdown in November 1992 and subsequent plant defueling, the production of radioactive gases and liquids associated with reactor operation has ceased. Radioactive decay has eliminated much of the inventory of such materials that existed at the time the reactor was permanently shutdown. Accidents associated with the radioactive release from subcomponents or systems have all been determined to be a small fraction of the 10 CFR 100 limits. Operator action for these accidents in the permanently defueled condition are not required to protect the health and safety of the public.

2. Fuel Handling Accident

As a result of the permanently defueled condition of Trojan the consequences of a fuel handling accident could be based on actual maximum conditions present in the fuel pool rather than assuming worst case conditions that could result based on continued plant/reactor operation. The accident analysis performed demonstrated doses that were a small fraction of 10 CFR 100 limits. The accident did not credit operator action for consequence mitigation.

3. Loss of Spent Fuel Pool Cooling

The heat load in the spent fuel pool is much less than the design value due to the decay time that has elapsed (assuming 40 weeks after shutdown) since the reactor has shutdown and the number of irradiated fuel assemblies stored in the pool is less than previously assumed.

This safety analysis assumes the following: 40 weeks have occurred since plant shutdown; Loss of cooling occurs with a pool starting temperature of 140 degrees Fahrenheit; A chlorine gas event occurs coincident with the loss of cooling, which precludes operator action.

Based on the above assumptions the pool would take 35.8 hours to start boiling and 218.6 hours (approximately 9 days) to boil off pool inventory to approximately 10 feet above the fuel. These results show that there is sufficient time to affect repairs to the cooling system or to establish make-up flow prior to uncovery of the fuel without operator actions. Spent fuel pool cooling only requires level be maintained above fuel.

<u>Safety Evaluation Number</u> 93-141

<u>Subject</u>
Chapter 6 of the DSAR, "Accident Analyses"

<u>Summary</u>
The permanent defueling of the Trojan Facility has resulted in many accidents that were analyzed in Chapter 15 of the FSAR are no longer applicable. The permanent defueling has resulted in the permanent removal of the reactor operation including heat generation and removal by the secondary system. The following accidents therefore have been deleted from discussion from the Defueled Safety Analysis:

15.1, Increase in Heat Removal by the Secondary System
15.2, Decrease in Heat Removal by the Secondary System
15.3, Decrease in Reactor Coolant System Flow Rate
15.4, Reactivity and Power Distribution Anomalies
15.5, Increase in Reactor Coolant System Inventory
15.6, Decrease in Reactor Coolant Inventory
15.8, Anticipated Transients Without Scram
15.9, Evaluation of Safety Analyses for Mixed Fuel Core Design and Increased
Steam Generator Plugging

Three Accidents Remain: Radioactive Release from a Subsystem or Component, Fuel Handling Accident, and Loss of Spent Fuel Cooling. the Radioactive Release from a Subsystem or Component and the Fuel Handling

Accident discussion are relatively unchanged and remain bounded by that previously provided in the FSAR. The loss of spent fuel cooling is an addition to the discussion previously provided in the accident analysis (was previously provided in Chapter 9) and the discussion has been expanded relative to its importance to the safe storage of irradiated fuel.

<u>Safety Evaluation Number</u> 93-142

<u>Subject</u>
Chapter 5 of the DSAR, "RadWaste Systems and ALARA"

<u>Summary</u>
Chapter 5 describes the radiation protection features pertinent to the Plant after issuance of the Possession Only License and after 40 weeks since the Plant was shutdown. The limited scope of Chapter 5 now covers the radwaste systems and radiation monitoring equipment needed to support irradiated spent fuel storage. the only remaining safety-related components are the Spent Fuel Pool, liner (including sealing surfaces of the fuel transfer tube), and racks, with all other previously safety-related components now either removed from service or reclassified and maintained as sources of long term makeup water for the Spent Fuel Pool. Charcoal filtration to remove iodine in release paths has been deleted from consideration since the credible release sources have been reduced such that filtration is not required.

<u>Safety Evaluation Number</u> 93-143

<u>Subject</u>
Chapter 4 of the DSAR, "Operation Systems"

<u>Summary</u>
This LDCR is part of the effort to create a Defueled Safety Analysis Report. This LDCR develops the chapter that discusses the operation controls associated with the safe storage of irradiated fuel. Administrative controls such as maintaining isolation of operating equipment from deactivated equipment as well as the monitoring and operation of systems to ensure assumptions of accident analysis are maintained is included.

<u>Safety Evaluation Number</u> 93-144

<u>Subject</u>
Chapter 1 of the DSAR, "Introduction and Summary"

<u>Summary</u>
QA Regulatory Guides are being deleted from the QA Program and will be added to the Defueled Safety Analysis Report. No design, materials, construction standards, or any overall performance of a plant system in manner which could lead to an accident is changing. No procedure for mitigating accidents is being changed. No unreviewed safety question is involved.

<u>Safety Evaluation Number</u> 93-145

<u>Subject</u>
OI-T-73, Revision 1

<u>Summary</u>
Revision 1 to Temporary Operating Instruction (OIT) 73, Draining and Processing the Spray Additive Tank, introduces modifications to systems still described in the FSAR. This modification will be used to add water to the Spray Additive Tank (SAT) in order to flush out the remaining sodium hydroxide.

The Containment Spray System is no longer needed to support plant operations. the modifications necessary to place the SAT in a safe condition are temporary and will be removed upon completion of the evolution. Every effort has been made to ensure the tank is empty of sodium hydroxide in order to prevent a large exothermic reaction before water is introduced, but precautions will be taken to ensure personnel safety and to minimize equipment damage. In addition, the modifications installed by this procedure are temporary and will be removed upon completion of this evolution. A determination was made that no unreviewed safety question is involved.

Safety Evaluation Number 93-146

Subject
Change to PGE-1052 to reflect changes to other licensing documents as a result of the permanently defueled condition

Summary
Topical Report PGE-1052, Quality-Related List Classification Criteria for the Trojan Nuclear Plant, provides criteria used for determination of the quality-related classification of components. LDCR 93-077 will incorporate changes as reflected in the latest revision of the Quality Assurance Program, and changes that reflect the permanently defueled condition of Trojan.

The purpose of the Fire Protection Plan is to prevent the effects of fires from endangering the health and safety of the public and to provide plant property protection. The importance of the function of Fire Protection led the NRC to include specific recommended QA requirements into BTP 9.5-1 and into the subsequent Appendix A. These requirements were added to the Trojan QA Program PGE 8010 in Appendix A. The goal of this section was not to impose an entire Appendix B program as was required for Safety-Related items and activities, but to provide assurance that Fire Protection features would protect Safety-Related items and activities. The new QAProgram requirements address Quality-Related items and activities which properly address the Fire Protection Program.

Safety Evaluation Number 93-150

Subject
Rev. 5 TPP 17-1,

Summary
The proposed change is administrative in nature; it intends to reduce the complexity and level of detail of the procedural controls for the Trojan Corrective Action Program. The Corrective Action Program is applied to both activities and material items at Trojan.

The reason this safety evaluation is required is that the proposed change involves an exception to PGE's position on an NRC Regulatory Guide described in Table 3.5-1 of the DSAR. Specifically, the change involves a reduction in some procedural requirements for Corrective Action Requests (CARs). CARs that are evaluated to not involve a Significant Condition Adverse to Quality, nor a condition that is reportable nor a TTS violation; will no longer require a determination of cause. They will also not require action to preclude recurrence, or follow-up review of the item by Nuclear Oversight. This relaxation in requirements is consistent with the recent major change to the Nuclear Quality Assurance Program (PGE-8010, Rev. 16), but it is not entirely consistent with the requirements of ANSI N45.2.12-1977. Trojan is committed to this standard in the DSAR via our statement of compliance with R.G. 1.144. The standard requires cause determination, preventive action, and followup review for all audit-initiated CARs, regardless of the significance of the problem. Also, due dates for evaluation of audit findings will be controlled by the CAR process rather than an arbitrary requirement of 30 days by the standard. Our intent is that we not apply artificially higher requirements on these findings merely because they are identified in an audit. A determination was made that no unreviewed safety question is involved.