ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8807060382 DOC.DATE: 88/06/27 NOTARIZED: NO DOCKET #
FACIL:STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529
50-300 Easton Station, Niagara Mohawk Power Corp. 05000300
AUTH.NAME AUTHOR AFFILIATION
BRUNT VAN,E.E. Arizona Nuclear Power Project (formerly Arizona Public Serv
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: Annual rept of changes for CY87.

DISTRIBUTION CODE: IE47D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 64
TITLE: 50.59 Annual Report of Changes,Tests or Experiments Made W/out Approv

NOTES:Standardized plant. 05000528
Standardized plant. 05000529
Application withdrawn 8/22/68. 05000300

See Reports

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD5 LA	1	0	PD5 PD	5	5
	LICITRA,E	1	0	DAVIS,M	1	0
INTERNAL:	ACRS	10	10	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	NRR/DLPQ/HFB 10	1	1
	NRR/DOEA/EAB 11	1	1	NRR/DREP/RPB 10	2	2
	NRR/DRIS/SGB 9D	1	1	NUDOCS-ABSTRACT	1	1
	REG FILE 02	1	1	RGN1 FILE 01	1	1
	RGN5 FILE 01	1	1			
EXTERNAL:	LPDR	1	1	NRC PDR	1	1
	NSIC	1	1			
NOTES:		1	1			

R I D S / A D D S

R I D S / A D D S

TOTAL NUMBER OF COPIES REQUIRED: LTTR 33 ENCL 30

Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

161-01142-EEVB/PGN
June 27, 1988

Docket Nos. STN 50-528/529/530

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

References: (1) Letter from J. G. Haynes (ANPP) to NRC dated 6/1/87 (161-00256). Subject: 10CFR50.59 Annual Report for the 1986 Calendar Year.
(2) Letter from J. G. Haynes (ANPP) to J. B. Martin (NRC) dated 11/19/86 (PP39097). Subject: Supplemental 10CFR50.59 Annual Report.

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
10CFR50.59 Annual Report for the 1987 Calendar Year
File: 88-A-056-026; 88-032-404

Attached please find a report of those changes at PVNGS Units 1, 2 and 3 which were made pursuant to the requirements of 10CFR50.59 during the 1987 calendar year. The attached report contains a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each change.

If you have any additional questions on this matter, please contact Mr. A. C. Rogers at (602) 371-4041.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
Executive Vice President
Project Director

EEVB/PGN/dlm
Attachment

cc: J. B. Martin (all w/a)
T. J. Polich
G. W. Knighton
M. J. Davis
A. C. Gehr

IE47
1/1

8807060382 880627
PDR ADOCK 05000528
R PDC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION V
1450 MARIA LANE, SUITE 210
WALNUT CREEK, CALIFORNIA 94596

June 24, 1988

Docket Nos. 50-528, 50-529 and 50-530

Arizona Nuclear Power Project
P.O. Box 52034
Phoenix, Arizona 85072-2034

Attention: Mr. E. E. Van Brunt, Jr.
Executive Vice President

566
RBN

Gentlemen:

SUBJECT: COMMENT ON ANPP AUXILIARY FEEDWATER SYSTEM DESIGN BASES MANUAL

By letter, dated March 10, 1988, you supplied for review a copy of the ANPP Design Bases Manual section on the Auxiliary Feedwater System and the associated sections from the Design Criteria Manual and System Descriptions Manual. My staff and members of the NRR staff have reviewed these documents.

First, a few general comments are in order. Your Detailed Design Criteria and System Description are heavily oriented toward the mechanical engineering aspects, and even so, do not provide any real detail regarding the means employed by the designers to implement, in the plant features, the very general design criteria. In addition, we note that these contain little specific information regarding electrical and instrumentation and control system features implemented by the designers to effect the general design criteria. These situations seem to make these documents useful mainly as a reference to other documents, of limited usefulness to engineers performing modification design, and little usefulness to engineers engaged in monitoring and maintaining system performance in conformance to the intent of the designer.

I am providing for your information, by attachment, a copy of the Portland General Electric Company Design Bases Document for the Trojan Nuclear Power Plant Auxiliary Feedwater System. The scope and depth of this document were arrived at after much discussion between PGE and NRC. We feel this represents a creditable effort to capture the system design bases and provides a substantial reference for engineers engaged in designing system modifications or monitoring and maintaining system performance.

I would urge you to: critically compare your concept of a Design Bases Manual to the attached PGE concept, other industry benchmarks, and the products of other utilities engaged in similar efforts; and come to an ANPP conclusion regarding the desired scope and depth of the ANPP Design Bases Manual.

DESIGNATED ORIGINAL

Certified By [signature]

IE-01

RECEIVED
NRC
REGION V

I anticipate further discussion of this issue during one of our future periodic management meetings.

88 JUL 1 P1:06

John B. Martin
Regional Administrator

Attachment:
As stated

cc: J. G. Haynes, Vice President
W. F. Quinn, Director, Nuclear Safety and Licensing
R. Papworth, Director, Quality Assurance
T. D. Shriver, Manager, Compliance

bcc w/enclosures:
RSB/Document Control Desk (RIDS)
Resident Inspector

CONCURRENCES:

RPate/nw 6/23/88	DKirsch 6/23/88	JCrews 6/23/88	JMartin 6/24/88	
REQUEST COPY YES / NO	REQUEST COPY YES / NO	REQUEST COPY YES / NO	REQUEST COPY YES / NO	REQUEST COPY YES / NO
				SEND TO PDR YES / NO

DESIGNATED ORIGINAL

Certified By

IE-01

[illegible]

ATTACHMENT

This attachment contains a listing of the changes, tests, and experiments which were made pursuant to the requirements of 10CFR50.59 during the 1987 calendar year at Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3.

A brief description of each change and a summary of the safety evaluation for each change is presented below. The listed changes did not: (1) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or (ii) create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report; or (iii) reduce the margin of safety as defined in the basis for any technical specification.

(1) Description

The purpose of this change was to update and incorporate item II.F.2 of the LLIR into Chapter 18 of the FSAR. This item concerns instrumentation for detection of inadequate core cooling and provides the PVNGS response to reflect current plant status. This change was provided to the NRC in Amendment 17 to the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change reflects plant design already considered in the accident analyses and proposes no new structural equipment, and is made consistent with the guidance of NUREG-0737.

(2) Description

The purpose of this change was to update and incorporate item II.K.1.5 of the LLIR into Chapter 18 of the FSAR. This item concerns review of engineered safety feature valves and is revised to reflect commitments that have been satisfied and a change that was previously approved by the NRC in SSER 7. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change was largely administrative in nature and reflects a change previously approved by the NRC. The change was consistent with the guidance of NUREG-0737.

(3) Description

The purpose of this change was to update and incorporate item III.A.1.2 of the LLIR into Chapter 18 of the FSAR. This item concerns updating of the emergency support facilities and is revised to reflect the fact that certain commitments have been satisfied. This change was transmitted to the NRC in Amendment 17 to the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change is largely administrative in nature, and reflects the fact that certain commitments have been satisfied. The change is consistent with the guidance of NUREG-0737.

(4) Description

This change revised Sections 1.8, 6.2.4.2.1, 9.3.4.1, Figure 6.2.4-1 and Table 6.2.4-2 of the FSAR to reflect the fact that the power supply for containment isolation valve CHA-HV-524 was removed during normal operations. This valve is the charging line isolation valve outside containment and does not close on CIAS. The power was removed to prevent inadvertent operation of the valve. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The valve is normally open and remains open during accident conditions. In addition, the valve is not taken credit for in any accident analysis. The change ensures that the valve will not inadvertently close and that charging and/or auxiliary spray flow will be available when required.

(5) Description

FSAR Sections 8.3.1.1.6, Figure 8.3-4 and 14B.8 were revised to reflect the replacement of the manual transfer switch in the inverters with an automatic static transfer switch. The inverters are part of the 120V Class 1E vital instrumentation and control power supplies. This change was transmitted to the NRC in Amendment 17 to the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. This change will prevent the temporary loss of power to the vital instrumentation and control system loads (i.e., reactor protection and ESF Actuation System) resulting from the time it takes to manually transfer to the alternate power source supplied by the 480VAC motor control center. The safety analyses assume the loss of one redundant set of onsite AC and DC power sources, which the 120V vital AC bus is a part of. Also, the transfer switch is not credited in the analyses. If the automatic transfer switch fails, the 120V Vital AC bus is declared inoperable and the action requirements of TS 3.8.3 are applicable.

(6) Description

FSAR Table 11.4.3 was changed to reflect the as built capacity of the resin transfer/dewatering pump, which is 90 gal/min. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The Spent Resin Transfer Design Guide was reviewed and the pump design determined acceptable for the system. In addition, this pump is not a safety related piece of equipment.

(7). Description

A total of thirty (30) changes to the FSAR were identified that were editorial in nature. These changes involved one of four actions: 1) correction of typographical errors, 2) clarification of the description regarding the as-designed/as-built status/function of the plant, 3) correction of inconsistencies within the FSAR or between the FSAR and other design basis documents, and 4) relocation of information within the FSAR. Three of these changes were transmitted to the NRC in Amendment 17 and the remaining twenty-seven were transmitted in the USAR, Rev. 0.

Summary of Safety Evaluation

These changes did not introduce an unreviewed safety question. They were editorial in nature and had no impact on the design or operation of the plant.

(8) Description

FSAR Section 9.3.3.2.1.4 was revised to reflect the addition of a package sewage treatment plant and the airlift pump with associated pipe and fittings. This change brought the number of package sewage treatment plants to three. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The equipment is not safety related and has no impact on any safety related equipment.

(9) Description

This change amended FSAR Table 11.2-1 to reflect the replacement of the anti-foam pump in the liquid radwaste system with a pump having slightly different operating characteristics. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The pump is not safety related and is of a comparable and improved design for the pump it is replacing.

(10) Description

This change deletes the reference in FSAR Section 11.5.2.1.14 to a radiation monitoring system (RMS) channel failure signal causing a BOP/ESFAS. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. No changes have been made to the RMS or ESFAS which would prevent either system from performing its intended function. A RMS failure will still cause a control room alarm, and the operators will determine whether the ESFAS should be manually activated or if other actions are to be taken.

(11) Description

This change amends FSAR Section 9.3.4 and Table 9.2-8 regarding the pressurizer sample cooler and the gas stripper. The change revises the cooling water requirement for the pressurizer surge sample line from 26 gpm to 30 gpm max/14 gpm min, and for the gas stripper from 700 gpm to 500 gpm. In addition, the operating pressure in the reactor drain tank is 0.5 psig. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change made the FSAR consistent with Combustion Engineering interface requirements which were instituted to assure the proper functioning of the equipment.

(12) Description

This change revised Figure 1.2-5 of FSAR Section 1 to reflect that the Radwaste Compactor (#40) was moved to the plant SE corner of the 55 gallon drum storage area. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The equipment is now in a contained space instead of in an open area, and the new location provides essentially the same orientation to HVAC and fire protection as the old location. In addition, exposure to equipment important to safety is also essentially the same.

(13) Description

FSAR Section 1.8, 3.7 and Table 5.2-2 were revised to reflect the approval of ASME Code Case N-411 by the NRC and its acceptance in Reg. Guide 1.84. This change was transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

The use of this code case by ANPP does not introduce an unreviewed safety question since the code case has been approved for use by the NRC and has been accepted in Reg. Guide 1.84.

(14) Description

The purpose of this change was to modify the QA program described in FSAR Sections 17.1A and 17.2 to incorporate administrative control and organization changes. These changes did not reduce licensing commitments and were transmitted to the NRC in Amendment 17 of the FSAR.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change involves only the description of the QA program, and does not impact important to safety equipment.

(15) Description

FSAR Tables 8.3-3 and 9.1-1 were revised to reflect the correct data for the fuel pool cooling pumps. The changes reflect actual pump data from vendor information and nameplate inspections. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The fuel pool cooling pumps still deliver their rated flow and the increased load of the pumps does not significantly reduce the diesel generator margin. The operation of the pumps is not affected.

(16) Description

FSAR Section 11.5.2.1.1.7.2 and Table 11.5-1 were revised to reflect the replacement of the moving filter mechanisms of radiation monitors J-SQN-RE-8, 14, 13A and 13B with fixed filter mechanisms. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The monitors were modified due to deficiencies in the original design, and the modification simplified operation and maintenance of the monitors. In addition, the monitors are non-safety related equipment.

(17) Description

This change revised FSAR Section 2.2, 18.III.D.3.4 and Table 6.4-3 to note that the PVNGS control room does not have chlorine detectors and that CRVIAS can only be initiated manually. In addition, the reference that PVNGS has a Type B control room per Reg. Guide 1.78 was deleted. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change reflects current design and has no impact on any FSAR accident analyses.

(18) Description

FSAR Sections 3.2 and 12.5.2.2.7 were revised to reflect the replacement of the Radiation Exposure and Management (REM) system by the Radiological Records and Access Control (RRAC) System. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The REM/RRAC system is not important to safety and operation of the system will not impact any safety related system. The REM/RRAC system is also not taken credit for in any safety analyses.

(19) Description

FSAR Section 8.3.1.4.1.1 is revised to take exception to the separation requirements of Reg. Guide 1.75 for the new fuel handling crane and containment refueling machine. Similar exceptions have been accepted by the NRC as documented in SER Supplement 7. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change documents the as-built condition of the plant. The cranes are used infrequently and operate only during short periods. When not in use, they are deenergized. Since the cranes are locally controlled, any malfunction would be detected by the operator and appropriate action initiated.

(20) Description

FSAR Section 9.5.4.2.1 was revised to provide additional clarification regarding the diesel fuel oil storage tank capacity, to eliminate continued confusion relating to the volume provided for periodic testing. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Clarification was provided regarding the diesel fuel oil storage tank capacity, but the system or its mode of operation was not altered in any way.

(21) Description

FSAR Section 5.1.5.H.4 was revised to change the required feedwater temperature accuracy from ±1°F to ±5.5°F. This was necessary because the existing temperature elements for measuring feedwater temperature did not meet the ±1°F accuracy. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The analysis for the uncertainty in calculating secondary calorimetric power assumes a feedwater temperature accuracy of ±5.5°F.

(22) Description

The pressure requirement for the auxiliary building below the 100' elevation was changed from 1/4" wg to "measurable negative" pressure. This required a change to FSAR Section 9.4.2.2. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Requiring a measurable negative pressure still ensures that the auxiliary building below 100' elevation will be maintained negative to ambient in case of a LOCA.

(23) Description

FSAR Section 1.8 was revised to update ANSI N509 (referenced by Reg. Guide 1.52) from the 1976 revision to the 1980 revision. Reg. Guide 1.52 concerns design, testing and maintenance of post accident air cleanup equiment. The NRC has accepted ANSI N509-1980 in the Standard Review Plan, but has not yet revised Reg. Guide 1.52 to reflect the acceptance. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The charcoal filters affected by this change must still meet the requirements consistent with the assumptions made by the safety analysis.

(24) Description

FSAR Section 11.5.2.1.3.13 was revised to document the replacement of the RU-1 discharge solenoid isolation valve with a manual isolation valve. The modification allows RU-1 to be isolated more conveniently and will ensure that maintenance on RU-1 will not release containment atmosphere into the Auxiliary Building. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The design basis accident applicable for RU-1 is a containment overpressurization. RU-1 is not taken credit for in the accident analysis since the CIAS valves perform the safety related function in this situation.

(25) Description

FSAR Section 6.4.5, 6.4.7, 6.5.1.4, and Table 9A-1 were revised to reflect the approval by the NRC of the 1980 revision to ANSI N509, concerning charcoal filters. Reg. Guide 1.52 has not yet been revised and still references ANSI N509-1976. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The inservice test requirements for the charcoal are not changed and the decontamination efficiencies required by the FSAR will still be met.

(26) Description

FSAR Table 8.3-6 describes Class 1E DC System Loads, however, the format of the FSAR table is not consistent with the format of the Bechtel Calculation that identifies the data used in the table. Therefore, FSAR Table 8.3-6 was revised to make the format consistent with the calculation, and to incorporate results that were originally omitted. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes were editorial in nature and did not impact the operation or functional requirements of the Class 1E DC power system.

(27) Description

FSAR Sections 1.8, 5.2 and 11.5 were revised to document the correct location of the RU-1 sample point, ANPP's position on Reg. Guide 1.45 (taking exception to the capability of RU-1 to quantify leakage) and ANPP's methods for leakage detection. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. RU-1 will alarm on abnormally high radiation levels and the control room operators can respond appropriately to increasing trends. The function of the detection system remains the same, and other methods exist to quantify the leakage.

(28) Description

FSAR Sections 9.5, 9B.2, and the response to question 9A.106 were revised to reflect ANPP's exception to the gap requirements and latch throw requirements of a U.L. tested fire door. FSAR Section 9.5 requires that all fire doors conform to the U.L. tested configuration. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The existing deviations do not change the ratings of the doors or the capability of the door to perform its intended function.

(29) Description

FSAR Sections 1.9.2.4 and 9.1.4 were revised to change the containment polar crane drop limit above the reactor vessel flange from 17 feet to 18 feet. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The original safety analyses were performed based on an 18 foot drop limit.

(30) Description

FSAR Section 9B.2 was revised to change the fire rating of doors A-134, A-135, and A-228 from Class "A" (3 hour rated) to Class "B" (1-1/2 hour rated). The fire barriers between the zones are 1 hour rated and only a 1-1/2 hour rating is necessary for the doors, per the requirements of 10CFR50 Appendix R. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. A Class "B" door exceeds the requirements of a 1 hour barrier. The rating of the fire barrier is not affected by this change.

(31) Description

The response to FSAR question 9A.106 was revised to document that Control Room Security doors J303 and J319 are not U.L. labeled. The doors are not labeled since the latch throw is 1/2 inch, while NFPA-80, 1975 requires a 3/4 inch latch throw. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The doors are capable of performing their intended function with the existing deviation, because the fireloading in the zones adjacent to the control room are low.

(32) Description

FSAR Section 1.8 was revised with respect to Reg. Guide 1.52 to allow the use of silicone type sealants for sealing electrical and piping penetrations in the fuel building HVAC (HF) and control building HVAC (HJ) essential air filtration units. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The use of the silicone sealant will not affect the ability of the HF and HJ systems to perform their safety related function. Any degradation of the sealant would not result in a leakage path for contamination into the control room enevelope.

(33) Description

The range of the main steam support structure effluent monitors described in FSAR Table 11.5-1 is changed from 10^{-3} to 10^{4} R/hr to 10° to 10^{5} mR/h. This is consistent with Reg. Guide 1.97 requirements. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change documents the as-built condition of the monitors, which still meet the Reg. Guide 1.97 requirements.

(34) Description

FSAR Section 1.1 is revised to reflect the transfer of 5.7% of undivided ownership from Salt River Project (SRP) to the Los Angeles Department of Water and Power. Pursuant to an agreement dated 8/18/77, this ownership was transferred when PVNGS Unit 1 was placed in commercial service. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The ownership does not impact the operation of the plant or the function of safety related equipment.

(35) Description

FSAR Table 11.5-1 was revised as follows: The listing of the gas charcoal alarm setpoint values for monitors RU-12, RU-141, RU-142, RU-143, RU-144, RU-145, and RU-146 was deleted and replaced by a footnote referencing the tech spec. The tech spec requires that the setpoints be determined in accordance with the Offsite Dose Calculation Manual (ODCM) and could change periodically as the ODCM is revised. In

addition, the particulate and iodine channel monitors listing for RU-142, RU-143, RU-144 and RU-146 alarm setpoint values was deleted and replaced by a footnote referencing ALARA limits. The values for these monitors are defined by ALARA limits, which change due to varying radiation levels in the monitor areas. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes makes the FSAR consistent with tech spec requirements and ALARA considerations.

(36) Description

The Onsite Power System descriptions of FSAR Section 8 were revised to reflect as built conditions. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes do not alter any important to safety equipment or assumptions considered in the FSAR accident analyses.

(37) Description

FSAR Section 9A was revised to document ANPP's exception to Reg. Guide 1.140, allowing the use of silicone sealant on containment preaccess filter units. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The pre-access filter units provide no safety related function and their failure or decreased performance cannot initiate an accident or result in the release of an offsite dose.

(38) Description

FSAR Section 6.4.4.2 was revised to accurately reflect sources of chlorine gas onsite. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. For the most conservative chlorine gas release, the maximum concentration of chlorine gas at the control room air intake structure is below the toxicity limits specified in Reg. Guide 1.78. The source of the chlorine is the electrolytic cells in the water reclamation facility and as such no safety related systems are impacted.

(39) Description

FSAR Section 9.2.3.1.2.D and 1.9.2.4.21 were revised relative to the discussion of the water quality specifications for the demineralized water system. The change clarifies that the FSAR water quality specifications are design values and not operating specifications, and takes exception to the CESSAR water quality specifications for the demineralized water system. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The water limits for the RCS, steam generators and feedwater systems are unchanged. In addition, the operation of the demineralized water system is not changed and the quality of the secondary coolant is not reduced.

(40) Description

FSAR Table 8.3-3 was revised to reflect the maximum diesel generator load during pump runout conditions per revision to the calculation. In addition, FSAR Table 8.3-1 was revised to be consistent with Table 8.3-3. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change verifies that the increased load due to a pump runout will not exceed the diesel generator rating. The diesel generator is still capable of delivering its rated output without being overloaded.

(41) Description

This change revised FSAR Table 9A-1 and FSAR Section 14B.41, paragraph 3.4, to note that in-place leak testing for HEPA filters per position C.5.c of Reg. Guide 1.140 will not be performed for the containment preaccess units. Visual inspection will be performed periodically and at acceptance. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The preaccess AFUS provide no safety related function. The units are 100% recirculation, located in the containment building. No off-site radioactive release could result from leakage through a HEPA bank.

(42) Description

FSAR Sections 9.5 and 11.4 were revised to reflect the addition of the Dry Active Waste Processing and Storage (DAWPS) Facility. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The DAWPS facility is an independent structure, remotely located, with no interaction with safety related equipment. In addition, the functions carried out at the DAWPS facility are not required for the safe shutdown of the PVNGS unit. The change does affect an important to safety system, the fire protection (FP) system. The FP water for the DAWPS facility is supplied by a branch from the main FP piping for the plant and all FP piping has been designed in accordance with all applicable NFPA requirements. This modification has also been evaluated for compliance with the applicable NRC regulations in accordance with IE Circular 80-18, "10CFR50.59 Safety Evaluations for Changes to Radioactive Waste Treatment Systems," and Generic Letter 81-38, "Storage of Low-Level Radioactive Waste at Power Reactor Sites".

(43) Description

This change revises FSAR Table 6.2.5-1 to reflect the as-built design parameters of the containment hydrogen control system. The correct hydrogen analyzer scale is 0-10% hydrogen, and the correct accuracy is ±3.0% full scale. In addition, FSAR Table 7.5-1 was modified to reflect as-built system capabilities and to be consistent with FSAR Table 6.2.5-1. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The hydrogen analyzer retains its safety related seismic category I design to preclude failure of the analyzer function or other important to safety equipment. The analyzer is redundant so that a single failure will not remove hydrogen indication. The point at which the recombiners are actuated (3.5%) falls in the 0-10% scale of the analyzer.

(44) Description

This change revises FSAR Section 1.8 to correct the design class for the primary safety relief valve position indicator from Q9E to R9E and the power supply from 1E to non-1E. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change is consistent with the guidance of Reg. Guide 1.97, Rev. 2 and NUREG-0737, II.D.3. The function of the indicator is not changed, which is to provide post accident monitoring.

(45) Description

FSAR Sections 9B.2.11.5 and 9B.2.11.8 were revised to clarify the discussion of the radiant energy shield that the pressurizer auxiliary spray valve (Train A) is provided with. Some sections described it as a "radiant energy shield" and others as a "coating of 1 hour rating". All references were changed to "radiant energy shield" to be consistent. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The requirements of 10CFR50, Appendix R, Section III.G.f are met. The change was a wording change only.

(46) Description

FSAR Section 9.2.1.9 was revised to delete the high conductivity alarm for the spray pond. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Chemistry monitors the concentrations of constituents which plate out, and the concentrations are held within allowable levels to not exceed the assumptions of the design basis analysis. The conductivity is still monitored, only the alarm is removed.

(47) Description

FSAR Section 1.9.2.4 was revised to make the following changes to the initial test program for PVNGS Unit 3.

1. The low power physics test program was modified to utilize the CEA exchange method for determining CEA worths, rather than the boration/dilution method used on Units 1 and 2.

2. The variable T_{avg} test performed at the 50% power plateau was modified to consist of a ratio-type measurement, rather than an explicit measurement of both the temperature coefficient and power coefficient. The explicit measurement was performed at 95% and 100% power.

3. The CPC/COLSS verification test was modified to be performed at 0%, 50%, and 100% power, and deleted from the 20% and 80% power plateaus.

4. The Loss of Offsite Power (LOP) test was deleted.

These changes were transmitted to the NRC by letter dated 8/31/87 (161-00474) and in the USAR, Rev. 0.

Summary of Safety Evaluation

1. This change did not introduce an unreviewed safety question. The alternate test was performed within the conditions of the Technical Specifications. The conditions during the test subjected the plant to significantly less abnormal conditions since more shutdown margin was available, less boration and dilution of the RCS was performed, and time required was significantly shorter. In addition, the NRC had previously evaluated and approved the use of the CEA exchange method.

2. This change did not introduce an unreviewed safety question. Plant conditions for the modified test will not be less conservative than the more extensive test and the modified test took significantly less time. Plant operation after the test was totally unaffected. Based on measurements from Units 1 and 2, the modified test method was shown to provide good results. Unit 3 was shown to respond essentially identically and the MTC was verified to be within the limits assumed in the safety analysis.

3. This change did not introduce an unreviewed safety question. The change did not require any modification to the facility and does not cause the facility to be operated in a less conservative manner than assumed in the safety analyses. The COLSS and CPC systems are identical on all three PVNGS units and underwent extensive testing at the vendor facility and on Units 1 and 2 during startup testing. The change deleted redundant testing which was deemed unnecessary based on experience during Units 1 and 2 startup testing and performance of the Unit 3 test at 0%, 50%, and 100% power.

4. This change did not introduce an unreviewed safety question. Proper operation of the safety systems and plant design features utilized for response to the LOP test were verified through other tests. The general plant responses and operator control capabilities were successfully demonstrated during the power ascension test programs on Units 1 and 2. Review of Unit 1 and 2 test results revealed no abnormal safety system design/construction configurations that would not be detected by scheduled routine surveillance testing and preventative maintenance. This justified deletion of the LOP test which otherwise would unnecessarily challenge the plant safety systems.

   Footnote: In February of 1988, the NRC informed ANPP that the staff believed the LOP test should be performed for Unit 3. ANPP provided additional information to support not doing the LOP test by letters dated 3/17/88 (161-00887) and 6/13/88 (161-01105) and is awaiting further response from the NRC.

(48) Description

FSAR Section 7.2.1.1.2.3 was revised to clarify the fact that the excore log power channels monitor from $2x10^{-7}$ to 200% power, not

$2x10^{-8}$ to 200% power as stated in CESSAR. The change was made to reflect the actual capability of the monitor. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The operation of the monitor remains the same and the parameters previously evaluated in the FSAR do not change.

(49) Description

This changed revised FSAR Sections 1.9.2.4, 4.5 and 5.2.3 to relax the requirements for flushing the RCS with inhibited water if the duration of the flushing is less than 5 days. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. It takes at least one week to produce fluorine-induced stress corrosion cracking (SCC) in sensitizing weldments. Therefore, if the flushing time with uninhibited water is limited to a maximum of five days, the weldments are not sensitized and the possibility of SCC is precluded. RCS leakage has been previously addressed and is not a concern using this procedure.

(50) Description

The description of the radwaste solidification system in FSAR Sections 11.4.2.3.1 and 11.4.3 was revised to delete the reference to "Portland Cement" as the solidification medium. This was replaced by "cement" to allow the use of any type of cement as a solidification medium that is compatible with PVNGS design and 10CFR61 requirements. In addition, reference is made to the U.S. Gypsum Company Topical Report for their Envirostone product. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The accident analyses in the FSAR make no assumptions concerning the solid radwaste system, and the solid radwaste system has no interface with equipment important to safety. Only the specific make of cement will change, but the PVNGS design criteria and 10CFR61 requirements will continue to be met.

(51) Description

The combustible loading and equivalent fire severity for Zones 73, 74A and 74B described in FSAR Sections 9B.2.12.3F.4, 9B.2.12.4F and 9B.2.12.5F were revised to reflect revisions to the Fire Hazards Analysis calculations for the MSSS. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The combustible loading is decreased considerably, and no design or hardware changes are being made.

(52) Description

FSAR Section 11.4.2.4 was revised to clarify that wet wastes and spent filter cartridges are stabilized (not solidified) in high integrity containers (HICs) which are approved containers for shipping radioactive material. The HICs meet the stabilization requirements of 10CFR61. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The containers meet the requirements of DOT and the NRC (10CFR61) and do not impact safety related equipment or systems.

(53) Description

FSAR Table 11.5-1 was revised to change the reference on the "B" Containment Building Atmosphere monitor and the Movable Airborne Monitor from moving filter to fixed cartridge filter. In addition, a typographical error in the "field concentration" column of the "B" Containment Building Atmosphere monitor was corrected from $1x10^9$ ci/cm^3 to $1x10^{-9}$ ci/cm^3. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The monitors are not taken credit for in any accident analyses. The moving filter has a long history of maintenance problems, and the fixed filter assembly is a simple assembly with no moving parts. The only failure mechanisms the fixed filter assemblies have are inleakage and nonfiltering, which are also associated with the moving filter assemblies.

(54) Description

FSAR Sections 14.A and 15.A were revised to reference CESSAR in the response to several questions. The revisions indicate that the responses were provided on the CESSAR docket. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change was completely editorial in nature and does not affect operation of the plant or safety related equipment or systems.

(55) Description

The description of the Train A safe shutdown raceways for fire zone 10B in FSAR Section 9B.2.1.1.C.6 were changed from a 3 hour rated wrapping to a 1 hour rated wrapping. In addition, the fire severity times were corrected accordingly. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Zone 10B has both fire detectors and an automatic fire suppression system, therefore, only a 1 hour rated wrapping is required per 10CFR50, Appendix R, Section III.G.2.

(56) Description

FSAR Section 12.5.2.2.5 was revised to define the calibration frequency for flow measurement devices used in conjunction with portable air sampling instruments and equipment. This information was previously not discussed in the FSAR. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. This change provides information not previously discussed in the FSAR. The portable air sampling equipment has no impact on accident analyses or equipment important to safety.

(57) Description

The FSAR Section 13.1.1.2.3 description of the activities of the Nuclear Construction group was updated to reflect the responsibilities of the group once the construction phase is complete. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change will not affect the ability of the plant to operate safely or the operation of equipment important to safety. Safety related activities still fall under the QA programs, and construction will be done with appropriate procedures.

(58) Description

FSAR Sections 9B.2.12 and 9B.2.15 were revised to add deviations for fire zones 73, 37B, 37D, 39B, 74A, and 74B to reflect the actual thermolag installation in those fire zones. The fire barriers are in untested configurations. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. This change allows a deviation to 10CFR50, Appendix R, Section III.G.2 for the configuration; however, the configuration provides the equivalent protection required by III.G.2.

(59) Description

FSAR Section 13.1.1.2.2.2 was revised to delete emergency planning from the responsibilities of the Technical Services Group. The Assistant Vice President, Nuclear Production is responsible for emergency planning, and the removal of the responsibility from Technical Services was overlooked when the responsibility description for the Assistant Vice President, Nuclear Production was changed in Amendment 17. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The emergency planning group still exists and their function has not changed. The reporting was changed to increase the attention given to the group, and this change was administrative, to correctly reflect the reporting chain.

(60) Description

FSAR Section 13.1.1.2.2.3 was revised to reflect the following organizational changes within the Nuclear Engineering Department. Onsite liaison was renamed Resident Engineering. Operations Support and Scheduling was deleted. The Projects Supervisor position and the Methods, Training and Compliance group were added to the department. In addition, the Resident Engineering and Projects Supervisor report to the Nuclear Engineering Production Manager. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The Nuclear Engineering Department will continue to provide the same support, fulfill the same responsibilities, and perform the same function as before. The change was made to enhance the performance of the department.

(61) Description

FSAR Section 13.2.1.3.1 was revised to replace the requirement for "lectures" in the modified systems course for non-licensed operators to "training". This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The training consists of a self-study guide in the areas mentioned and classroom training is provided as part of continuing non-licensed operator training. The substitution of training meets the intent of the lectures, and is consistent with all applicable NRC guidance.

(62) Description

FSAR Sections 9.2.2.1.5, Table 9.2-5 and 9.2.2.1.7F were revised to reflect that a nitrite/borate corrosion inhibitor, not a chromate inhibitor, is used for the Essential Cooling Water System (ECWS). This resulted in increasing the allowable halogen limit to account for the different inhibitor. In addition, the change allowed the use of the inhibition treatment chemical for pH control instead of potassium hydroxide. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. There are no accidents previously analyzed in the FSAR which could be caused by a failure of the ECWS. The design basis of the system requires that a single failure of any component in the ECWS will not impair the ability of the ECWS to meet its functional requirements. There are two physically separate, full-capacity ECWS trains, and this change does not affect their number, independence, separation or capacity. The use of the nitrate/borate based inhibitor is in accordance with the vendor's recommendation.

(63) Description

FSAR Table 6.2.4-2 was revised to change the notation that the atmospheric dump valves (ADVs) will be closed post accident, to the ADVs being open or closed post accident. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Numerous FSAR accident analyses assume that the ADVs will be used following an accident for heat removal. This change provides consistency throughout the FSAR.

(64) Description

FSAR Table 6.2.4-1 was revised to indicate that Type C leakage testing is not required for valves SIA-PSV151, SIA-UV708, and SIB-PSV140. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Per the PVNGS Technical Specifications (TS) these valves are exempt from Type C testing because they are below the minimum submergence level and are not required for post accident operation. The operation or purpose of the valves does not change. The change results in consistency between the FSAR and TS.

(65) Description

This changed revised FSAR Chapter 4 to incorporate NRC questions and ANPP responses into the text. In addition, the statement that the initial signature traces for the Loose Parts Monitoring System (LPMS) will be taken during the pre-core hot functional test was revised to reflect that the signature traces were actually taken during post-core hot functional testing. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The LPMS will still operate in accordance with technical specifications. Data obtained without the core in place would be meaningless, therefore, the data obtained during post-core hot functional testing will provide the required signature traces.

(66) Description

FSAR Appendix 10A was revised to incorporate, either by reference or inclusion, responses to the NRC questions contained in Appendix 10A. The changes are editorial in nature. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes were editorial in nature, and had no effect on plant operation or equipment and systems important to safety.

(67) Description

FSAR Section 16 was revised to identify the specific documents which contain the technical specifications for each PVNGS unit. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes were editorial and did not affect the operation of the plant or equipment and systems important to safety.

(68) Description

FSAR Section 3.3 was revised to note that the refueling water tank (RWT), a Seismic Category I structure, is designed for tornado effects. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The RWT was designed for tornado effects, and the operation of the tank will not change.

(69) Description

FSAR Section 1.8 was revised to allow the diesel generator governor oil cooler to be cooled with the jacket water cooling system. The same change was made to FSAR Section 9.5.5.2 and submitted to the NRC in Amendment 17 to the FSAR. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change was administrative, to provide consistency throughout the FSAR. The function and operation of the system has not changed.

(70) Description

FSAR Table 6.2.4-1 was revised to reflect that the outside flange on containment penetration 58 does not require Type B testing. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The function and operation of the flange remains the same. Containment integrity is based on the double gasketed inner flange, which must meet the acceptance criteria for Type A testing.

(71) Description

The response to FSAR question 15A.59 was augmented to explicitly note that the operator action to keep the steam generator tubes covered in conjunction with a stuck open atmospheric dump valve (ADV) results in a RCS cooldown rate that exceeds technical specification requirements during a Steam Generator Tube Rupture (SGTR) with a stuck open ADV. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The SGTR event with a stuck open ADV has been evaluated in the existing FSAR

safety analysis, and this change further explains the results. The operation or design of plant equipment does not change, and the change does not alter the assumptions used in the safety analyses.

(72) Description

This change revised the maintenance program for the switchyard, described in FSAR Section 8A.5, to reflect current practice. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The maintenance program was updated to be less prescriptive, but allowing flexibility to meet the needs of the switchyard. Maintenance testing and inspection will still be performed. The requirements of 10CFR50, Appendix A, GDC 18 and the surveillance requirements of the technical specifications will continue to be met.

(73) Description

FSAR Section 9A.68 was revised to delineate requirements for fire barrier penetration seals that cannot be installed in the same manner as tested. The requirements are consistent with the NRC guidance found in Generic Letter 86-10. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The penetration seals will be evaluated to ensure that an equivalent level of protection will be provided, and the barriers will be maintained to prevent the spread of fire. The installations must meet the NRC guidance found in Generic Letter 86-10. In addition, backup measures such as detection and suppression are provided in case of seal failure.

(74) Description

FSAR Section 9.1.3.3.1.1 was revised to reflect the lowering of the spent fuel pool high level alarm PCN-LSHL-003 from elev. 138'-6" to 138'-2". This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Reducing the distance between the normal pool level and the high alarm restricts the fluctuation of the water level to a smaller band, producing earlier warning of potential problems. The function of the alarm has not changed.

(75) Description

This change deleted FSAR Table 3E-2, which provided a list of equipment requiring environmental and seismic qualification. The table was not used by the NRC during the PVNGS review process. This information is contained in other places, such as the equipment qualification (EQ) list. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The information contained in the table was not used by the NRC in their review of the EQ program. The NRC review was based on ANPP's NUREG-0588 report and other associated submittals. In addition, the information is kept current in other sources. Deletion of the table does not affect the operation of any plant equipment or the EQ requirements/specifications applied to any equipment.

(76) Description

FSAR Section 9.3.2 and 18.II.B.3 were revised to reflect the as-built design of the PVNGS Post Accident Sampling System (PASS) and to note the differences between the Unit 1 and the Units 2 and 3 configuration. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The PASS is not required for safe operation or shutdown of PVNGS, nor does it interface with important to safety components.

(77) Description

FSAR Sections 9.2.4 and Table 9.2-13 were updated to reflect modifications to the domestic water system. The changes included replacement of 4 inch reverse osmosis vessels and membranes with 8 inch vessels and membranes, replacement of the cleaning and other miscellaneous piping, valves, and fittings, and updating the piping material and filter cartridge size. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The malfunction or failure of the domestic water system has no impact on safety related or important to safety systems or components. The domestic water system serves no safety function and is not taken credit for in any previously evaluated FSAR accident analyses.

(78) Description

FSAR Section 17.1B was revised to reflect that ANPP would retain and maintain original design documentation, as opposed to it being part of Bechtel's Quality Assurance (QA) program. The documentation will be maintained in accordance with the ANPP QA operations program for document control and retention. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The commitments of record retention continue to be met, just by a different organization. The documentation continues to be maintained and is retrievable for reference when making design decisions.

(79) Description

FSAR Section 8.3.1.1.4.6 was revised to clarify the description of the load shedding circuits. In addition, Table 8.3-4 was updated to reflect the 120VAC Vital Power System Loads. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The additions to the 120VAC Vital Power System Loads ensure that the operators have the necessary instrumentation to respond to an accident condition. The change to the load shed description clarifies that the primary and secondary breakers remain closed on a load shed signal to avoid overloading the breakers. The changes are made to enhance equipment performance and/or availability and do not impact the ability of the equipment to perform its intended function.

(80) Description

FSAR Section 17.2.6, on Quality Assurance document control, was revised to clarify that each department is responsible for the preparation, review, approval and distribution of its own departmental directives, guidelines, and instructions. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change was administrative, to clarify departmental responsibilities. It did not result in a reduction of QA commitments.

(81) Description

FSAR Table 7A-3 was revised to reflect that the Auxiliary Feedwater Actuation Signal (AFAS) logic was changed to remove the "AFAS-1

priority over AFAS-2" feature of the auxiliary feedwater pump turbine steam supply valve. In addition, the manual override was deleted. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change will not impact the containment post accident equipment qualification temperature. The purpose was to minimize operator error and improve operational flexibility.

(82) Description

FSAR Section 9.2.8.5 was revised to reflect that calcium analyzer #AIT-075 was abandoned in place, and that chlorine detection instruments (calcium appears in fixed proportion to chlorine and is used to measure chlorine) are no longer provided to detect inleakage from the plant cooling water system (PCWS) to the turbine cooling water system (TCWS). This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The TCWS is not an important to safety system and is not credited in any FSAR accident analyses. A malfunction of the TCWS would not affect any system or component important to safety. A grab sample point downstream of the TCWS pumps provides for periodic water analysis.

(83) Description

FSAR Sections 12.5.2 and 12.5.3 were revised to clarify information regarding radiation protection activities and accurately reflect dosimetry calibration and reading intervals. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. No changes are made to equipment or systems important to safety and procedures for safe shutdown or accident mitigation are not affected. The calibration checks for the radiation monitors are governed by technical specification requirements.

(84) Description

FSAR Section 8.3.1.1.6 was revised to delete the requirement to replace the manual transfer switch for the class inverters with static transfer switches prior to restart from the first refueling outage. This change was submitted to the NRC by letter dated 10/15/87 (161-00589), and also in the USAR, Rev. 0.

Summary of Safety Evaluation

The change did not introduce an unreviewed safety question. The change did not alter the design of Unit 1 or assumptions made in the FSAR safety analyses. The current manual transfer switches comply with all regulatory requirements.

(85) Description

FSAR Sections 11.5.2, 12.5.2 and Table 9.3-3 were revised to clarify the description of radiation monitoring equipment, location, operation, and calibration criteria. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes can improve the ability of Radiation Protection personnel to respond to and mitigate the consequences of an accident.

(86) Description

FSAR Section 7.4.1.1 was revised to reflect changes to the undervoltage protection system for the 4.16kV ESF bus. In response to an NRC requirement in the PVNGS SER, an additional protective trip was installed at approximately 90% of nominal bus voltage, with a time delay to avoid spurious trips due to short duration transients. This was in addition to the 70% voltage setpoint used to detect loss of offsite power. Also, FSAR Section 7.4.1.1 was clarified to accurately describe the permissive signal to the diesel generator automatic starting sequence. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Because offsite power will be tripped and the diesel generator started on a higher undervoltage, potential damage to equipment due to a sustained undervoltage will be prevented. The diesel generator will not be impeded from starting during a loss of offsite power, nor have the trip setpoints changed. This change was found acceptable by the NRC in PVNGS SER Supplement 5.

(87) Description

FSAR Section 9B was revised to reflect the actual thermolag installation in fire zones 10B, 42B, 42C, 46A, 46B, 46E, 47B, and 52D. This represents a deviation from Section III.G.2 of Appendix R to 10CFR50. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The existing installation is equivalent to that required by Section III.G.2 of Appendix R.

(88) Description

FSAR Section 9.5.1 and Appendices 9A and 9B were revised to reflect the as-built condition of the plant. Combustible loading figures were updated to reflect combustibles added as a result of plant changes and occupancy. In addition, reporting responsibilities and department titles were revised to reflect the existing organization. This change was transmitted to the NRC in USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change does not decrease the fire protection commitments or reduce the level of protection as presently described. The change also does not affect separation criteria or spurious actuation analyses and the existing fire hazards analysis remains valid.

(89) Description

FSAR Tale 11.2-1 was revised to allow the use of 316 stainless steel (SS) in addition to 316L SS for the liquid radwaste (LR) recycle monitor pump LRN-P03. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The 316 SS physical properties are compatible with the LR chemistry and are comparable to 316L SS. The pump is a non-safety related, non-ASME Section 3 pump and its electrical or mechanical function is not affected by the use of a higher carbon content in the SS material.

(90) Description

FSAR Section 8.3 was revised to reflect changes to the Degraded Electrical Power procedure that allow for connecting both ESF buses (S03 and S04) to a single operating diesel generator after a loss of power when one diesel generator is inoperable. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change did not result in any system modification. The effectiveness of operator recovery actions to mitigate a possible core damage event are increased by reducing the amount of time necessary to perform them. In addition, the conditional core damage probability is reduced by approximately a factor of 2.1 when the revised procedure is utilized.

(91) Description

FSAR Sections 6.2.4 and 6.2.6 were revised to modify the description of the containment isolation system to reflect as-built conditions. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. All changes were in accordance with 10CFR50 Appendix A. Testing per 10CFR50 Appendix J will still occur.

(92) Description

Question 7A.4 of FSAR Appendix 7A was revised to describe the impact of loss of power to panels E-NNN-D11 and E-NNN-D12 on pressurizer heater operation depending on the position of hand switches HS-100 and HS-100-3. Also, the impact on the pressurizer level control and the pressurizer pressure control systems, and the operation of the charging pumps was clarified. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The loss of backup heaters is within the accident analyses in the FSAR. All actions are detailed in existing plant procedures.

(93) Description

FSAR Table 3.2-1 and Section 17A.62 were revised to make the following changes:

1. Add ERFDADS to Table 3.2-1 (Quality Classification of Structures, Systems and Components).

2. Add accident monitoring instrumentation to Table 3.2-1.

3. Modify response to questions 17A.62 C.1, 2, 4, 5, 11, 12 and 17 to reflect completed design.

4. Modify footnote u in Table 3.2-1 to reflect Reg. Guide 1.97 QA requirements.

This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. No changes are made to the facility or procedures. This change clarifies the QA requirements.

(94) Description

FSAR Sections 9B.2.12 and 9B.2.15 were revised to make editorial changes and update the combustible loading calculation and fire department response time figures, for consistency with the changes reported in item #88 of this report. These changes also clarified the deviations added by item #58. This change was transmitted to the NRC in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The analysis for the deviations is not affected by the minor changes in the combustible loading values and shows that equivalent protection is provided.

(95) Description

FSAR Section 1.9.2.4 was revised to note an exception to CESSAR Section 14.2.12.1.15. The change revised the preoperational test method for the holdup subsystem test to provide greater flexibility in obtaining the required test data. This change was transmitted to the NRC by letter dated 10/8/87 (161-00561) and in the USAR Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The holdup subsystem is a non-safety related portion of the CVCS and is not credited for mitigating the consequences of any FSAR accident analyses. The change does not affect the design or operation of any equipment.

(96) Description

FSAR Section 3.8.1.6.6.1 was revised to delete AWS D1.1-1972, Rev. 1, 1973 (Structural Welding Code) and replace with the acceptance criteria of Nuclear Construction Issues Group (NCIG) document NCIG-01, Rev. 2. This approach is technically acceptable and has been approved by the NRC by letter dated 6/26/85 to the NCIG. This change was submitted to the NRC by ANPP for prior approval by letter dated 9/22/87 (161-00519).

Per 10CFR50.54(a)(3)(iv), this change is regarded as accepted since more than 60 days passed since the submittal with no NRC response. The change was included in the USAR, Rev. 0.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The new acceptance criteria is industry and regulatory approved for safety-related structural applications.

(97) Description

The PPS functional test - RPS/ESFAS logic procedure, which is described in FSAR Section 7.2.2.3.3, was modified to clarify the acceptance criteria for testing matrix logic with an input parameter bypassed. When a parameter is bypassed the test verifies that the bistable relay operates properly and that the not normally closed contacts operate. The normally open contacts are not tested, as there is no testing method or design change practical to allow this. This procedure is applicable to all three units.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The only failure of this portion of the system that could prevent proper actuation is the normally open contacts failing closed. This is analyzed in CESSAR Table 7.2-5. Since only one channel at a time can be bypassed, this failure could not prevent a trip from occurring.

(98) Description

A procedure change was initiated to modify the normal makeup process for Unit 1 to batch boric acid to the refueling pool using the boric acid makeup pumps (BAMPs) when the refueling water tank level is less than 73%. The performance of this procedure involves a jumper installation and is therefore a change to the facility.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The BAMPs are not required in Mode 6 with the refueling canal level less than 23 feet. In addition, the suction header taps off the RWT below its normally expected level when the refueling canal is filled. The RWT level and concentration cannot be significantly reduced by this change.

(99) Description

The dynamic MOVATS testing procedure was modified, which involves clipping test leads onto safety related valve motor operators (auxiliary feedwater discharge valves) for taking test data. This satisfies the requirements of IEB 85-03 and involves a change to the facility for all three units.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Should a single valve fail, the auxiliary feedwater pump (AFWP) would be tripped. Performance of secondary heat removal and plant cooldown with one AFWP inoperable is discussed in the FSAR and is within allowed technical specification limits. The operability of the valves is not affected by the test leads. If the valves must be adjusted as a result of the testing, ASME Section XI testing is performed prior to returning the valves to service.

(100) Description

A new test procedure was written to allow for MOVATS testing for the Safety Injection System. These are tests not described in the FSAR and are applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. All valves being tested will still respond properly to a

safety injection actuation signal. In addition, testing is only performed on one train of the emergency core cooling system at a time, with a fully redundant train intact.

(101) Description

A procedure described in the FSAR was modified to provide instructions for alignment of containment spray pumps to provide shutdown cooling flow. This procedure is applicable to Unit 1.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The flow and discharge pressure capability of a containment spray pump meets or exceeds that of a Low Pressure Safety Injection pump, therefore, this configuration is within the bounds of the safety analyses. In addition, the containment spray pump materials are compatible with RCS fluids. No valve lineups are changed or valves isolated, so the Iodine Removal System remains operable.

(102) Description

A new procedure was written for the implementation of antimony removal in the RCS and interconnecting systems for Unit 3. This involved changes to the facility since N_2 and O_2 rigs will be connected to existing tap-offs and will be removed as part of normal Mode 5 recovery.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. All technical specification limits relating to activity and chemistry will be complied with. Implementation of this procedure will not change the plant operating configuration. Based on evaluations by EPRI, Combustion Engineering, and experience at European plants, this procedure will not pose a safety concern.

(103) Description

A new procedure was written to change the Unit 1 Cycle 1 core to the Cycle 2 configuration. This results in a change to the facility.

Summary of Safety Evaluation

The implementation of this procedure did not introduce an unreviewed safety question. All associated fuel handling equipment is operated in accordance with approved procedures. The procedure does not require the use of any important to safety equipment. Fuel handling accidents inside and outside containment have been previously evaluated in the FSAR, and their probability or consequences will not be increased.

(104) Description

The Unit 3 initial fuel loading procedure was modified to install a modification to the fuel pool cooling system, lower the reactor vessel level requirement to conform to the FSAR, make various administrative changes, and incorporate additional information. The modification to the fuel pool cooling system consists of a temporary elbow added to the pool cooling system cleanup discharge pipe into the refueling canal, redirecting flow downward into the canal, rather than into the side.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The safety related functions of the pool cooling system (cooling as opposed to cleanup) will not be required during the time this modification is in place since no irradiated fuel will be present in the pool. With respect to the reactor vessel level requirement, the core will remain fully moderated, and with shutdown cooling in operation, the RCS boron will remain well mixed.

(105) Description

To support power ascension testing for Unit 3, a procedure was written to perform movable incore amplifier gain adjustment/detector bias voltage determination and demonstrate operability of the Movable Incore Detector System. To permit the use of manual control boxes for positioning the movable detectors in the active core region, the cables in panel 3E-RIN-J02 are temporarily disconnected to permit connection of the interconnect cables between the manual control boxes and the control panel. This constitutes a change to the facility.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. Plant equipment and systems are not compromised or degraded by the test, and will be operated in accordance with approved PVNGS Station Manual procedures during its performance. The procedure is within all technical specification limits.

(106) Description

A new test procedure was written to calibrate the turbine power algorithm in the Core Operating Limit Supervisory System (COLSS) for Unit 3. This is a test not described in the FSAR.

Summary of Safety Evaluation

Implementation of this test procedure did not introduce an unreviewed safety question. COLSS is not an important to safety system, but provides only a monitoring function.

(107) Description

The low power physics test procedure for Unit 3 was revised to reflect that the Plant Monitoring System database values are temporarily modified during the test to allow movement of the CEAs in group mode to the mechanical limits. This involved a change to the facility.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change has no effect on equipment operation, all systems will perform as described in the FSAR, and all assumptions in the safety analyses remain valid.

(108) Description

A new procedure was written to perform a check of the reactivity computer following its installation. This resulted in the uncompensated ion chambers to be disconnected from the startup/control channel drawers to provide input to the reactivity computer. In addition, other non-class plant instrumentation was connected to the reactivity computer system strip chart recorder. These were changes to the facility and are applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The potentially affected systems are not required to perform safety functions.

(109) Description

The procedure for determining Moderator Temperature Coefficient (MTC) was revised. The current procedure uses the conventional test method of measuring core average temperature changes while using CEA movement to maintain power essentially constant, and balancing temperature changes against core power changes. The revised procedure balances transient xenon worth against changes in core average temperature. This involved a change to procedures described in the FSAR, and was applicable to Units 1, 2, and 3.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. All equipment and systems are operated in accordance with approved plant procedures during the test. Technical Specification (TS) Special Test Exception 3/4.10.4 is applicable if the TS 3.1.1.3 limit of 568°F is exceeded during the test. The action statements associated with TS 3/4.10.4 ensure that the margin of safety is not reduced.

(110) Description

The procedure for performing Engineering Evaluation Requests was revised to incorporate an NRC commitment and additional administrative controls. This was a change to a procedure described in the FSAR and was applicable to Units 1, 2, and 3.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Any action resulting from this change will require a separate safety review and evaluation to be performed.

(111) Description

A new procedure was written to perform pre-operational carryover testing of the Train B diesel generator. This procedure requires temporary test connections to be made at the local control panels and a temporary tygon tubing connection to be made at the B fuel oil day tank. This involved a change to the facility and is applicable to Unit 3.

Summary of Safety Evaluation

This change did not result in an unreviewed safety question. The performance of this test is prior to declaring the diesel generator operable, therefore, plant safety is not impacted.

(112) Description

A new procedure was written to monitor the QSPDS-ERFDADS Data Link. This is a test not described in the FSAR and is applicable to Unit 3.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The procedure involves a monitoring test and has no active interfaces with plant components.

(113) Description

A new procedure was written to test valve motor operators to increase reliability and to assist in ensuring operability. This is a test not described in the FSAR and is applicable to Units 1, 2, and 3.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. All testing meets the requirements of IEB 85-03 and the initial startup test program, and is conducted under the work control program.

(114) Description

The procedure to perform ASME Section XI valve stroke time testing was revised to include valves SIA-UV698 and SIB-UV699. This was a change to procedures described in the FSAR, and was applicable to Unit 1.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Adding additional valves to the surveillance program is a prudent measure but has no safety significance.

(115) Description

A new procedure was written to perform an evaporator vibration test for the Unit 2 liquid radwaste system (LRS). This was a test not described in the FSAR. In addition, the procedure requires a change to the facility by attaching temporary strain gauges and accelerometers to the evaporator body.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The LRS is not taken credit for in the safety analyses. The temporary test equipment has no impact on the operation of the evaporator.

(116) Description

A new procedure was written to gather data from the final lineup of the main turbine for Unit 3. This was a test not described in the FSAR.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The turbine control system performs no safety related function and is not required to mitigate the consequences of an accident. The test ensures the proper operation of the turbine control system.

(117) Description

A new procedure was written to perform feedwater control system (FWCS) tuning. Since the procedure allows temporary provisional adjustments to electronic components within the FWCS for dynamic system evaluation, this is a change to the facility applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The FWCS is not considered in any action involved with accident analyses and contains no important to safety equipment.

(118) Description

This procedure change was to add a welding specification for making seal welds between steam generator tubes and tube plugs. This will require a change to the facility and is applicable to all three units.

Summary of Safety Evaluation

This change did not result in an unreviewed safety question. Welded plugs are the best method of plugging. The procedure is done in an approved manner, qualified to ASME Section XI requirements, and recommended by the CE technical manual. The integrity of the welds are determined by pressure testing. Plugging of the tubes results in decreased primary to secondary leakage.

(119) Description

The Bulk Chemical Receipt procedure was cancelled since these requirements cannot be directed by Chemistry procedures. The requirements are contained in Purchasing procedures. This was a change to a procedure described in the FSAR and is applicable to all three units.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The requirements of administrative procedures concerning purchase of bulk chemicals are covered in purchasing documentation. The cancellation has no effect on FSAR accident analyses, or equipment and systems important to safety.

(120) Description

A new procedure was written to perform a postcore hot functional test (HFT) chemistry test for Unit 3. The new procedure deviates from some chemistry specifications described in CESSAR, and therefore is a change to a procedure described in the FSAR.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The limits for fluoride, chloride and oxygen remain unchanged from CESSAR and Tech. Spec. 3/4.4.6 limits. The hydrogen and other specifications in CESSAR are not appropriate for the operating conditions under which this test is performed. The limits in the procedure were adopted from EPRI-SGOG secondary water chemistry guidelines, with concurrence by CE.

(121) Description

A procedure for operation of the Post Accident Sampling System (PASS) was revised to delete boronmeter and collimator, add liquid system sump

flush, update the valve verification list, and update references. This was a change to a procedure described in the FSAR and was applicable to Unit 2.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. PASS is a monitoring system designed to provide information only and is independent of safety related or safe shutdown equipment, systems or components.

(122) Description

The Post Accident Sampling System Surveillance procedure was revised to delete inline analyses. This was a change to a procedure described in the FSAR and was applicable to Unit 2.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. PASS is a monitoring system designed to provide information only and is independent of safety related or safe shutdown equipment, systems or components.

(123) Description

A procedure for operation of the Post Accident Sampling System (PASS) was revised to delete the boronmeter and collimator and update the valve verification list accordingly. In addition, prerequisites were aligned with the associated operating instructions. This was a change to a procedure described in the FSAR and was applicable to Unit 3.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. PASS is a monitoring system designed to provide information only and is independent of safety related or safe shutdown equipment, systems or components.

(124) Description

A change was made to the Post Accident Sampling System (PASS) Functional Test Procedure to delete boronometer, containment radwaste sump, auxiliary building radwaste sump and inline samples. Parallel sample time requirements were changed to one hour. In addition, changes were made to clarify "inoperability", correct typographical errors, and update the format. This was a change to a procedure described in the FSAR and was applicable to Unit 2.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. PASS is a monitoring system designed to provide information only and is independent of safety related or safe shutdown equipment, systems or components.

(125) Description

New procedures were written to perform the reactor makeup water tank (RMWT) and holdup tank surveillance test and the Liquid Radwaste System recycle monitor tank and total dissolved solids tank surveillance test for Unit 3. These are temporary changes to the FSAR statement "There are no provisions or pathways for the release of radioactive liquids to the environment."

Summary of Safety Evaluation

Implementation of these procedures did not introduce an unreviewed safety question. The activity of the RMWT is limited by Tech. Spec. 3.11.1.3 and the maximum inventory of the Refueling Water Tank is described in FSAR 2.4.13.2 and 15.7.3, which ensures that the consequences of an accident would be bounded by the safety analyses. No safety related equipment is involved.

(126) Description

A new procedure was written to allow transfer of contaminated resin from the condensate (CD) system to vendor receiving tanks for shipment offsite for Unit 1. This involves adding a temporary spool piece to the CD system to facilitate resin transfer from the service vessel to the vendor receiving tank, which is a change to the facility.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. A spill of contaminated resin and/or water is bounded by the steam generator tube rupture described in CESSAR. Since the procedure deals with cleanup after a tube leak, the probability of a spill is decreased by removal of the contaminated resin water. There is no safety related equipment involved in the performance of this procedure.

(127) Description

The Liquid Radwaste System (LRS) demineralizer procedure was revised to process LRS evaporator distillate through a portable filter(s). This is a temporary change to the facility with the installation of a portable filter in the LRS and is applicable to Unit 1.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. No important to safety equipment is involved, and there are no liquid

release points in the LRS system. Any potential leakage from any components involved with the change would be minimal and would be collected by the radioactive drain system.

(128) Description

The portable spent resin transfer system procedure was revised to reflect installation of a 3-way valve (SR-HV420) downstream of SR-V967 to allow spent resin to be routed to the truck bay flange connection or to a flange connection in the High-Level Storage Area. This was a change to the facility and is applicable to Unit 1.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The valve is installed inside the High-Level Storage area and any spent resin spill accident would be contained within this area. No important to safety equipment is involved.

(129) Description

A new procedure was written to perform rod shoulder gap measurements during irradiated fuel inspection. This is a test not described in the FSAR and is applicable to all three units.

Summary of Safety Evaluation

The implementation of this procedure did not introduce an unreviewed safety question. Fuel will be handled using normal operating procedures and equipment. The equipment involved is not important to safety. In the event of a fuel assembly falling out of the stand, this accident would be less severe than the fuel handling accident evaluated in the FSAR since the stand will be located in the cask loading pit where other assemblies would not be damaged. All technical specification load limits, water levels and ventilation requirements are adhered to.

(130) Description

A new procedure was written to perform eddy current inspection of CEA guide tubes. This is a test not described in the FSAR and is applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. All fuel movement will be via normal operating procedures. The equipment involved is not important to safety. All technical specification load limits, water levels and ventilation requirements are adhered to.

(131) Description

A new procedure was written for setting up and using the fuel inspection stand in the spent fuel pool, for use during irradiated fuel inspection. This is a test not described in the FSAR and is applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The stand is restricted by interlocks so it cannot be moved over the fuel assemblies. Movement of fuel to and from the stand is done via normal operating procedures. No important to safety equipment is involved. All technical specification load limits, water levels and ventilation requirements are adhered to.

(132) Description

A new procedure was written to perform guide tube length measurements during irradiated fuel inspection. This is a test not described in the FSAR and is applicable to all three units.

Summary of Safety Evaluation

Implementation of this procedure did not introduce an unreviewed safety question. The procedure does not deviate from normal fuel handling procedures. No important to safety equipment is involved. All technical specification load limits, water levels and ventilation requirements are adhered to.

(133) Description

This change involved removing the flow indication and associated valves for the full flow recirculation lines on auxiliary feedwater pumps AFA/AFB/AFN-P01 and capping the end connections. In addition, the change involved removing the blind flanges from the vent and drain lines, capping the end connections, and stiffening the supports on the suction and discharge lines of the auxiliary feedwater pumps. The changes are made to the existing piping to prevent fatigue failure due to excessive vibrations. This change was implemented in Unit 2 during the reporting period and affected drawing 13-M-AFP-001 (FSAR Section 10.4).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The changes do not affect FSAR accident analyses and reduce the exposure of the piping to stress fatigue.

(134) Description

This change added discharge tail piping from the ASN-PSV-334 discharge flange to the floor. This change was implemented in Unit 1 during the reporting period and affected drawing 13-M-ASP-001 (FSAR Section 10.3).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change has no impact on any safety related equipment and eliminates a potential personnel safety hazard of being badly burned by steam blow-off from the open discharge flange on ASN-PSV-334.

(135) Description

This change replaced existing 8 inch globe valve CDN-V064 in the condensate system with a 6 inch throttling valve of the pressure seal type. The change was implemented in all three units during the reporting period and affected drawing 13-M-CDP-001 (FSAR Section 10.4).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The throttling, pressure seal type globe valve installed is consistent with the existing system design of throttling operation, and will eliminate the valve seat damage which occurred with the existing 8 inch valve blowing the packing out. The valve is not safety related and the change will improve system operation and personnel safety.

(136) Description

This change added graduated sight glasses on the diesel fuel oil day tanks by modifying the existing level switch standpipe. This change was implemented in all three units during the reporting period and affected drawing 13-M-DGP-001 (FSAR Section 9.5).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The gauge assembly is maintained shut off by the root valves and is opened to measure tank level only during infrequent calibration. The gauge has no operating parts or movement and provides increased measurement accuracy.

(137) Description

This change added drainage provisions for the gaseous radwaste system (GRS) header, moved the H_2/O_2 analyzer sample line, and rerouted the GRS surge tank drain line. These changes were implemented in Unit 2 during the reporting period. Drawings 13-N-GRP-001 (FSAR Section 11.3), 13-M-RDP-002 (FSAR Section 9.3) and 13-M-RDP-004 (FSAR Section 9.3) were affected by the change.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The only connection to an important to safety component is the connection of the sampling piping to the surge tank, which was done in accordance with

Reg. Guide 1.143. All piping leading to the RD system, as installed by this change, is equipped with two isolation valves in series to preclude the uncontrolled intrusion of radioactive gas to the RD system.

(138) Description

This change provided an interlock between heaters HFA-E01 and HFB-E01 and the fans in the fuel building essential air handling units, to shut down the heaters when the fans stop. This change was implemented in Unit 1 during the reporting period. Drawings 13-E-HFB-004 and 13-E-HFB-006, which are incorporated into the FSAR by reference, were affected by this change.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. This change corrects a potential failure mode for the heaters and prevents damage to the heaters from overtemperature. The heaters do not interact with any other safety systems.

(139) Description

For the fuel building HVAC system, this change installed a new auxiliary relay for the Safety Equipment Inoperable Status (SEIS) (heater power) alarm independent of the AFU fan. In addition, the control push button was made a spare so that heater power is available automatically when the AFU fan starts or stops. This change was implemented in Unit 1 during the reporting period and affected drawing 13-E-HFB-006, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change ensures that the heaters energize when required and that the SEIS alarm correctly indicates a problem, correcting a potential failure mode for the heaters.

(140) Description

This change installed annubar coupling in the hydrogen purge supply line, replaced control valve JHPNPCV0025, and changed the setpoints of instruments at the hydrogen purge exhaust unit to control the purge unit at the design flowrate of 50 scfm. This change was implemented in Units 1 and 2 during the reporting period and affected drawing 13-M-HPP-001 (FSAR Section 6.2.5).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change does not deviate from the original design, but corrects the setpoints and instrumentation to meet the original design.

(141) Description

This change installed a drag valve type flow restriction as a replacement to a flow orifice in the miniflow return line of the non-essential auxiliary feedwater pump (AFN-P01). This change was implemented in Units 1 and 2 during the reporting period and affected drawing 13-M-AFP-001 (FSAR Section 10.4).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The system operates in the same manner as previously evaluated, only the miniflow is increased during normal operation. When the pump is used during an emergency, the miniflow is turned off from the control room.

(142) Description

This change added a depressurization (or vent) mode to the containment power access purge system and added a 10 inch flow regulating valve with an 8 inch bypass line with flow orifice downstream of valve UV-5B. This change was implemented in Unit 2 during this reporting period. Drawings 13-J-ZAF-013, 13-J-ZAF-014, 13-J-CPL-001, 13-E-ZAC-015, 13-E-ZAC-016, 13-E-ZAC-026, and 13-E-ZAC-071, which are incorporated into the FSAR by reference, and drawings 13-M-HCP-001 and 13-M-CPP-001 (FSAR Section 9.4), were affected. In addition, procedures described in the FSAR were changed to allow the system to operate as modified.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The addition of the vent mode is required to ensure that the intended design of the containment power access purge system is met and that no unmonitored backflow will occur. This ensures that 10CFR100 offsite dose limits are not exceeded.

(143) Description

This change added an emergency start pushbutton on the Train A diesel generator local panel to allow a simulated start on loss of power in the event of test mode start failure. This change was implemented in Units 1 and 2 during this reporting period and affected FSAR Section 8.3.1.1.4.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The design of the diesel generator remains the same. The modification already exists on the Train B diesel generator and should enhance the availability of the diesel generators.

(144) Description

This change installed two new radio base stations, battery bank, antenna tower, and associated cabling to improve security radio coverage. This change was implemented in Units 2 and 3 during this reporting period and affected drawing 13-E-NNA-002, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change has no effect on any safety related systems.

(145) Description

The following improvements were made to the security radio system: (1) added a signal quality selection system, (2) removed secondary squelch capability, (3) installed four-wire audio circuit, (4) improved receiver preamplifier specifications, (5) installed a telephone in the main steam support structure, (6) provided human factors improvements for the central and secondary alarm stations, (7) reduced unauthorized access to radio consoles, and (8) extended radio coverage by additional antenna installations. Those changes were implemented in all three units during this reporting period. Drawings 13-E-ZAC-004, 13-E-ZAC-015, 13-E-ZAC-016, 13-E-ZAC-017, and 13-E-ZAC-018, which are incorporated into the FSAR by reference, were affected by this change.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The equipment in question is not important to safety nor does it affect the operation of a safety system. In addition, the system is not identified as a safe shutdown system in the spurious actuation study, and is not used by the operators to safely shutdown the plant. The cable separation study is also not affected. Additions to combustible loadings are negligible and do not appreciably affect the current fire hazards analysis.

(146) Description

This change removed insulation from the pressurizer relief valves and from the common header to the reactor drain tank (RDT). The change was implemented in Unit 2 during this reporting period. Drawings 13-M-CHP-003 (FSAR Section 9.3) and 13-M-RCP-001 (FSAR Section 5.1) were affected.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change was made to enhance pressurizer relief valve reliability in the event of possible infrequent weepage. The line in question has little or no effect on containment heat loads and the insulation has no safety impact.

(147) <u>Description</u>

This change distributed annunciator window #7A10A inputs to windows #7A09A, #7A09B, #7A10A, and #7A11A. The change was implemented in Units 1 and 2 during this reporting period. Drawings 13-J-HCL-001, 13-J-HCL-002, 13-J-HCL-003, and 13-J-HCL-004, which are incorporated into the FSAR by reference, were affected.

<u>Summary of Safety Evaluation</u>

This change did not introduce an unreviewed safety question. The operation and function of the system will remain the same.

(148) <u>Description</u>

This change installed a Typernette 340 printer in place of the alarm typer for the plant computer, and installed the demand typer in a new location. The change was implemented in Unit 3 during this reporting period. Drawings 13-E-ZJC-006, 13-E-ZJC-007 and 13-E-ZJC-009, which are incorporated into the FSAR by reference, were affected.

<u>Summary of Safety Evaluation</u>

This change did not introduce an unreviewed safety question. The change is consistent with the system design bases, and does not increase the failure probability of the Plant Monitoring System. The equipment is not safety related.

(149) <u>Description</u>

For the Blowdown Demineralizer System, the following changes were made: (1) blowdown control valves were modified, (2) conical orifices were installed, (3) the flash tank pressure controller was modified and (4) balancing globe valves and flow indicators were installed on the Blowdown Heat Exchanger cooling water line. These changes were implemented in Unit 3 during this reporting period and affected drawing 13-M-SCP-004, which is incorporated into the FSAR by reference.

<u>Summary of Safety Evaluation</u>

This change did not introduce an unreviewed safety question. The modifications do not change the operation of the system. No safety related components are affected.

(150) <u>Description</u>

This change involved installation of a spectacle flange and test connection on shutdown cooling relief valves. This affected FSAR Figure 6.3-1 and drawing 13-M-SIP-002 (FSAR Section 6.3). The change was implemented in Unit 3 during this reporting period.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The function and design of the valves remains the same.

(151) Description

This change moved the power feeds to the Radiation Monitoring System (RMS) control room cabinets J-SQA/B-C01 and J-SQA/B-C05 from distribution panels D31 and D32 to distribution panels D25 and D26. In addition, larger secondary cables were provided from E-PNA-V25 and E-PNB-V26 to distribution panels D25 and D26. This change was implemented in Unit 3 during this reporting period. Drawings 13-E-PNA-001, 13-E-PNA-002, 13-E-PHA-001, 13-E-PHA-002, 13-E-ZJC-006, 13-E-ZJC-007, 13-E-ZJC-009, 13-E-ZJC-037, 13-E-ZJC-038, 13-E-ZJC-052 and 13-E-ZJC-053, which are incorporated into the FSAR by reference, were affected.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Providing more reliable power to the RMS cabinets will ensure that communications with the PMS microcomputer will not be interrupted. There is no impact on equipment important to safety.

(152) Description

This change added backup power to the Post Accident Radiation Monitors J-SQN-RU-139, 140, 141, 142, 143 and 144 to be fed from Train B, which is diesel backed up. This change was implemented in all three units during this reporting period. FSAR Tables 1.8-1, 8.3-3 and 11.5, and drawings 13-E-ZAC-016 and 13-E-PHA-006, which are incorporated into the FSAR by reference, were affected.

Summary of Safety Evaluation

This change did not introduce an unrevived safety question. The modification does not affect important to safety equipment and will enhance compliance with technical specifications.

(153) Description

This change installed a condensate pot on the drain line from the H_2/O_2 analyzer to prevent leakage of gases into the Radwaste Drain System. The change was implemented in Units 1 and 2 during the reporting period and affected drawing 13-N-SSP-001 (FSAR Section 9.3).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change enhances the operability of the system.

(154) Description

This change modified pipe supports on the steam generator blowdown sampling lines to be consistent with ASME requirements. This change was implemented in Unit 3 during this reporting period and affected drawing 13-M-SGP-002 (FSAR Section 10.3). A related change was made to the FSAR (see item #13) to incorporate the use of ASME Code Case N-411, which was used in the supporting analyses.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The modifications made the pipe supports consistent with ASME code requirements. The sampling lines are not required for safe shutdown of the plant.

(155) Description

This change added heated junction thermocouples and core exit thermocouples per Reg. Guide 1.97, as well as associated cables, raceways, and supports. The change was implemented in all three units during this reporting period. Drawings 13-E-ZAC-065, 13-E-ZAC-066, 13-E-ZCC-012, 13-E-ZCC-015, 13-E-ZCC-016, 13-E-ZCC-030, 13-E-ZCC-041, 13-E-ZCC-042, 13-E-ZCC-053, 13-E-ZCC-056, and 13-E-ZCC-057, which are incorporated into the FSAR by reference, were affected.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The instrumentation was added for post accident monitoring and is not required for safe shutdown of the plant. The addition of the instrumentation will enhance monitoring of in-core conditions.

(156) Description

This change relocated the excore startup channel speaker in containment to approximately midpoint in the refueling pool on the south wall above the 140 foot elevation. The change was implemented in Unit 1 during this reporting period and affected drawing 13-E-ZCC-020, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The excore startup system is not considered in the FSAR accident analyses and is not essential to plant safety.

(157) Description

This change replaced the motors and changed the gear ratios of two motor operated valves (J-SIA-HV-0604 and J-SIB-HV-0609) on the safety injection system. This change was implemented in Units 2 and 3 during

this reporting period. FSAR Table 8.3-1 and drawings 13-E-PHA-003 and 13-E-PHA-004, which are incorporated into the FSAR by reference, were affected.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The basic operation and function of the systems remains the same. The change improves the system reliability. The potential for mechanical binding and loss of electrical power malfunctions is unchanged.

(158) Description

This change provided extended cable length for Ru-27 and Ru-28, and relocated detectors for Ru-37 and Ru-38. The change was implemented in Unit 3 during this reporting period and affected drawing 13-E-ZAC-026, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change results in increased sensitivity and shorter detection time, which increases the margin of safety. Any potential failure of the monitors has been previously evaluated.

(159) Description

This change replaced butterfly valve TV-183 on the turbine cooling water system with a vee-ball valve. The change was implemented in Unit 2 during this reporting period and affected drawing 13-M-TCP-002 (FSAR Section 9.2).

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. The change provides minimum flow and temperature control during periods of low level loads. The turbine cooling water system has no safety function.

(160) Description

This change relocated the trash compactor from the truck bay area to the low level storage area. The change was implemented in Units 1 and 2 during this reporting period and affected FSAR section 11.4.2.4 and Figure 9B.24.

Summary of Safety Evaluation

This change did not introduce an unreviewed safety question. Wet pipe sprinkler protection, as described in the FSAR, will be maintained in the new location.

(161) Description

This modification rewired the limit switches at motor operators for valves 1,2,3J-AFB-UV-0034, 1,2,3J-AFB-UV-0035, 1,2,3J-AFC-UV-0036 and 1,2,3J-AFA-UV-0037, to permit limit close operation. The change was implemented in all three units during this reporting period. Drawings 13-E-AFB-005, 13-E-AFB-010, and 13-E-AFB-011, which are incorporated into the FSAR by reference, were affected. In addition, valve operability verification testing, which is not described in the FSAR, will be performed.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The operation and function of the valves remains the same and the design capability of the auxiliary feedwater system to remove decay heat remains unchanged. The potential for auxiliary feedwater pump steam binding due to possible seat leakage will be monitored by scheduled verification that the auxiliary feedwater piping in the main steam support structure is not heating up.

(162) Description

This modification plugged the low pressure turbine 8th stage extraction pressure sensing lines and the bowl pressure sensing line at the associated coupling on the hot reheat line. A total of three plugs per turbine section were done for a total of nine plugs per unit. The change was implemented in all three units during this reporting period and affected drawing 13-M-EDP-002 (FSAR Section 10.2).

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The change does not affect the ability of the turbine generator or main condenser to function as designed. Neither system performs a safety function.

(163) Description

A modification was made to remove the thermal insulation on steam traps and replace with personnel protective barriers. This change was implemented in Unit 1 during this reporting period and affected drawings 13-M-SGP-001 and 13-M-SGP-002 (FSAR Section 10.3).

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The change will improve the reliability of the steam traps, and the increase in heat removal is bounded by the FSAR safety analyses.

(164) Description

A temporary modification was made in Unit 1 to install a portable filter skid for removal of radioactivity from the chemical waste neutralizer tanks. Drawing 01-M-CMP-002 (FSAR Section 9.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The equipment is not safety related. Pressure boundary integrity is monitored and maintained. Any leakage is contained within tank walls or drip trays.

(165) Description

A temporary modification was made to all three units to install a disc on the Low Pressure Safety Injection (LPSI) A and B and Containment Spray (CS) A and B motor shafts above the top of the mechanical seal gland and below the lower motor bearings. This modification was made to prevent water from spraying in the vicinity of the lower LPSI/CS pump bearing housing in the event of a failed mechanical seal. Drawing 13-M-SIP-001 (FSAR Section 6.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The disc will not impair pump operation since it is not pressure retaining and will be adequately fastened to the shaft to prevent removal by normal operation. Worst case failure consequences are bounded by existing FSAR accident analyses.

(166) Description

A temporary modification was made in Units 1 and 2 to install NALCO acti-brome feed skids and required appurtenances at the metering pump house, east of the spray pond. This modification provided Unit 1 with bromine injection capability, and affected drawing 13-M-SPP-001 (FSAR Section 9.2). In addition, the description in FSAR Section 10.4.5.2 was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The change enhances the chemistry control required by the spray pond program of Tech. Spec. 6.8.4.f. Chemistry evaluation shows acti-brome is compatible with system materials. The bromine is in solution in the circulating water system and has no impact on important to safety equipment or the rest of the plant.

(167) Description

A temporary modification was made in Unit 2 to install a spool piece in place of spray pond hypochlorite addition element 02-JCIN-FQI-0124. Drawing 13-M-SPP-001 (FSAR Section 9.2) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. This equipment is not addressed in the FSAR accident analyses.

(168) Description

A temporary modification was made in Unit 2 to install instrument cables for pipe whip restraint measurement at electrical containment penetration 87. Drawing 13-E-ZAC-066, which is incorporated into the FSAR by reference, was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The penetration does not contain any safety related cabling. Conformance to Reg. Guide 1.63 guidance is maintained.

(169) Description

A temporary modification was made in Units 2 and 3 to attach a stainless steel swageloc plug on the outlet of the low pressure N_2 system rupture disc (2JGANPSE147 and 3JGANPSE147) bodies. This modification was made to prevent N_2 from blowing out of the rupture disc opening, allowing system pressure to be maintained. Drawing 13-M-GAP-002 (FSAR Section 9.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The change does not affect safety-related system operation. This portion of the N_2 system is still protected from over-pressure by relief valve PSV-80.

(170) Description

Temporary modifications were made in Units 1 and 2 to install a 60" diameter blind flange on the discharge bell of 1M-HCN-A02C, 2M-HCN-A02B, and 2M-HCN-A02A, to provide control element drive mechanism (CEDM) cooling in the event of fan failure. Drawing 13-M-HCP-002 (FSAR Section 9.4) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. CEDM ACU operation is not required for safe shutdown of the plant and has no safety related function. The flange is designed to preclude failure of important to safety components.

(171) Description

A temporary modification was made in Unit 2 to allow a splice connection to be made in the condulet at valve 2JSGAHY179A. The modification affected drawing 13-E-SGB-021, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The splice is functionally equivalent to the original design.

(172) Description

A temporary modification was made to the Unit 3 condensate (CD) system to replace valve CDN-V140 (a 6" globe valve) with a Conval 1500# 2" globe valve. Drawing 13-M-CDP-002 (FSAR Section 10.4) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The only effect of a failed valve in this location would be that the CD system could not be isolated if maintenance was required. It has no effect on the operation of the system after it is put into service. The CD system is not required to respond for any accident scenarios evaluated in the FSAR.

(173) Description

A temporary modification was made to the charging pump discharge line vent valves (V81, V960 and V961) to add local pressure gauges to aid in the determination of bladder integrity. Drawing 13-M-CHP-002 (FSAR Section 9.3) was affected. This modification was also made in Unit 1 during the 1986 reporting period but was inadvertently omitted from the 1986 report.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The maximum possible leakage due to piping rupture is bounded by the FSAR accident analyses.

(174) Description

A temporary modification was made in Unit 3 which removed pressure gauge CHN-PI-219 and installed a 0-200 psi pressure gauge. Drawing 13-M-CHP-004 (FSAR Section 9.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The temporary gauge is comparable with the design of CHN-PI-219 and does not impact the operation of any other equipment.

(175) Description

A temporary modification was made to Unit 3 to jumper out cells 4 and 6 of battery 3EPKDF14, which were defective. Drawing 13-E-PKA-007, which is incorporated into the FSAR by reference, was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. Surveillance testing will determine system operability and will ensure the technical specification and FSAR requirements are met.

(176) Description

A temporary modification was made in Unit 3 to allow the pressurizer spray lines to be temperature monitored during post-core hot functional testing. The modification consisted of placing thermocouples on the spray line (018-BCAA-4") and routing wires to the PERMDAS scanners on the 80 foot elevation of containment. Drawing 13-M-RCP-001 (FSAR Section 5.1) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The thermocouples are external to the RCS pressure boundary and are not readily adjacent to any important to safety equipment.

(177) Description

A temporary modification was made in Unit 3 to install a jumper bypassing the output contact status of Ru-31 RIC to BOP-ESFAS (FBEVAS Channel A) for modes that FBEVAS is not required by Tech Spec 3.3.3.1 (i.e., until irradiated fuel is in the fuel building). This affected FSAR Sections 3.1, 7.3, 9.4 and 11.5, which discusses automatic initiation of FBEVAS Channel A to BOP-ESFAS.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. Ru-31 is not taken credit for in the FSAR accident analyses. In addition, during this modification, the other load group is available and manual device control is fully operable. Ru-31 output to BOP-ESFAS is to be fully restored prior to placing irradiated fuel in the fuel building.

(178) Description

A temporary modification was made in Unit 3 to place a Masscomp 5500 computer in the computer room for archiving data taken during power ascension. Associated cables for data taking will also be installed. Fire seals in areas discussed in FSAR sections 9B.2.2.14, 9B.2.3.2, and 9B.2.2.3 will be removed to facilitate cable installation.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. This change did not affect any safety related equipment or systems. Proper compensatory measures are taken to ensure no loss of fire protection effectiveness while the fire seals are removed.

(179) Description

A temporary modification was made in Unit 2 to reroute the acid injection inlet line from the chemical waste neutralizing tank inlet nozzle to a 3" blind flange on the top, plant west side of tanks 2CMN-T01A and 2CMN-T01B. Chemical spray shields were installed on all affected mechanical flanges. Drawing 13-M-CMP-002 (FSAR Section 9.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The chemical waste neutralizing tanks are not addressed in the FSAR accident analyses. The equipment has no impact on plant safety.

(180) Description

A temporary modification was made in Units 1 and 2 to install a jumper to defeat the automatic trip of the radwaste evaporator from high level in the concentrate tanks. This was to allow continued processing of radwaste influents while waiting for arrival of portable solidification equipment. Drawings 13-N-LRP-002 and 13-N-LRP-003 (FSAR Section 11.2) were affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. During the time the trip was bypassed, the concentrate monitor tanks were isolated from any possible fluid addition, maintaining the same degree of equipment safety. If any overflow were to occur, spillage would be contained within the radwaste building.

(181) Description

A temporary modification was made in Unit 1 to install a new air compressor, air receivers, and associated piping, electrical and control systems in the 100' level of the turbine building for a breathing/service air system until the permanent plant change package could be closed. This affected drawings 13-M-IAP-001 and 13-M-TCP-003 (FSAR Section 9.2).

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. There is no effect on normal plant operation or shutdown. Failure of the system will not adversely impact any safety related systems.

(182) Description

A temporary modification was made in Unit 3 to provide a temporary panel, pressure gauges, and sensing lines to various Train B low pressure safety injection (LPSI), high pressure (HPSI), and containment spray (CS) valves. Drawing 13-M-SIP-001 (FSAR Section 6.3) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The change added pressure measurement capability at existing connections which have isolated valves.

(183) Description

A temporary modification was made in Unit 3 to provide alternative cooling water supply to air compressors IAN-C01B and C01C, and aftercoolers IAN-E01B and E01C during a turbine cooling water system outage. This affected drawings 03-M-TCP-003 and 03-M-DSP-002 (FSAR Section 9.2).

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. No safety related or important to safety equipment was affected. The change was made to facilitate maintenance on the turbine cooling water system.

(184) Description

A temporary modification was made in Unit 1 to install blanks in isolation dampers 1M-HJA-M57, M59, M15, M16, M56, and M58, to provide isolation of the control room pressure boundary during maintenance activities on isolation dampers. Drawing 01-M-HJP-001 (FSAR Section 9.4) was affected. In addition, the description of the dampers in FSAR sections 6.4 and 9.4.1, and the post fire actions described in FSAR sections 9.4.1.2.2.2, 6.4.2.2.2.N, 6.4.3.4, and 18.III.D.3.4 were affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The replacement of redundant active components with passive blanks has no effect on important to safety equipment or systems. The control area envelope is partially placed into its essential (isolated) alignment, which is protective in nature.

(185) Description

A temporary modification was made in Unit 1 to install a blank in place of isolation damper 1M-HJB-M56, to provide ductwork integrity during maintenance activities in the damper. FSAR section 6.4 and drawing 01-M-HJP-001 (FSAR Section 9.4) were affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The smoke removal system on the 140' elevation of the control building is affected, but portable smoke exhaust fans are stored in this area which should compensate for this. The blank is fabricated to the same standards as the rest of the control room essential ventilation system.

(186) Description

A temporary modification was made in Unit 1 to provide non-class, 480V, 3 phase, AC power from weld receptacle motor control center bucket 1-E-NHN-M1002 to class swing battery charger 1-E-PKB-H16 for control room indication during the Train B outage scheduled on class 4160V AC bus 1-E-PBB-S04. This affected drawings 13-E-PKA-001 (FSAR Section 8.3) and 13-E-PKA-005, which is incorporated into the FSAR by reference.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The equipment powered as a result of this change is utilized as "functional use only" and in no way will be used to secure the plant in the event of an accident. Loads and breaker sizes were considered in the design of this modification.

(187) Description

A temporary modification was made in Unit 2 to install a strip chart recorder to monitor SV-12 (trip solenoid) and LS-16 (trip latch position switch) of the feedwater pump turbine trip circuitry to determine the cause of unannunciated trips. Drawing 13-E-FTB-005, which is incorporated into the FSAR by reference, was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The equipment was installed to monitor performance of the feedwater pump turbine trip circuitry and did not affect performance or function of the system. No safety related or important to safety equipment was affected.

(188) Description

A temporary modification was made to Unit 3 to provide the capability for discharge of uncontaminated water from the liquid radwaste system (LRS) holdup and recycle monitor tanks to the evaporation pond, via the oil/water separator. Drawing 03-N-LRP-001 (FSAR Section 11.2) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. Discharges to the evaporation pond are controlled by administrative

means to ensure compliance with Tech. Spec. 3/4.11.1 limits. In addition, caution and/or danger tags are installed on valves as appropriate. The pressure boundary of the LRS is not affected.

(189) Description

A temporary modification was made to Unit 1 to install a portable ion exchanger skid for recirculation of the Reactor Makeup Water Tank. Drawings 13-M-CHP-002, 13-M-CHP-003 and 13-M-RDP-005 (FSAR Section 9.3) were affected. This modification was implemented during the 1986 reporting period but was inadvertently omitted from the 1986 report.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The operation of the tank did not change. The modification was installed on the overflow line to the RD sump, not on an active piece of safety related piping.

(190) Description

A temporary modification was made in Unit 1 to provide for hookup/ interface with the ATI Transportable Volume Reduction System (ATI-TVR-III). Drawings 01-M-DSP-002 (FSAR Section 9.2), 01-M-FPP-003 (FSAR Section 9.5), 01-M-IAP-002 (FSAR Section 9.3), 01-N-SRP-002 (FSAR Section 11.4) and 01-N-SRP-003 (FSAR Section 11.4) were affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. FSAR accident analyses make no assumptions concerning the affected systems. All potential radioactive drainage is returned to the radwaste system floor drains and means are provided to transfer radioactive fluid from the ATI trailer to the radwaste system storage tanks in case of emergency. A separate fire protection analysis ensures that fire protection capabilities are maintained.

(191) Description

A temporary modification was made in Unit 2 to install a jumper to cause unit cooler 2M-HAA-Z04 to run any time the breaker is closed. This provides cooling to the Train A auxiliary feedwater pump room with essential ACU (2M-HAA-Z04) during maintenance of the normal AHU (2M-HAN-A01A). Drawing 13-E-HAB-006, which is incorporated into the FSAR by reference, was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. By placing the ACU in continuous operation, the temperature in the auxiliary feedwater pump room will be maintained and the ACU will perform its safety related function by being in operation in the event of an auxiliary feedwater pump actuation.

(192) Description

A temporary modification was made in Unit 3 to provide temporary power to essential lighting panels 3E-QBN-D90 and D91 to maintain emergency lighting battery integrity during a mini outage. Drawings 13-E-PGA-005 and 13-E-PGA-006, which are incorporated into the FSAR by reference, were affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The lighting and associated power supplies are non safety related.

(193) Description

A temporary modification was made to Unit 1 to add fire protection alarm check valves and associated piping to the entry/exit trailer, the radiation protection trailer, and the connection for the integrated leak rate test skid. Drawing 13-M-FPP-003 (FSAR Section 9.5) was affected. This modification was implemented during the 1986 reporting period but was inadvertently omitted from the 1986 report.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The system is designed with three 50% capacity fire pumps. The design bases of the fire protection system are not changed.

(194) Description

A temporary modification was made in Unit 1 to install a one inch rubber hose from valve APDSNV550 on the demineralized water makeup strainer, to drain valve APDSNVA37 on the feedwater pump cooling water supply line. Valve APDSNVA37 was closed and valves V550 and V414 were opened to provide cooling water to the auxiliary boiler feedwater pumps during a demineralized water system outage. Drawing AO-M-DSP-001 (FSAR Section 9.2) was affected.

Summary of Safety Evaluation

This modification did not introduce an unreviewed safety question. The equipment involved is not discussed in the FSAR and any malfunction will not impact plant safety.

(195) Description

An engineering evaluation was performed to address the poor vacuum performance experienced during a Post Accident Sampling System (PASS) performance test, while evacuating the off-gas septum flash via eductor VX-28. The 3/8" discharge tubing was replaced with 1/2" tubing, which affected drawing 13-N-SSP-003 (FSAR Section 9.3). The change was implemented in Unit 3 during this reporting period.

Summary

This change did not introduce an unreviewed safety question. PASS has no function in safely shutting down the plant; it serves to obtain and analyze coolant and area samples following an accident to aid in core damage assessment. The change has no impact on PASS pressure boundary integrity.

(196) Description

An engineering evaluation was performed to determine how to secure the opening where containment isolation valve 3J-SIA-UV-0672 was temporarily removed. The resolution consisted of installing a temporary blank flange over the pipe opening, utilizing a steel flat plate, gasketing material, tie rods, and pipe saddle clamps. This was implemented in Unit 3 during this reporting period and affected drawing 13-M-SIP-001 (FSAR Section 6.3).

Summary

This change did not introduce an unreviewed safety question. The change was implemented to ensure compliance with Tech. Spec. 3.9.4.C.1 for containment isolation during fuel load. The piping system is to be restored prior to Mode 4 entry. Seismic evaluation is not compromised and a pressure test will be performed to ensure isolation integrity.

(197) Description

An engineering evaluation was performed to disposition the discovery that portions of lines RC-A-051-BCAA-16", SI-A-240-BCAA-16" and SI-160-BCAA-14" were installed out of tolerance, creating the existence of a low point. The resolution consisted of removing the drain valves from the system and capping the pipe. This change was implemented in Unit 3 during this reporting period and affected drawing 13-M-SIP-002 (FSAR Section 6.3).

Summary

This change did not introduce an unreviewed safety question. The change will maintain RCS integrity and is bounded by FSAR accident analyses.

A number of changes implemented in one unit during the 1987 reporting period were implemented during the 1985 or 1986 reporting period in another unit. As a result, the description of the change and the summary of the safety evaluation were included in the 1985 or 1986 report.

In addition, some changes implemented during the 1987 reporting period were inadvertently included in the 1986 report.

These changes are tabulated below.

Unit In Which Change was Implemented During 1987	Unit For Which Change Was Previously Reported	1986 Report Item No. (Reference 1)	1985 Report Item No. (Reference 2)
2	1	150	-
3	1,2	79	-
1	2	208	-
3	1,2	158	-
1	2	211	-
2,3	1	161	-
1	2	213	-
1	2	214	-
2,3	1	173	-
3	1,2	233	-
3	1	251	-
1	2	-	164
3	1,2	228	-
2	1	190	-
3	1,2	191	-
1	2	229	-
2	1	195	-
3	2	223	-
3	1	-	143
3	1,2*	-	147
2	1	-	149
2	1	-	54
3	1	300	-
1	2	225	-
2	1	202	-
3	1	-	161
2,3	1,2	118	-
2	1,2	119	-
2	1,2	120	-
1,3	1,2	123	-
1,3	1,2	124	-
1,2	1,2	126	-
2	1,2	128	-
1,2,3	1,2	129	-
2	1,2	130	-
1,2	1**	137	-
1	1	141	-

*This change was reported for Unit 1 in the 1985 report. The Unit 2 change was implemented during the 1986 reporting period and was inadvertently omitted from the 1986 report. The Unit 3 change was implemented during the 1987 reporting period.

**This change was a temporary modification to provide breathing air to the containment during an outage.

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

8808020382

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

List of Figures

TROJAN NUCLEAR PLANT

AUXILIARY FEEDWATER SYSTEM
DESIGN BASIS DOCUMENT

CONTENTS

List of Tables

TROJAN NUCLEAR POWER PLANT
AUXILIARY FEEDWATER SYSTEM
DESIGN BASES DOCUMENT

1.0 FUNCTIONAL DESCRIPTION

1.1 PURPOSE

The safety function of the Auxiliary Feedwater (AFW) System is to supply water to the secondary side of the steam generators for reactor decay heat removal when normal feedwater sources are unavailable. The AFW System also supplies feedwater to the steam generators during startup, hot standby, and shutdown.

1.2 SYSTEM BOUNDARIES

The AFW System boundary from suction to discharge (including the water source and heat sink) includes those portions of the system required to accomplish the AFW System function and connected branch piping up to and including the second valve, which is normally closed or capable of automatic closure when the safety function is required. The AFW System boundary also includes (as described in a Nuclear Regulatory Commission [NRC] Safety Evaluation Report [SER] dated August 4, 1981) any portion of branch piping that is structurally coupled to the AFW System boundary in a way that causes the seismic response of the branch piping to transmit loads to the AFW System. As a minimum, this includes the branch lines outside the AFW System boundary to a point of three orthogonal restraints. All mechanical and electrical equipment, piping (eg, instrument air), conduits, and cable trays that are necessary for, or contain items that are necessary for, the operation of the AFW System are also considered within the bounds of the AFW System. In addition, the structures housing these systems and components are included. Similar constraints apply to alternative

means of decay heat removal. The mechanical, electrical, and instrument and control components that are considered a part of the AFW System boundaries are listed in the following subsections. Figure 8-1 illustrates the AFW System boundaries.

1.2.1 MECHANICAL

(1) AFW Pumps P-102A and 102B.

(2) DeLaval oil pump.

(3) Turbine Driver K-107A.

(4) Turbine Driver K-107A drain coolers.

(5) Turbine Driver K-107A and P-102A lube oil cooler.

(6) Diesel Driver K-107B.

(7) Diesel Driver K-107B speed increaser X-173.

(8) AFW pump diesel fuel oil day tank (T-152).

(9) Pump P-102B coolers.
   (a) Diesel lube oil cooler.
   (b) Jacket water cooler.
   (c) Speed increaser gear lube oil cooler.
   (d) Diesel intercooler.
   (e) Pump P-102B lube oil cooler.

(10) Electric AFW Pump P-182.

(11) Lube Oil Pump P-183.

(12) Pump P-182 shaft-driven oil pump.

(13) Manual valves.

(14) AFW flow control isolation valves (CV-3004 A1 through D1 and A2 through D2).

(15) Turbine steam supply isolation valves (CV-1451 through CV-1454 and MO-3170).

(16) Solenoid Valves SV-1451 through SV-1454.

(17) Accumulators T-166A through T-166D.

(18) Turbine Trip and Throttle Valve MO-3071.

(19) Service Water System Isolation Valves (MO-3045A, MO-3045B, MO-3060A, and MO-3060B).

(20) P-182 motor-operated discharge isolation valves (MO-2947A and MO-2947B).

(21) P-182 differential pressure control valve (CV-2967).

1.2.2 ELECTRICAL

(1) Remote shutdown panel (C-160).

(2) Turbine Driver K-107A governor.

(3) Diesel Driver K-107B starter battery and battery charger.

(4) Diesel Driver K-107B governor.

(5) Pump P-182 ac motor.

(6) Valve motor operators.

1.2.3 INSTRUMENT AND CONTROL

(1) Limit switches for valve position indication in control room and Panel C-160.

(2) Flow Switches FIS-3004 A1 through D1 and FIS-3004 A2 through D2 and automatic flow control valve closure and lockout controls.

(3) Condensate storage tank level instrumentation.

(4) AFW pump differential pressure control circuits.

(5) Valve position switches.

(6) Turbine Driver K-107A steam inlet and exhaust pressure monitoring loops.

(7) Pump P-102A suction and discharge pressure monitoring loops.

(8) Pump P-102B suction and discharge pressure monitoring loops.

(9) Diesel Fuel Oil Tank T-152 level controls.

(10) AFW flow monitoring instrumentation.

(11) Turbine Driver K-107A speed monitor loop and electrical overspeed trip.

(12) Pumps P-102A and P-102B, and Speed Increaser X-173 bearing temperature monitoring loops.

(13) Turbine Driver K-107A local temperature and pressure gages.

(14) P-102A local temperature and pressure gages.

(15) Diesel Driver K-107B local temperature, pressure, and speed gages, and local temperature and pressure gages.

1.2.4 SYSTEM INTERFACES

The following paragraphs describe the systems that interface with the AFW System.

1.2.4.1 Diesel Fuel Oil System

The Diesel Fuel Oil System supplies fuel oil to the diesel-driven AFW pump fuel oil day tank (T-152).

1.2.4.2 Condensate Makeup Water System

The SCII condensate storage tank (CST) is the preferred source of water for the diesel-driven and turbine-driven AFW pumps. It is the only source of water for the electric AFW pump. Recirculation flow from each AFW pump is directed back to the CST.

1.2.4.3 Service Water System

The Service Water System is the emergency water supply to the suction of the diesel-driven and turbine-driven AFW pumps. The Service Water System also supplies cooling water to the diesel-driven AFW Pump. Service water cools the diesel jacket cooler; the diesel engine intercooler; the diesel engine and pump lube oil coolers; and the gear

lube oil cooler on the speed increaser. Service water is used as the backup coolant for the turbine-driven AFW pump lube oil cooler and the turbine bearing heat exchangers.

1.2.4.4 Feedwater and Condensate System

The AFW System supplies emergency feedwater to the steam generators through Feedwater System piping at the inlet to the steam generators. The electric AFW pump supplies feedwater through the Feedwater System piping to the steam generators during normal plant startup and shutdown. If both main feedwater pumps trip, a signal automatically starts the ESF AFW pumps to supply feedwater to the steam generators. The Condensate Chemical Injection System inlet isolation valves and pumps are manually opened and are started at Chemistry Department request to supply hydrazine to main feedwater lines entering the Containment when using the electric AFW pump to place the steam generators in a wet layup condition.

1.2.4.5 Main Steam System

Steam is supplied to the Terry turbine of the turbine-driven AFW pump from the Main Steam System. Steam supply lines tap off each main steam header upstream of the main steam isolation valves. During AFW System operation, steam pressure signals from the main steam headers are compared to the discharge pressure signals from the AFW pumps to control AFW flow to the steam generators.

1.2.4.6 Reactor Protection System

The Reactor Protection System provides automatic starting signals to the diesel-driven and turbine-driven AFW pumps. These signals

(Channel A for the turbine-driven pump and Channel B for the diesel-driven pump) are:

(1) Safety injection (SI) (any SIS).

(2) Steam generator low-low level (11 percent, 2/3 any steam generator).

(3) Undervoltage on 4.16-kV Buses A1/A2 (undervoltage on degraded grid).

1.3 SYSTEM PERFORMANCE REQUIREMENTS

This subsection describes the design, protective, and operational outputs of the AFW System as a function of operating mode or condition. Included is a description of the physical conditions under which such operation is required to occur.

1.3.1 AMERICAN NUCLEAR SOCIETY CONDITIONS

Since 1970, Westinghouse has used the American Nuclear Society (ANS) classification of plant conditions, which divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are:

(1) Condition I: Normal operation and operational transients.

(2) Condition II: Faults of moderate frequency.

(3) Condition III: Infrequent faults.

(4) Condition IV: Limiting faults.

The AFW System is designed to maintain its functional capability when any of the following conditions occurs:

(1) Loss of normal feedwater.

(2) Major rupture of a main feedwater pipe.

(3) Induced malfunction in the Steam Generator PORV Control System.

(4) Induced malfunction in the Pressurizer PORV Control System.

1.3.1.1 Loss of Normal Feedwater

A loss of normal feedwater (from pump or turbine failures, valve malfunctions, or loss of off-site ac power) is an ANS Condition II event. If an alternative supply of feedwater were not supplied to the Plant during this accident, residual heat after the reactor trip would heat the primary system water enough to cause water relief from the pressurizer. Significant loss of coolant from the Reactor Coolant System (RCS) could lead to core damage. Analysis shows that after a loss of all off-site ac power simultaneous with a loss of normal feedwater, the AFW System can remove the stored and residual heat, and thus prevent overpressurization of the RCS and loss of coolant from the reactor core.

The BLKOUT code (see WCAP-7898, Long Term Transient Analysis for PWRs) was used in the analysis to model the plant transient after a loss of normal feedwater. Major assumptions were:

(1) The initial steam generator water level (in all steam generators) at the time of reactor trip is at a conservatively low level, ie, the lower narrow-range level tap.

(2) The Plant is initially operating at 102 percent of the engineered safeguards design rating.

(3) A conservative core residual heat generation is based upon long-term operation at the initial power level preceding the trip.

(4) A heat transfer coefficient in the steam generator is associated with the RCS natural circulation.

(5) Auxiliary feedwater flow is available 1 min. after the accident, at 426 gpm.

(6) Auxiliary feedwater is delivered to two steam generators.

(7) Secondary system steam relief is achieved through the code safety valves. Steam relief is through the power operated relief valves (PORVs) or steam dump valves for most cases of loss of normal feedwater. However, analysis assumed these to be unavailable.

(8) The initial reactor coolant average temperature is 4°F lower than the nominal value since this condition results in a greater expansion of RCS coolant during the transient and in a higher water level in the pressurizer.

The results of this analysis are presented in Figure 15.2-9 of the UFSAR, which shows plant parameters after a loss of normal feedwater. At no time is the tube sheet uncovered in the steam generators receiving AFW flow, and at no time is there water relief from the pressurizer. Thus, if the auxiliary feed delivered is greater than that assumed (426 gpm), the initial reactor power is less than 102 percent, or if the steam generator water level in at least one steam generator is above the low-low level trip point at the time of the trip, the result will be a steam generator minimum water level higher than shown and increased margin to the point at which reactor coolant water relief occurs.

1.3.1.2 Major Rupture of a Main Feedwater Pipe

A major feedwater line rupture (ANS Condition IV event) is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is inside Containment between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of AFW to the affected steam generator.

Analysis (see WCAP-7909, MARVEL-A Digital Computer Code for Transient Analysis for a Multiloop PWR System) showed that for the postulated feedline rupture, the assumed AFW System capacity is adequate to remove decay heat, prevent overpressurizing the RCS, and prevent uncovering of the reactor core. The AFW is assumed to be initiated 10 min. after the trip with a feed rate of 426 gpm. An additional 5 min. is assumed to elapse before the feed lines are purged and the relatively cold (120°F) AFW enters the unaffected steam generators. The assumed AFW flow rate can remove decay heat 2,100 seconds after a trip. After this time, core decay heat decreases below the AFW heat removal capacity, and reactor coolant temperature and pressure decreases.

1.3.1.3 Induced Malfunction in the Steam Generator PORV Control System

After a feedline rupture outside Containment, the steam generator PORVs are assumed to exhibit a consequential failure due to an adverse environment. Failure of the PORV in the open position results in the

depressurization of multiple steam generators, which are the source of steam for the steam turbine-driven AFW pump. This scenario was analyzed (see UFSAR Section 15.2.8.3) with the following assumptions:

(1) A break occurs outside Containment between the penetration and feedline check valve.

(2) An adverse environment resulting from the rupture affects the steam generator PORV control systems associated with the ruptured loop and the intact loops.

(3) A single active failure occurs in the diesel engine-driven AFW pump.

Analysis of this scenario demonstrated that postulated break locations at Trojan would be limited to 5- to 10-ft lengths of 14-in.-diameter feedwater pipe adjacent to the Containment penetrations. These piping runs are located entirely within the compartmentalized Main Steam Support Structure (MSSS). A main feedwater piping rupture of the class that leads to this scenario could not generate an adverse environment at the location of the steam generator PORV components associated with any of the intact loops. At most, the PORV control system for the affected steam generator could be subjected to an adverse environment. In this case, even if the affected PORV were to fail open, steam delivery to the AFW pump turbine driver from the three unaffected steam generators would continue to be available.

1.3.1.4 Induced Malfunction in the Pressurizer PORV Control System

As part of the follow-up efforts to the Three Mile Island 2 (TMI-2) accident, Westinghouse analyzed this class of accidents and reported the results in WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems. The analyses assumed a total loss of feedwater with various concurrent small primary pipe breaks. The

transients were analyzed to 5000 seconds, without operator action and with an assumption of no auxiliary feedwater, to determine when operator action would be required to ensure no core uncovery.

The WCAP-9600 analyses concluded that the worst-case situation would be an optimally sized break that just precluded delivery of safety injection fluid to the RCS. This break size (about 0.2 in. in diameter) is considerably smaller than the open area of one of the two pressurizer PORVs. Section 4.2.3.5 of WCAP-9600 indicates that no operator action would be needed if both pressurizer PORVs were to fail open since safety injection flow would then be sufficient to ensure no core uncovery. Therefore, the conclusions reported in WCAP-9600 are conservative with respect to the possibility of one or both pressurizer PORVs being stuck open because of consequential malfunction of the PORV control system. Furthermore, the WCAP-9600 analyses were based on a typical two-loop plant. This basis ensures a conservative minimum action time because of the small RCS water inventory relative to a four-loop plant such as Trojan. The conclusion reached in Section 4.2.4 of WCAP-9600 is that for the worst-case primary pipe break (0.2-in. equivalent diameter) concurrent with the loss of all feedwater, the plant can be brought to a fully stable situation without core damage, provided AFW flow is initiated within 3,500 seconds.

To apply these conclusions to the case of a concurrent feedline rupture, Westinghouse performed additional calculations conservatively, and assumed that all liquid inventory in the steam generator associated with the ruptured feed line would flow out of the rupture without removing any heat (ie, liquid blowdown). These calculations show that the heat removal capability of the liquid inventory blowdown requires operator action 1,200 seconds earlier than the time reported in WCAP-9600. Thus, if a feed line rupture is assumed to be coincident with the analyses performed in Section 4.2 of WCAP-9600, the operator has at least 2,300 seconds to cause injection

of AFW into the intact steam generators. UFSAR Section 15.2.8.1 assumes AFW initiation at 60 seconds; hence, the consequences of feed line rupture with the consequential failure of the pressurizer PORV control system are bounded by those reported in Section 15.2 of the UFSAR.

1.3.2 OTHER PERFORMANCE REQUIREMENTS

1.3.2.1 Control Room Inaccessibility

The remote shutdown station for AFW System operation is required to assure the Plant can be brought to a safe condition after a main control room evacuation (Westinghouse to Bechtel Letter POR-1472, October 20, 1972). Sufficient controls and indications are available at the Remote Shutdown Panel (C-160) to initiate and monitor AFW System performance.

1.3.2.2 Loss-of-Coolant Accident Concurrent with Leaking Steam Generator Tube

In this case, a barrier is maintained between the steam feed lines through preservation of a pressure differential that prevents leakage from the RCS into the secondary plant. Immediately after the loss-of-coolant accident, the steam pressure is above the RCS pressure. When the RCS pressure is low, the differential pressure is maintained by a static head above the leaking tube. The AFW System provides water to fill the steam generator and thus prevents leakage from the primary to secondary side.

1.3.2.3 System Cooldown

The AFW system must supply sufficient feedwater to the steam generators to remove decay heat and maintain hot standby conditions for 2 hr and subsequently cool down the primary system to the

temperature and pressure at which the Residual Heat Removal (RHR) System can operate (a cooldown to 350°F, which corresponds to a steam generator pressure of 125 psia) within 4 hr.

The design basis AFW flow rate to support this cooldown is 880 gpm. Westinghouse design criteria assume a minimum AFW flow rate of 440 gpm to envelop the accident analysis assumption of 426 gpm available flow. The minimum rated flow per pump required by Westinghouse to maintain sufficient heat transfer surface in the steam generators while limiting reactor coolant temperature rise to prevent relief of water through the pressurizer relief valve is 440 gpm (per Westinghouse to Bechtel Letter POR-121 dated June 4, 1969). However, a 440-gpm capacity would lengthen the time for reactor cooldown. Hence, Westinghouse recommended a total minimum available AFW flow rate of 880 gpm for the hot shutdown and turbine trip conditions to meet the 4-hr cooldown criteria (per Westinghouse to Bechtel Letter POR-113 dated June 3, 1969). The consequent system design specifies a minimum flow rate of 880 gpm per pump in order to envelop both this requirement and other single failure considerations.

The minimum usable condensate storage supply of feedwater required by original Westinghouse design (per Westinghouse to Bechtel Letter POR-728 dated March 29, 1971) to effect this cooldown was 190,000 gallons. The Westinghouse calculations supporting this volume are not available to PGE. However, the assumptions used to determine this volume were:

(1) Cooldown rate of 50°F per hour.

(2) Reactor coolant pumps shut down.

(3) No water being used except to feed the steam generators.

(4) No blowdown from the steam generators.

(5) No reserve tank volume available.

(6) Storage requirements for feedwater makeup or initial charging of the system not included.

(7) Plant to be cooled down from maximum calculated load to hot standby, held at hot standby for 2 hr, then cooled down to 350°F in 4 hr.

Reanalysis of the required condensate storage volume was provided by Westinghouse in 1973. This volume of 196,000 gallons is illustrated by Westinghouse Curve SSE-1119 dated December 11, 1972. Additional assumptions used in determining Curve SSE-1119 were:

(1) Steam generator levels are initially at the low-low level.

(2) Steam generators are refilled to no-load programmed level at completion of the cooldown to 350°F.

(3) Condensate water temperature is 100°F.

The design basis usable volume of 196,000 gallons is greater than that determined previously due to increased margin required by Westinghouse to account for uncertainty in plant metal mass (see Westinghouse to Bechtel Letter POR-1613 dated February 14, 1973).

Westinghouse evaluation of this requirement for Trojan (per Westinghouse to Bechtel Letter POR-85-579 dated May 24, 1985) assumes delivery of the required volume to the active steam generators and includes no allowances for unusable CST volume, AFW pump NPSH requirements, or any condensate water lost from a pipe break prior to isolation.

1.3.3 MODE OPERABILITY

OPERABILITY of the AFW System ensures that the RCS can be cooled to less than 350°F from the normal operating temperature in the event of a total loss of offsite power. Either the diesel-driven or the turbine-driven AFW pump has the capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS pressure and temperature to 400 psig and 350°F, respectively, at which point the Residual Heat Removal System may be placed into operation for continued cooldown. Limiting Condition for Operation (LCO) 3.7.1.2 requires that at least two independent AFW pumps and associated flow paths be OPERABLE in MODEs 1, 2, and 3 with:

(1) One feedwater pump capable of being powered by an OPERABLE diesel with ≥450 gal. of fuel in its day tank.

(2) One feedwater pump capable of being powered from an OPERABLE steam supply system.

The AFW System is not required to be OPERABLE in MODEs 4, 5, or 6.

1.3.4 SUMMARY

The AFW System is designed to supply feedwater at a flow sufficient to ensure adequate heat transfer area coverage in the steam generators to prevent a temperature rise in the reactor coolant that would cause release of coolant through the pressurizer relief valves. This condition is satisfied as long as the AFW System can provide feedwater at a minimum flow rate of 426 gpm total to two steam generators within 1 min. after receipt of a signal that automatically starts the AFW pumps.

AFW System flow is required to remove decay heat and cool down the RCS to 350°F corresponding to a steam generator pressure of 125 psia, at which time the Residual Heat Removal System can be operated. The

system must also be able to provide the required AFW flow for at least 2 hr from one AFW pump train, independent of any ac power source (per GS-5 of NUREG-0611). In addition, sufficient feedwater (196,000 gallons of usable water in the CST) must be available to maintain hot standby conditions for 2 hr after a reactor trip, then cool down the primary system at an average rate of 50°F per hour to the temperature and pressure at which the Residual Heat Removal System can operate (a cooldown to 350°F in 4 hr).

2.0 CODES, STANDARDS, AND REGULATORY DOCUMENTS

This section describes the regulatory documents, codes and standards, and general design criteria (GDC) applicable to the design, procurement, manufacture, installation, testing, operation, modification, and maintenance of the Auxiliary Feedwater (AFW) System and its components. With the exception of the GDC listed in 10 CFR 50 Appendix A and described in Updated Final Safety Analysis Report (UFSAR) Chapter 3, Section 3.1, regulatory documents, codes, and standards cite the specific component or portion of the AFW System to which they apply.

2.1 GENERAL DESIGN CRITERIA

The following GDCs are applicable to the AFW System and its components, electrical supplies, and instrumentation:

(1) Criterion 1: Quality Standards and Records.

(2) Criterion 2: Design Bases for Protection Against Natural Phenomena.

(3) Criterion 3: Fire Protection.

(4) Criterion 4: Environmental and Missile Design Bases.

(5) Criterion 13: Instrumentation and Control.

(6) Criterion 17: Electric Power Systems.

(7) Criterion 19: Control Room.

(8) Criterion 20: Protection System Functions.

(9) Criterion 34: Residual Heat Removal.

2.2 REGULATORY DOCUMENTATION

The following regulatory documents are applicable to the Auxiliary Feedwater (AFW) System or its components, electrical power supplies, and instrumentation as noted.

2.2.1 CODE OF FEDERAL REGULATIONS (CFR)

(1) 10 CFR 50.48. Fire Protection (AFW System and components).

(2) 10 CFR 50.49. Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants (AFW System electrical components, power supplies and instrumentation, and modifications after February 22, 1983).

(3) 10 CFR 50 Appendix A. General Design Criteria for Nuclear Power Plants (AFW System).

(4) 10 CFR 50 Appendix B. Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants (AFW System).

(5) 10 CFR 50 Appendix R. Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, Sections III.G, III.J, and III.L (AFW System).

(6) 10 CFR 100. Reactor Site Criteria (general for Trojan Nuclear Plant).

2.2.2 REGULATORY GUIDES

(1) Regulatory Guide 1.29. Seismic Design Classification (Rev. 3) September 1978 (AFW System).

(2) Regulatory Guide 1.30. Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment, August 1972 (AFW System instrumentation and electrical equipment).

(3) Regulatory Guide 1.62. Manual Initiation of Protective Actions, October 1973 (AFW System).

(4) Regulatory Guide 1.89. Qualification of Class 1E Electrical Equipment for Nuclear Power Plants, November 1974 (applicable as described in PGE-1025).

(5) Regulatory Guide 1.97. Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant and Environmental Conditions during and following an Accident (Rev. 3), May 1983 (AFW System instrumentation).

(6) Regulatory Guide 1.100. Seismic Qualification of Electrical Equipment for Nuclear Power Plants (Rev.1), August 1977 [applicable to future modifications per PGE-1028, In-House Position (IHP) No. 1.100-1-2, effective December 31, 1984].

(7) Regulatory Guide 1.106. Thermal Overload Protection for Electric Motors or Motor Operated Valves (Rev. 1), March 1977 (applicable to modifications involving electric motors or motor-operated valves per PGE-1028, IHP No. 1.106-1-1, effective September 1, 1982).

(8) Regulatory Guide 1.117. Tornado Design Classification (Rev.1), April 1978 (applicable as described in PGE-1028, IHP No. 1.117-1-1, effective September 1, 1982).

2.2.3 NUREGs

(1) NUREG-0578. TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, July 1979.

(2) NUREG-0588 (Category 1). Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment (AFW System Class 1E equipment purchased before February 22, 1983).

(3) NUREG-0611. Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants.

(4) NUREG-0737. Clarifications of TMI Action Plan Requirements, Section II.E.

(5) NUREG-0800. Standard Review Plan, Section 10.4.9, Auxiliary Feedwater System (PWR), (Rev. 2), July 1981 (used by the NRC for review of the AFW System for SERs).

2.3 CODES AND STANDARDS

This section lists the codes and standards applicable to the Auxiliary Feedwater (AFW) System. These sources of codes and standards are:

(1) American Society of Mechanical Engineers (ASME).

(2) Institute of Electrical and Electronics Engineers (IEEE).

(3) American National Standards Institute (ANSI).

(4) National Electrical Manufacturers Association (NEMA).

2.3.1 GENERAL

ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.

2.3.2 MECHANICAL

(1) ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, Division I, 1968 and Addenda through 1971 (AFW Pumps P-102A and P-102B).

(2) Draft ASME Code for the Inservice Testing of Pumps in Nuclear Plants, April 1970 (AFW Pumps P-102A and P-102B).

(3) Draft ASME Code for the Inservice Testing of Valves in Nuclear Power Plants, June 1970 (AFW System valves).

(4) ASME Boiler and Pressure Vessel Code, Section XI, 1977 edition, IWA-7210 (modifications to the AFW System).

(5) ASME Boiler and Pressure Vessel Code, Section XI, 1983 edition and Addenda through the summer of 1983 (applicable to testing of the AFW System and components).

(6) ASME Boiler and Pressure Vessel Code Section III, Class 2, Nuclear Power Plant Components (Containment Penetrations, 1971, and Addenda through Summer 1971).

(7) ANSI B31.1.0. Power Piping Code, 1973 (piping and valves in the AFW System up to the discharge piping connecting to the Main Feedwater System).

2.3.3 ELECTRICAL

(1) IEEE 279-1971. Criteria for Protection Systems for Nuclear Power Generating Stations (AFW System).

(2) IEEE 308-1971. Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations (power supplies to AFW System components and instrumentation).

(3) IEEE 323-1971. IEEE Trial Use Standard. Qualifying Class 1. Electrical Equipment for Nuclear Power Generating Stations (applicable to original AFW System equipment and instrumentation as described in PGE-1025).

(4) IEEE 323-1974. Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations (applicable to AFW System electrical equipment and instrumentation as described in PGE-1025).

(5) IEEE 338-1977. Standards for the Periodic Testing of Nuclear Power Generating Station Safety Systems (AFW System).

(6) IEEE 344-1971. IEEE Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations (applicable to original AFW System equipment and instrumentation).

(7) IEEE 344-1975. IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations (AFW System equipment and instrumentation modifications after 1975).

(8) NEMA Standard SM-22-1970. Single-Stage Steam Turbines for Mechanical Drive Service (K-107A).

3.0 SYSTEM DESIGN BASES

This section presents design criteria applicable at the system level. Although many criteria are generic in nature, the general and specific requirements applicable to the system are addressed.

3.1 PIPING PRESSURE AND TEMPERATURE CRITERIA

The piping in the Auxiliary Feedwater (AFW) System is designed according to Power Piping Code ANSI (American National Standards Institute) B31.1 and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. All piping within the boundaries of the system, as prescribed in Section 1.2, is constructed of carbon steel and designed according to the Power Piping Code ANSI B31.1, with the exception of the AFW pump discharge piping connecting to the Main Feedwater System. This piping is constructed of impact-tested carbon steel and designed according to ASME Boiler and Pressure Vessel Code Section III, Class 2, Nuclear Power Plant Components, Containment Penetrations. The piping classification coding and the design, normal and maximum pressure, and temperature ratings are found in M-301, Specification for Piping Materials and Standard Details, Revision 1.

3.2 SEISMIC CRITERIA

GDC 2, "Design Basis for Protection against Natural Phenomena", and Appendix A to 10 CFR 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants", require that nuclear power plant structures, components, and systems important to safety be designed to withstand the effects of earthquakes without loss of ability to perform their safety functions. Those structures, components, and systems that have

been designed to remain functional in the event of a safe shutdown earthquake (SSE) are designated safety related and as Seismic Category I (SCI). The Auxiliary Feedwater (AFW) System design must meet the requirements of GDC 2, and the safety-related portion of the AFW System (the portion necessary for safe shutdown of the plant) must be designed, constructed, and maintained to SCI requirements in accordance with Position C.1.g of Regulatory Guide 1.29. These requirements are included within the scope of IE Bulletins 79-02, 79-04, 79-07, 79-14, and 80-11, and IE Information Notice 80-21. The non-safety-related portions are designed in accordance with Position C.2 of Regulatory Guide 1.29.

Seismic response spectra curves for plant areas containing safety-related equipment have been developed for both the operating basis earthquake (OBE) and the SSE. The OBE is assumed to produce a ground acceleration of 0.15 g horizontal and 0.10 g vertical. The SSE is assumed to produce a ground acceleration of 0.25 g horizontal and 0.17 g vertical.

The entire AFW System is designed to meet SCI requirements with the following exceptions:

(1) The electric motor-driven AFW pump (P-182).

(2) The turbine and diesel-driven pump recirculation lines.

(3) The condensate storage tank (CST).

Seismic Category I requirements extend to the first seismic restraint beyond the SCI/SCII boundaries, meaning that the first restraint is to be a full anchor (restraining six degrees of freedom) where feasible. When a full anchor cannot be located within a reasonable distance from the SCI/SCII boundary, two 3-way directional restraints are to be located beyond the boundary.

3.2.1 P-182

The electric motor-driven AFW pump and associated piping are not safety-related and are not seismically qualified. This pump is not required for plant safe shutdown because the safety-related portion of the AFW System includes two redundant 100 percent capacity pumps. This classification was found to be acceptable to the Nuclear Regulatory Commission (NRC) in its review for the implementation of NUREG-0611 recommendations for Trojan under TMI Action Plan Item II.E.1.1 [see the Safety Evaluation Report (SER) transmitted in NRC-to-PGE letter dated February 18, 1981].

3.2.2 RECIRCULATION LINES

The recirculation lines from the safety-related pumps to the CST are not seismically qualified downstream of the pressure-reducing orifices and locked-open valves. A break in the recirculation lines would not adversely affect pump operation.

3.2.3 CST

The CST (the preferred water supply) is not seismically qualified. The backup water supply from the Service Water System to the safety-related AFW pumps is seismically qualified. Service water is supplied to each pump through a separate SCI piping system. Each service water line is isolated from the other so that failure of one does not affect the other. Each line is also isolated from the SCII CST by nonreturn valves. Procedures accomplish the switchover from the primary to the secondary supply upon loss of the primary supply. These measures were found acceptable by the NRC in its review for the implementation of NUREG-0611 recommendations for Trojan under TMI Action Plan Item II.E.1.1 (see the SER transmitted in NRC-to-PGE letter dated October 23, 1980).

The common AFW pump suction piping from the CST to the pumps up to the check valves was initially classified SCII. Failure of the suction piping while the CST water level is above the pump shutdown setpoint could cause the AFW pumps to run dry. To increase system reliability and prevent this occurrence, this piping was upgraded to SCI downstream of the expansion joint located near the CST nozzle as part of RDC 86-002. RDC 86-002 also added the expansion joint to replace MD-050, the common suction line isolation valve, which was removed. The expansion joint in the common suction piping allows relative movement between the CST and piping, and is required for seismic qualification of the pipe.

The SCI components of the AFW System are identified in the Trojan Q-List, Table VII.

3.3 QUALITY ASSURANCE CRITERIA

GDC 1, "Quality Standards and Records", requires that structures, components, and systems important to safety be designed, fabricated, created, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 3 of the Updated Final Safety Analysis Report (UFSAR) describes the quality classification system and quality standards that satisfy GDC 1 for water- and steam-containing components of the Trojan Nuclear Plant. Group 3 quality standards apply to the Seismic Category I portions of the Auxiliary Feedwater (AFW) System. Group 4 quality standards apply to the remaining portions of the AFW System.

Codes and standards used for components and systems specify the design and quality assurance (QA) requirements. Standards applicable to each quality group are given in Table 3.2-3 of the UFSAR. For modifications to the design of new systems, the latest effective edition of the applicable code, as listed in 10 CFR 50, may be used.

The AFW System at Trojan was designed, constructed, installed, and tested in accordance with a QA program that meets the requirements of 10 CFR 50, Appendix B. All modifications are performed in accordance with PGE-8010, Nuclear Quality Assurance Program.

3.4 REDUNDANCY/DIVERSITY CRITERIA

GDC 34, "Residual Heat Removal", requires redundancy of Auxiliary Feedwater (AFW) System components so that under accident conditions, the safety function can be performed despite a single active component failure. This requirement may be coincident with the loss of off-site power for certain events. In addition, NRC Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants", has guidelines that may be used to select the minimum diversity acceptable for AFW System pump drives and power supplies. Adequate diversity precludes system failures due to failure of a single type of motive power that can be subject to a failure of the driving component itself, its source of energy, or the associated control system. Branch Technical Position ASB 10-1 specifically recommends the following:

(1) The AFW System should consist of at least two full-capacity, independent systems that include diverse power sources.

(2) Other powered components of the AFW System should also use the concept of separate and multiple sources of motive energy. The required diversity, for example, might be two separate AFW trains, each capable of removing the residual heat load of the reactor system, with one separate train powered from either of two ac sources and the other train wholly powered by steam and dc power.

(3) The piping arrangement, both intake and discharge, for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. This arrangement should take into account pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One arrangement that would be acceptable is crossover piping containing valves that can be operated by remote manual control from the control room, by use of the power diversity principle for the valve operators and actuation systems.

(4) The AFW System should be designed with suitable redundancy to offset the consequences of any single active component failure; however, each train need not contain redundant active components.

(5) With respect to a high-energy line break, the system should be arranged to assure the ability to supply necessary emergency feedwater to the steam generators, despite the postulated rupture of any high-energy section of the system; a concurrent single active failure is assumed.

The above criteria envelop the generic recommendations identified in NUREG-0611.

The AFW System at Trojan meets both the redundancy requirements of GDC 34 and the guidelines of Branch Technical Position ASB 10-1. The system is designed with adequate redundancy to accommodate a single active component failure without loss of function. Diversity in pump motive power sources and essential instrumentation and control power sources is provided.

The AFW System provides two redundant and independent means of supplying feedwater to the steam generators for cooling the RCS under emergency conditions. Two full-capacity AFW pumps are provided with diversity in drivers. Pump P-102A, in AFW Train A, is driven by a steam turbine. Pump P-102B, in AFW Train B, is driven by a diesel engine. There are no valves in the common piping to the condensate storage tank (CST) to prevent AFW pump damage due to operation without water. The discharge piping of each AFW pump is separated from the other by motor-operated isolation valves and check valves until they join into a single line before connecting with the main feedwater line of each steam generator. Each steam generator is separately supplied with AFW through its main feedwater line.

The pumps take suction from two sources. The normal source is the SCII CST. If this source fails, water can be supplied to the pumps from the SCI Service Water System (SWS). Service water is supplied to each pump through a separate SCI piping system. Each SWS supply line is isolated from the other so that failure of one does not affect the other. Each service water line is also isolated from the CST by nonreturn valves.

An SCII, full-capacity AFW pump (P-182) is used during normal startup and shutdown to reduce wear on the SCI ESF AFW pumps. In an emergency, this pump may be aligned to AFW System Train A or B should both SCI pumps fail. This capability to provide increased AFW System availability extends beyond normal single failure criteria and thus provides increased reliability. However, operation of AFW Pump P-182 is not assumed for any accident analysis.

Complete physical and electrical separation is maintained throughout the pump control, control signals, electrical power supplies, and instrumentation for each safety-related AFW pump. Class 1E electrical components in the AFW System are powered by the Engineered Safety Features (ESF) Electrical Distribution System. AFW System Trains A and B are powered by ESF Instrumentation Channels A and B. SCII AFW Pump P-182 may be manually loaded on either 4.16-kV ESF bus if needed.

3.5 ENVIRONMENTAL QUALIFICATION CRITERIA

The environmental qualification (EQ) review of electric equipment important to safety is performed in accordance with 10 CFR 50.49.

GDC 4, "Environmental and Missile Design Bases", establishes the general requirement for environmental qualification of safety-related equipment. It states in part that "structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents".

IEEE 323-1974, "IEEE Standards for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", is the current industry standard for environmental qualification of safety-related electrical equipment. This standard was first issued as a trial-use standard, IEEE 323-1971. The 1974 standard includes specific requirements for aging, margins, and document maintenance that were not included in the 1971 trial-use standard. The 1974 standard was endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.89 for plants with existing construction permit applications.

Design of the Trojan Nuclear Plant was initiated before the issuance of IEEE 323-1971. PGE committed to the Westinghouse Qualification Program, which used IEEE 323-1971 as a design standard for many Nuclear Steam Supply System (NSSS) components, especially those inside Containment.

In 1979, the Division of Operating Reactors (DOR) published "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors", containing definitive criteria for reviewing the EQ of safety-related electrical equipment. The intent of the DOR guidelines is to provide a basis for judgments required to

confirm that operating reactors are in compliance with GDC 4. It provides a number of NRC staff positions on selected areas of the qualification issue. Category 1 positions apply to equipment qualified in accordance with IEEE 323-1974. Category 2 positions apply to equipment qualified in accordance with IEEE 323-1971. In June 1984, the NRC issued Revision 1 to Regulatory Guide 1.89, which describes a method acceptable to the NRC staff for complying with 10 CFR 50.49. The guide endorses IEEE 323-1974, subject to certain clarifications and conditions. The DOR guidelines, NUREG-0588, and 10 CFR 50.49 are implemented at Trojan in accordance with the following: Equipment ordered prior to February 22, 1983 is qualified in accordance with DOR guidelines or NUREG-0588 (Category 1). Equipment ordered after February 22, 1983 but before June 1, 1984 is qualified in accordance with NUREG-0588 (Category 1), unless special provisions are applicable as delineated in PGE-1025. Qualification in accordance with NUREG-0588 is construed to be equivalent to meeting the provisions of 10 CFR 50.49. Equipment ordered after June 1, 1984 should be qualified in accordance with 10 CFR 50.49, unless special provisions are applicable. The provisions of Revision 1 to Regulatory Guide 1.89 are complied with in the qualification review of this equipment, except as noted in PGE-1028, Regulatory Guide Policy Manual.

The EQ Program also includes electric equipment not currently in the scope of 10 CFR 50.49, including electric equipment important to safety in mild environments and certain postaccident monitoring equipment in harsh environments specified in Category 3, Regulatory Guide 1.97. These equipment categories are included in the EQ Program to ensure that all equipment qualification efforts are effectively integrated throughout plant design.

The AFW components important to safety in harsh environments are:

(1) Various terminal boards in the electrical penetration and main steam support structure (MSSS).

(2) The four turbine steam supply solenoid valves and associated limit switches.

(3) The eight CV-3004 series flow control valves and associated flow indication switches.

(4) The eight 3043 series flow transmitters.

(5) Power, instrumentation, and control cabling.

The Engineered Safety Features (ESF) AFW pumps are located in a mild environment in the Turbine Building. Coolers have been provided on K-107A drain lines to reduce humidity, and enclosures for three SCI components have been provided with additional sealing to increase reliability because of high humidity in the P-102A pump room. Supply and exhaust fans for each ESF AFW pump room automatically start upon a pump start.

High energy line breaks in the AFW pump rooms were excluded from consideration, as described in PGE-1004, because the redundant trains are separated from each other, as well as from the effects of a main steam or feedwater line break.

Table 3-2 of PGE-1025 is the master list of electrical equipment important to safety at Trojan within the scope of 10 CFR 50.49(b) and is generated from a sort of the Component Summary Sheets (CSS) for equipment in harsh environments. The identification of equipment important to safety also documents electrical equipment important to safety in mild environments.

Table 4-1 of PGE-1025 lists the normal and accident environment conditions for each plant location.

3.6 FIRE PROTECTION CRITERIA

GDC 3, "Fire Protection", requires that structures, systems, and components important to safety be designed and located to minimize the probability and effect of fires and explosions. 10 CFR 50.48(a) further requires that the Trojan Nuclear Plant implement a fire protection program. The Auxiliary Feedwater System is a safe shutdown system per 10 CFR 50 Appendix R.

PGE-1012, "Trojan Nuclear Plant Fire Protection Plan", describes the fire protection features at the Trojan Nuclear Plant that implement compliance with the above requirements. The NRC SERs dated March 9, 1978, March 25, 1980, and October 6, 1980 document NRC acceptance of features meeting requirements of Branch Technical Position (BTP) APCSB9.5-1, Appendix A, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976. This has since been incorporated in BTP CMEB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants and GDC 3. Also, an NRC SER dated October 15, 1985 documents NRC acceptance of features meeting the requirements of 10 CFR 50, Appendix R, Sections III.G.3 and III.L. These requirements, and NRC-granted exceptions, form the basis for the Trojan Nuclear Plant fire protection design and are described in detail in PGE-1012.

3.7 ENVIRONMENTAL PROTECTION CRITERIA

GDC 2, "Design Bases for Protection Against Natural Phenomena", requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of ability to perform their safety functions. GDC 4, "Environmental and Missile Design Bases", requires that structures, systems, and components important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, and testing, and with

postulated accidents. It also requires that they be appropriately protected against dynamic effects, including missiles, pipe whips, and discharging fluids. This section describes those criteria addressing flooding protection, missile protection, tornado protection, pipe whip, and jet impingement.

3.7.1 FLOODING PROTECTION

The design basis for flooding protection of the AFW considers two types of flooding events: external flooding, and internal flooding due to rupture of fluid system piping.

3.7.1.1 External Flooding

The design basis natural flooding event for the Trojan site is the probable maximum flood (PMF) for the Columbia River, established by the U.S. Army Corps of Engineers. The maximum water level at the Plant site for the PMF is 39.2 ft mean sea level (MSL), based on a flood elevation of 36 ft above MSL corresponding to a river discharge of 2.2 million cubic feet per second (cfs), plus a wave run-up of 3.2 ft.

The design basis artificial flood for the Trojan site is a flood resulting from a catastrophic failure of the Grand Coulee Dam due to the massive effects of a nuclear weapon. This postulated event would result in a short-duration river discharge of 3.6 million to 4.4 million cfs at the Trojan site approximately 2 days after dam failure. The maximum flood elevation for 4.4 million cfs, on the basis of artificial flood hydrographs, calculated energy slopes, and Manning's formula, is 41 ft MSL, plus a wind wave run-up of 1.75 ft, for a water level at Plant site of 42.75 ft.

The flooding protection criterion is implemented at Trojan by virtue of the fact that all components of the AFW System are at least 45 ft above MSL, and therefore above any credible flood level.

3.7.1.2 Internal Flooding

The design basis internal flooding event for the AFW System is a massive circulating water pipe rupture in the Turbine Building. The assumed maximum flood rate is based on a design flow rate of 210,000 gpm for each of the two Circulating Water System pumps.

Protection for the design basis event is implemented by the flood relief dampers on the west wall of the Turbine Building and a concrete and steel dike that surrounds the SCI AFW pumps and remote shutdown station rooms. The 2-ft-high flood relief dampers are assumed to accommodate a discharge rate of 500,000 gpm to the Plant yard. Therefore, a 2-ft-high dike is sufficient to prevent flooding of the SCI AFW pumps and remote shutdown station rooms. Drain line protection is provided in the rooms to prevent backflooding of circulating water into the rooms.

3.7.2 MISSILE PROTECTION

As required by GDC 4, the AFW System must be appropriately protected against the dynamic effects of missiles. The design basis turbine missiles are those from the last-stage wheel of the low-pressure turbine of the turbine generator, and a bucket vane from the low-pressure turbine of the turbine generator. Maximum assumed shield penetration would result from a low trajectory (predominantly horizontal) missile from the 120-degree segment of the last-stage wheel of the low-pressure turbine, which would penetrate 3 ft 0 in. into 3,000 psi reinforced concrete, or 1 ft 9 in. into 6,000 psi reinforced concrete.

Protection from turbine missiles is implemented through the combined thickness of slabs above the AFW pumps. No other internally generated missiles are assumed to present a significant hazard to the AFW System. The SCI portion of the AFW pump suction piping located outside the Turbine Building is protected by a concrete slab.

3.7.3 TORNADO PROTECTION

As required by GDC 2, the AFW System must be designed to withstand the effects of tornadoes without a loss of ability to perform its design safety function. The design bases reflect appropriate consideration of the most severe of the natural phenomena reported for the site, appropriate combination of the effects of normal and accident conditions with the effects of the natural phenomena, and the importance of the safety functions to be performed.

The design wind velocity is 105 mph 30 ft above ground elevation of 45 ft above MSL. The design basis tornado load for the AFW pump rooms is a 200-mph tornado. This load is based on analysis per American Society of Civil Engineers (ASCE) Paper 3269, Wind Forces on Structures, and reflected in the Trojan Nuclear Plant SER dated October 1974. The design basis missiles for the 200-mph tornado are equivalent to:

(1) A 4 in. by 12 in. by 12-ft long wood plank traveling end on at a velocity of 200 mph at any elevation of the structure.

(2) A 3 in. diameter by 10-ft long ASA Schedule 40 steel pipe traveling end on at a velocity of 75 mph at any elevation of the structure.

(3) A 4000-lb passenger car striking with a velocity of 40 mph not more than 25 ft above grade.

Combined tornado loading takes into account the wind loading, the tornado-induced pressure differential, and tornado-generated missiles to produce the most critical loading condition.

The SCI portions of the AFW System are protected against tornadoes and tornado missiles. Exposed sections of AFW pump suction piping and

condensate storage tank (CST) level instrumentation located outdoors are protected by enclosures. The CST level instrumentation is required to provide level indication in the control room and the remote shutdown station, and to stop the SCI AFW pumps upon a low CST level. The redundant level transmitters are on opposite sides of the CST and are provided with missile-proof enclosures. The output of the transmitters "fails low" to trip the AFW pumps if the transmitters break away from the CST.

3.7.4 PIPE WHIP AND JET IMPINGEMENT

As required by GDC 4, the AFW System design must include protection against dynamic effects, including the effects of pipe whipping and discharging fluids, that may result from equipment failures and events and conditions outside the nuclear power unit.

Bechtel Power Corporation Topical Report BN-TOP-2, "Design for Pipe Break Effects", describes analytical techniques to analyze fluid jet impingement loads and to establish design criteria for pipe restraints provided to prevent damage from pipe whip effects resulting from postulated pipe breaks in nuclear power plants. PGE-1004, "Trojan Nuclear Plant Analyses of Pipe System Breaks, Outside Containment", describes the analysis performed to verify the acceptability of plant design. High-energy lines are defined as piping systems whose operating temperatures exceed 200°F and/or whose operating pressure exceeds 275 psig.

The AFW System lines analyzed in PGE-1004 are the steam supply lines to the AFW turbine driver and the AFW discharge lines to the steam generators.

Request for Design Change (RDC) 80-054 added a guard pipe around the turbine-driven AFW pump discharge line in the diesel-driven pump room so that a rupture in a turbine pump discharge line would not damage the

diesel-driven AFW pump. This RDC was based on recommendations from NRC-to-PGE letters, dated October 3, 1979 and May 14, 1980. RDC 80-061 connected the discharge of the motor-driven AFW pump to the discharge of the turbine-driven AFW pump to improve system reliability in accordance with NUREG-0737 and recommendations from NRC-to-PGE letters dated October 3, 1979 and May 5, 1980.

3.8 AUTOMATIC INITIATION

GDC 20, "Protection System Functions", requires that a protection system be designed to initiate protective action automatically to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. GDC 34 requires that the safety function of the designed system, that is, residual heat removal by the Auxiliary Feedwater (AFW) System, can be accomplished even in the case of a single failure. In addition, to improve the reliability of the AFW System, licensees are required to upgrade the system to ensure timely automatic initiation when necessary. The recommendations of NUREG-0578, Section 2.1.7.a, are:

(1) The design shall provide for the automatic initiation of the AFW System.

(2) The automatic initiation signals and circuits shall be designed so that a single failure does not result in the loss of AFW System function.

(3) Testability of the initiating signals and circuits shall be a feature of the design.

(4) The initiating signals and circuits shall be powered from the emergency buses.

(5) Manual capability to initiate the AFW System from the control room shall be retained and shall be implemented so that a single failure in the manual circuits does not result in the loss of system function.

(6) The AC motor-driven pumps and valves in the AFW System shall not be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiating signals and circuits shall be designed so that their failure does not result in the loss of manual capability to initiate the AFW System from the control room.

In the long term, these signals and circuits are required to be upgraded in accordance with safety-grade requirements. Specifically, in addition to the items mentioned above, the following is required: automatic initiation signals and circuits must have independent channels, use environmentally qualified components, have system bypassed/inoperable status features, and conform to control system interaction criteria, as stipulated in IEEE 279-1971.

The initiation signals, logic, and associated circuitry of the Trojan Nuclear Power Plant AFW automatic initiation system comply with the long-term safety-grade requirements of NUREG-0578, Section 2.1.7.a, and the subsequent clarification issued by the Nuclear Regulatory Commission (NRC), with the exception that the initiation circuits for loss of both main feedwater pumps are not safety grade. The NRC has determined that because the trip of both main feedwater pumps is a secondary AFW System automatic initiation signal for Westinghouse plants and because there are other diverse safety-grade AFW System automatic initiation signals provided in the Trojan design, the circuitry providing initiation on the tripping of both main feedwater

pumps does not have to comply with safety-grade requirements (see NRC-to-PGE letter dated April 8, 1982). No credit is taken for this initiation signal in the accident analyses.

At Trojan, logic channels supply initiation signals to the control circuitry and automatically initiate AFW flow under any of the following conditions:

(1) Turbine-driven AFW pump.

(a) Safety injection (Train A).
(b) Steam generator (SG) low-low level (two of three level transmitters in any one SG).
(c) Undervoltage on 4.16-kV Bus A1.
(d) Trip of both main feed pumps.

(2) Diesel-driven AFW pump.

(a) Safety injection (Train B).
(b) SG low-low level (two of three level transmitters in any one SG).
(c) Undervoltage on 4.16-kV Bus A2.
(d) Trip of both main feed pumps.

Section 4.17 of IEEE 279-1971 requires that protection systems include means for manual initiation of each protective action at the system level and that the single-failure criterion as set forth in Section 4.2 of IEEE 279-1971 be met.

Manual operation of the AFW System is provided in the Control Room and at Panel C-160. Each control circuit is independent so that a single failure in one train does not affect the redundant train. In addition, the automatic initiating circuits are designed to be electrically independent from the Control Room manual start circuit so that failure

of the automatic initiating signals does not affect the control room manual capability of AFW pumps. Manual initiation may be accomplished at the following locations:

(1) Turbine-driven AFW pump.

(a) Local (C-160).
(b) Control room.

(2) Diesel-driven AFW pump.

(a) Local (C-160).
(b) Control room.

(3) Motor-driven AFW pump.

(a) Local (at the 4.16-kV A5 switchgear).
(b) Control room.

System valves are aligned for normal operation (normally open); therefore, flow is initiated upon the startup of either pump. The operation of either automatically initiated pump provides the capacity to remove decay heat from the steam generators at a rate sufficient to prevent overpressurization of the Reactor Coolant System (RCS) and to maintain steam generator levels. Initiation of AFW flow within 1 min. of receipt of an automatic start signal provides sufficient capacity to remove decay heat, prevent overpressurizing the RCS, and prevent uncovering the reactor core under the postulated accident conditions. This condition was an assumption of accident analysis (see Section 1.3) and thus no margin may be assumed. Consequently, the AFW System is capable of automatically initiating appropriate protective action, with precision and reliability, whenever a condition monitored by the system reaches a preset level.

The protection system is designed to be independent of the control system. In certain applications, the control signals and other nonprotective functions are derived from individual protective channels through isolation amplifiers. The design meets the requirements of GDC 34. The automatic initiation signals and circuitry for the AFW System at Trojan comply with the single-failure criteria of IEEE 279-1971.

The Trojan Plant has the ability to test the AFW initiation system while at power. The automatic start circuits are designed with provisions for both periodic testing and calibration during normal plant operations (see PGE-to-NRC letter dated October 17, 1979). Trojan's 31-day and 92-day interval Periodic Operating Tests (POTs) 5-1 and 5-2 and 18-month interval POTs 5-3 and 25-2e outline the methods and reporting procedures to be used in testing the pumps, valves, and their initiating circuits required by the Technical Specifications (Paragraphs 4.7.1.2.1 and 4.7.1.2.2). All locked valves (FW 087-094, FW 103-114, and FW 119-120) in the auxiliary feed flow path are checked during the monthly POT.

The auto-start signals for the steam turbine-driven and diesel-driven AFW pumps are powered from 120-V preferred instrument ac Buses Y11 and Y22, respectively.

4.0 COMPONENT DESIGN BASES

This section consolidates and addresses information regarding design basis requirements for individual (or groups of) components within the system. Discussion concentrates on component-level design criteria, which may or may not relate directly to the system performance requirements discussed in Section 3.0. Where appropriate, earlier discussions are cited to avoid repetition.

4.1 AFW PUMPS P-102A and P-102B

The two engineered safety features (ESF) auxiliary feedwater (AFW) pumps in the AFW System automatically start to supply feedwater into the steam generators as required in Section 1.3.4. The design requirements concerning AFW Pumps P-102A and P-102B are described below.

4.1.1 DESIGN REQUIREMENTS

(1) AFW Pumps P-102A and P-102B must be qualified as SCI components per Section 3.2 as required by Position C.1.g of Regulatory Guide 1.29.

(2) The two AFW pumps, P-102A in Train A and P-102B in Train B, must meet redundancy and single-failure criteria as described in Section 3.4.

(3) The AFW pumps must have diverse drivers to meet the criteria stated in BTP ASB 10-1 relating to GDC 34. AFW Pump P-102A is driven by Turbine Driver K-107A, and AFW Pump P-102B is driven by Diesel Driver K-107B.

(4) The net positive suction head (NPSH) required at the suction of the AFW pumps at the design flow rate of 960 gpm is

25 ft. The required NPSH is established by the pump supplier and verified in Bechtel Calculation 16-42. Bechtel Calculations 16-44 and 16-47, and Nuclear Plant Engineering (NPE) Calculation TE-115 also address NPSH on the basis of the condensate storage tank (CST) level during both one- and two-pump operation.

(5) AFW Pumps P-102A and P-102B must meet design pressure and temperature ratings specified in Section 3.1.

(6) AFW Pump P-102A must operate independently of ac power as required by Recommendation GL-3 of the NRC-to-PGE letter dated October 23, 1980.

4.1.2 CONFIGURATION

The two ESF AFW pumps are identical six-stage, horizontal, centrifugal pumps. Both are equipped with radial, pressure-lubricated bearings, which are journal bearings with a Kingsbury thrust bearing on the outer end. Bearing lubrication is supplied by a shaft-mounted DeLaval oil pump.

The lubricating oil is cooled by an internal lube oil cooler. Cooling water for the P-102A lube oil cooler is provided by the pump itself to allow operation independent of ac power. Service water was used to cool the lube oil cooler of P-102A until RDC 78-020 modified the cooling water flow path to allow pump discharge water to cool the lube oil. Service water can be used as a backup source of lube oil cooler cooling water flow. For P-102B, cooling water flow to the lube oil cooler is supplied from the Service Water System.

Mechanical seals are located on each end of the pump shaft. The seals are cooled by a portion of the pump discharge flow, which is then returned to the CST. A minimum flow orifice is located in each of the AFW Pumps P-102A and P-102B recirculation lines to prevent overheating

of the pump during minimum flow operation. The orifice is rated for a flow of 80 gpm (minimum recirculation flow rate) with a pressure drop of 1,470 psi.

Applicable AFW Pumps P-102A and P-102B design and operating data provided by the pump supplier, Bingham Willamette Company, are listed in Table 9-1.

4.1.3 MARGIN EVALUATION

Design margin for AFW flow rate for accident analyses is held by Westinghouse and is not available to PGE. However, the specified pump ratings of 880 gpm exceed the minimum flow rate of 426 gpm assumed for the worst-case accident analysis. The pump supplier minimum recirculation flow is 80 gpm, and no margin is provided. The margin between available NPSH (38.3 ft) and required NPSH (25 ft) is 13.3 ft as verified by Bechtel Calculation 16-42.

4.2 TURBINE DRIVER K-107A

Turbine Driver K-107A receives steam from each main steam header to drive Auxiliary Feedwater (AFW) Pump P-102A. It automatically starts to mechanically power P-102A to send feedwater into the steam generators as required in Section 1.3.4.

4.2.1 DESIGN REQUIREMENTS

(1) Turbine Driver K-107A must be qualified as a Seismic Category I component per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(2) The turbine driver must be capable of automatic startup as required by GDC-20 and acceleration to design speed within the required time (60 seconds) to meet the required flow conditions per Section 1.3.4.

(3) The control power for Turbine Driver K-107A must be from a Class 1E 125-V dc power source via inverters to ensure that P-102A operates independently of ac power for at least 2 hr as required by Recommendation GL-3 of NUREG-0611 (see the NRC-to-PGE letter dated October 23, 1980).

(4) Turbine Driver K-107A operates on steam pressure supplied between 125 psia and 1,220 psia (lowest setpoint of main steam safety valves with full accumulation).

(5) Sufficient and redundant instrumentation and controls must be located in the control room and at the Remote Shutdown Panel (C-160) to allow for proper operation and monitoring of turbine Driver K-107A as required by GDC 13 and GDC 19.

4.2.2 CONFIGURATION

Turbine Driver K-107A is a single-stage, noncondensing turbine mechanically coupled to AFW Pump P-102A. Turbine Driver K-107A is located within a reinforced concrete enclosure at the east end of the 45-ft level of the Turbine Building.

K-107A automatically starts and accelerates to 4,560 rpm within 60 seconds if any of the following start signals is present:

(1) Safety injection signal Channel A.

(2) Undervoltage on 4.16-kV ESF Bus A1.

(3) Low-low water level (11 percent) from two of three level transmitters in any steam generator.

(4) Loss of both main feedwater pumps (2/2 coincidence).

Steam is supplied to K-107A from each of the four main steam headers. The supply lines tap off each header upstream of the main steam isolation valves. Steam is supplied to K-107A from each main steam header through an air-operated control valve (CV-1451 through CV-1454). The four supply lines converge to a common header containing MO-3170 (steam stop valve) and MO-3071 (turbine trip and throttle valve). The air-operated control valves are normally closed, but the steam lines downstream of them are kept warm by a small amount of steam flowing through bypass orifices. These orifices allow steam flow to K-107A when it is idle to ensure a quick starting capability. Drain coolers, cooled by service water, are installed on lines draining condensed steam to reduce the humidity in the P-102A pump room. The steam pressure required to operate K-107A was verified by PGE Calculation TM-006 in response to License Change Request LCR-76-16.

Controls for operation of K-107A are located in the control room and locally at Panel C-160. Normally, speed is automatically controlled by a governor that compares a required differential pressure signal between the P-102A discharge pressure and the main steam pressure to an actual differential pressure signal. This feature is described in Section 4.11. The Woodward Electric Governor System provides reliability to prevent overspeed upon startup. Overspeed trips protect both the pump and the driver, and limit the pump discharge pressure to the maximum allowable by the piping design (Section 3.1). The electrical trip is set at 5,100 rpm to protect the machine from damage and limit the discharge pressure of the pump. PGE Calculation TM-002 verified that no damage would occur with P-102A operating at just under the original electrical trip setpoint of 5,480 rpm. The mechanical overspeed trip, set at 5,500 rpm, a backup for the electrical overspeed trip, also prevents equipment damage. When testing the mechanical overspeed trip, Turbine Driver K-107A is uncoupled from AFW Pump P-102A to prevent overpressurizing the pump discharge piping.

A level switch trips the turbine driver upon a 30 percent water level in the condensate storage tank (CST). The trip setpoint was determined as part of PGE Calculation TE-115, AFW Pump/CST Level Control Modification for Request for Design Change (RDC) 84-096. An override allows operators to manually restart K-107A at a lower AFW System flow rate if it was shut down on the CST low-level trip. Control power for the K-107A Woodward electric governor is supplied from 24-V dc from JQ-2079A in C-160, which is supplied from a Class 1E 120-V ac power source panel Y22. K-107A is designed to operate for at least 2 hr independently of ac power. This compliance to NUREG-0737, Item II.E.1.1, is verified by Memorandum ANR-172-84, which cites the following:

(1) PGE Calculation TE-028: Evaluated the station battery capacity under various temperature and battery age conditions.

(2) PGE Calculation TE-029: Evaluated the load shed necessary to maintain an operable station battery for a minimum of 2 hr after a loss of ac event.

(3) E. L. Davis to H. E. Williams Memorandum ELD-03-82M, dated December 21, 1982: Summarized the station battery capabilities and recommended load shedding from the A train in order to meet the 2-hr requirement.

(4) R. L. Steele to C. P. Yundt Memorandum RLS-03-83M, dated January 4, 1983: Recommended a procedure change to maintain an operable A train battery and AFW pump for 2 hr after a station blackout.

(5) C. P. Yundt to R. L. Steele Memorandum CPY-443-83, dated June 15, 1983: Documented completion of the Nuclear Plant Engineering (NPE) recommended procedure change.

Cooling water to the turbine lube oil cooler is supplied from P-102A. This flow keeps the K-107A lube oil temperature between a minimum of 110°F (cooler outlet) and a maximum of 185°F (cooler inlet). Applicable Turbine Driver K-107A design and operating data provided by the turbine supplier, Terry Steam Turbine, are listed in Table 9-2.

4.2.3 MARGIN EVALUATION

The design steam inlet pressure range of 125 psia to 1,320 psia provides a margin that envelops the steam pressure range of 125 psia to 1,220 psia, at which the AFW System is designed to operate.

The maximum K-107A inlet steam moisture condition of 0.5 percent by weight is twice the design moisture content of steam exiting the steam generators. The internal steam generator moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.25 percent by weight.

4.3 DIESEL DRIVER K-107B

Diesel Driver K-107B is designed to automatically start and mechanically power P-102B to supply feedwater to the steam generators as required in Section 1.3.4.

4.3.1 DESIGN REQUIREMENTS

(1) Diesel Driver K-107B must be qualified SCI per Section 3.2 as required by Position C.1.g of Regulatory Guide 1.29.

(2) Diesel Driver K-107B must be capable of automatic startup as required by GDC 20 and acceleration to design speed within the required time (60 seconds) to meet the required flow conditions per Section 1.3.4.

(3) Control power for the diesel driver is from the Diesel Driver K-107B nickel-cadmium 28-V battery.

(4) Sufficient and redundant controls and instrumentation must be located in the control room and at the Remote Shutdown Panel (C-160) to allow for proper operation as required by GDC 13 and GDC 19.

(5) The auxiliary feedwater (AFW) pump diesel fuel oil day tank must provide sufficient fuel to operate K-107B for 10 hr under design flow conditions (960 gpm at 3,400-ft head).

4.3.2 CONFIGURATION

Diesel Driver K-107B is a 12-cylinder Waukesha diesel engine, supplied with No. 2 diesel fuel and mechanically connected to P-102B through a speed increaser (X-173). The speed increaser ratio of 1:3.8 allows K-107B optimum speed (1,200 rpm) to match P-102B speed (4,560 rpm). A 12-cylinder diesel engine is used to develop a maximum-rated horsepower of 1,579 bhp at 1,200 rpm to ensure proper operation of P-102B.

K-107B automatically starts and accelerates to 1,200 rpm within 60 seconds if any of the following start signals is present:

(1) Safety injection signal Channel B.

(2) Undervoltage on 4.16-kV engineered safety features (ESF) Bus A2.

(3) Low-low water level (11 percent) from two of three level transmitters in any steam generator.

(4) Loss of both main feedwater pumps (2/2 coincidence).

Diesel fuel is gravity fed from T-152, the AFW pump diesel fuel oil day tank. The tank has a capacity of 500 gallons. The Technical Specifications require at least 450 gallons of fuel in T-152 for P-102B to be operable. This level has been verified sufficient by Bechtel Calculation 12-22.

Controls for the operation of K-107B are located in the control room and at Panel C-160. Normally, speed is automatically controlled by a governor that compares a required differential pressure signal between P-102B discharge pressure and main steam pressure to an actual differential pressure signal. This feature is described in Section 4.11. A Woodward electric governor system prevents diesel engine overspeed during startup. An overspeed trip device trips K-107B if engine speed exceeds 1,350 rpm to protect the diesel engine. The setpoint of the trip ensures that pump discharge pressure does not exceed allowable limits as specified in Section 3.1. This protection has been verified by Bechtel Calculation 16-52. The quick starting capability of K-107B is also enhanced by jacket water system heaters to keep jacket water warm and a soak back pump to circulate lube oil when K-107B is idle.

A level switch trips K-107B at 35 percent water level in the condensate storage tank (CST). The trip setpoint was determined as part of PGE Calculation TE-115, AFW Pump/CST Level Control Modification for RDC 84-096. An override allows operators to manually restart K-107B at a lower AFW System flow rate if the pump shuts down on the CST low-level trip.

Both control and starting power for K-107B are supplied from a Class 1E dedicated 24-V battery.

The Service Water System supplies cooling water to the diesel lube oil cooler, intercooler, jacket water cooler, and speed increaser gear lube oil cooler. Service water flow to K-107B and X-173 is normally

isolated, but when any of the four K-107B automatic start signals is present, MO-3060B opens to initiate cooling water flow.

Applicable K-107B design and operating data provided by the diesel supplier, Waukesha, are listed in Table 9-3.

4.3.3 MARGIN EVALUATION

The margin between actual T-152 capacity of 500 gallons and the volume required to operate P-102B for 10 hr under design flow conditions (960 gpm at 3,400-ft head) has not been verified.

4.4 AFW PUMP P-182

Auxiliary feedwater (AFW) Pump P-182 is a non-safety-related pump. It is used to supply feedwater to the steam generators during normal plant shutdown and startup. It can also be used as a backup to the two engineered safety features (ESF) AFW pumps, P-102A and P-102B.

4.4.1 DESIGN REQUIREMENTS

(1) Pump P-182 must meet design pressure and temperature ratings as specified in Section 3.1.

(2) Pump P-182 must be capable of meeting system design flow requirements if both safety-related pumps are inoperable.

(3) Pump P-182 must be qualified SCII according to Position C.2 of Regulatory Guide 1.29 per NUREG-0800 as described in Section 3.2.

(4) Pump P-182 must be capable of being manually loaded on either emergency diesel generator (EDG) feeding Bus A1 or A2 under emergency conditions if both safety-grade AFW pumps fail to start when there is a coincident loss of off-site power (PGE-to-NRC letter dated December 12, 1980 and NRC-to-PGE letter dated February 18, 1981).

(5) The net positive suction head (NSPH) required for P-182 at the design flow rate of 1,020 gpm with a design head of 3,400 ft is 20 ft as determined by the pump supplier, Bingham-Willamette.

4.4.2 CONFIGURATION

AFW Pump P-182 is an eight-stage, horizontal, centrifugal pump with a double volute and a single suction first-stage impeller. A pump with eight stages is used to obtain the desired system flow rate and pressures with a 3,600 rpm motor. The pump is driven by a 1,250-hp ac motor supplied from 4.16-kV Bus A5. Pump design and operating data are provided in Table 9-4.

Initial bearing lubrication for P-182 is provided by P-183, an installed motor-driven lube oil pump. Power for P-183 is supplied from 480-V ac Bus B-30 (after RDC 86-004 is implemented, power will be supplied from B-43). P-183 starts first to circulate pump lubricating oil when a start of P-182 is initiated. P-182 starts when lube oil pressure is greater than 6 psig. As P-182 comes up to speed, a shaft-driven lube oil pump begins to circulate P-182 lubricating oil. When P-182 reaches normal operating speed and adequate oil pressure is attained by the shaft-driven oil pump (approximately 8 psig), P-183 automatically shuts off. P-183 also automatically starts whenever P-182 is running and lube oil pressure drops to 3 psi. A manual override (43B) on C-341 can be used to allow a local start of P-182 if

P-183 fails to automatically start during a P-182 startup. Cooling water flow to the pump bearing lube oil cooler is supplied from a portion of the discharge of P-182.

Controls and indications are located in the control room and on Panel C-341. P-182 may be started from the control room or locally at 4.16-kV A5 switchgear. It may also be stopped by using the emergency stop push button on Panel C-341.

P-182 automatic trips occur on low suction pressure or low lube oil pressure. The low suction pressure trip prevents pump operation with inadequate NPSH. An override allows operators to manually restart P-182 at a lower flow rate if it has shut down on the low suction pressure trip. The low lube oil pressure prevents pump and bearing damage due to low circulating lube oil pressure.

Under emergency conditions (both ESF AFW pumps have failed to start coincident with a loss of off-site power), power to start and operate P-182 can be supplied from 4.16-kV ESF Bus A1 or A2. An unloaded ESF bus supplied from an EDG is required to start P-182 because of the large power requirement and high starting current of the pump motor. Electrical loads may be added to the ESF bus after P-182 has started, but the total must be kept less than the maximum as specified in Trojan Off-Normal Instruction ONI-55. There are no automatic starting features associated with this pump because of the large power requirement and high starting current of the pump motor.

4.5 MANUAL VALVES

Manual valves in the Auxiliary Feedwater (AFW) System provide positive isolation for portions of the system to allow maintenance during plant operation.

4.5.1 DESIGN REQUIREMENTS

(1) Manual valves in the flow path from the condensate storage tank (CST) to the tie-in point with the Main Feedwater System must be locked open as required by Recommendation GS-2 of NRC-to-PGE letter dated October 3, 1979.

(2) There must be no isolation valves in the common AFW pump suction piping from the CST.

(3) Manual valves that serve a safety-related function or could interfere with a safety function must be locked in the position required for automatic initiation.

4.5.2 CONFIGURATION

RDC 79-064 provided limit switches for open or open/closed position indication in the control room and the Remote Shutdown Panel C-160, for manual valves in the flow path from the CST to the tie-in point with the Main Feedwater System. RDC 86-002 removed the common AFW pump suction isolation valve (MD-050). Alarms are provided at C-160 for closed valves in the flow path from the CST to the Main Feedwater System. Monthly inspections are performed to ensure all manual valves in the AFW System flow path are in their proper position per Periodic Operating Test (POT) 5-2.

4.6 AUXILIARY FEEDWATER FLOW CONTROL ISOLATION VALVES

Each of the eight flow control valves, CV-3004 A1 through D1 and A2 through D2, are required to automatically isolate a ruptured auxiliary feedwater (AFW) feed header to maintain the required flow to the intact steam generator feed headers. The valves are also used to control AFW flow to individual steam generators.

4.6.1 DESIGN REQUIREMENTS

(1) The valves must be qualified SCI in accordance with Position C.1.g of Regulatory Guide 1.29.

(2) The valves must meet the redundancy and single-failure criteria requirements discussed in Section 3.4.

(3) The valves must be safety-related and powered by separate Class 1E 125-V dc power supplies in accordance with GDC 17.

(4) The valves must be environmentally qualified for a harsh environment per PGE-1025 as described in Section 3.5.

(5) The valves have remote status indication and are operable from the control room and Panel C-160 in accordance with GDC 19.

(6) The valves can be opened or closed against the differential pressure (2,170 psid) of the AFW pump running at shutoff head at the overspeed trip setpoint with the corresponding steam generator depressurized.

4.6.2 CONFIGURATION

The eight AFW flow control/isolation valves are 3-in., SCI globe valves operated by Class 1E 125-V dc motor actuators. Controls and open/close indications allow remote manual throttling for each valve from the control room or the Remote Shutdown Panel, C-160. CV-3004 A1 through D1 are powered from 125-V dc Train A; CV-3004 A2 through D2 are powered from 125-V dc Train B.

During normal (standby) operation, all CV-3004 valves are full open. High AFW flow rate in a branch line, as sensed by flow switches FIS-3004 A1 through D1 and FIS-3004 A2 through D2, isolates AFW flow

through the branch line by closing the associated flow control/ isolation valve. The setpoint for the flow switches is 500 gpm. If a valve in one train automatically closes upon high flow, interlocks prevent automatic closure of remaining CV-3004 valves in the same train. A reset capability is provided to reopen automatically closed valves after the high flow fault has been cleared. The reset circuitry is powered by Class 1E 120-V ac power supplies.

The valves are on the 59-ft level of the Main Steam Support Structure and are qualified for operation in a harsh environment per PGE-1025, as described on Drawing E-2.

4.7 TURBINE STEAM SUPPLY ISOLATION VALVES

The Turbine Driver K-107A steam supply isolation valves are CV-1451 through CV-1454 and MO-3170. These valves isolate steam supplied from the four main steam headers to Turbine Driver K-107A. Control Valves CV-1451 through CV-1454 operate automatically to supply steam to this turbine driver. MO-3170 is normally open and does not automatically operate.

4.7.1 DESIGN REQUIREMENTS

4.7.1.1 CV-1451 through CV-1454

(1) Limit switches for Control Valves CV-1451 through CV-1454, and Solenoid Valves SV-1451 through SV-1454, must be environmentally qualified per PGE-1025 as described in Section 3.5.

(2) CV-1451 through CV-1454, SV-1451 through SV-1454, and Accumulators T-166A through T-166D must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(3) CV-1451 through CV-1454 must operate independently of ac power as required by Recommendation GL-3 of NUREG-0611, as described in NRC-to-PGE letter dated October 23, 1980. SV-1451 through SV-1454 de-energize on a loss of ac power and direct air to the bottom sides of the actuators of CV-1451 through CV-1454, allowing them to fail open on a loss of ac power.

(4) SV-1451 through SV-1454, which operate CV-1451 through CV-1454, are powered from Class 1E 120-V ac power supply Panel Y11 as required by GDC 17.

(5) CV-1451 through CV-1454 each have a dedicated accumulator (T-166A through T-166D) to ensure valve operation upon loss of instrument air. The accumulators are sized (15 gallons) to allow three valve motions (open-close-open).

(6) CV-1451 through CV-1454 must be shut during normal operation.

(7) CV-1451 through CV-1454 must meet piping pressure requirements per Section 3.1.

<u>4.7.1.2 MO-3170</u>

(1) MO-3170 must be environmentally qualified per PGE-1025 as described in Section 3.5.

(2) MO-3170 must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(3) MO-3170 must be required to be in the open position to ensure that K-107A and P-102A can be operated independently of ac power.

(4) The MO-3170 must be powered by a Class 1E 480-V ac power supply as required by GDC 17.

(5) The MO-3170 valve body pressure rating must meet piping pressure requirements per Section 3.1.

(6) Redundant control switches and valve position indicators are located in the control room and on the Remote Shutdown Panel (C-160) as required by GDC 13 and GDC 19.

4.7.2 CONFIGURATION

4.7.2.1 CV-1451 Through CV-1454

Steam supply lines to K-107A from each of the four main steam headers are isolated by normally shut control valves. These control valves, CV-1451 through CV-1454, are 3-in., air-operated gate valves that automatically open when K-107A receives a start signal. Air is directed from the Instrument Air System to the pistons of CV-1451 through CV-1454 by operation of Solenoid Valves SV-1451 through SV-1454. The solenoid valves receive power from 120-V ac Panel Y11. On a loss of power, SV-1451 through SV-1454 are automatically positioned to direct instrument air to the control valve actuators to open CV-1451 through CV-1454.

A 15-gallon accumulator in the air supply line to each control valve allows operation of the valve if instrument air pressure is lost. The sizing of the accumulators (T-166A through T-166D) allows three valve motions per control valve (open-close-open). The accumulator size is based on a calculation performed as part of RDC 80-003.

Control switches for operation of SV-1451 through SV-1454 and position indication for CV-1451 through CV-1454 are located in the control room and at the Remote Shutdown Panel (C-160).

The control valves are designed to open with a maximum differential pressure of 1,270 psid. The maximum pressure ensures that the control valves can open with the maximum pressure in the steam generators and atmospheric pressure downstream of the valves.

4.7.2.2 MO-3170

MO-3170 is the AFW Pump P-102A turbine driver stop valve. It is a normally open, 4-in., motor-operated globe valve, located downstream of Control Valves CV-1451 through C-1454 and upstream of MO-3071, the turbine trip and throttle valve. The valve is in the open position to ensure that steam is supplied to K-107A to allow P-102A to operate independently of ac power.

MO-3170 is designed to open with a maximum differential pressure of 1,270 psid. The maximum pressure across the valve ensures that it can open with the maximum pressure in the steam generators and atmospheric pressure conditions downstream of the valves.

A control switch (common for MO-3170 and MO-3071) and position indication light are located in the control room. A separate control switch and position indication are located on Panel C-160. Electrical power for MO-3170 is supplied from 480-V ac ESF Bus B-23 (Class 1E power supply).

4.8 TURBINE TRIP AND THROTTLE VALVE

Turbine Trip and Throttle Valve MO-3071 is located in the steam supply line to K-107A downstream of MO-3170. The valve automatically opens to provide steam flow to K-107A upon startup and rapidly closes under an overspeed condition to shut off steam flow.

4.8.1 DESIGN REQUIREMENTS

(1) MO-3071 must be qualified SCI per Section 3.2, as required by Position C.1.g. of Regulatory Guide 1.29.

(2) MO-3071 and the associated limit switches must be environmentally qualified per PGE-1025, as described in Section 3.5.

(3) MO-3071 is supplied with power from 125-V dc Distribution Panel D10, a Class 1E power supply as required by GDC 17. A dc power supply ensures that P-102A can operate independently of ac power for at least 2 hr as required by Recommendation GL-3 of NUREG-0611, described in NRC-to-PGE letter dated October 23, 1980.

(4) Redundant control switches and valve position indicators for MO-3071 are located in the control room and at the Remote Shutdown Panel (C-160), as required by GDC 13 and GDC 19.

(5) MO-3071 is shut during normal operation, and automatically opens on any P-102A start signal.

(6) MO-3071 must meet the piping pressure requirements of Section 3.1.

4.8.2 CONFIGURATION

MO-3071 is a 4-in., motor-operated globe valve that provides steam flow to K-107A upon a startup of P-102A. The valve has a relatively slow stroke time (20 seconds) to provide control of steam flow until the K-107A governor valve can adequately control steam flow. It also automatically closes upon a K-107A overspeed condition to protect K-107A and P-102A from damage.

MO-3071 can open with a maximum differential pressure of 1,270 psid so the valve can open with maximum pressure in the steam generators and atmospheric pressure downstream of the valve.

MO-3071 automatically opens upon any of the automatic start signals listed for K-107A in Section 4.2.2. To ensure it opens as required, the valve receives dc power independently of ac power. A control switch (common for MO-3071 and MO-3170) and position indication light are located in the control room. A separate control switch for MO-3071 and valve position indication is located on Panel C-160.

4.9 SERVICE WATER SYSTEM ISOLATION VALVES

There are four Service Water System isolation valves for the Auxiliary Feedwater System. Two valves, MO-3045A and MO-3045B, isolate the SCI water supply to the suction of engineered safety features AFW Pumps P-102A and P-102B. The remaining two valves, MO-3060A and MO-3060B, isolate service water cooling flow to the auxiliary coolers of P-102A/K-107A and P-102B/K-107B/X-173, respectively.

4.9.1 DESIGN REQUIREMENTS

4.9.1.1 MO-3045A and MO-3045B

(1) MO-3045A and MO-3045B must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(2) MO-3045A and MO-3045B, and their associated limit switches, must be environmentally qualified per PGE-1025, as described in Section 3.5.

(3) MO-3045A and MO-3045B must be powered from Class 1E 480-V ac supplies as required by GDC 17.

(4) MO-3045A and MO-3045B must meet piping pressure requirements per Section 3.1.

(5) The two valves, MO-3045A in Train A and MO-3045B in Train B, must meet redundancy and single-failure criteria described in Section 3.4.

4.9.1.2 MO-3060A and MO-3060B

(1) MO-3060A and MO-3060B must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(2) MO-3060A, and MO-3060B and the associated limit switch, must be environmentally qualified per PGE-1025, as described in Section 3.5.

(3) MO-3060B must be powered from a Class 1E 480-V ac power supply as required by GDC 17.

(4) MO-3060B automatically opens upon any of the start signals associated with K-107B, as listed in Section 4.3.2.

(5) MO-3060A and MO-3060B must meet piping pressure requirements per Section 3.1.

4.9.2 CONFIGURATION

4.9.2.1 MO-3045A and MO-3045B

MO-3045A and MO-3045B are 6-in., motor-operated gate valves. They are normally shut to isolate the Service Water System supply to the suction of ESF AFW Pumps P-102A and P-102B. Both valves have position indication and control switches in the control room. MO-3045A receives electrical power from 480-V ac ESF Bus B-25, and MO-3045B receives power from 480-V ac ESF Bus B-26.

The valves can open upon a maximum differential pressure of 100 psid; this capability is based on the Service Water System being in operation (both service water pump and service water booster pump in service in one train) with one pump operating at the shutoff head, as verified in Bechtel Calculation 16-50.

These valves have no automatic control functions associated with their operation. They are only opened, from the control room, during emergency AFW System operation after the water level of the condensate storage tank (CST) has decreased to less than the low-low level alarm setpoint (<9 percent). The operators then have a maximum of 30 min. to open the valves from the time the low-low level alarm is actuated before vortexing in the CST occurs. The level setpoint was determined as part of PGE Calculation TE-115, AFW Pump/CST Level Control Modification for RDC 84-096. A commitment to the time requirement was made by PGE on the basis of the Safety Evaluation Report (SER) transmitted in the NRC-to-PGE letter dated October 23, 1980.

4.9.2.2 MO-3060A and MO-3060B

MO-3060A and MO-3060B are both motor-operated valves that isolate cooling water from the Service Water System to the auxiliary coolers of P-102A, K-107A, and X-173. MO-3060A is a 2-in. globe valve; MO-3060B is a 6-in. butterfly valve. The valves can open with a maximum differential pressure of 100 psid; this capability is based on the Service Water System being in operation (both service water pump and service water booster pump in service in one train) with one pump operating at the shutoff head, as verified in Bechtel Calculation 16-50.

MO-3060A is electrically disabled and normally shut. It is not required to be open since the lube oil coolers for P-102A and K-107A are cooled by a portion of the P-102A discharge flow. The automatic operating power was disabled by modifications performed under

RDC 78-020. Because MO-3060A had been powered from an ac power source and was operated to allow cooling water from the Service Water System to the coolers for P-102A and K-107A, it was required to be disabled to meet NRC requirements of Recommendation GL-3 of NUREG-0611, as described in NRC-to-PGE letter dated October 23, 1980. This lineup allows operation of P-102A and K-107A to be independent of the ac power for at least 2 hr.

MO-3060B receives electrical power from 480-V ac ESF Bus B-24. It is normally shut when diesel-driven K-107B is idle, to allow the jacket waterheaters to preheat K-107B. MO-3060B automatically opens upon any of the K-107B automatic start signals listed in Section 4.3.2.

4.10 CONDENSATE STORAGE TANK LEVEL INSTRUMENTATION

Redundant condensate storage tank (CST) level instrumentation is provided for:

(1) Engineered safety features auxiliary feedwater pump trips upon a low CST level.

(2) Level indication in the control room and at the Remote Shutdown Panel.

(3) Level alarms to alert the operators that a level limit is being approached.

4.10.1 DESIGN REQUIREMENTS

(1) Level Transmitters LT-5201 (Train A) and LT-5265 (Train B) trip the associated AFW pump upon a low CST level or after failure of the CST.

(2) The level transmitters and pump trip circuitry must be qualified SCI; level indications and alarms must be SCII.

(3) Redundant level transmitters are located on opposite sides at the bottom of the CST, and are protected by enclosures.

(4) Redundant CST level indication and low level alarms must be provided in the control room to allow operators to anticipate the need to make up water or to transfer to an alternate water supply. The low level alarm setpoint should allow at least 20 minutes for operator action. This requirement was promulgated in NRC to PGE letter dated October 13, 1979, NRC Requirements for Auxiliary Feedwater System at Trojan Nuclear Plant, Additional Short-Term Recommendation No. 1, and acknowledged in PGE to NRC letter dated December 31, 1979, Responses to October 3, 1979 NRC Questions Concerning the Trojan Auxiliary Feedwater System.

4.10.2 CONFIGURATION

RDC 84-096 upgraded the CST level instrumentation from nonsafety grade/single channel to safety grade/redundant channel, to improve system reliability and to change the AFW pump trip from a low suction pressure to a low CST level. The output of the transmitters indicates "low" if they break away from the CST. The diesel AFW pump trips at 35 percent; the turbine AFW pump trips at the 30 percent level. Override capability is provided to allow controlled operation below these levels. A low-low level alarm at a 9 percent tank level allows the operator at least 30 minutes to switch to the backup suction source before vortexing and loss of pump suction occurs. The basis for the trip and alarm setpoints was determined as part of PGE Calculation TE-115, AFW Pump/CST Level Control Modification for RDC 84-096.

4.11 DIFFERENTIAL PRESSURE CONTROL

Operation of the two ESF AFW Pumps P-102A and P-102B, and electric AFW Pump P-182, is controlled by separate differential pressure control circuits. The circuits are similar since each compares the difference between steam generator pressure and the respective pump discharge pressure to a preset differential pressure to control the pump flow rate. The design requirements listed in Section 4.11.1 concern only the ESF AFW pumps. Section 4.11.2 describes both the ESF AFW pumps and the P-182 differential pressure control circuits.

4.11.1 DESIGN REQUIREMENTS

(1) Instrumentation for the differential pressure control of ESF AFW Pumps P-102A and P-102B must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(2) Instrumentation for the differential pressure control of ESF AFW Pumps P-102A and P-102B must be powered from separate Class 1E 120-V ac ESF power sources, as required by GDC 17.

(3) Instrumentation for the differential pressure control of ESF AFW Pumps P-102A and P-102B must be environmentally qualified per PGE-1025 as described in Section 3.5.

(4) Two steam generator pressure inputs are available to each ESF AFW pump pressure control circuit to meet redundancy and single-failure criteria requirements described in Section 3.4.

(5) Redundant instrumentation for operation of ESF AFW Pumps P-102A and P-102B differential pressure control are

located in the control room and at the Remote Shutdown Panel to meet the requirements of GDC 13 and GDC 19.

4.11.2 CONFIGURATION

The differential pressure control circuits for ESF AFW Pumps P-102A and P-102B maintain a preset differential pressure (100 psid) between steam generator pressure and the pump discharge pressure by controlling the speed of K-107A and K-107B, respectively. The differential pressure setpoint of 100 psid is based on ensuring that sufficient pressure is maintained in the steam generators to prevent leakage of reactor coolant through a U-tube (Westinghouse to Bechtel Letter POR-1710 dated April 25, 1973). The steam generator pressure signals to the P-102A differential pressure control circuit are supplied from Main Steam Headers A (PT-514) and C (PT-536). The steam generator pressure signals for the P-102B differential pressure control circuit are supplied from Main Steam Headers A (PT-516) and D (PT-545). The two steam generator pressure input signals to the circuits are auctioneered to ensure that the pump pressure control circuits receive the highest pressure signal. Auctioneering of the two steam generator input signals is required to prevent a low steam generator pressure signal due to a circuit fault or a main steam line break from reducing AFW flow to the intact steam generators. This auctioneering was installed as part of RDC 80-115.

The P-102A differential pressure control circuit receives power from 120-V ac Preferred Instrument Bus Y11. The P-102B differential pressure control circuit receives power from 120-V ac preferred Instrument Bus Y22.

The P-182 differential pressure control circuit compares a preset differential pressure signal to the difference between Steam Generator C pressure and the P-182 pump discharge pressure. An output signal then varies the position of CV-2967 to maintain the preset

differential pressure. PDC-2967 is located in the control room for controlling the differential pressure of P-182.

4.12 FLOW INDICATION

Flow instrumentation is provided for indication of AFW System flow to each steam generator.

4.12.1 DESIGN REQUIREMENTS

(1) Redundant safety-grade AFW System flow indication is required to monitor operation of the AFW System from the control room and the Remote Shutdown Panel (C-160) in accordance with NRC-to-PGE letters dated September 13, 1979 and October 3, 1979 to implement NUREG-0578 recommendations.

(2) AFW System flow instrumentation must be environmentally qualified and powered by Class 1E 120-V preferred instrument ac power supplies.

(3) AFW System flow instrumentation must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

4.12.2 CONFIGURATION

RDC 79-099 provided redundant SCI instrumentation to read combined flow from the A and B trains to each steam generator. FT-3043A through 3043D provide indication at the Remote Shutdown Panel (C-160) and in the control room at Panel C-15. FT-3043E through FT-3043H provide indication on the engineered safety features vertical bench board (Panel C-19). Both channels share the same flow elements.

4.13 REMOTE SHUTDOWN STATION

AFW System controls are provided in the control room. In addition, controls are provided at the Remote Shutdown Panel (C-160) for use should the control room become inaccessible.

4.13.1 DESIGN REQUIREMENTS

(1) GDC 19 requires that equipment be provided at appropriate locations outside the control room so that the plant may be shut down and cooled down. Remote operation of the AFW System is required to bring the plant to cold shutdown if the control room becomes inaccessible.

(2) Panel C-160 must be powered from Class 1E power supplies.

(3) Panel C-160 must meet 10 CFR 50, Appendix R, requirements for separation and operability in the event of a fire.

4.13.2 CONFIGURATION

The Remote Shutdown Panel C-160 provides sufficient controls and indication to allow operation of the AFW System. Local control for AFW Pump P-182 is not available at Panel C-160. In addition, Service Water System (SWS) Isolation Valves M-3045A and M-3045B cannot be operated from C-160. They can be opened manually within prescribed time limits (30 min.) if required (refer to Section 4.9).

C-160 is in a mild environment and is thus exempt from 10 CFR 50.49 qualification (see PGE Drawing E-2).

Table 9-5 lists AFW System controls and instrumentation available at the remote shutdown station, and their functions.

The C-160 panel is presently located in Room 89 (Fire Area T4) on the east side of the 45-ft level of the Turbine Building, which is completely isolated from the Turbine Building general area. According to the Scope Review of RDC 85-053, Appendix R reviewer determined that the C-160 panel could become overheated in the event of a fire in the Turbine Building general area (Fire Area T1). This overheating could possibly lead to the inoperability of both trains of the AFW System. RDC 85-052 will relocate Panel C-160 to the 45-ft level of the Control Building across the hallway from the elevator in the old whole-body counting room to meet Appendix R separation and remote operability requirements.

4.14 AFW PUMP P-182 DISCHARGE ISOLATION VALVES

The two 6-in., motor-operated Gate Valves MO-2947A and MO-2947B form a boundary between SCII (upstream) and SCI (valve and downstream) portions of the AFW System.

4.14.1 DESIGN REQUIREMENTS

(1) MO-2947A and MO-2947B must be qualified SCI per Section 3.2, as required by Position C.1.g of Regulatory Guide 1.29.

(2) MO-2947A and MO-2947B are normally closed to isolate the SCII AFW Pump P-182 discharge to the SCI trains.

4.14.2 CONFIGURATION

The P-182 discharge isolation valves provide isolation between SCI and SCII portions of the AFW System. They are provided with a Class 1E 480-V ac power supply for improved reliability. Controls and indications for operation of the valves are provided in the control room. Only one valve is opened when P-182 is used to supply feedwater to the steam generators.

4.15 AFW PUMP P-182 DIFFERENTIAL PRESSURE CONTROL VALVE

The differential pressure control valve for AFW Pump P-182, CV-2967, operates to maintain a predetermined differential pressure, as compared to the actual differential pressure between Steam Generator C and the P-182 discharge pressure.

4.15.1 DESIGN REQUIREMENTS

(1) CV-2967 must be SCII per Position C.2 Regulatory Guide 1.29, as described in Section 3.2.

(2) CV-2967 must meet piping pressure requirements per Section 3.1.

4.15.2 CONFIGURATION

CV-2967 is a 6-in., air-operated globe valve installed in the discharge piping of P-182. The differential pressure control of CV-2967 is described in Section 4.11. Control room operators can take manual control of CV-2967. The valve may also be locally operated with a manual handwheel, if required.

5.0 SYSTEM OPERATION

The OPERABILITY of the Auxiliary Feedwater (AFW) System ensures that the Reactor Coolant System (RCS) can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power. The OPERABILITY of the condensate storage tank with the minimum usable water volume (196,000 gal.) and maximum temperature (100°F) ensures that sufficient water is available to maintain the RCS under hot standby conditions for 2 hr after a reactor trip, followed by a cooldown to 350°F in 4 hr, with steam discharge to atmosphere, concurrent with a total loss of off-site power.

5.1 NORMAL OPERATION

Under normal plant operating conditions, the AFW System is aligned to allow automatic initiation of flow to the steam generators. The eight flow control valves are in the full open position. Steam header isolation valves are shut, the steam stop valve is open, and the turbine trip and throttle valve is shut. The pump AUTO-MANUAL controllers are in the automatic position with a 100 psid setpoint. The diesel-driven and turbine-driven AFW pump switches are in the automatic position. The maintenance lockout switches and low-CST level block switches on C-160 are in the NORMAL position. The C-160 remote-local selector switches are in the REMOTE position.

5.2 NORMAL TRANSIENT OPERATION

The turbine-driven AFW pump is Train A. Upon a START signal, the four steam supply valves (CV-1451 through CV-1454) and the turbine trip and throttle valve (MO-3071) open to supply steam to the Terry turbine.

The diesel-driven AFW pump is Train B. Upon a START signal, the diesel starts and comes up to speed. The motor-operated Service Water System cooling water supply valve opens to provide flow to the diesel and pump lube oil coolers, the diesel engine jacket cooler, the turbo intercooler, and the speed increaser gear lube oil cooler.

When either Engineered Safety Features (ESF) AFW pump starts, the associated room supply and exhaust fans start to maintain room temperatures.

Normal supply of makeup is from the SCII condensate storage tank. If the normal supply is lost, motor-operated valves are opened from the control room to provide service water as the SCI supply.

A flow indication switch is located upstream of each motor-operated control valve (CV-3004-series valves) in the two supply lines to each steam generator. A high flow in any line trips the FIS to close the respective motor-operated valve and lock the trips from the remaining valves in that train.

During normal startup and shutdown, the electric AFW pump is used to supply feedwater from the CST to the steam generators to maintain desired steam generator levels or to place the steam generator in wet layup. The electric AFW pump is not qualified for design basis events, and it functions to reduce wear on the safety-grade pumps and drivers.

5.3 ABNORMAL AND EMERGENCY OPERATIONS

The ESF AFW System is automatically initiated upon any of the following signals:

(1) Safety injection (SI).

(2) Low-low level on two of three level transmitters in any steam generator.

(3) Undervoltage on 4.16-kV Bus A1 or A2.

(4) Trip of both main feedwater pumps (unless bypassed).

Initiation of AFW isolates the blowdown and sampling lines for each Steam Generator to prevent further water inventory loss.

P-182 can be manually loaded onto 4.16-kV ESF Bus A1 or A2 under emergency conditions.

6.0 INSPECTION AND TESTING

This section summarizes design requirements for the inspection and testing of the AFW System, and describes those inspections and tests performed to verify proper operation of major components to ensure system compliance with the design basis.

6.1 DESIGN INSPECTION AND TESTING REQUIREMENTS

Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves must be performed according to Section XI of ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC according to 10 CFR 50.55a(g)(6)(i).

The AFW Pumps P-102A and P-102B and system valves (from Sections 4.6, 4.7, 4.8, and 4.9) are currently tested according to PGE-1048, Inservice Testing Program for Pumps and Valves, Second Ten-Year Interval. This program is based on requirements called out in Subsections IWP and IWV of Section XI of ASME Boiler and Pressure Vessel Code, 1983 edition and addenda through the summer of 1983. System piping is currently inspected according to PGE-1049, Inservice Inspection Program, Second Ten-Year Interval. This program is also based on meeting the standards of ASME Code Section XI, 1983 edition and addenda through the summer of 1983.

The Trojan inservice inspection and testing programs contained in PGE-1048 and PGE-1049 are implemented according to PGE-8010, Nuclear Quality Assurance Program. Compliance with the applicable requirement of ASME Code Section XI and appropriate addenda is ensured by an inspector as required by Paragraph IWA-2120.

6.2 SYSTEM INSPECTION AND TESTING

Inspection and testing of the AFW System and its components is performed according to PGE-1048 and PGE-1049. The Technical Specifications specify the mode operability requirements, as described in Section 1.3.3, and the surveillance requirements for the AFW System and its components. The surveillance requirements list testing for the following AFW System components:

(1) AFW Pumps P-102A and P-102B (4.7.1.2.1).

(2) Diesel Driver K-107B (4.7.1.2.2).

(3) AFW Pump P-182 (4.7.1.2.3).

(4) CST level requirements (4.7.1.4.1).

(5) Service Water System requirements when supplying the AFW System (4.7.1.4.2).

The testing of the operability of the AFW System is performed according to the following Periodic Operating Tests (POTs):

(1) POT-5-1, Pump and Valve Inservice Test.

(2) POT-5-2, Valve Lineups and Inservice Testing.

(3) POT-5-3, System Performance and Valve Inservice Test.

(4) POT-25-2e, Switches S826 and S841 - K-609 and K-633, Trains A and B (SIS).

The tests conducted according to POT-5-1 satisfy the requirements for the inservice testing of the following AFW components as defined by the Technical Specifications and PGE-1048:

(1) Turbine-driven AFW pump and the associated steam supply check valves.

(2) Diesel-driven AFW pump, lube oil cooler water supply check valves, and associated suction check valves.

(3) Electric AFW pump operability.

The tests conducted according to POT-5-2 satisfy the Technical Specifications requirements verifying AFW System valve lineups and diesel fuel oil level for K-107B, and the inservice testing for time-cycle requirements of the following valves as defined by PGE-1048:

(1) CV-3004A1, B1, C1, and D1.

(2) CV-3004A2, B2, C2, and D2.

(3) MO-3045A and MO-3045B.

(4) MO-3071 and MO-3170.

The tests conducted according to POT-5-3 satisfy requirements from both the Technical Specifications and PGE-1048. One section of the POT satisfies the 18-month surveillance requirement for operating the diesel-driven AFW pump as required by the Technical Specifications. Other sections of the POT satisfy the inservice testing requirements for the turbine-driven AFW pump steam supply check valves and both AFW Pumps P-102A and P-102B suction and discharge valves. Additional testing conducted per POT-5-3 verifies AFW pump design flow, proper position of the AFW pump discharge valves as indicated by position status lamps, and

proper automatic startup of AFW Pumps P-102A and P-102B when a trip is inserted for both main feedwater pumps. Calculations are then performed from data during the previously described tests for pump suction conditions to ensure proper NPSH for the AFW pumps under all operating conditions.

The test conducted according to POT-25-2e verifies that Protection Relays K-609 and K-633 in the Safety Injection System operate properly. This test is a concern for the AFW System, since these relays automatically start AFW Pumps P-102A and P-102B when a Safety Injection Signal is present. These relays are tested to ensure that AFW Pump P-102A starts, that steam supply valves CV-1451 through CV-1454 open, and that AFW Pump P-102B starts.

7.0 DESIGN BASES EVOLUTION

The following Trojan Nuclear Plant modifications affect the configuration of the Auxiliary Feedwater System. A summary of each, with an explanation of its effect on the system design bases (if any), is provided.

7.1 COMPLETED AND CLOSED OUT RDCs

RDC 75-191 SUMMARY: Modified the AFW flow instrument wiring to prevent a ground fault from disabling the indication.

EFFECT OF BASES: None.

RDC 75-276 SUMMARY: Added interlocks to service water valves for AFW turbine and diesel drivers.

EFFECT ON BASES: None.

RDC 76-124 SUMMARY: Replaced AFW turbine wheel due to wear damage and high backpressure problems.

EFFECT ON BASES: None.

RDC 76-164 SUMMARY: Modified the AFW diesel exhaust line from 8 in. to 12 in. to reduce crankcase backpressure.

EFFECT ON BASES: None.

RDC 76-175 SUMMARY: Provided maintenance lockout switches for the ESF AFW pumps.

EFFECT ON BASES: None.

RDC 76-210 SUMMARY: Modified valve lineup for the Terry turbine drain line.

EFFECT ON BASES: None.

RDC 76-217 SUMMARY: Added oil reservoir to the Terry turbine governor actuator.

EFFECT ON BASES: None.

RDC 76-225 SUMMARY: Relocated the governor cabinet for the turbine-driven AFW pump behind Panel C-160 for temperature considerations.

EFFECT ON BASES: None.

RDC 76-226 SUMMARY: Added interlocks for service water valves for AFW turbine and diesel.

EFFECT ON BASES: None.

RDC 76-227 SUMMARY: Installed jacket water heaters on K-107B to improve quick startup capability by maintaining jacket water warm when K-107B is idle.

EFFECT ON BASES: None.

RDC 76-241 SUMMARY: Reinstalled original turbine AFW pump ramp generator/signal converter.

EFFECT ON BASES: None.

RDC 76-261 SUMMARY: Replaced AFW turbine trip and throttle valve relay enclosure with a watertight, corrosion-resistant enclosure.

EFFECT ON BASES: None.

RDC 76-265 SUMMARY: Added drain coolers to K-107A steam line drains to reduce humidity in P-102A pump room. Coolers are supplied cooling water from Service Water System.

EFFECT ON BASES: None.

RDC 76-277 SUMMARY: Added ramp generator to K-107B speed control circuitry to prevent overspeed during diesel startup.

EFFECT ON BASES: None.

RDC 76-278 SUMMARY: Added a soakback pump for Diesel Driver K-107B to improve quick startup capability by circulating lubricating oil when K-107B is idle.

EFFECT ON BASES: None.

RDC 76-285 SUMMARY: Raised elevation of the diesel AFW pump speed increaser auxiliary oil pump to prevent flooding of the motor.

EFFECT ON BASES: None.

RDC 76-293 SUMMARY: Redesigned K-107B diesel annunciator circuitry to indicate when diesel "NOT READY FOR AUTOSTART".

EFFECT ON BASES: None.

RDC 76-345 SUMMARY: Provided cutout for AFW autostart on loss of both main feedwater pumps, without activating maintenance lockout.

EFFECT ON BASES: None.

RDC 76-515 SUMMARY: Provided cutout for AFW autostart on loss of both main feedwater pumps, without activating maintenance lockout.

EFFECT ON BASES: None.

RDC 77-139 SUMMARY: Major modification installed electric motor-driven AFW Pump P-182, installation included:

- P-182 suction piping from CST.
- P-182 discharge piping to the diesel-driven AFW Pump P-102B discharge piping.
- AFW Pump P-182, associated valves, and 1,250-hp ac motor.
- Additional cubicle (A510) and circuit breaker in the 4.16-kV Bus A5 switchgear.
- Local instrumentation, transmitters, and associated sensing lines; instrument air supply lines for P-182 discharge control valve, CV-2967.
- Auxiliary lube oil Pump P-183 and associated pressure switches.
- Local temperature monitoring Panel C-341, for P-182 pump and motor.

EFFECT ON BASES: Implementation established design baseline for AFW Pump P-182. No effect on system design bases.

RDC 77-162 SUMMARY: Installed manometer to monitor AFW diesel crankcase pressure.

EFFECT ON BASES: None.

RDC 78-020 SUMMARY: Installed piping and associated equipment to turbine-driven AFW Pump P-102A to allow cooling of P-102A and K-107A lube oil coolers from P-102A discharge flow. Also removed electrical automatic actuation of MO-3060A, isolating Service Water System cooling water flow from P-102A and K-107A lube oil coolers.

EFFECT ON BASES: This modification enhanced the redundancy design basis for the AFW system, and specifically the electrical independence recommendations of BTP ASB-10 (see Section 3.4), by removing the ac-powered service water booster pumps as a required source of lube oil cooling for operation of P-102A and K-107A.

RDC 79-032 SUMMARY: Added pipe supports for AFW diesel breather tube assembly.

EFFECT ON BASES: None.

RDC 79-043 SUMMARY: Installed flex base on AFW diesel fuel return line to avoid vibration cracking.

EFFECT ON BASES: None.

RDC 79-063 SUMMARY: Added low suction pressure trips for the ESF AFW pumps.

EFFECT ON BASES: None.

RDC 79-064 SUMMARY: Added limit switches for manual valves in the flow paths of the two ESF AFW pumps from the CST to the inlet piping of the Main Feedwater System. Installed indications in the control room and at Panel C-160.

EFFECT ON BASES: None.

RDC 79-099 SUMMARY: Upgraded power supplies for AFW flow instrumentation to Class 1E. Installed redundant, safety-grade flow indicators in the control room to allow monitoring of AFW System flow rate to all four steam generators.

EFFECT ON BASES: Implemented redundancy requirements (see Section 4.12).

RDC 80-003 SUMMARY: Provided for 2-hr operation of AFW Pump P-102A independent of ac power. Included:

- Replacement of motor operators for the K-107A steam inlet valves with air operators (CV-1451 through CV-1454).
- Installation of Solenoid Valves SV-1451 through SV-1454, Air Accumulators T-166A through T-166D, and associated piping, tubing, and supports for the air supply to CV-1451 through CV-1454.

EFFECT ON BASES: Established design basis for the 2-hr operation of the turbine-driven AFW Pump P-102 independent of ac power.

RDC 80-024 SUMMARY: Changed power supply for AFW Pump P-102A discharge pressure transmitter PT-3083A from 120-V Instrument ac Bus Y02 to 120-V Preferred Instrument ac Bus Y11.

EFFECT ON BASES: None.

RDC 80-054 <u>SUMMARY</u>: Installed a guard pipe around AFW Pump P-102A discharge piping located in the P-102B pump room.

<u>EFFECT ON BASES</u>: This modification implemented BTP ASB-10 redundancy recommendations with regard to a high-energy line break of the P-102A discharge piping. It also implemented PGE-1004 analyses for pipe whip and jet impingement.

RDC 80-061 <u>SUMMARY</u>: Installed a second discharge piping connection for AFW Pump P-182 to allow discharge to P-102A discharge piping (as well as to P-102B discharge piping). Also installed check valves and motor-operated gate valves (MO-2947A and MO-2947B) for isolation of P-182 discharge headers from the ESF AFW trains.

<u>EFFECT ON BASES</u>: None.

RDC 80-086 <u>SUMMARY</u>: Installed new battery and battery charger for Diesel Driver K-107B starting system.

<u>EFFECT ON BASES</u>: None. Installed to improve reliability.

RDC 80-115 <u>SUMMARY</u>: Provided a redundant steam generator pressure signal input to each ESF AFW pump differential pressure control circuit. Also, entailed auctioneering circuitry to account for spurious low steam generator pressure signals or main steam line breaks.

<u>EFFECT ON BASES</u>: None. Installed to enhance system reliability.

RDC 80-106 SUMMARY: Modified loss of control power annunciator to actuate "NOT READY FOR START" for AFW diesel and turbine.

EFFECT ON BASES: None.

RDC 81-070 SUMMARY: Added new power feeders for FIS-3004 A1 through D1 and A2 through D2.

EFFECT ON BASES: None.

RDC 81-073 SUMMARY: Replaced AFW flow transmitters with environmentally qualified units.

EFFECT ON BASES: Implemented environmental qualification requirements.

RDC 81-090 SUMMARY: Modified turbine AFW pump tubing to eliminate governor oil overflow problems.

EFFECT ON BASES: None.

RDC 82-016 SUMMARY: Modified AFW Pump P-182 circuitry to provide emergency stop at C-341; installed new lube oil pump start/stop switch at C-341.

EFFECT ON BASES: None.

RDC 82-059 SUMMARY: Provided pipe support modifications to AFW instrument lines.

EFFECT ON BASES: None.

RDC 82-062 SUMMARY: Modified AFW diesel alarm circuitry to annunciate on loss of soakback pump.

EFFECT ON BASES: None.

RDC 82-063 SUMMARY: Modified AFW turbine alarm circuitry to indicate turbine "NOT READY FOR START".

EFFECT ON BASES: None.

RDC 83-014 SUMMARY: Modified pipe supports for AFW diesel cooling jacket drain.

EFFECT ON BASES: None.

RDC 83-025 SUMMARY: Modified control circuitry for MO-3071, MO-3170.

EFFECT ON BASES: None.

RDC 83-046 SUMMARY: Added an enclosure for exposed AFW piping which was not in a vital area enclosure.

EFFECT ON BASES: This modification was implemented for safeguard reasons.

RDC 84-100 SUMMARY: Relocated selected redundant circuit raceways.

EFFECT ON BASES: This modification implemented Appendix R review design requirements.

RDC 84-104 SUMMARY: Replaced eight CV-3004 valves.

EFFECT ON BASES: Implemented environmental qualification requirements.

RDC 84-105 SUMMARY: Replaced eight AFW flow differential pressure switches for the CV-3004 valves.

EFFECT ON BASES: Implemented environmental qualification requirements.

RDC 84-108 SUMMARY: Installed seals on AFW steam inlet valve limit switches and AFW flow transmitters.

EFFECT ON BASES: Implemented environmental qualification requirements.

7.2 RDCs INSTALLED BUT NOT CLOSED OUT

RDC 84-096 SUMMARY: Installed redundant Level Transmitters LT-5201 and LT-5265 on the CST, powered from 120-V preferred instrument ac. Replaced low suction pressure trips for the ESF AFW pumps with low CST level trips for each ESF pump.

EFFECT ON BASES: None. Installed to enhance system reliability.

RDC 86-002 SUMMARY: Removed the common ESF AFW Pump Isolation Valve MD-050 and replaced it with an expansion joint.

EFFECT ON BASES: This modification implemented the design requirement for no manual isolation valves in the common suction piping from the CST to the ESF AFW pumps (see Section 4.5.1) to preclude common mode failure of AFW pump flow. It also implemented an upgrade of the piping from the expansion joint at the CST to the AFW pumps to SCI.

RDC 86-024 SUMMARY: Added a pressure barrier to the diesel AFW pump room.

EFFECT ON BASES: None. Implemented to maintain diesel AFW pump room as a mild EQ environment.

7.3 RDCs INCOMPLETE OR PENDING

RDC 84-044 SUMMARY: Changes scale on AFW flow instrumentation to provide linear indication.

EFFECT ON BASES: None.

RDC 84-091 SUMMARY: Installs decouple switch for P-102A. Modifies circuit design to prevent fire-induced circuit damage.

EFFECT ON BASES: Implements Appendix R requirements to allow circuit operation without fuse replacement.

RDC 85-029 SUMMARY: Provides ATWS mitigation system actuation circuitry (AMSAC).

EFFECT ON BASES: Not yet evaluated.

RDC 85-051 SUMMARY: Changes power supply to AFW flow indication sensing lines heat tracing.

EFFECT ON BASES: None.

RDC 85-052 SUMMARY: Relocates Panel C-160 to the 45-ft level of the Control Building. Installs additional instrumentation at C-160 to upgrade it to a remote shutdown station per Appendix R requirements.

EFFECT ON BASES: When implemented, will redefine design baseline for remote shutdown requirements (see Section 4.13).

RDC 86-003 SUMMARY: Implements various AFW reliability improvements, including:

- Bypass of diesel engine jacket water high temperature trip on autostart (P-102B).
- Resetting thermal overload relay for MO-3060B to maximum current rating allowed by Appendix R requirements (P-102B).
- Removal of one timing relay in P-102B trip circuitry.
- Bypass of thermal overload relays for MO-3071 (P-102A).
- Sealing of Transmitters PT-3083A, PT-3072A, and ATB-701 in P-102A pump room from high-moisture environment.
- Common improvements to replace the AFW pump suction and discharge transmitters.

EFFECT ON BASES: None.

RDC 86-004 SUMMARY: Modifies AFW Pump P-182 to provide common power supply for Auxiliary Lube Oil Pump P-183, local indication of suction pressure, local override of the low suction pressure trip, and local indication for CST level.

EFFECT ON BASES: None.

RDC 86-036 SUMMARY: Installs DC monitor relays for the governor control power circuitry for both K-107A and K-107B. Provides indication to operators if any fuses have blown in the governor control power circuits causing machines to slow down.

EFFECT ON BASES: None.

RDC 86-039 SUMMARY: Replaces AFW turbine steam supply check valves with newer, upgraded-type check valves.

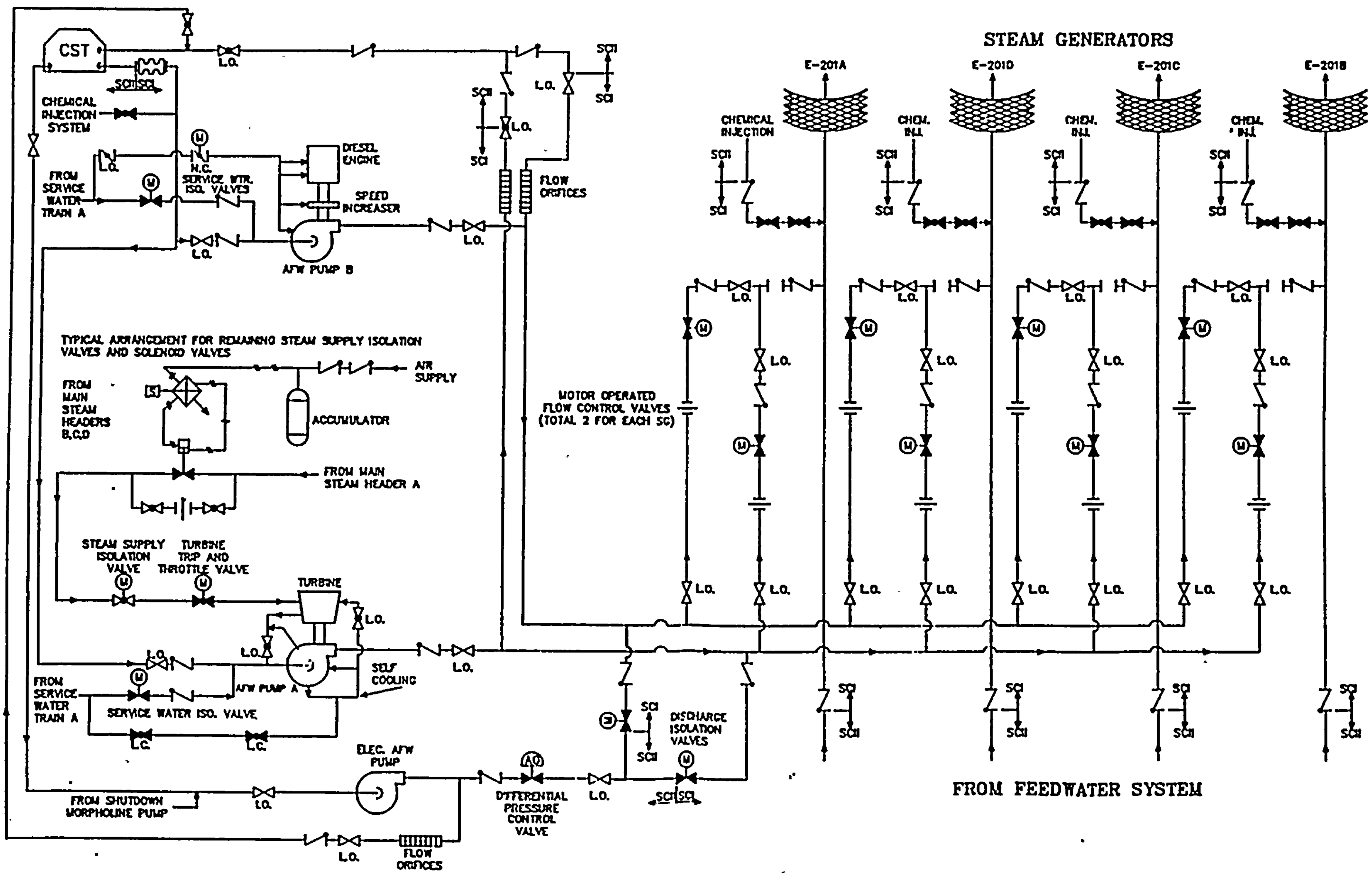
EFFECT ON BASES: None.

RDC 86-043 SUMMARY: Modifies control circuitry for MO-3071 and MO-3170.

EFFECT ON BASES: None.

RDC 86-047 SUMMARY: Installs dc monitor relay coil for K-107A electronic speed monitor.

EFFECT ON BASES: None.



SU045014

FIGURE 8-1
AUXILIARY FEEDWATER SYSTEM

TABLE 9-1
AUXILIARY FEEDWATER PUMPS P-102A AND P-102B
DATA SHEET

Type	Horizontal centrifugal
Flow rate (design)	960 gpm (including 80 gpm recirculation)
Pump casing pressure (design)	2,000 psig
Rated head at design flow rate	3,400 ft
Shutoff head	3,910 ft
NPSH required at design flow rate	25 ft
Minimum NPSH available	26 ft
Suction temperature (normal)	70°F
Speed	4,560 rpm
Discharge pressure at shutoff head	1,700 psig
<u>Lube Oil</u>	
Circulation	3.2 gpm
Temperature at bearing outlet	150°F
Heat transferred to oil	12,000 Btu/hr
Cooling water flow to lube oil cooler	5 gpm
Maximum cooling water inlet temperature	85°F
Reservoir capacity	17.7 gal.

TABLE 9-2
TURBINE DRIVER K-107A

Type	Single-stage, noncondensing
Steam inlet pressure	
Design	125 to 1,320 psia
Operating (normal)	900 to 1,100 psia
Maximum steam inlet temperature	590°F
Backpressure on casing	
Maximum	165 psig
Normal	0 psig
Steam moisture condition (maximum)	1/2%
Rated bhp (at 1,305 psig and 580°F steam inlet temperature)	1,045
Rated speed	4,560 rpm
Speed operating range	2,660-4,560 rpm
Electrical trip speed	5,100 rpm
Mechanical trip speed	5,500 rpm

TABLE 9-3
DIESEL DRIVER K-107B
DATA SHEET

Rated bhp	1,579
Rated speed	1,200 rpm
Operating speed range	450 to 1,200 rpm
Diesel engine overspeed trip	1,350 rpm
Diesel fuel type	No. 2 diesel fuel
Diesel fuel oil day tank volume	500 gallons
Normal oil pressure	45 ± 5 psig

<u>Service Water System Flow Rates to K-107B (Design)</u>

Diesel engine jacket cooler	230 gpm
Speed increaser lube oil cooler	15 gpm
Diesel engine lube oil cooler and intercooler	67 gpm

<u>Speed increaser</u>

Ratio (rpm)	1:3.8
Lube oil flow	16.75 gpm

<u>Lube oil cooler temperatures</u>

Cooling water flow (inlet/outlet)	90°F/98°F
Lube oil flow (inlet/outlet)	150°F/132.5°F

<u>Flow orifice</u>

Differential pressure (maximum)	1,470 psid
Flow (maximum)	150 gpm

TABLE 9-4
AFW PUMP P-182
DATA SHEET

Flow rate (design)	1,020 gpm (including 140 gpm recirculation)
Speed	3,560 rpm
Head at design flow rate	3,400 ft
NPSH required at design flow rate	20 ft
NPSH available	25 ft
Pump casing pressure (design)	2,000 psig
Suction temperature (design)	90°F
Cooling water flow to lube oil cooler	10 gpm
Maximum cooling water inlet temperature	90°F
<u>Driver</u>	
Type	Electric motor, 1,250 hp
Motor service factor	1.15
Speed	3,560 rpm
<u>Pump trips</u>	
Low oil pressure	<3 psig
Low suction pressure	<12 psia

TABLE 9-5
AUXILIARY FEEDWATER SYSTEM INSTRUMENTATION
AT REMOTE SHUTDOWN STATION (C-160)

Instrument	Function
Train A	
Controls	- Manual/auto control for P-102A - Manual/auto control for CV-1451 through CV-1454 - Manual control of CV-3004A1, B1, C1, D1 - Manual differential pressure control for P-102A
Indication	- Position indication for valves operated from RSS - Position indication for manual AFWS valves - P-102A suction and discharge pressure - K-107A steam supply and exhaust pressure - P-102A/SG differential pressure - K-107A operating speed - CST level - SG pressure - SG level - AFW flow to SGs
Train B	
Controls	- Manual/auto control for P-102B - Manual control for CV-3004A2, B2, C2, D2 - Manual differential pressure control for P-102B
Indication	- Position indication for valves operated from RSS - Position indication for manual AFWS valves - P-102B suction and discharge pressure - P-102B/SG differential pressure - K-107B operating speed - K-107B diesel fuel oil day tank level - CST level - SG pressure - SG level - AFW flow to SGs

10.0 REFERENCES

10.1 CODES, STANDARDS, AND REGULATORY REFERENCES

10.1.1 CODES AND STANDARDS

10.1.1.1 ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, Division I, 1968 and Addenda through 1971.

10.1.1.2 ASME B&PV Code, Section III, Class 2, Nuclear Power Plant Components, Containment Penetrations, 1971 and Addenda through summer of 1971.

10.1.1.3 ASME B&PV Code Section XI 1977 Edition, IWA-7210, and 1983 edition and Addenda through summer of 1983.

10.1.1.4 Draft ASME Code for the Inservice Testing of Pumps in Nuclear Plants, April 1970.

10.1.1.5 Draft ASME Code for the Inservice Testing of Valves in Nuclear Plants, June 1970.

10.1.1.6 ANSI B31.1.0, Power Piping Code, 1973.

10.1.1.7 ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.

10.1.1.8 IEEE 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations.

10.1.1.9 IEEE 308-1971, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

10.1.1.10 IEEE 323-1971, Trial Use Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations.

10.1.1.11 IEEE 323-1974, Standard for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations.

10.1.1.12 IEEE 338-1977, Standards for the Periodic Testing of Nuclear Power Generating Station Safety Systems.

10.1.1.13 IEEE 344-1971, Guide for Seismic Qualification of Class 1E Electrical Equipment for Nuclear Power Generating Stations.

10.1.1.14 IEEE 344-1975, Recommended Practices for Seismic Qualification of Class 1E Electrical Equipment for Nuclear Power Generating Stations.

10.1.1.15 NEMA Standard SM-22-1970, Single-Stage Steam Turbine for Mechanical Drive Service.

10.1.2 REGULATORY REFERENCES

10.1.2.1 Code of Federal Regulations, Title 10 (10 CFR).

10.1.2.1.1 10 CFR 50.48, Fire Protection.

10.1.2.1.2 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.

10.1.2.1.3 10 CFR 50.55a, Codes and Standards.

10.1.2.1.4 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

10.1.2.1.5 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plant and Fuel Reprocessing Plants.

10.1.2.1.6 10 CFR 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, Sections III.G, III.J, and III.L.

10.1.2.1.7 10 CFR 100, Reactor Site Criteria.

10.1.2.2 USNRC Regulatory References.

10.1.2.2.1 USNRC Regulatory Guide 1.29, Seismic Design Qualification (Rev. 3), September 1978.

10.1.2.2.2 USNRC Regulatory Guide 1.30, Quality Assurance Requirements of the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment, August 1972.

10.1.2.2.3 USNRC Regulatory Guide 1.62, Manual Initiation of Protective Actions, October 1973.

10.1.2.2.4 USNRC Regulatory Guide 1.75, Physical Independence of Electric Systems (Rev. 2), September 1978.

10.1.2.2.5 USNRC Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, November 1974.

10.1.2.2.6 USNRC Regulatory Guide 1.93, Availability of Electric Power Sources, December 1974.

10.1.2.2.7 USNRC Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (Rev. 3), May 1983.

10.1.2.2.8 USNRC Regulatory Guide 1.100, Seismic Qualification of Electrical Equipment for Nuclear Power Plants (Rev. 1), August 1977.

10.1.2.2.9 USNRC Regulatory Guide 1.106, Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1), March 1977.

10.1.2.2.10 USNRC Regulatory Guide 1.117, Tornado Design Classification (Rev. 1), April 1978.

10.1.2.2.11 NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, Section 2.1.7, July 1979.

10.1.2.2.12 NUREG-0588, (Category 1), Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

10.1.2.2.13 NUREG-0611, Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants, January 1980.

10.1.2.2.14 NUREG-0737, Clarification of TMI Action Plan Requirements, Section II.E.1.2, November 1980.

10.1.2.2.15 NUREG-0800, Standard Review Plan, Section 10.4.9, Auxiliary Feedwater System (PWR), (Rev. 2), July 1981.

10.1.2.2.16 USNRC Branch Technical Position (BTP) APCSB 10.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976", August 23, 1976.

10.1.2.2.17 USNRC Branch Technical Position (BTP) ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants".

10.1.2.2.18 USNRC Division of Operating Reactors (DOR) "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors", 1979.

10.1.2.2.19 USNRC IE Bulletin No. 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts, March 8, 1979.

10.1.2.2.20 USNRC IE Bulletin No 79-04, Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation, March 30, 1979.

10.1.2.2.21 USNRC IE Bulletin No. 79-07, Seismic Stress Analysis of Safety-Related Piping, April 14, 1979.

10.1.2.2.22 USNRC IE Bulletin No. 79-14, Seismic Analyses for As-Built Safety-Related Piping Systems, July 2, 1979.

10.1.2.2.23 USNRC IE Bulletin No. 80-11, Masonry Wall Design, May 8, 1980.

10.1.2.2.24 USNRC IE Information Notice No. 80-21, Anchorage and Support of Safety-Related Electrical Equipment.

10.1.2.2.25 USNRC IE Information Notice No. 84-54, Deficiencies in Design Base Documentation and Calculations Supporting Nuclear Power Plant Design.

10.2 PGE INTERNAL DOCUMENTS AND TECHNICAL MANUALS

10.2.1 PGE/TROJAN REPORTS

10.2.1.1 Trojan Updated Final Safety Analysis Report (UFSAR), Docket No. 50-344.

10.2.1.2 PGE-1004, Trojan Nuclear Plant Analyses of Pipe System Breaks, Outside Containment.

10.2.1.3 PGE-1012, Trojan Nuclear Plant Fire Protection Plan.

10.2.1.4 PGE-1022, Inservice Testing Program for Pumps and Valves.

10.2.1.5 PGE-1025, Trojan Nuclear Plant Environmental Qualification Program Manual.

10.2.1.6 PGE-1028, Regulatory Guide In House Positions.

10.2.1.6.1 IHP 1.28-2-1, Regulatory Guide 1.28, Quality Assurance Program (Design and Construction), December 31, 1981.

10.2.1.6.2 IHP 1.29-3-1, Regulatory Guide 1.29, Seismic Design Qualification, September 1, 1982.

10.2.1.6.3 IHP 1.30-0-2, Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment, September 1, 1982.

10.2.1.6.4 IHP 1.32-2-1, Regulatory Guide 1.32, Criteria for Safety-Related Electrical Power Systems for Nuclear Power Plants, September 1, 1982.

10.2.1.6.5 IHP 1.59-2-1, Regulatory Guide 1.59, Design Basis Floods for Nuclear Power Plants, September 1, 1982.

10.2.1.6.6 IHP 1.60-1-1, Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants, September 1, 1982.

10.2.1.6.7 IHP 1.62-0-1, Regulatory Guide 1.62, Manual Initiation of Protective Actions, September 1, 1982.

10.2.1.6.8 IHP 1.64-2-2, Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, September 1, 1982.

10.2.1.6.9 IHP 1.75-2-2, Regulatory Guide 1.75, Physical Independence of Electric Systems, December 31, 1984.

10.2.1.6.10 IHP 1.76-0-1, Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, September 1, 1982.

10.2.1.6.11 IHP 1.89-1-1, Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, March 1986.

10.2.1.6.12 IHP 1.93-0-1, Regulatory Guide 1.93, Availability of Electric Power Sources, September 1, 1982.

10.2.1.6.13 IHP 1.97-3-1, Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, December 31, 1984.

10.2.1.6.14 IHP 1.100-1-2, Regulatory Guide 1.100, Seismic Qualification of Electrical Equipment for Nuclear Power Plants, December 31, 1984.

10.2.1.6.15 IHP 1.102-1-1, Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants, September 1, 1982.

10.2.1.6.16 IHP 1.106-1-1, Regulatory Guide 1.106, Thermal Overload Protection for Electric Motors on Motor-Operated Valves, September 1, 1982.

10.2.1.6.17 IHP 1.115-1-1, Regulatory Guide 1.115, Protection Against Low-Trajectory Turbine Missiles, December 31, 1981.

10.2.1.6.18 IHP 1.117-1-1, Regulatory Guide 1.117, Tornado Design Classification, September 1, 1982.

10.2.1.6.19 IHP 1.124-1-1, Regulatory Guide 1.124, Service Limits and Loading Combinations for Class 1E Linear-Type Component Supports, September 1, 1982.

10.2.1.7 PGE-1043, Accident Monitoring Instrumentation Review for the Trojan Nuclear Plant.

10.2.1.8 PGE-1048, Inservice Testing Program for Pumps and Valves, Second Ten-Year Interval.

10.2.1.9 PGE-1049, Inservice Inspection Program, Second Ten-Year Interval.

10.2.1.10 PGE-8010, Nuclear Quality Assurance Program.

10.2.1.11 Trojan Safety-Related List.

10.2.2 TECHNICAL MANUAL REFERENCES

10.2.2.1 M12-54, Auxiliary Steam Generator Feedwater Pumps P-102A and P-102B (Rev. 4). Bingham-Willamette Company.

10.2.2.2 M12-55, Auxiliary Steam Generator Feedwater Pump Diesel Engine Driver K-107B (Rev. 1). Waukesha Motor Company.

10.2.2.3 M12-61, Auxiliary Steam Generator Feedwater Pump Turbine Driver K-107A (Rev. 1). Terry Steam Turbine Company.

10.2.2.4 N14440-10, Electric Auxiliary Feedwater Pump P-102, (Rev. 2). Bingham-Willamette Company.

10.2.3 TROJAN PLANT OPERATING MANUAL (POM) REFERENCES

10.2.3.1 Off-Normal Instruction (ONI)-55, Operation of Electric AFP Supplied by EDG.

10.2.3.2 Plant Operating Test (POT) 5-1, Pump and Valve Inservice Test.

10.2.3.3 POT-5-2, Valve Lineups and Inservice Testing.

10.2.3.4 POT-5-3, System Performance and Valve Inservice Test.

10.2.3.5 POT-25-2e, Switches S826 and S841 - K609 and K633 Trains A and B (SIS).

10.2.3.6 Plant Operating Instruction, OI-8-2, Auxiliary Feedwater.

10.3 PGE/TROJAN PRINTS, DRAWINGS, AND SPECIFICATIONS

10.3.1 SPECIFICATIONS

10.3.1.1 Trojan Nuclear Plant Technical Specifications.

10.3.1.2 PGE Electrical Design Guide No. 1.

10.3.1.3 E-2, Electrical Equipment Environmental Qualification.

10.3.1.4 TM-014 Auxiliary Feedwater Pump P-182 and Motor.

10.3.2 PIPING AND INSTRUMENTATION DIAGRAMS (P&IDs)

10.3.2.1 M-208 (Rev. 28), Main Steam System.

10.3.2.2 M-213 Sheet 1 (Rev. 30), Condensate and Feedwater System.

10.3.2.3 M-213 Sheet 2 (Rev. 8), Condensate and Feedwater System.

10.3.2.4 M-213 Sheet 4 (Rev. 0), Condensate and Feedwater System.

10.3.2.5 M-214 Sheet 1 (Rev. 37), Auxiliary Steam System.

10.3.2.6 M-218 Sheet 1 (Rev. 33), Service Water System.

10.3.2.7 M-226 (Rev. 18), Diesel Fuel Oil System.

10.3.2.8 M-228 (Rev. 33), Makeup Water Treatment System.

10.3.3 LOGIC DIAGRAMS

10.3.3.1 E-1133, Auxiliary Feedwater Pump P-102A Channel "A".

10.3.3.2 E-1133A, Auxiliary Feedwater Pump P-102A Channel "A".

10.3.3.3 E-1134, Auxiliary Feedwater Pump P-102B Channel "B".

10.3.3.4 E-1134A, Auxiliary Feedwater Pump P-102B Channel "B".

10.3.3.5 E-1150, Motor Operated Valves.

10.3.3.6 E-1159, Steam Turbine Driven Auxiliary Feedwater Pump Steam Inlet Valves.

10.3.3.7 E-1183, Service Water System Motor Operated Valves.

10.3.3.8 E-1266, Heating and Ventilation Fans, Auxiliary Feedwater Pump Room.

10.3.3.9 E-1266A, Heating and Ventilation Fans, Auxiliary Feedwater Pump Room.

10.3.3.10 E-1285, Auxiliary Feedwater Pump Valves.

10.3.3.11 E-2300, Start-up Auxiliary Feedwater Pump P-182.

10.3.3.12 E-2316, Start-up Auxiliary Feedwater Pump P-182 Discharge Valves.

10.3.4 ELECTRICAL PRINTS

10.3.4.1 E-1 Plant Single Line Diagram (Rev. 14).

10.3.4.2 E-32 12.47-kV System (Rev. 8).

10.3.4.3 E-33 4.16-kV System, Sheet 1 (Rev. 12).

10.3.4.4 E-34 4.16-kV System, Sheet 2 (Rev. 11).

10.3.4.5 E-35 480-V Load Centers, Sheet 1, Engineered Safety Feature System (Rev. 6).

10.3.4.6 E-37 480-V Motor Control Centers, Sheet 1, Engineered Safety Features System (Rev. 21).

10.3.4.7 E-38 480-V Control Centers, Sheet 2, Engineered Safety Feature System (Rev. 20).

10.3.4.8 E-44 125-V, 250-V dc, and 208/120-V Instrument and Preferred ac System (Rev. 19).

10.3.4.9 E-44A 125-V, 250-V dc, and 208/120 Instrument and Preferred ac System (Rev. 0).

10.3.4.10 E-45 120-V Preferred ac Panels Y11, Y13, Y22, and Y24.

Sheet No.	Rev.
1 of 13	12
2 of 13	15
5 of 13	13
7 of 13	13
11 of 13	0
12 of 13	0

10.3.4.11 E-47 125-V dc Distribution Panels D10, D20, D30, and D40.

Sheet No.	Rev.
3 of 14	7
6 of 14	8
9 of 14	9
12 of 14	7

10.3.4.12 E-54 480-V Motor Control Centers, Sheet 3, Engineered Safety Features System (Rev. 24).

10.3.4.13 E-55 480-V Motor Control Centers, Sheet 4, Engineered Safety Features System (Rev. 18).

10.4 VENDOR REPORTS, SPECIFICATIONS, AND MISCELLANEOUS REFERENCES

10.4.1 Westinghouse Commercial Atomic Power WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems.

10.4.2 WCAP-7898, Long Term Transient Analysis for PWRs.

10.4.3 WCAP-7909, MARVEL - A Digital Computer Code for Transient Analysis for a Multiloop PWR System.

10.4.4 Franklin Research Center Technical Evaluation Report (TER) C5257-296, Auxiliary Feedwater System Automatic Initiation and Flow Indication (Trojan Nuclear Plant), March 3, 1982.

10.4.5 Bechtel PC Topical Report BN-TOP-2, Design for Pipe Break Effects.

10.4.6 Bechtel Specification M-12, SCI Auxiliary Feedwater Pumps, Drivers, and Auxiliaries.

10.4.7 Bechtel Specification M-301, Specification for Piping Materials and Standard Details, Rev. 1.

10.4.8 Bechtel Specification M-541, Sheets 11, 12, 13, Control Valve Data Sheets, Rev. 4.

10.4.9 Bechtel Specification M-545-1, Pressure Safety Valve Data Sheet, Rev. 2.

10.4.10 Bechtel Specification M-12-53, Auxiliary Feedwater Pump Diesel Engine Connection Diagram, Rev. 6.

10.5 CORRESPONDENCE REFERENCES

10.5.1 VENDOR CORRESPONDENCE

10.5.1.1 Westinghouse Letter to Bechtel, POR-113, June 3, 1969, Auxiliary Feed Pumps.

10.5.1.2 Westinghouse Letter to Bechtel, POR-121, June 4, 1969, Auxiliary Feed Pumps.

10.5.1.3 Westinghouse Letter to Bechtel, POR-151, July 22, 1969, Plant Startup and Shutdown.

10.5.1.4 Westinghouse Letter to Bechtel, POR-519, August 6, 1970, Auxiliary Feedwater System.

10.5.1.5 Westinghouse Letter to Bechtel, POR-728, March 29, 1971, Condensate Storage.

10.5.1.6 Westinghouse Letter to Bechtel, POR-751, April 14, 1971, Auxiliary Feedwater Initiation Time.

10.5.1.7 Westinghouse Letter to Bechtel, POR-1472, October 20, 1972, Steam Systems Design Manual Transmittal.

10.5.1.8 Westinghouse Letter to Bechtel, POR-1613, February 14, 1973, Auxiliary Feedwater System Pump Capacities and Assured Storage Quality.

10.5.1.9 Westinghouse Letter to Bechtel, POR-1710, April 25, 1973, Auxiliary Feedwater System.

10.5.1.10 Westinghouse Letter to Bechtel, POR-1888, September 27, 1973, Auxiliary Feedwater System.

10.5.1.11 Westinghouse Letter to Bechtel, POR-2111, June 11, 1974, Auxiliary Feedwater System.

10.5.1.12 Westinghouse Letter to PGE, POR-85-579, May 24, 1985, Condensate Storage Tank Volume Requirements.

10.5.1.13 Bechtel Letter to PGE, BP-8650, November 1, 1977, RDC 77-139, Startup Auxiliary Feedwater Pump.

10.5.1.14 Bechtel Letter to PGE, BP-10485, July 11, 1980, RDC 80-003, Auxiliary Feedwater ac Independence.

10.5.1.15 Bechtel Letter to PGE, BP-10570, August 20, 1980, Motor-Driven AFWP Separate Flow Path.

10.5.1.16 Bechtel Letter to PGE, BP-12373, February 24, 1986, CST Level Requirements.

10.5.1.17 Bechtel Letter to PGE, BP-12422, March 27, 1986, AFW Suction Piping Analysis.

10.5.1.18 Bechtel Telephone Call to PGE, December 30, 1986, Verification of AFW Setpoints and Operation.

10.5.2 REGULATORY CORRESPONDENCE

10.5.2.1 Trojan Nuclear Plant Safety Evaluation Report (SER), October 1974, Initial SER of Trojan Nuclear Plant.

10.5.2.2 NRC Letter to PGE, March 9, 1978, SER, Amendment No. 22 to NPF-1.

10.5.2.3 NRC Letter to PGE, October 3, 1979, NRC Requirements for Auxiliary Feedwater System at Trojan Nuclear Plant.

10.5.2.4 PGE Letter to NRC, October 17, 1979, Response to NRC to PGE Letter Dated September 13, 1979.

10.5.2.5 PGE Letter to NRC, November 26, 1979, Response to NRC to PGE Letter Dated October 3, 1979.

10.5.2.6 NRC Letter to PGE, September 13, 1979, Follow-up Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident.

10.5.2.7 PGE Letter to NRC, December 31, 1979, Response to October 3, 1979 NRC Questions Concerning the Trojan Auxiliary Feedwater System.

10.5.2.8 NRC Letter to PGE, March 25, 1980, SER, Auxiliary Feedwater System Requirements.

10.5.2.9 NRC Letter to PGE, May 14, 1980, Auxiliary Feedwater System Requirements.

10.5.2.10 PGE Letter to NRC, July 25, 1980, Response to NRC to PGE Letter Dated May 15, 1980.

10.5.2.11 NRC Letter to PGE, October 23, 1980, SER, Implementation of Recommendations for Auxiliary Feedwater Systems.

10.5.2.12 PGE Letter to NRC, December 12, 1980, Supplemental Response to SER Dated October 23, 1980.

10.5.2.13 NRC Letter to PGE, February 18, 1981, SER Supplement, Implementation of Recommendations for the Auxiliary Feedwater System.

10.5.2.14 NRC Letter to PGE, August 4, 1981, SER, Seismic Qualification of the Auxiliary Feedwater System.

10.5.2.15 NRC Letter to PGE, April 8, 1982, NUREG-0737 Item II.E.1.2, Auxiliary Feedwater System Automatic Initiation and Flow Indication.

10.5.2.16 NRC Letter to PGE, May 5, 1982, Testing of the Motor-Driven Auxiliary Feedwater Pump.

10.5.3 INTERNAL CORRESPONDENCE

10.5.3.1 PGE Telephone Call to Westinghouse, DIH-T22-80, September 5, 1980, Westinghouse Design Criteria for Auxiliary Feedwater Systems.

10.5.3.2 Memorandum ELD-03-82M, December 21, 1982, 125-V dc Battery Capacity.

10.5.3.3 Memorandum RLS-1125-82M, December 16, 1982, Alternating Current Independence of Turbine-Driven AFW Pump Limitations due to Battery Capacity of Inverter Failure.

10.5.3.4 Memorandum RLS-03-83M, January 1, 1983, Recommended Procedure Change to Maintain Operable A Train 125-V dc Battery.

10.5.3.5 Memorandum CPY-443-83, June 15, 1983, Documentation of NPE-Recommended Procedure Changes.

10.5.3.6 Memorandum ANR-172-84M, A. N. Roller to J. W. Lentsch, July 11, 1984, Verification Program Disposition Form Auxiliary Feedwater System Evaluation, NUREG-0737, Item II.E.1.1.

10.5.3.7 Memorandum RLS-1072-84M, R. L. Steele to J. W. Lentsch September 28, 1984, IE Information Notice No. 84-54.

10.5.3.8 Memorandum RLS-732-85M, R. L. Steele to J. W. Lentsch, June 5, 1985, Trojan Nuclear Plant Condensate Storage Tank (CST) Minimum Volume.

10.5.3.9 Memorandum RLS-453-86M, April 16, 1986, Auxiliary Feedwater System Reliability Improvements.

10.5.3.10 Memorandum JWL-297-86M, May 28, 1986, Minimum Required CST Volume.

10.5.3.11 Memorandum RLS-691-86M, R. L. Steele to W. S. Orser, June 24, 1986, CST Level Instrumentation and STS Minimum Volume.

10.6 CALCULATION REFERENCES

10.6.1 PGE CALCULATIONS

10.6.1.1 NSRD Calculation No. TNP-86-27, AFW Flow Rate at End of CST Design Basis Volume, November 14, 1986.

10.6.1.2 NPEEB Calculation No. TE-028, 125-V dc Station Battery Capacity, November 10, 1982 (superseded).

10.6.1.3 NPEEB Calculation No. TE-029, Station Battery Required Load Shed During Station Blackout (superseded).

10.6.1.4 NPEEB Calculation No. TE-105, 120-V ac Preferred Instrument Bus.

10.6.1.5 NPEEB Calculation No. TE-115, AFW Pump/CST Level Control Setpoints, July 24, 1986.

10.6.1.6 NPEEB Calculation No. TE-119, Station Battery Capacity, September 11, 1986.

10.6.1.7 NPEMB Calculation No. TM-002, AFW Pump Overspeed Trip Test Discharge Pressure, January 10, 1981.

10.6.1.8 NPEMB Calculation No. TM-006, AFW System Steam Pressure, January 10, 1981.

10.6.1.9 NPEMB Calculation No. TM-082, AFW Panel Room Ventilation, February 14, 1983.

10.6.1.10 NPEMB Calculation No. TM-158, AFW Reliability/CST Suction Piping, June 19, 1986.

10.6.1.11 NPEMB Calculation No. TM-185, Stress Analysis of AFW Turbine Exhaust Drain Line, November 13, 1986.

10.6.1.12 NPEMB Calculation No. TM-191, Seismic Analysis of AFW Pump Lube Oil Drain Piping, December 16, 1986.

10.6.1.13 NPECB Calculation No. TC-055, AFW Pump Foundation Design, November 5, 1985.

10.6.1.14 NPECB Calculation No. TC-258, FIS 3004 A1-D2 Rigid Mounting Configuration, Rev. 1, June 5, 1985.

10.6.1.15 NPECB Calculation No. TC-265, Seismic Qualification of CV-3004 Replacement Valve Operators, February 9, 1985.

10.6.1.16 NPECB Calculation No. TC-296, Pipe Support Modifications, September 11, 1985.

10.6.1.17 NPECB Calculation No. TC-335, Condensate Storage Tank Evaluation, April 17, 1986.

10.6.1.18 NPECB Calculation No. TC-336, CST Level Transmitter Enclosures, April 25, 1986.

10.6.1.19 NPECB Calculation No. TC-348, AFW Pump Reliability Improvements, June 2, 1986.

10.6.1.20 NPECB Calculation No. TC-378, AFW Room Exhaust Flappers, October 23, 1986.

10.6.1.21 NPECB Calculation No. TC-380, C-160 Room Exhaust Ductwork, October 28, 1986.

10.6.1.22 NPECB Calculation No. TC-399, AFW Ductwork, December 15, 1986.

10.6.1.23 Accumulator Sizing Calculation (RDC 80-003), March 31, 1981.

10.6.2 BECHTEL CALCULATIONS

10.6.2.1 12-22, Fuel Oil Day Tank T-152 Sizing, September 26, 1975.

10.6.2.2 16-14, AFW Pump Turbine Drain System Flow Restrictions, December 28, 1976.

10.6.2.3 16-17, AFW Pump Turbine Drain Condenser Sizing, April 29, 1977.

10.6.2.4 16-42, AFW Pump NPSH Available, June 23, 1978.

10.6.2.5 16-43, P-182 Low Suction Pressure Trip Setpoint, October 9, 1978.

10.6.2.6 16-44, Minimum CST Level for AFW Pump NPSH Requirements, July 3, 1979.

10.6.2.7 16-47, NPSH Available for Two AFW Pump Operation, April 22, 1980.

10.6.2.8 16-48, AFW Isolation Following Main Steam Line Break, June 30, 1980.

10.6.2.9 16-49, Maximum AFW Flow to Faulted Steam Generator, March 26, 1982.

10.6.2.10 16-50, Safety-Related MOV Design Basis, November 1, 1986.

10.6.2.11 16-52, AFW System Pressure due to Pump Overspeed.

10.6.2.12 AFW Pumps, Check AFWP Sizing in Relation to As-Built System, September 17, 1969.

APPENDIX A
OPEN ITEMS

The following items require some level of resolution to be incorporated as part of the design basis for the Auxiliary Feedwater system.

1. The assumed maximum steady state flow rate for internal flooding for the AFW system needs verification. Design flow rate for each Circulating Water System train is 210,000 gpm; but some documents imply an assumed flooding rate of 500,000 gpm. (Section 3.7.1)

2. The design basis for piping pressure and temperature ratings throughout the system needs further explanation. Specification M-301 is referenced for actual pressure and temperature ratings, but the functional basis for selection of required parameters for each piping spool is not available. Analysis of piping classification code schedules, system P&IDs, and some walkdowns are necessary to resolve. (Section 3.1)

3. The following Bechtel calculations are needed to support design basis information:

   12-22, Fuel Oil Day Tank Sizing, 9/26/75.
   16-14, AFW Pump Turbine Drain System Flow Restrictions, 12/28/76
   16-17, AFW Pump Turbine Drain Condenser Sizing, 4/29/77
   16-49, Maximum AFW Flow to Faulted SG, 3/26/82.

4. The design margin included in AFW System flow rate requirements is not available. Accident analyses assume a minimum available flow of 426 gpm, while Westinghouse-supplied design criteria (and, specifically, Westinghouse letters to Bechtel POR-113, dated June 3, 1969, and POR-121, dated June 4, 1969) require a minimum flow of 440 gpm. Any basis for the margin between these two figures, or for any margin included in the accident analysis flow rate of 426 gpm, should be provided by Westinghouse. (Section 4.1.3)

5. Calculations supporting the basis for the change in September 1986 to the K-107A trip set points need verification. Trojan Plant Engineering performed calculations to support the change, but these have not been incorporated into NPE or NSRD controlled calculation files. (Section 4.2.2)

6. Diesel driver K-107B design horsepower is known, but an evaluation is needed to determine actual pump horsepower required for proper operation of P-102B at rated speed. (Section 4.3.2)

7. A calculation is needed (Bechtel Calculation 12-22, Fuel Oil Day Tank Sizing, 9/26/75, may suffice when received) to verify the sizing criteria for diesel fuel oil day tank T-152. This is necessary to confirm that the tank capacity of 500 gallons and operability requirement of 450 gallons are based on K-107B fuel consumption requirements. (Sections 4.3.2, 4.3.3)

8. High AFW flow in a faulted branch line will result in isolation of that branch at 500 gpm. NQAD has expressed a concern that the basis for this flow rate selection be verified. Considering the AFW pump rating of 880 gpm, less 500 gpm lost through the faulted branch line, would leave only 380 gpm deliverable to the steam generators. This flow rate would be less than the 426 gpm assumed in the worst-case accident analyses. Bechtel Calculation 15-48, AFW Isolation Following Main Steam Line Break, June 30, 1980, does not resolve this item, and further evaluation is needed. Bechtel Calculation 15-49, Maximum AFW Flow to Faulted SG, March 26, 1982, may be germane when received. (Section 4.6.2)

9. Westinghouse letter POR-1613 to Bechtel dated February 14, 1973, and Curve SSE-1119 established the minimum usable CST volume requirement of 196,000 gallons. However, no Westinghouse calculations are available regarding the basis for determining this value. Required CST volume assuming rated AFW pump flow for two hours at hot standby followed by four hours of cooldown far exceeds 196,000 gallons. It would appear that the average flow required over the six hour period does not exceed 544 gpm. An evaluation is needed to determine assumptions for flow requirements at various stages during plant cooldown. NSRD calculation TNP-86-27, AFW Flow Rate at End of CST Design Basis Volume, dated November 14, 1986, refers to this item but does not resolve it. (Section 5.0)

10. Diesel driver K-107B battery sizing design basis needs to be verified. RDC 80-086 is pertinent. (Section 4.3.2)

11. Westinghouse letter POR-751 to Bechtel dated April 14, 1971, established a requirement that the AFW pumps be operating at rated speed and delivering rated flow within one minute following actuation of any automatic signal which starts the pumps. The design basis assumptions or calculations supporting the figure of one minute are needed. Also needed are any margin calculations or assumptions included. (Section 1.3)

12. Design basis for the 100 psid set point for AFW pump differential pressure control is unavailable. Determination would require Westinghouse input. (Section 4.11.2)

13. Design margins for cooling water to AFW pumps P-102A and P-102B auxiliaries are unavailable. Resolution will require pump vendor input. (Section 4.1.3)

14. Basis for diesel driver K-107B lube oil cooler cooling water requirements are unavailable. Resolution will require Waukesha input. (Section 4.3.2)

15. Design basis, if any, for AFW pump P-182 flow rate and recirculation flow rate characteristics are unavailable. (Section 4.4.2)

16. A calculation is needed to support the design basis for sizing of accumulators T-166A through T-166D. (Section 4.7.1.1)

17. Design basis for valve MO-3170 stroke time is needed. (Section 4.7.1.2)