TUELECTRIC

Log # TXX-92114
File # 10112 (clo & Att 2)
10250 (clo & Att 1)
Ref. # 10CFR50.59(b)(2)
10CFR50.36

William J. Cahill, Jr.
Group Vice President

February 28, 1992

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)-UNIT 1
DOCKET NO. 50-445
ANNUAL OPERATING REPORT AND ANNUAL 10CFR50.59 REPORT

Gentlemen:

Attacnment 1 is the second Annual Operating Report prepared and submitted pursuant to Specification 6.9.1.2 of Appendix A (Technical Specifications) to the Comanche Peak Steam Electric Station Operating License Unit 1, NPF-87. This attachment also complies with the annual operating report guidance provided in position C.1.b of U.S. NRC Regulatory Guide 1.16 Revision 4.

Attachment 2 is the annual report required by 10CFR50.59(b)(2) for 1991. This report contains descriptions of the changes, tests and experiments completed on CPSES Unit 1 under the provisions of 10CFR50.59(a), including a summary of the safety evaluation of each. Items in this report are referenced by their 50.59 evaluation numbers. This report covers the period from December 31, 1990 through December 31, 1991.

If you have any questions, please contact Mr. Jorge L. Rodriguez at (214) 812-8323.

Sincerely,

William J. Cahill, Jr.
William J. Cahill, Jr.

By Roger D. Walker
Roger D. Walker
Manager of Nuclear Licensing

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JLR/gj
Attachments

c - Mr. R. D. Martin, Region IV
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Mr. T. A. Bergman, NRR
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COMANCHE PEAK STEAM ELECTRIC STATION

ANNUAL OPERATING REPORT
1991

TEXAS UTILITIES ELECTRIC COMPANY

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1.0 SUMMARY OF OPERATING EXPERIENCE

The Comanche Peak Steam Electric Station is a pressurized water reactor licensed at 3411 Megawatt thermal (MWt). It is located in Somervell County in North Central Texas about 65 miles southwest of the Dallas-Fort Worth Metropolitan area. The nuclear steam supply system was purchased from Westinghouse Electric Corporation and is rated for a 3425 MWt output.

The Comanche Peak nuclear power plant achieved initial criticality on April 3, 1990. Initial power generation occurred on April 24, 1990, and the plant was declared commercial on August 13, 1990. Since being declared commercial, Comanche Peak Unit 1 has generated 7,895,564 MW hours of electricity as of December 31, 1991, with a net plant capacity factor of 56.6 (using net MDC). The unit and reactor availability was 61.0 and 82.2%, respectively.

On March 20, 1991, the unit was removed from service because of condenser tube failure and on March 22, 1991, the unit entered a Mid-Cycle Outage. With the exceptions of the unplanned turbine repairs, outage activities were completed in support of the original outage schedule. Turbine repairs were completed with less than a two percent (2%) impact on turbine output. The unit was returned to service on May 27, 1992.

On October 3, 1991, the unit was removed from service for its first refueling outage. Overall, the outage was successful in its implementation with a duration of 68 days. This duration is significantly lower than the average for first refueling Westinghouse 4-loop plants of 105 days. Fifty-six fresh fuel assemblies were loaded for Cycle 2. The unit was returned to service on December 11, 1991.

Figure 1 provides a histogram of the average daily electrical output of the unit for 1991. Table 2.1 is a compilation of the monthly summaries of the operating data and Table 2.2 contains the yearly and total summaries of the operating data.

2.0 OUTAGES AND REDUCTION IN POWER

Table 2.3 describes plant shutdowns and provides explanations of significant dips in average power levels.

3.0 PERSONNEL EXPOSURE AND MONITORING REPORT

The personnel exposure and monitoring report is provided in Table 3.0.

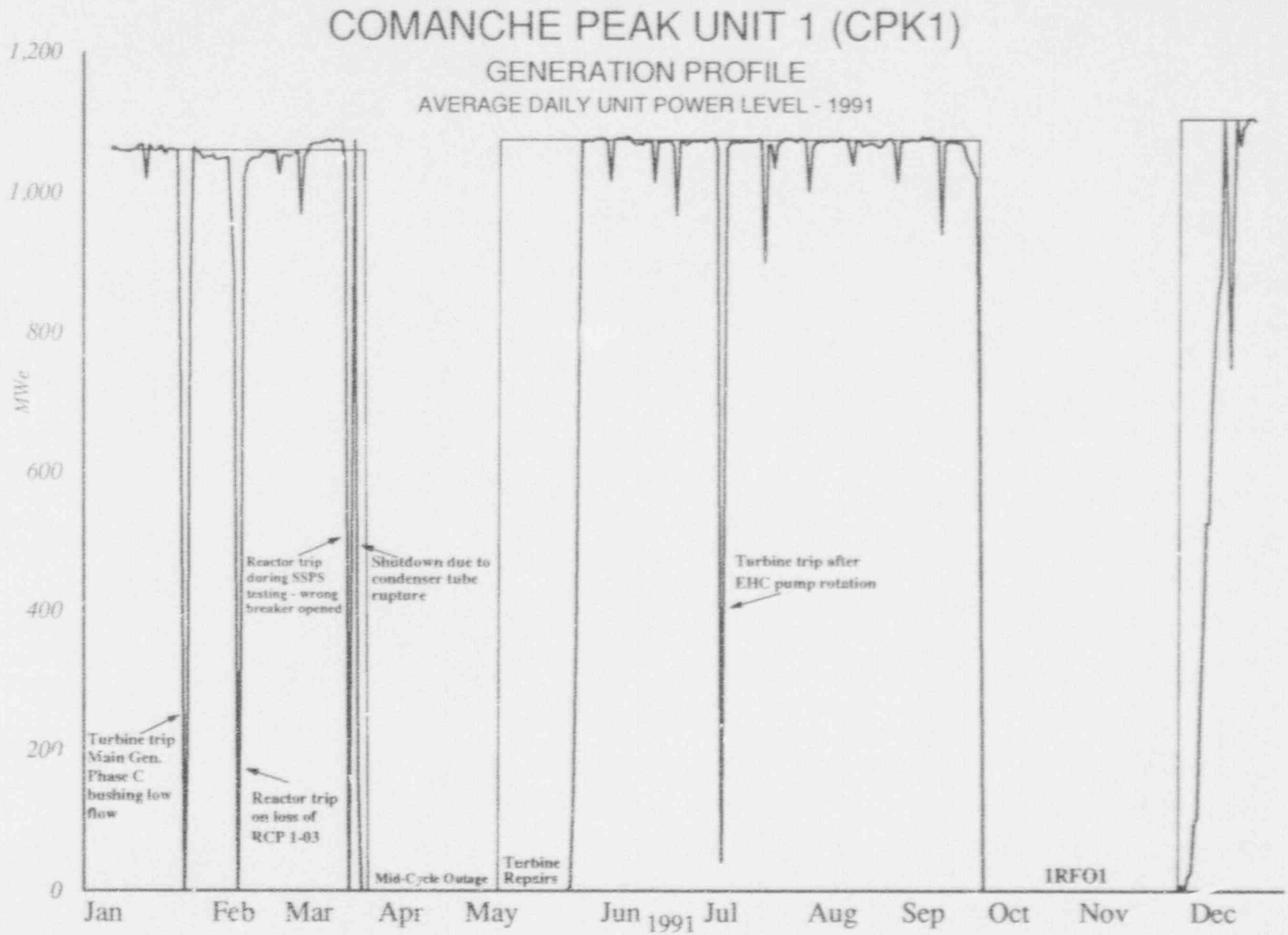
COMANCHE PEAK UNIT 1 (CPK1)
GENERATION PROFILE
AVERAGE DAILY UNIT POWER LEVEL - 1991
MWe
1,200
1,000
800
600
400
200
0
Turbine trip Main Gen. Phase C bushing low flow
Reactor trip on loss of RCP 1-03
Reactor trip during SSPS testing - wrong breaker opened
Shutdown due to condenser tube rupture
Mid-Cycle Outage
Turbine Repairs
Turbine trip after EHC pump rotation
1RFO1
Jan
Feb
Mar
Apr
May
Jun
1991
Jul
Aug
Sep
Oct
Nov
Dec

TABLE 2.1 (PAGE 1 OF 2)

ELECTRIC POWER GENERATION DATA (1991)

MONTHLY

	January	February	March	April	May	June
Hours RX was critical	710.5	652.4	445.3	0	124.25	720
RX Reserve Shutdown Hours	33.5	19.6	298.7	719	619.75	0
Hours Generator On-Line	702.5	648	437.2	0	106.5	720
Unit Reserve Shutdown Hours	0	0	0	0	0	0
Gross Thermal Energy Generated (MWH)	2,259,446	2,054,786	1,375,315	0	261,965	2,439,547
Gross Elec. Energy Generated (MWH)	762,318	685,222	459,846	0	81,397	803,116
Net Elec. Energy Generated (MWH)	728,251	654,883	434,571	0	61,416	769,150
RX Service Factor	95.5	97.1	59.9	0	16.7	100
RX Availability Factor	100	100	100	100	100	100
Unit Service Factor	94.4	96.4	58.8	0	14.3	100
Unit Availability Factor	94.4	96.4	58.8	0	14.3	100
Unit Capacity Factor (using MDC net)	85.2	84.7	50.8	0	7.2	92.9
Unit Capacity Factor (using DER net)	85.2	84.7	50.8	0	7.2	92.9
Unit Forced Outage Rate	5.6	3.6	14.4	0	84.4	0
Hours in Month	744.0	672.0	744.0	719.0	744.0	720.0
Net MDC (Mwe) estimated	1150.0	1150.0	1150.0	1150.0	1150.0	1150.0

TABLE 2.1 (PAGE 2 OF 2)

ELECTRIC POWER GENERATION DATA (1991)

MONTHLY

	July	August	September	October	November	December
Hours RX was critical	725	744	720	65.4	0	581.95
RX Reserve Shutdown Hours	19	0	0	0	0	0
Hours Generator On-Line	720	744	720	65.35	0	479.92
Unit Reserve Shutdown Hours	0	0	0	0	0	0
Gross Thermal Energy Generated (MWH)	2,398,615	2,521,411	2,431,361	196,474	0	1,236,146
Gross Elec. Energy Generated (MWH)	784,052	830,208	802,524	63,638	0	399,677
Net Elec. Energy Generated (MWH)	748,785	794,872	768,809	52,562	0	368,751
RX Service Factor	97.4	100	100	8.8	0	78.2
RX Availability Factor	100	100	100	8.8	0	78.2
Unit Service Factor	96.8	100	100	8.8	0	64.5
Unit Availability Factor	96.8	100	100	8.8	0	64.5
Unit Capacity Factor (using MDC net)	87.5	92.9	92.9	6.1	0	43.1
Unit Capacity Factor (using DER net)	87.5	92.9	92.9	6.1	0	43.1
Unit Forced Outage Rate	3.2	0	0	0	0	6.2
Hours in Month	744.0	744.0	720.0	745.0	720.0	744.0
Net MDC (Mwe) estimated	1150.0	1150.0	1150.0	1150.0	1150.0	1150.0

TABLE 2.2

ELECTRICAL POWER GENERATION DATA
1991

	YEAR	CUMULATIVE
Hours RX was critical	5488.8	8415.2
RX Reserve Shutdown Hours	1709.55	1982.45
Hours Generator On-Line	5343.47	8209.17
Unit Reserve Shutdown Hours	0	0
Gross Thermal Energy Generated (MWH)	17,175,066	25,331,994
Gross Electric Energy Gen. (MWH)	5,671,998	8,336,998
Net Elec. Energy Generated (MWH)	5,382,050	7,895,564
RX Service Factor	62.7	69.3
RX Availability Factor	82.2	85.7
Unit Service Factor	61.0	67.6
Unit Availability Factor	61.0	67.6
Unit Capacity Factor (using MDC net)	53.4	56.6
Unit Capacity Factor (using DER net)	53.4	56.6
Unit Forced Outage Rate	12.6	11.0
Hours in Reporting Period	8760	12137

Table 2.3

UNIT SHUTDOWNS AND POWER REDUCTIONS

NO.	DATE	TYPE F:FORCED S:SCHEDULED	DURATION (HOURS)	REASON	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER	CORRECTIVE ACTIONS/COMMENTS
001	910123	F	41.5	A	3	Reactor Trip on Turbine Trip due to low primary water pressure to main generator, unit entered MODE 3. (LER 91-002)
002	910210	F	24	G	3	Reactor tripped on Undervoltage Trip of #3 Reactor Coolant Pump. Trip caused by operator error when switch board cabinet was opened disconnecting UV relays, unit entered MODE 3. (LER 91-004)
003	910317	F	25.7	G	3	Reactor Trip caused by personnel error during surveillance testing.
004	910320	F	48.1	A	1	Steam generator chemistry at action level 3 due to high sodium, caused by tube failure in the main condenser. Began power reduction/shutdown to correct problem.
005	910322	S	233	B/F	1	Planned outage to extend fuel cycle past summer peak.

Table 2.3

UNIT SHUTDOWNS AND POWER REDUCTIONS

NO.	DATE	TYPE F:FORCED S:SCHEDULED	DURATION (HOURS)	REASON	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER	CORRECTIVE ACTIONS/COMMENTS
006	910401	S	719	B/F	1	Planned Outage. Major work accomplished during outage: o Feedwater Orifice Installation o RCP Seal Replacement o D/G Work and Surveillances o Secondary Plant Material Condition/ Reliability o MOV Testing o Turbine-Generator Inspection/ Repairs. o Electrical Bus Outages o Condenser Tube Repairs o Feedwater Elbow Thinning o BW/IP Check Valve Sticking o Switchyard Modifications
007	910501	S	60	B/F	1	Planned Outage
008	910503	F	577.5	A	1	Forced Outage due to Damage to LP turbine.
009	910713	F	24	A	3	Reactor tripped on Turbine trip due to EHC fluid pressure fluctuations. (LER-91-020)

Table 2.3
UNIT SHUTDOWNS AND POWER REDUCTIONS

NO.	DATE	TYPE F:FORCED S:SCHEDULED	DURATION (HOURS)	REASON	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER	CORRECTIVE ACTIONS/COMMENTS
010	911003	S	1632.27	C	1	First Refueling Outage. Major work items: RCP-02 seal work, Replace L-5 blades LP-2 turbine, Generator rotor upgrade, Diesel Generator overhaul, Snubber inspection, Steam Generator In-Service Inspections
011	911210	F	12.68	A	1	Turbine Shutdown required to repair HP control valve oscillations, unit entered MODE 2.
012	911211	F	18.78	A	1	Turbine Shutdown required to repair Manual FW valve disc/stem separation, unit entered MODE 2.
013	911222	F	35.5	G	4	Automatic Turbine Runback to 60% power initiated when troubleshooting a heater drain valve.

1) REASON
- A: EQUIPMENT FAILURE (EXPLAIN)
- B: MAINT OR TEST
- C: REFUELING
- D: REGULATORY RESTRICTION
- E: OPERATOR TRAINING AND LICENSE EXAMINATION
- F: ADMINISTRATIVE
- G: OPERATIONAL ERROR (EXPLAIN)
- H: OTHER (EXPLAIN)

2) METHOD
- 1: MANUAL
- 2: MANUAL SCRAM
- 3: AUTOMATIC SCRAM
- 4: OTHER (EXPLAIN)

GENERAL NOTE:

There are no single release of radioactivity or single radiation exposure (i.e., single entry into a radiation controlled area) specifically associated with the outage which accounts for more than 10 percent of the allowable annual values.

TABLE 3.0

Personnel Exposure and Monitoring Report

Work & Job Function	# Personnel (>100 mrem) Station	Utility	Contract	Total Man-Rem Station	Utility	Contract
Reactor Operations & Surveillance						
Maintenance & Construction	1	0	1	0.171	0.000	0.518
Operations	13	[illegible]	1	4.386	0.178	0.957
Health Physics & Lab	4	0	14	1.233	0.000	4.132
Supervisory & Office Staff	0	0	0	0.099	0.000	0.104
Engineering Staff	0	0	0	0.123	0.000	0.134
Routine Plant Maintenance						
Maintenance & Construction	22	0	110	8.602	0.020	31.151
Operations	2	0	7	0.967	0.000	3.019
Health Physics & Lab	3	0	5	0.900	0.000	1.749
Supervisory & Office Staff	0	0	0	0.223	0.000	0.017
Engineering Staff	0	0	6	0.242	0.000	1.887
Inservice Inspection						
Maintenance & Construction	0	0	65	0.165	0.000	30.306
Operations	0	0	5	0.122	0.000	1.165
Health Physics & Lab	2	0	10	1.717	0.000	2.202
Supervisory & Office Staff	0	0	0	0.004	0.000	0.027
Engineering Staff	0	0	18	0.021	0.000	3.525
Special Plant Maintenance *						
Maintenance & Construction	0	0	11	0.066	0.000	3.784
Operations	0	0	7	0.080	0.000	2.358
Health Physics & Lab	6	0	10	1.363	0.000	3.624
Supervisory & Office Staff	0	0	0	0.000	0.000	0.000
Engineering Staff	0	0	0	0.001	0.000	0.152
Waste Processing						
Maintenance & Construction	0	0	2	0.009	0.000	0.557
Operations	0	0	1	0.142	0.000	0.429
Health Physics & Lab	5	0	3	1.973	0.000	1.370
Supervisory & Office Staff	0	0	0	0.000	0.000	0.000
Engineering Staff	0	0	0	0.000	0.000	0.056
Refueling						
Maintenance & Construction	19	0	16	6.994	0.000	5.476
Operations	8	0	0	2.253	0.000	0.201
Health Physics & Lab	1	0	6	0.239	0.000	1.287
Supervisory & Office Staff	1	0	0	0.563	0.000	0.000
Engineering Staff	3	0	0	0.656	0.023	0.198
Totals						
Maintenance & Construction	35	0	205	16.007	0.020	71.792
Operations	31	1	23	7.950	0.178	8.129
Health Physics & Lab	20	0	55	7.425	0.000	14.364
Supervisory & Office Staff	1	0	0	0.888	0.000	0.148
Engineering Staff	3	0	27	1.043	0.023	5.952
	90	1	310	33.313	0.221	100.385

*Special Plant Maintenance includes all work activities associated with implementation of Unit 1 design modifications.

4.0 A REPORT OF RESULTS OF SPECIFIC ACTIVITY ANALYSIS IN WHICH THE PRIMARY COOLANT EXCEEDED THE LIMITS OF TECHNICAL SPECIFICATION 3.4.7.

Technical Specification 6.9.1.2.b requires the results of specific activity analyses in which the primary coolant exceeded the limits of Technical Specification 3.4.7.

During the year ending December 31, 1991 the specific activity of the reactor coolant was less than 1 microcurie per gram dose equivalent I-131 and was also less than 100 divided by E-Bar microcuries per gram of gross radioactivity.

5.0 IRRADIATED FUEL INSPECTION RESULTS

During October 1991, CPSES Unit 1 entered the first refueling outage with indications of 2 or possibly 3 fuel failures. During the outage, all 193 fuel assemblies were off-loaded from the reactor vessel into Spent Fuel Pool #1. Ultrasonic Testing (UT) was performed on all 193 fuel assemblies (50,952 fuel rods) to locate individual failed fuel rods. UT identified two failed fuel rods; one failed fuel rod in assembly C30, rod location G5, and the other failed fuel rod in assembly A34, rod location H8. During UT of fuel assembly A03, a metallic spring was observed to be protruding from the bottom nozzle. Subsequent underwater TV camera examination confirmed that the object was a fuel rod plenum spring caught in one of the bottom nozzle flow holes and extending into the assembly to a position just below a fuel rod bottom end plug. It was noted that about one-third of the spring (approximately 3 inches) was missing. Several small fretting marks were observed on the edge of the bottom nozzle as a result of reactor coolant flow induced vibration of the spring against the bottom nozzle. No other unusual indications were observed. Assembly A03 was scheduled for discharge during this refueling.

Full-face underwater TV camera examinations were performed on all fuel assemblies scheduled for reload along with fuel assemblies A03, C30, and A34. Visual inspection of fuel assembly C30 revealed a fuel rod top end plug wedged in a flow hole in the top nozzle just above the location of the failed fuel rod. The top end plug in fuel rod position G5 was observed to be missing. This discovery confirmed that assembly C30, fuel rod location G5 was the likely source of the plenum spring observed in the bottom nozzle of assembly A03. During examination of the bottom nozzle area, it was observed that the gap between the bottom of all fuel rods and the bottom nozzle appeared normal. No other unusual indications were observed on this fuel assembly.

Visual examination of the top nozzle area of assembly A34 revealed that the failed fuel rod in location H8 was extending approximately three-quarters of an inch above the height of the other fuel rods in the assembly. The top end plug of the failed rod in location H8 appeared intact. Because the location of the failed fuel rod was near the center of the assembly, the position of the bottom of the failed fuel rod could not be observed during examination of the bottom nozzle area. No other unusual indications were observed on this assembly.

No significant indications were observed during the visual inspections of the remainder of the fuel assemblies scheduled for reload. In general, the fuel assemblies appeared to be in very good condition with only a very light coating of residue ("crud") observed on the surface of the fuel rods.

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COMANCHE PEAK UNIT 1
ANNUAL 10CFR50.59 REPORT

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This report contains a description and a summary of the following 10CFR50.59 evaluations.

Evaluation Number
SE-89-092

Activity Title:

Discontinued use of Circulating Water System siphon bleed line to the Safe Shutdown Impoundment (SSI)

Description of Change(s):

This activity revises the status of the isolation valve for the Circulating Water bleed-off connections to the Safe Shutdown Impoundment (SSI) from normally open to normally closed. This connection was previously described in the FSAR as a means for providing continuous make-up to the SSI to prevent excessive dissolved solids concentrations, and a source of chemical treatment for the Service Water System.

Summary of Evaluation:

The bleed-off connection serves no safety function. No change in the reliability of the Service Water System (SSW) or any other safety system will result from the discontinuation of the use of this line. The volume of the SSI and the maximum sediment buildup is specified and assured via Technical Specification 3.7.5. A number of programs and procedures exist to ensure the reliabilty of the SSW system. The use of the bleed connection as a means for chemical treatment is insignificant since the SSW system has an independent, dedicated source of chemical injection. There are no credible failure modes associated with this activity and no analyzed accident/malfunction is impacted.

Evaluation Number
SE-89-125

Activity Title:

Temporary modification for installation of load sensing pins in snubber rear brackets for transient testing.

Description of Change(s):

This safety evaluation was performed on a Temporary Modification(TM) that was installed prior to Unit 1 licensing in January, 1990. The TM consisted of replacement of existing load pins with load sensing clevis pins in the snubber rear bracket assemblies on the Feedwater and Main Steam piping in containment. The purpose of this TM was to verify the thermal/dynamic loads in the subject piping during power ascension as part of transient testing. The use of the load sensing pins provided flexibility for verification of calculated loads and validating the model used in the load calculations.

The load sensing pins were designed and manufactured to the intent of ASME B&PVC Class 1 requirements. The load sensing pins are equal or better than the permanent pins.

The load sensing pins were removed during the first refueling outage in October, 1991. The original load pins were installed when the load sensing pins were removed.

Summary of Evaluation:

The load sensing pins were designed to be a "drop-in" replacement to the existing load pins. The load sensing pins were as good as or better with respect to the design requirements of the original pins.

The evaluation of the load sensing pins was performed and determined to not affect the accidents as evaluated in the licensing based documents, nor create any new accident. The replacement of the load sensing pins with the original pins restores the assemblies to the original design as described in the licensing based documents.

Evaluation Number
SE-90-018

Activity Title:

Revise Safety Analysis Report (SAR) to document the safety analysis for accessing vital area of Safeguards Bld. Sump Drain Panel post-LOCA

Description of Change(s):

The SAR, Section II.B.2, was revised to further document the safety analysis of operator action outside the Control Room post-accident as required by post-TMI requirements (NUREG-0737). This SAR change adds documentation of the analysis for accessing the Safeguards Building Sump Drain Panel post-LOCA.The Safeguards Building Vents and Drains System is associated with this change. The change is required to show acceptability of the system design and document the estimated total dose received, potential dose rates encountered, and time required by the operator for the access task.

Access to the Safeguards Building Sump Drain Panel Room (#79) is required to allow the cperator to diagnose passive failures in the ESF systems (e.g. pump seal failure). This accessibility is evaluated in context with the design basis leak of 50 gpm in the ESF recirculation loop from the containment sump at 24 hours post- LOCA. The mitigation of offsite consequences is based on detection and isolation of the ESF leak within 30 minutes.

Summary of Evaluation:

A radiological impact is associated with this post-accident operator action outside of the Control Room; however, the action results in an estimated whole body dose of only 0.11 rem which is well within the acceptable (NUREG-0797) design criteria of 5 rem provided in GDC-19 and NUREG-0797. Therefore, this access does not represent an increase in radiological consequences which would require NRC approval based on USNRC letter dated May 10, 1989, from C.E. Rossi to T.E. Tipton of NUMARC.

There are no accidents for which this access can be an initiating event since it is performed post-accident. The operator actions have no effect on the performance of the Safeguards Building Vents and Drains or other safety systems; therefore, there are no failure modes and no effect on the probability of impacting a previously identified accident or creation of a new accident.

There are no Technical Specifications associated with this access or the Safeguards Building Vents and Drains System. Although the ESF systems are subject to Technical Specifications, leak detetion and passive failure mitigation are not defined as essentail auxiliary supporting systems by NUREG-0800 or ESF support systems by the SAR. Nevertheless, this activity satisfies the acceptance criteria of NUREG-0737 and does not decrease the margin of safety.

Evaluation Number
SE-90-019

Activity Title:

Revise SAR to clarify functional requirements for Control Room air intake dampers and provide for related vital area access post-LOCA

Description of Change(s):

This SAR change clarifies the sizing and system requirements for air accumulators pertaining to the Control Room air intake dampers. The change also adds an analysis for accessing the vital area of rooms 150 and 150A in order to manually manipulate the Control Room air intake dampers post-accident. The Control Room Air Conditioning System (CRACS) and Instrument Air systems are associated with this acitivty.

Summary of Evaluation:

The accessibility to the vital area was evaluated with respect to the relevant design basis accident, i.e., LOCA. A radiological impact is associated with the post-accident operator action outside of the Control Room; however, the action results in an estimated whole body dose of only 0.3 rem which is well within the acceptable (NUREG-0797) design criteria of 5 rem provided in GDC-19 and NUREG-0797. Therefore, this access does not represent an increase in radiological consequences which would require NRC approval based on USNRC letter dated May 10, 1989, from C.E. Rossi to T. E. Tipton of NUMARC.

This activity evaluated the acceptabilty of criteria for safety related air accumulators provided for the Control Room air intake dampers. Minimum system requirements were determined to support testing in place of using sizing criteria. Tank sizing criteria (SAR 9.3.1) (30 days) were previously used conservatively due to lack of minimum acceptance criteria for system requirement. Minimum system requirement is 2 hours based on NUREG-0800 criteria and vital area accessability evaluated in accordance with the SAR, engineering calculations and NUREG-0737. Therefore, any time between 2 hours and 30 days is acceptable for the accumulator acceptance criteria.

The operator access action has no effect on equipment/system failure modes for accidents described in the licensing basis documents. The air accumulators are associated with Technical Specification 3/4.7.7, Control Room HVAC, because they are required for the system to be OPERABLE unless the air intake dampers are locked open. Post-LOCA, it would be acceptable to lock open the dampers in event Instrument Air cannot be restored. This provision was anticipated in the design and the SAR (Section 6.4). Because OPERABILITY is assured both before and after an accident, there is no decrease in the margin of safety as defined in the basis for Technical Specification 3/4.7.7.

Evaluation Number
SE-90-041
Revision 3

Activity Title:

Allowing additional aluminum and/or zinc materials in containment during modes 1-4 to support maintenance and/or surveillance activities

Description of Change(s):

This evaluation has been updated to accomodate the revised analysis performed to support increased aluminum and zinc inventory in Containment as presented in FSAR Section 6.2.5A. This revised evaluation takes into account the effects of aluminum and zinc in solutions on containment radiation levels. There is no design change associated with this re-analysis of hydrogen production.

Summary of Evaluation:

The possibility of increased radiation levels due to additional aluminum or zinc inside Containment depends primarily on the possibility of parts or materials containing these elements being exposed to an intense neutron flux during power operation, thereby becoming irradiated It is not expected that such parts or materials could inadvertently be allowed to enter the Reactor Coolant system, thus passing through the core neutron flux region, while the Reactor is at power.

. potential increase in dose rates due to Al-28 would be of short ation after reactor shutdown and have insignificant radiological pact. Even an unreasonable large quantity of irradiated zinc assumed to be released to the Containment would yield post-accident dose rates which are negligible in comparison to those from the fission products postulated to be released per Regulatory Guide 1.4.

Assuming that all of the Al and Zn in Containment dissolves into the aqueous phase, this would produce a very dilute solution of metal cations in the water. A reaction would then be required to deposit the Al or Zn in the stainless steel grain boundaries (the chance of metal ion undergoing the electrochemical reduction directly on a grain boundary are very small). Since the RCS is cool (below 200 F) and depressurized, the effect of embrittling would not be a problem. The temperature needed to "soften" the grain boundary is not present and the stress needed to propagate a crack is not present.

The direct effect to the Containment atmospheric pressure will be a net increase of less than 1.0% following a LOCA; therefore, negligable.

Evaluation Number
SE-90-078

Activity Title:

Diesel generator start logic change.

Description of Change(s):

The design modification involves a change in the diesel generator start logic such that the diesel generator will start with the loss of the preferred and alternate power sources or if the plant is on the alternate source when power is lost. The affected undervoltage relays are contained within the 6.9kV switchgear, 1EA1 and 1EA2. This change consists of removing time delay relays 27AX2/ST2 and 27BX2/ST2 which start the diesel on loss of preferred offsite power; removing the diesel start signal off of bus undervoltage time delay relays 27-2X/1EA1, 2; adding the start signal to a new bus undervoltage time delay relay, 272X-1/1EA1 which is controlled by bus undervoltage relays 27-2/1EA1, 2; and changing the time delays on relays 27AX1/ST1, 27BX1/ST2, 27-2X/1EA1, 2 and the sequencer time delay relays (27-1A, B, C, D/1EA1, 2).

Summary of Evaluation:

The implemented design modification removed the relaying associated with starting the diesel generator upon loss of preferred offsite power. The result of the change will eliminate unnecessary starts of the diesel when the alternate power source is available. The addition of the new time delay relay of 1.0 sec will allow the alternate offsite power source breaker to close and power the bus. If the alternate power source fails to power the bus, then the diesel will receive a diesel start signal on 6.9 kV bus undervoltage. In addition to bus undervoltage and manual starts, the diesel generator will still receive a start signal on a safety injection (SI) signal and an SI in conjunction with a loss of offsite power signal(s). Although this change appears to have reduced the flexibility and margin of safety for having an electrical power source power the safeguard busses, it was determined, based on operating experience, that the change was still necessary to eliminate unnecessary starts which results in excessive wear and tear on the diesel.

The plant has experienced several trips, due to external disturbances, of the preferred offsite power source. Following each trip, the diesel started, but the plant loads were successfully picked up by the alternate power source. Had this design modification been in place, the diesel generator would have never started on loss of preferred offsite power. An analyses of the consequences of the additional delay time in powering the loads in the event the alternate power source is not available. Determined that the delay is still within the 2 second allowances of the accident analysis which also takes into account the voltage decay following the loss of either offsite source and breaker and relay operation times. The NRC approved the License Amendment Request (LAR 90-003) associated with this change to the CPSES Technical Specification on October 11, 1990.

Evaluation Number
SE-90-082

Activity Title:

Installation of sample coolers, sinks, demin. water cooling/flushing lines, isolation valves and sink drain locations.(LDCR 90-132, 158)

Description of Change(s):

Add the sample lines, drains, coolers, sample sinks, isolation valves and appurtenances for taking samples from the Floor Drain, Waste and Boron Recycle Evaporators. The addition of the piping/tubing is below the existing normally closed valves which also function as Radioactive Waste Mangement System boundries. These valves will only be opened during sampling operations.

Summary of Evaluation:

Installation of sample coolers, sample sinks and demieralizer water cooling and flushing lines will improve personnel safety and decrease contamination while taking samples from Floor Drain, Waste and Boron Recycle Evaporators. No safety systems were affected by this design change. However, system important to safety(e.g., Radioactive Waste Management System) will not be affected by implementation of the design modification as the installation will be downstream of normally closed valves, which form the Radioactive Waste Management System boundaries.

Evaluation Number
SE-90-085

Activity Title:

Replace the existing Water Treatment System with a new high capacity system.

Description of Change(s):

Replace the existing Water Treatment System with a more efficient, high capacity and reliable Water Treatment System to meet the plant demands. Also provide electrical, control, water, air and drain services to the new system from existing plant systems.

Summary of Evaluation:

The existing Water Treatment System for the plant is undersized in capacity for the pretreatment section of the system. Existing clarification/filtration units are limited in capacity and reliability. This design modification will replace the pretreatment section of the system with the higher capacity, more reliable with a storage capabilty for the future usage of the pretreatmwnt water. Also provided to this new Water Treatment System is the necessary electrical, control and instrument, water, drain and air supply from various plant systems.

The new equipment will upgrade the capacity and reliability of the Pretreatment section of the Water Treatment System. These systems are neither safety related nor systems important to safety and their failure would not affect the safe operation of the plant.

Evaluation Number
SE-90-095

Activity Title:

Additional shielding for the fuel transfer tube (in containment). LDCR SA-90-148.

Description of Change(s):

Provide additional shielding around the fuel transfer tube to reduce the dose rate when spent fuel is move between the reactor containment building and the spent fuel pool.

Summary of Evaluation:

The analysis of this activity (placing additional shielding around the spent fuel transfer tube from the containment side) does not impact on any accident previously analyzed. Nor does implementation of this activity result in any accident not previously considered or evaluated. The results of the seismic considerations indicate that there is no reduction in the margin of safety previously considered; however, the additional shielding and access control features do result in greater dose protection features.
Therefore no unreviewed safety question exists.

Evaluation Number
SE-90-101

Activity Title:

Provisions for allowing the instrument air system to be a source for breathing air.

Description of Change(s):

The Instrument Air System was changed to allow its use as a source for breathing air. Hensen fittings and lock boxes were added at each breathing air station to exlude non use - of breathing air from the connections.

Summary of Evaluation:

The use of the Instrument Air system as a source of respirable air is acceptable as Instrument Air is clean, particulate "free", oil free and capable of reliably supplying uninterrupted Grade "D" quality air. Evaluation indicates that operation of Breathing Air at the Breathing Air Stations will not adversely affect the safety related portions of the system or operation of other Instrument Air users.

This modification presents no new failure modes for the plant or any plant systems. The additional load of 150 SCFM upon the Instrument Air system is within the capacity of an Instrument air Lead/Lag compressor combination.

The modified piping including the added weight of the Hansen fittings and lock boxes have been evaluated by Pipe stress and found acceptable. The use of Breathing Air equipment is controlled by Stations Operations Procedures (Ref STA-659, STA-211) and control of equipment in Seismically design buildings/areas comply with STA-661. Work practices involved in the implementation of this DM comply with STA-661 (use of welding machines and other equipment in Seismic areas). No credible failure modes are associated with implementation of this DM.

Evaluation Number
SE-90-203
Revision 1

Activity Title:

Add the alternate RCA access, tool room, mens and women changing and shower areas. LDCR 90-161,208,222, and 91-18.

Description of Change(s):

Addition has been made for the office and service area HVAC sytem to provide the HVAC facility to the newly added tool room, alternate RCA access and mens and womens changing and shower area modifications. The exhaust from the tool room and alternate RCA area is routed to the Primary Plant Ventilation Exhaust System. A rebalancing and revision of the project process air flow diagram for the office area is completed.

Summary of Evaluation:

New facilities such as tool room and alternate RCA access are added to faclitate the Unit 2 construction personnel flow while Unit 2 is under construction. Also converted existing mens locker into womens changing and shower facilities. The HVAC services for this facilities are provided by modifying the existing office and service area HVAC system. The office and service area HVAC system modification includes the air flow balancing, adding fan coil units and routing certain exhaust to the Primary Plant Ventilation System(PPVS). The project air flow diagrams are revised to reflect this change in the system. Reroute of exhaust through the PPVS will not impact the negative pressure requirement of the negative pressure boundary.

Evaluation Number
SE-90-213

Activity Title:

Conversion of hot shop, room 39, in the switchgear building II to an alternate Radiation Control Area access point (LDCR SA--90-192).

Description of Change(s):

The hot shop (Room 39 in the Unit 2 switchgear building) was removed and the area was modified to provide a second (or alternate) access point to the Radiation Controlled Area (RCA). The principal reason for this change was to enhance access to the RCA for construction and contractors during future outages.

Summary of Evaluation:

Implementation fo this change does not affect any Safety related system, equipment, structure or parameter. The Non-Safety related parameters, systems, etc. that were affected/considered, include: CIVIL/STRUCTURAL LAYOUT OF ROOM 39, HVAC SYSTEM, FIRE PROTECTION, SECURITY PROGRAM/PLAN, and RADIATION ZONES. The overall impact to this change, collectively and indivually, does not impact the performace of any plant system or structure.

Evaluation Number
SE-90-214

Activity Title:

Add temperature indicators to the EDG jacket water coolers and lube oil coolers. (LDCR-90-199)

Description of Change(s):

Add the temperature indicators and ASME III, Class 3 thermowells for the Emergency Diesel Generator Jacket water coolers and lube oil coolers. TI-3415-2A,2B and TI-3416-2A,2B are installed on lube oil coolers and TI-3415-3A,3B and TI-3416-3A,3B are installed on the jacket water coolers for Train A and B of the Emergency Diesel Generator Systems..

Summary of Evaluation:

Installation of the temperature indicators in the inlet and outlet of the jacket water and lube oil heat exchangers provide the temperature difference across the heat exchangers. This parameters is required to monitor the performance of the heat exchangers and provide the input to the heat exchanger performance monitoring program. The existing piping on the diesels contain plugged 3/4" threaded couplings in acceptable locations. Threaded ASME III, Class 3 thermowells will be installed to maintain pressure boundry of the diesel generator jacket water and lube oil systems. No cutting and welding is required for the installation of this TI's, hence the integrity and cleanness of these systems is not effected. Installation of the thermowells and the temperature indicators does not introduce any new failure modes and does not impact the safety of the plant.

Evaluation Number
SE-90-217

Activity Title:

Provides 2-out-of-3 coincident logic for Main Feedwater Pump low suction pressure trip. DM 89-034

Description of Change(s):

Provides 2-out-of-3 coincident logic for Main Feed Pumps (MFP) 1A and 1B low suction pressure trip. This is accomplished by adding two additional low suction pressure switches, using the same taps as the pressure transmitter for the existing pressure instrument loop. For MFP 1B the new pressure switches will have the same setpoint as the existing trip setpoint. Although the installation for MFP 1A is the same, the trip setpoint is set lower to provide for staggered low suction presssure tripping of the pumps. In addition, a 4 second time delay is added to the trip logic to prevent unnecessary pump trips due to low pressure spikes.

Summary of Evaluation:

Single point failure analyses studies identified the Main Feedwater Pump suction pressure switch as a component whose failure can initiate a sequence of events leading to a reactor trip. Addition of pressure switches to provide for 2-out-of-3 coincident logic eliminates this single point failure probability. The studies also recommended adding a time delay into the low suction trip logic and to stagger the trip setpoints of the pumps to lessen the impact of a feedwater transient in the event of a feedwater low pressure spike.
The staggered trip values ensure the existing turbine runback control design is effective to prevent a reactor trip on falling suction pressure. The new trip set point for MFP 1A is still well above the NPSH requirement for the pumps at valve-wide-open flow conditions.

The modification was evaluated for its effect on Chapter 15 accident analyses:

15.1.1 Feedwater system malfunction that results in decrease in the feedwater temperature, and,

15.2.7 Loss of normal feedwater flow,

and was found not to impact the analyses. Neither does it affect the ability of the Auxiliary Feedwater System to operate.

The modification provides for the reduction in probability for reactor trips while still maintaining adequate feedwater pump low suction pressure protection.

Evaluation Number
SE-90-224

Activity Title:

Addition of tool room, 42A, to the access control area in switchgear building (LDCR SA-90-205).

Description of Change(s):

With conversion of Room 39 from a hot shop to an access control point, the associated hot tool room was also eliminated. Consequently, a hot tool room is being added. Access to the tool room is located in the alternate access control point. The tool room itself extends into the area of the Unit 2 Switchgear Room.

Summary of Evaluation:

Implementation of this activity does not affect any safety related system, structure, equipment, or parameter. The Non-safety related systems, structures, etc. affected by this change include: HVAC, CIVIL/STRUCTURAL, SECURITY PROGRAM/PLAN, and RADIATION ZONES.

The individual and cumulative affects of these changes resulted in no unreviewed safety questions/concerns being identified.

Evaluation Number
SE-90-227

Activity Title:

Revision to the turbine lube oil low pressure trip logic to reduce the possibility of turbine trips due to pressure switch failure.

Description of Change(s):

This change revises the turbine protection logic for the low lube oil pressure trip from single signal actuation per channel, to 2-out-of-3 pressure switch actuation per channel. It also provides an indication of pressure switch output mismatch and power signal loss.

Summary of Evaluation:

No new failure modes are introduced since the low lube oil pressure trip setpoint is unaffected. Only the protection logic and the number of devices initiating the trip signal are changed. This change will maintain turbine protection while reducing the number of plant transients caused by a turbine trip due to a spurious low pressure signal or a failure of a lube oil pressure switch.

Evaluation Number
SE-90-231

Activity Title:

Addition of 480 V power receptacles in the Unit 1 containment for HEPA units, welding units and power tools.

Description of Change(s):

The implemented design modification added 480V power receptacles inside Unit 1 containment. This resulted in the parallelling of existing electrical penetration conductors, to maximize power, and the addition of a distribution panel and aluminum power receptacles. The receptacles are intended to power equipment used during maintenance and surveillances, such as HEPA units, welding units and power tools.

Summary of Evaluation:

The modification involves the addition of a distribution panel and aluminum receptacles inside containment via an existing electrical penetration. To maximize power, electrical penetration conductors were paralled. The parallelling of conductors changed the primary and backup protection requirements for the circuit. Primary and backup protection is provided by two breakers in series which is in accordance with RG 1.63, "Electrical Penetration Protection."

During accident conditions, aluminum produces hydrogen gas. Control of hydrogen and other combustible gases formed inside containment during an accident has been evaluated in the containment analysis. An analysis for the additional aluminum inside containment determined that the hydrogen produced due to the additional aluminum would not result in an appreciable increase in containment pressure nor degrade the ability of the Combustible Gas Control System in preventing the hydrogen concentration from reaching its flammability limit. Based on the above, the addition of the aluminum is determined not to be an unreviewed safety question .

Evaluation Number
SE-90-232

Activity Title:

Deletion of Mild Environment from the Equipment Qualification Program

Description of Change(s):

FSAR Sections 3.10N, 3.10B, 3.11N, 3.11B, Appendix 3A, 7.5 and 17A were revised to exclude mild environment from the EQ program.

Summary of Evaluation:

A mild environment is characterized by the absence of environmental conditions, during and following design basis events that could pose significant challenges to redundant systems, and thus, induce common mode failures. In the absence of common cause mechanism, failures are expected to occur randomly (i.e., in an unpredictable and unavoidable manner. The redundancy of safety-related systems prevents random failures from adversely affecting the execution of a safety function. The occurrence of random failures is therefore tolerable and fully accounted for by the safety system design.

At CPSES, the existing mild program relies on the procurement documents and surveillance programs to maintain the equipment for its installed life. Replacement of this equipment is based on the existing maintenance and surveillance programs in conjunction with a trending program to avoid unneccessary replacement of equipment based on aging predictions, when by definition, equipment located in a mild environment will not experience environmental extremes that are worse than its normal environment; and therefore a significant aging mechanism does not exist.

The inclusion of the mild environment program as part of the maintenance and surveillance programs is consistent with the Standard Review Plan, Section 3.11.

Evaluation Number
SE-90-234

Activity Title:

Installation of Isolation Valves on Condensate Polishing Vessels

Description of Change(s):

The modification consists of the installation of 14" manual isolation valves on each condensate polishing demineralizer influent/effluent line.

Summary of Evaluation:

The existing butterfly valves did not adequately isolate the polishing demineralizer against a 600 psig line pressure. The new isolation valves allow for the removal from service individual demineralizers without depressurizing the entire Condensate System.

The modification has no effect on the functioning of the Condensate System and does not impact any existing accident analyses or create the potential for any new accident.

Evaluation Number
SE-90-235

Activity Title:

Changes in Condensate pump trip logic and backup control systems to reduce unit tripping because of component single failures.

Description of Change(s):

The modifications include the following:

1. Replaces existing Condenser hotwell low-low level switches (with switches having more contacts) and adds an additional switch to provide for 2-out-of-3 coincident logic to trip both Condensate pumps when the 2-out-of-3 logic is actuated. Previously, each pump tripped separately from individual low-low level switches.

2. Replaces the 2 High-High/Low-Low level alarms in the control room with 2 new alarms. One alarm monitors condenser high water level. The other alarm will be operated from the new condenser low water level switches and will provide annunciation before condensate pump trip on low water level.

3. Adds a solenoid valve to the low flow condensate makeup valve. This solenoid bypasses the I/P converter, permitting the existing air supply to open the condensate makeup valve upon receipt of a low-low condenser hotwell level signal. This modification permits additional time for operator action (before condensate pump trip) in the event of a falling hotwell level due to a failure of the I/P converter.

4. Adds a second level switch, electrically connected in series with the original, to close the Condensate Reject to Condensate Storage Tank valve. This modification increases the reliability for valve closure in the event of condenser hotwell draining due to the failure of one level switch.

Summary of Evaluation:

All the modifications are implemented to reduce the probability of transient in the condensate system causing a plant trip due to a single component failure in the condensate system.

The effect of the modification was evaluated for two events (and their potential impact on accident analyses): the loss of both condensate pumps, and the loss of condenser vacuum. In neither event did the modification cause an increase in the probability or consequences of a previously analyzed accident or create the possibilty of a new accident.

Evaluation Number
SE-90-238

Activity Title:

Installation of a metal sided building to provide freeze protection for the Chilled water Surge Tank and associated piping.(LDCR-90-218)

Description of Change(s):

Installation of a new insulated building with heaters provides the freeze protection of the chilled water surge tank equipment and piping. This heated building maintains interior of the building at a temperature such that the Plant Chilled Water system remains in operation in case of a severe winter. Heat tracing provided earlier is no longer required ; therfore, it is deleted from the figure.

Summary of Evaluation:

Freeze protection of the chilled water surge tank equipment and its piping is provided by the new insulated building with heaters. This alternative arrangement to the heat tracing will ensure continued operation of the Plant Chilled Water System in a severe winter. Addition of the metal sided building on the Fuel Building roof will impose an additional load on the fuel building roof; however, it does not exceed the design load of the fuel building structure. Removal of the heat tracing from the chilled water surge tank equipment and piping will not effect the continued operation of the Plant Chilled Water System.

Evaluation Number
SE-90-239

Activity Title:

Install permanent assemblies to provide Service Air to the Control Room, and Unit 1 & 2 Cable Spreading Rooms.(LDCR-90-233)

Description of Change(s):

Piping is installed through existing penetrations in Cable Spreading Rooms 133 and 134, Auxiliary Building, Corridor 207 and the Control Room. Service Air is provided through this piping to operate Bisco pumping unit which is used for sealing the Control Room penetrations.

Summary of Evaluation:

Piping will be installed through existing penetrations in Cable Spreading Rooms 133 and 134, Auxilliary Building Corridor 207 and the Control Room. Service Air will be supplied to this piping via a flexible hose in corridor 207 as maintainence activities require. This feed will supply the Control Room as well as Unit 1 lnd 2 Cable Spreading Rooms.

Service Air is needed in the Control Room to provide service to the air operated Bisco pumping unit which is used for penetration seal work. This modification will allow penetration seal work to be performed without breaching the Control Room Positive Pressure boundary. This modification will also allow penetration work to be performed in the unit 1 and 2 Cable Spreading Rooms without breaching the security barriers. Implementation of this activity will not affect the Control Room positive pressure, since the Control Room seal material being removed will not exceed the allowable open area as defined in WOS-9017.

Evaluation Number
SE-90-240

Activity Title:

Replacement of temporary fill connection to Demineralized Water Storage Tank with permanent fill connection. (LDCR-90-234)

Description of Change(s):

Replace 3" temporary makeup fill connection for Demineralized Water Storage Tank with permanent fill connection with stainless steel piping and adequate permanent supports.

Summary of Evaluation:

Replaced 3" temporary makeup fill connection with permanent fill connection with stainless steel piping and adequate permanent supports. This design change provides the supply of makeup water to the demineralized water storage tank from mobile water tratment skid.This design change is implemted on the non-safety portion of the Reactor makeup and Deminerlizer system and hence their is no impact on the safety portion of the system.

Evaluation Number
SE-90-241

Activity Title:

Enhanced monitoring and alarm capability for the RHR System during Mid-Loop operation.

Description of Change(s):

Additional monitoring and alarm capability for RHR pump suction pressure and motor current as well as linear pump discharge flow indication have been provided by the following installations to enhance RHR system monitoring capability per the requirements of Generic Letter 88-17:

a. Suction pressure Transmitters with associated instrument tubing and isolation valves.
b. Current transformers and transducers in the RHR pump switchgear.
c. New cables routed from above transmitters and transducers to the 7300 analog racks to the main control board.
d. Suction pressure and current indicators on the main control board.
e. Replacement of FI-988 and scale plates for FI-618 and 619 on the main control board.

Summary of Evaluation:

The suction pressure transmitter mechanical design is consistent with existing criteria. The installation and testing will be in accordance with established administrative processes.

This change departs from previous practice by the use of the 1E current transformer (CT) as an isolation device between the 1E switchgear (CT primary side) and the non-1E portion of the current indication circuitry (CT secondary side).
However, the failure of the 1E circuit on the CT primary side caused by a malfunction in the non-1E circuit on the CT secondary side is not a credible failure mode based on the following two examined possibilities:
1- The basic design of a current transformer is such that a short circuit of the secondary windings (non-1E circuitry) does not affect the CT primary windings (1E portion of the switchgear).
2- In the case of an open circuit in the non-1E circuitry connected to the CT secondary, high voltages will occur in the CT winding. This could result in a partial discharge, however such a discharge is minimized by the CT design which has its windings encapsulated in a rigid casing with a conductive coating to short out voids; and by a conducting coating applied to the external casing surface and connected to the conductor to short out air gaps.
The absence of partial discharges at maximum operating voltage is verified by a factory test performed on each CT. Per test report, an open circuiting condition on the secondary does not violate the integrity of the CT, although the high voltage present under this condition would present a personnel hazard. Risk for personnel injury is minimized by installing the CT within the switchgear enclosure which is only entered by plant personnel when the switchgear is deenergized.

Evaluation Number
SE-90-241

The remaining aspects of the electrical design are consistent with existing design criteria. The CTs and current transducers installed in the switchgear cabinets are seismically installed. All non-1E wiring is installed in accordance with established separation criteria. Therefore, no new failure modes are created for the RHR system.

Evaluation Number
SE-91-001

Activity Title:

Temporary revision to condenser steam dump permissive C9 to allow two circulating water pumps to be shutdown during winter months.

Description of Change(s):

The modification changes the logic of the condenser available Steam Dump permissive C9 to allow two out of four circulating water pumps to be shutdown during the winter months.

Summary of Evaluation:

The Safety Evaluation reviewed the proposed change against the Turbine Trip and the Loss of Condenser Vacuum accident analyses in sections 15.2.3 and 15.2.5 respectively of the FSAR. The SE found that the loss of vacuum which would occur if a third circ water pump were to trip was bounded by the analysis of 15.2.5.

Evaluation Number
SE-91-002

Activity Title:

Evaluation for the WEXTEX process for steam generator tube expansion.

Description of Change(s):

This evaluation assesses the potential safety impact of the WEXTEX process for tube expansion on the Comanche Peak Unit 1 Model D4 steam generators. During the latter manufacturing stages of the Model D-4 steam generators, a number of tubes had to be removed to facilitate repairs/modifications. This activity took place after the channel heads were in place; thus, access limitations precluded the use of step-wise mechanical expansion as was used for expansion of the tube ends not affected by the modifications. Therefore, subsequent replacement and expansion of these tubes into the tubesheet was effected by the use of explosive expansion methods (WEXTEX). The records indicate a total of 3839 tube ends were involved in this reexpansion. Hot leg and cold leg tube ends were expanded in all four of the steam generators, representing approximately ten percent of the tubes in the four-loop Unit 1 installation.

Summary of Evaluation:

The tube ends in the steam generators have been shot peened as a means of providing enhanced resistance against primary water stress corrosion cracking(PWSCC). This measure has been implemented on virtually all steam generators in which the tube-sheet crevice closures were effected by full depth mechanical rolling. Historically, explosive transitions (the WEXTEX process) have exhibited a significantly lower susceptability to PWSCC than mechanically rolled transitions due to the lower residual stresses associated with the explosive expansion process.
Therefore, operation of Unit 1 with a portion of the steam generator tubes expanded by the WEXTEX process is not expected to result in an unreviewed safety question pursuant to 10 CFR 50.59 criteria.

Evaluation Number
SE-91-003

Activity Title:

Addition of Unit 2 computer programs for piping and pipe supports in the FSAR.

Description of Change(s):

Add the following Bechtel computer programs for piping and pipe supports to the FSAR, Section 3.9B and Appendix 3B: ME101, LEAP; ME150, FAPPS; ME035, BASEPLATE; ME153, MAPPS; ME149, SIGNIT; ME148, CAPPS; and ME214, LSAPS. These computer codes are being used at CPSES and have been used at other nuclear power plants successfully.

Summary of Evaluation:

This change adds Bechtel computer programs for piping and supports that will be used at CPSES. These programs are being used to analyze piping stress and piping support design in order to qualify the designs of class 2 and 3 piping and class 1, 2, and 3 piping supports. These program names and descriptions need to be added to the FSAR to reflect their use. This change documents this use.

These computer codes have been successfully used and qualifed for use at other nuclear power plants by Bechtel.

The computer programs are: ME101, LEAP; ME150, FAPPS; ME035, BASEPLATE; ME153, MAPPS; ME149, SIGNIT; ME148, CAPPS; and ME214, LSAPS.

The safety evaluation SE-91-003 indicated that there is no unreviewed safety question associated with the addition of these computer codes to the FSAR.

Evaluation Number
SE-91-004

Activity Title:

Temporary installation of a test blind flange at the 48" containment isolation valves to confirm compliance with T/S 3/4.6.1.7.

Description of Change(s):

u test blind flange was installed to individually leak rate test the valves in Containment penetrations MV-1 and MV-2. Subsequent to the test a flange remained in place, downstream of CIV HV-5536 and outside Contaiment Air supply penetration, although it was not required since the test provided acceptable results.

The test methodology used for determining leakage rates has been changed by this temporary modification to comply with the corrective actions described in LER 90-024 submitted in TU Electric letter logged TXX-90311 on September 24, 1990. Previously, the Containment isolation valves were tested simultaneously and the leakage rate reported was the total leakage rate measured. This change assesses which valve in each penetration has the highest leak rate with stem leakage properly considered to confirm that the penetration is in compliance with Technical Specification 3/4.6.1.7.

Summary of Evaluation:

The failure modes of the Containment ventilation penetrations are unaffected by the addition of the blind flange at the penetration. The installation of a passive device such as a blind flange cannot contribute to or affect the failure mode of the Containment isolation valves in any manner except the weight of the flange with respect to a seismic event. This installation has been analyzed for pressure retention, structural and seismic consideration and determined to be acceptable. With the blind flange in place, all stresses for the penetration and valves are within FSAR and Code allowables.

Since the valve is already locked closed in its safe position it does not have an active function. The safety function is simply to prevent excess leakage through the penetration when both valves are locked closed. Therefore, leaving a blind flange in place provides an additional boundary. Since there is no credible event that a blind flange can initiate, there is no increase in probability of occurrence.

Evaluation Number
SE-91-006

Activity Title:

Temporary provision of clarified water supply line for Unit 2 flushing

Description of Change(s):

This activity involves the temporary modification of the water treatment system to provide clarified flushing water for Unit 2.

Summary of Evaluation:

This activity does not involve any safety related equipment and does not affect any previously analyzed accident/malfunction nor does it create the potential for any new accident/malfunction.

Evaluation Number
SE-91-007

Activity Title:

Clarification of Plant Ventilation Chilled Water System design and normal operation configuration. (LDCR 91-027)

Description of Change(s):

The plant Ventilation Chilled Water System as defined in the FSAR section 9.4E.2.1 (a) requires operation of all five chillers during all plant operating modes. The plant conditions vary with the different indoor and outdoor conditions as well as different cooling loads in the plant. The Plant Ventilation Chilled Water System should be operated based on the plant cooling loads. The plant operators should have the flexibility to operate the system most efficiently and economically for the system and its components reliability and ensure the system meets the area temperature requirements of the plant Technical Specification limits.

Summary of Evaluation:

The FSAR section 9.4E.2.1(a) at present requires operation of five of the six chillers at all times. Five chillers are required to operate to provide maximum design capacity (3082 tons) of the system necessary to maintain area temperatures within Technical Specification limits. This capacity is based on the maximum outdoor temperature of 110 F. However, during the winter it is only necessary to run one chiller in the Auxiliary Building and one chiller in the Turbine Building due to the decreased heat load on the system. This will also allow conducting the slave relay test(OPT-463), which results in the deenergizing of two of the four Auxiliary Building Chillers. Also two chillers are required to operate during loss of offsite power to provide chilled water to the Unit 1 containment cooling units and positive displacement pumps fan coil units. Flexibilty to operate the system with less than five chillers is necessary to ensure the system and its components reliability.

Evaluation Number
SE-91-008

Activity Title:

Addition of engine roll test capability in each diesel generator channel 2 control circuit.

Description of Change(s):

Added the capability to perform an engine roll test from channel 2 of the diesel generator control circuit. This feature exists in channel 1 of each diesel generator. Channel 2 is being modified to be similar. New "Engine Roll Channel 2" pushbuttons have been added to the diesel control panels.

Summary of Evaluation:

The design modification involved the addition of an engine roll test circuit for Channel 2 of the diesel generator. The spring-return pushbuttons installed to conduct the test and the associated wiring is within the diesel generator control panel. The engine roll test is designed such that during maintenance/repair, the engine can be rolled and the starting air block valves tested without starting the diesel engine. Previously, this test feature only existed on Channel 1 of the diesel generators. The same feature is being added to Channel 2 of the diesel generators to allow additional flexibility when testing.

Evaluation Number
SE-91-009

Activity Title:

Administrative change to exclude exempt radioactive quantity sources from inventory control (LDCR SA-91-022).

Description of Change(s):

This change to the FSAR, paragraph 12.5.3.7, clarifies that TU Electric will inventory and control radioactive by-product sources which exceed the concentrations listed in 10CFR30.

Summary of Evaluation:

This change affects the adminstration process for the inventory of radioactive sources. As such, it does not affect any previously defined accident as there are no accidents or malfunctions associated with this change; it does not create any new type of accident not previously evaluated since there are no failure modes associated with this change; and it does not affect the margin of safety of any operational parameters.

Evaluation Number
SE-91-010

Activity Title:

Replacement of EDG Start air pressure switches with auxiliary valve in conjuction with different type of pressure switch(SA-91-025).

Description of Change(s):

Replace the Diesel Generator air pressure switch, which is unavailable, with an new type of air pressure switch and addition of an auxiliary valve in the Diesel Generator Air Starting System. The air pressure switch is required to start the air compressor at 220 psig and stop at 250 psig. Implementation of this change only differs in that the two components i.e., the auxiliary valve and the new air pressure switch are required to operate the air compressor in lieu of only one previous air pressure switch. The air starting system are redundant per each train of the Diesel Generator System.

Summary of Evaluation:

Diesel Generator Air Starting System air compressors are controlled by air pressure switches, which is not available from the vendor. Replacement of the existing air pressure switch requires the system modification to add auxiliary valve along with the new air pressure switch. The Diesel Generator Air Starting System is redundant per each train of the Diesel Generator System. Failure of the auxiliary valve or air pressure switch will not impact the operation of the Diesel Generator system. The air receivers of the Diesel Generator Air Starting System stores the air for 5 start of the Diesel Generator System. The failure of the air compressors does not render the Diesel Generator inoperable. If the new control devices leak excessively the compressor will run more frequently to maintain pressure in the receiver. If the leakage and additional operation of the air compressor went unnoticed, low pressure alarm on the air receiver intiate at 210 psig, which alert the operator for the appropriate corrective action. Implementation of this design modification for addition of auxiliary valve and a new air pressure switch will not impact the safety function of the Diesel Generator System.

Evaluation Number
SE-91-011

Activity Title:

Additional services in the Fuel Building to allow decontamination activities in Rooms 250 and 250-B (LDCR-91-012,013,014,0?0 & 024).

Description of Change(s):

Added the following equipment and services for rooms 250 and 250B in the fuel building to allow the decontamination activity:
1). Addition of piping for demineralizer water, instrument and service air, floor drains.
2). Exhaust duct connection to the drying cabinet.
3). Train C, non-safety conduits for lighting and power outlets.
4). Addition of handrails on platform at E.L.832'-6" and ladder to provide access to this platform from floor at E.L. 838'-9".
5). Added dryer, washing machine, sinks, sorting tables and other miscellenuous equipments in room 250B.

Summary of Evaluation:

Services in the form of air, water, electrical power outlets, lighting, access ladders and platforms, and ventilation supply and exhaust ducts are being provided to allow the decontamination activity in rooms 250 and 250B. Also added in the room 250B are dryer, washing machine, sinks, sorting tables etc. to provide the facility required for the decontamination activity. All the services provided to the decontamination areas in the fuel building are non-safety related and non-seismic class II and will not impact safety of the plant.

Evaluation Number
SE-91-013

Activity Title:

Installation of Fuel Building Hot Shop
LDCR SA-91-021.

Description of Change(s):

Remove the existing radioactive waste solidification equipment from rooms 251 and 252 and replace it with equipment and tools which allow the area to be used as a hot shop.

Summary of Evaluation:

Rooms 251 and 252 in the Fuel Building are will be used as a hot machine shop rather than a radioactive waste solidification system. the waste solidification system (as discussed in the Process Control Program) is handled by vendor equipment. This change then does not change any process previously discussed but allows the rooms to used more effectively. Consequently, since the addition of the hot machine shop equipment does not interconnect with the existing rad waste processing system and removes equipment which is no longer utilized, implementation of this change does not constitute an unreviewed safty question.

Evaluation Number
SE-91-014

Activity Title:

Installation of Temporary Gantry Crane and Holding Platform for Installation of Spent Fuel Storage Racks.

Description of Change(s):

The temporary modification provided for the installation of a temporary gantry crane to install spent fuel storage racks into Spent Fuel Storage Pool No. 1 and No. 2 in the Fuel Building and the installation of a temporary holding platform over the wet cask pit area on the 860' elevation of the Fuel Building. The temporary gantry crane used the crane rails of the Fuel Handling Bridge Crane.

The installation of the temporary gantry crane was required because the Fuel Building Overhead Crane cannot operate over the spent fuel pools and the Fuel Handling Bridge Crane does not have sufficient capacity to handle a fuel rack. The installation of the temporary holding area platform was required to provide an area to transfer the spent fuel rack from the Fuel Building Overhead Crane to the temporary gantry crane.

Summary of Evaluation:

Credible accidents or equipment malfunctions associated with this temporary modification are those related to heavy load drop considerations. During rack installation, no fuel was in the proximity of installation activities. Equipment required for safe shutdown or decay heat removal could not be damaged by a heavy load drop. The Fuel Building Overhead Crane and the temporary gantry crane were used for installation of the racks. The guidelines of NUREG-0612 relating to the control of heavy loads are satisfied for the fuel rack installation activities. The use of the existing crane rails has been evaluated and is acceptable. No spent fuel is in the fuel building, so personnel radiation exposure is not a credible consideration.

Evaluation Number
SE-91-015

Activity Title:

Changes to TRM section 3.1 "Snubbers".

Description of Change(s):

This change revises the Technical Requirements manual with respect to snubber testing. The changes include replacing the visual examination frequency table and wording in section 3.1b with the proposed table and wording in Generic Letter 90-09.

Delete the reference to the "Reject Line" in the "37" sample plan in section 3.1b(2) and delete the "Reject Line" in Figure 3.1-1.

Delete references to the "55" sample plan in section 3.1e(3) as the plan will not be used at CPSES.

Delete reference to snubbers specifically required not to displace under continuous load in section 3.1f(4).

Summary of Evaluation:

The described changes to the Technical Requirements Manual retain the confidence level for snubber operability resulting in no affect on any accidents of malfunctions of equipment evaluated in the licensing based documents.

Since no change in the confidence level for snubber reliability exists, the potential for a new type of unanalyzed event is not created.

No structures, systems, components, or system parameters are affected by the changes to the TRM. Therefore, no reduction in the margin of safety for any Technical Specification basis will be effected.

Evaluation Number
SE-91-016

Activity Title:

Blind flange installation at the 48" Containment exhaust penetration to maintain Containment integrity and compliance with T/S 3/4.6.1.7.

Description of Change(s):

A temporary modification was made on the 48" Containment Exhaust penetration to install a blank flange outside Containment for CIV HV-5538. This change was required to bring this penetration in compliance with Technical Specification 3/4.6.1.7 by way of a blank flange and subsequent demonstration to meet the specified leakage limits.

Summary of Evaluation:

The failure modes of the 48" Containment Exhaust ventilation penetration is unaffected by the addition of the blind flange in lieu of the Containment isolation valve. The installation of a passive device such as a blank flange cannot contribute to or affect the failure mode of the Containment isolation valves in any manner except the weight of the flange with respect to a seismic event. This installation has been analyzed for pressure retention, structural and seismic consideration and determined to be acceptable. With the blank flange in place, all stresses for the penetration and valves are within FSAR and Code allowables.

Since the valve is already locked closed in its safe position it does not have an active function. The safety function is simply to prevent excess leakage through the penetration when both valves are locked closed. Therefore, leaving a flange in place provides an additional boundary. Since there is no credible event that a blind flange can initiate, there is no increase in probability of occurrence.

The penetration sealing surface is being transferred from the valve resilient seat to the blind flange with its resilient gasket material. The leakage integrity tests with a maximum allowable leakage rate for the Containment isolation valves and attached blind flange assures that the valves are operable.

Evaluation Number
SE-91-017

Activity Title:

Add the temperature indicator, rebalance the air flow and increase the chilled water flow for the MS & FW pene.area HVAC System(SA-90-221).

Description of Change(s):

Added temperature indicator, increased air and chilled water flow, balance the air flow, and added insulation on the piping, equipment and supports in the Main Steam and Feedwater Penetration area and its HVAC system.

Summary of Evaluation:

The Main Steam and Feed Water Penetration areas were experiencing high temperatures due to minor steam leaks and insufficient insulation on pipes, pipes supports and valve bodies located in this area. In order to reduce the heat loads, piping, pipe supports, valve bodies, and equipment in this area is insulated. To efficiently cool this areas, chilled water flow through the cooling coils of the MS & FW Penetration HVAC System is increased. The air flow through this areas is increased which resulted in balancing the air flow of the system. The temperature indicator is added in the exhaust duct of the MS & FW Penetration HVAC system, which effectively monitor the bulk average temperature of the Main Steam and Feed Water areas, and can be used to demonstrate compliance with the Plant Technical Specification Limits.

Evaluation Number
SE-91-019

Activity Title:

Steam Generator 90/10 Flow Split modification.

Description of Change(s):

Flanges and flow restriction orifices are installed in the four main feedwater lines inside containment downstream of the 6" branch lines to auxiliary feedwater steam generator nozzles. The modification provides additional flow resistance in the main feedwater lines to ensure that a maximum of 90% of the total feed flow (at 100% power) enters the steam generator via the main feed nozzle. This maximum flow is imposed to satisfy the Westinghouse criteria to limit tube vibration in the Model D4 steam generators. The remaining 10% flow enters the steam generator via the auxiliary feedwater nozzle, thus bypassing the preheater section of the steam generator (the location of tube vibrations).

Summary of Evaluation:

An evaluation was performed to assess the effects for the addition of the orifice plates and flange assembly. The evaluation examined the following:

Piping, flange, and orifice stress analysis, break postulation, component loads and movements.

Jets, pipe whip and blowdown loads.

Mass and energy releases from postulated breaks inside and outside containment.

Containment pressure and temperature analysis due to main steamline or feedwater line breaks.

Environmental Qualification impacts

FSAR accident analyses

The evaluation concluded the original analyses performed continue to be limiting and envelope the effects of this change.

Evaluation Number
SE-91-021

Activity Title:

Installation of instrument air supply to the UPS HVAC System condenser throttling control valve .

Description of Change(s):

Installation of the instrument air supply from the Instrument Air System to the condenser throttling valve of the UPS HVAC System. The existing UPS system air supply source will remain as a backup source.

Summary of Evaluation:

The existing air supply, dedicated compressors, requires extreme amount of maintenance and provides poor quality air to the condensing water control throttling valve. By installing plant instrument air directly to the UPS HVAC units, the quality of air will improve, thereby increasing the reliability of the valve and the system. The existing air compressor will operate under a very light load, which will reduce its maintenance and the amount of particles and dirt around the system. This design modification will improve the quality of the air, reduce the maintenance and enhance the performance of the UPS HVAC System. The existing dedicated air compressor, which is connected to a class 1E power supply, will remain as a backup source of air supply and will be available in case of the failure of the plant instrument air supply.

Evaluation Number
SE-91-022

Activity Title:

Use of Copper Sheathed Cable Inside Containment in Lighting Systems

Description of Change(s):

The change expands the scope of Metal Clad cable to include Copper Sheathed (CS) cable. The CS cable will be used in non-safety related 120VAC lighting systems and 12VDC emergency lighting systems inside containment. The installation of conduits and pulling of lighting cable through conduit is a lengthy and manhour intensive compared to the direct installation of CS cable. The above applications of CS cable will provide ease of installation during Unit 2 construction and for considerations of ALARA requirements during Unit outages.

Summary of Evaluation:

For electrical separation purposes, test and analyses support that CS cable is considered equivalent to conduit. In addition, Wyle test report #53575, for South Texas Project, demonstrates that 1" electrical separation for CS cable provides adequate protection against electrical faults. UL crush tests demonstrate that the mechanical integrity of CS cable is better than Aluminum sheathed cable, which has been previously reviewed and approved by the NRC in similar cable applications. CS cable meets the seismic Category II requirements. CPSES applications of CS cable are limited to non-safety lighting systems inside containment. The copper sheath does not react with the containment accident environment; therefore hydrogen generation is not a concern. In addition, since cables are shielded by the copper sheath and are not exposed, the cables are not considered as combustible loadings or intervening combustibles. Based on the above, the use of CS cable is not considered an unreviewed safety

question for the specific applications identified.

Evaluation Number
SE-91-C23

Activity Title:

Addition of pressure switches to provide two-out-of-three coincidence logic for Main Feed pumps 1A & 1B low lube oil pressure trip.

Description of Change(s):

Provides 2-out-of-3 coincident logic for the low lube oil pressure trip for Main Feed pumps 1A and 1B and for Main Feed pump Turbines 1A and 1B. This coincident logic also incorporates a short time delay so that spurious transients will not cause unnecessary tripping of the Feedwater pumps.

In addition, this change provides for a similar 2-out-of-3 logic for the Main Feed Pump Turbine stop valve control oil pressure except that the control oil logic has no time delay

Summary of Evaluation:

No new failure modes associated with the implementation of this change have ben identified. The modification improves the reliability of the tripping function of the Main Feed Pumps, both to avoid unnecessary tripping, and to avoid a single component failure which could disable the trip function when a trip is necessary.

Chapter 15 events, Loss of normal Feedwater flow (FSAR 15.2.7) and Feedwater system malfunction that results in a decrease in Feedwater temperature (FSAR 15.1.1) were evaluated for impact by this change. Implementation of this change does not affect the feedwater pump trip setpoints or the instruments described as being necessary in the accident analyses. Nor does it affect the ability of the Auxiliary Feedwater System to operate as intended. The probability or consequences of any accident previously analyzed is not impacted.

Evaluation Number
SE-91-024

Activity Title:

RHR suction valve auto closure interlock removal.

Description of Change(s):

This activity involves the removal of the RHR suction isolation valve autoclosure interlock (ACI) circuitry and the addition of a "Valve-not-closed" alarm which alarms when any of the four RHR suction isolation valves are not fully closed and RCS pressure is above the alarm set point. This change was implemented to reduce the potential for spurious closure of the RHR valves during non-power cooling operations or when the RHR system is used for overpressure protection.

Summary of Evaluation:

WCAP-11736 provides a generic safety evaluation for ACI removal. The NRC reviewed this WCAP and in the SER indicated that it was acceptable for referencing in individual plant safety evaluations. A technical specification change related to the removal of the ACI was submitted to the NRC in TXX-91151 dated June 28, 1991, and the NRC approved an amendment to the technical specifications by letter dated October 18, 1991. TXX-91151 provided detailed information regarding potential safety concerns associated with this activity. The primary concern was the potential for increasing the probability of an interfacing system LOCA (i.e., LOCA caused by the failure of low pressure system due to overpressurization from a connecting high pressure system). Based on the referenced WCAP-11736 analyses, and implementation of the WCAP recommended compensatory measures, the overall probability for an intersystem LOCA was decreased. In addition, the probability for loss of the RHR System, during non-power operations or when used for overpressure protection, has been reduced.

Evaluation Number
SE-91-026

Activity Title:

Deletion of Unit 2 dampers in Primary Plant Ventilation System (PPVS). LDCR SA-91-045.

Description of Change(s):

Delete the dampers in the Primary Plant Ventilation System for Unit 2 based on the engineering analysis. The corresponding dampers in the Unit 1 PPVS is either deleted or abandoned in-place (no instrumentation or operators removed), or put in locked open position..

Summary of Evaluation:

The deletion of the Unit 2 dampers in the supply and exhaust side of the PPVS will enhance the operability of the system by reducing the system air pressure drops and increased air flows.

The deletion of these Unit 2 dampers will enhance the overall operation of the PPVS and therefore, will ensure that the spread of airborne contamination is limited by maintaining air pressure gradients and airflows from areas of 'ow potential airborne contamination to areas of higher potential airborne contamination, and is filtered prior to being released to the atmosphere.

The corresponding dampers in Unit 1 have been deleted, abandoned in place (no instrumentation or actuators remain) or do not serve any safety related function and have not been removed or lock open. Their existence do not adversely affect the safe operation of the plant.

Deletion of the Unit 2 dampers will not have any effects and/or failure modes that would potentially impact the radiological consequences already in existence in the areas served by the PPVS.

Evaluation Number
SE-91-028

Activity Title:

Vacuum deaerator vacuum pumps seal water recycle system.

Description of Change(s):

This activity involves the installation of a recycle loop on the seal water side of the Vacuum Deaerator Vacuum Pumps, using Turbine Plant Cooling Water for seal water cooling. The modification was performed in order to provide an improved method for processing seal water discharge. The previous method, via the Condensate Return Unit, provided insufficient pumping capacity, resulting in clean water being dumped to drain.

Summary of Evaluation:

The components associated with this activity are all non-safety related. The activity does not affect any previously analyzed malfunction or accident nor does it create the potential for any new accident or malfunction.

Evaluation Number
SE-91-029
Revision 1

Activity Title:

Revision to the MSR drain tank controls.
LDCR SA-91-149

Description of Change(s):

This activity performs the following for Reheater Drain Tanks 1-A1, 1-A2, 1-B1 and 1-B2; Shell Drain Tanks 1-A and 1-B; and Separator Drain Tanks 1-A and 1-B:

Replaces hard pipe connections with flexible metal hoses for the heater drain level transmitters to isolate the transmitters from the drain tank vibration.

Adds additional isolation valves to provide double isolation during maintenance of transmitters without isolating other functional transmitters on the same taps.

Deletes the heat tracing associated with Moisture Separator-Reheater (MSR) level control transmitters. Problems have been encountered with the existing bridle heat tracing. The revised MSR level control system includes a modified bridle insulation system which uses the drain tank heat to maintain instrumentation above freezing during plant operation. Draining of the system is required in the event of an extended plant outage.

Adds a local pneumatic manual/auto control stations for reheater drain tanks normal drain path control valves. This improves system operation and eliminates potential transients when the plant is operating outside of normal level control loop settings.

Revises alternate drain control valves to be reverse acting (i.e., low transmitter signal input equates maximum air output to the diaphragm) such that loss of signal or power will not result in unnecessary heater drain transients.

Adds new normally energized solenoid valves between the positioner and diaphragm of the alternate drain control valves so that the drain valve fails open on high level or loss of power to protect against MSR damage or water induction to the turbine.

Summary of Evaluation:

The above modifications were made to correct heater drain system instrument problems which can result in unnecessary plant transients. The heater drain system is not safety related and independent of any protective function.

The modifications were evaluated for potential impact on Chapter 15 event 'Sudden Decrease in Feedwater System Temperature' - and it was found to be bounded by the existing analysis.

Evaluation Number
SE-91-030

Activity Title:

Exception to NRC R.G. 1.137, Part C.2.d, Sample for and removal of condensate water at one day after the new fuel oil added.(LDCR-91-052)

Description of Change(s):

Exception to NRC R.G. 1.137, Draft (Revision 0) Part C.2.d:

Accumulated condensate should be removed from storage tanks on a quarterly basis or on a monthly basis when it is suspected or known that the groundwater table is equal to or higher than the bottom of the buried storage tanks.

This exception is consistent with Part C.2.d of NRC R.G. 1.137, Revision 1 and CPSES Plant Technical Specification Sections 4.8.1.1.2.c, d, and e; and 4.8.1.2 requirements.

Summary of Evaluation:

The accumulated condensate will be removed quarterly or monthly in lieu of one day after the new fuel oil is added to the fuel oil storage tank. This requirement is consistent with the requirement of Part C.2.d of NRC R.G. 1.137, Revision 1. The CPSES Technical Specifications 4.8.1.1.2.c, d, and e; and 4.8.1.2. requires the following:
a) Requires sampling and removal of the condensate every 31 days, which is proven to be a satisfactory sample frequency given the size of the clearance sump.
b) Requires testing for accumulated water (clear and bright test) on each truckload of diesel fuel oil delivered and unloaded into the storage tank.
The water introduced in the receiving activity would take longer than one day to settle out and find its way to the clearance sump. The current Technical Specification requirement to sample for and remove accumulated water every 31 days is adequate to protect the diesel generator fuel oil system. The removal of the requirement to sample one day after new fuel oil delivery would not adversely effect the system or it's components, would have no radiological consequence, and will not have any effect on the operability of the diesel generator.

Evaluation Number
SE-91-031

Act'vity Title

Blind flange installation at the 48" Containment supply penetration to maintain Containment integrity and compliance with T/S 3/4.6.1.7

Description of Change(s):

A temporary modification was made on the 48" Containment Supply penetration to install a blank flange outside Containment for CIV HV-5536. This change was required to bring this penetration in compliance with Technical Specification 3/4.6.1.7 by way of a blank flange and subsequent demonstration to meet the specified leakage limits.

Summary of Evaluation:

The failure modes of the 48" Containment Exhaust ventilation penetration is unaffected by the addition of the blind flange in lieu of the Containment isolation valve. The installation of a passive device such as a blank flange cannot contribute to or affect the failure mode of the Containment isolation valves in any manner except the weight of the flange with respect to a seismic event. This installation has been analyzed for pressure retention, structural and seismic consideration and determined to be acceptable. With the blank flange in place, all stresses for the penetration and valves are within FSAR and Code allowables.

Since the valve is already locked closed in its safe position it does not have an active function. The safety function is simply to prevent excess leakage through the penetration when both valves are locked closed. Therefore, leaving a flange in place provides an additional boundary. Since there is no credible event that a blind flange can initiate, there is no increase in probability of occurrence.

The penetration sealing surface is being transferred from the valve resilient seat to the blind flange with its resilient gasket material. The leakage integrity tests with a maximum allowable leakage rate for the Containment isolation valves and attached blind flange assures that the valves are operable.

Evaluation Number
SE-91-032

Activity Title:

Provisions of a feed for three hermetic water chillers being added to the common Chilled Water System in the Unit 2 Turbine Building.

Description of Change(s)

The modification involves disconnecting the electrical power feed for the 7000 kW electrical auxiliary boiler and revision of the breaker wiring to allow feeding the chillers by this breaker. The raceways for the electrical boiler are reused to feed the new chillers.

Summary of Evaluation:

The three new chillers are part of the nonsafety chilled water system, and are fed from the breaker which was feeding the electrical auxiliary boiler. The electrical boiler is no longer being used and has been retired in place. The chiller feeder cables and raceways are nonsafety related, routed in nonseismic areas, and routed in accordance with the electrical separation specification, per RG 1.75. The cables are 4/0 in size and meet the flame test of IEEE-383 as described in Section 1A(B) of the FSAR. The combustible loading in the Unit 1 fire area is less than that of the electrical boiler with a cable size of 350 MCM. Based on the above, the design modification is not considered an unreviewed safety question for the application described above.

Evaluation Number
SE-91-056

Activity Title:

Deletion of diesel generator valves from the Inservice Testing Plan.

Description of Change(s):

Delete 18 valves from the scope of the Inservice Testing Program. These valves are not within the scope of ASME B&PVC Section XI, Subsection IWV, 1986 Edition.

Summary of Evaluation:

Eighteen valves in the Diesel Generator system are to be deleted from the scope of the Inservice Testing Program Plan. Either the valves are not an ASME Code Class 1,2, or 3 valve; or the valves are not "active" valves performing a safety function in accordance with the Code; or both.

Therefore, this change does not involve an unreviewed safety question.

Evaluation Number
SE-91-057

Activity Title:

Provision of one hour fire rated cable in fire safe shutdown circuits as an alternate to a one hour fire barrier.

Description of Change(s):

The change expands the scope of one hour fire rated barries to include one hour fire rated cable for electrical separation purposes. One hour fire rated cable will be used in power and control fire safe shutdown circuit applications in areas outside containment where the total radiation dose is less than or equal to 50 MRADS gamma.

Summary of Evaluation:

One hour fire rated cable is Rockbestos cable constructed of a continuously welded stainless steel sheath, 12 mils thick, with organic and inorganic materials. The outer sheath is similar to the outer sheath of metal clad lighting cable and is therefore considered equivalent to conduit for electrical separation purposes. The jacket material is a glass braid with a layered silicone rubber insulation. The conductors are high temperature nickel-cladded copper sized for up to 1700 degrees F. The inorganic insulation in conjunction with the glassbraid is considered similar to a wrap of woven silicone dioxide, which is considered equivalent to conduit. The cable is Class 1E qualified per IEEE 323-1974 and IEEE 383-1974 for flame retardancy (unaged cables only). The cable is considered equivalent to conventional cable enclosed within a one hour fire barrier (e.g. TSI thermo-lag) since it maintained its electrical integrity for one hour when exposed to the ASTM E-119-1971 source and hose stream test after a fire. Because of the continuous construction, there are no exposed combustibles that would representa combustible loading impact. Based on the above, the one hour fire rated cable is not considered an unreviewed safety question for fire safe shutdown or electical separation applications.

Evaluation Numb ~
SE-91-058

Activity Title:

Clarification of the inspection and testing requirements of the Containment Preaccess Filtration system. (LDCR 91-068)

Description of Change(s):

The clarification to the exception listed under Regulatory guide 1.140 is required as table 1 (see notes 2 and 5) of ASME/ANSI N510-1980 allows a deletion of periodic in-place testing of filtration systems that are recirculating and located inside containment. The Containment Preaccess Filtration System is a recirculating system inside containment building and falls under this category. However, sections C.5.c and C.5.d of Regulatory Guide 1.140 state that the in-place testing shall be performed as described in ASME/ANSI N510-1980, Sections 10 and 12. Neither Sections 10 and 12 of ASME/ANSI N510-1980, nor Regulatory Guide 1.140 direct the reader to table 1 of the standard. Therefore, it is necessary to describe the exception taken to the Regulatory Guide under this section of the FSAR.

Summary of Evaluation:

The preaccess filtration unit is a non-safety, 100% recirculation unit which is located inside containment. The unit is utilized to reduce airborne contamination inside containment prior to personnel entry. Deletion of periodic in-place testing of HEPA and charcoal adsorber may result in bypass of unfiltered air. Excessive bypass of unfiltered air would only extend the duration of the operation of the Containment Preaccess Filtration System prior to allowing personnel to enter the containment. In addition, any unfiltered bypass air will remain inside the containment building and therefore, will not cause any uncontrolled release of airborne contamination.

Evaluation Number
SE-91-060

Activity Title:

Convertion of Chemical Feed Building to water production support lab.

Description of Change(s):

This activity involves the conversion of existing (but no longer used) Chemical Feed Building to a Water Production Laboratory to be used in support of the Unit 2 start-up program and water production when both units are operative. The conversion includes removal of unused equipment, and the installation of new electrical and mechanical support systems (e.g., instrument ai:, demineralized water, HVAC, electrical power, sewage, etc.).

Summary of Evaluation:

The above activity does not involve any safety related systems. It does not impact any analyzed accident/malfunction or create the potential for a new accident/malfunction.

Evaluation Number
SE-91-061

Activity Title:

Removal of resin sampler in the Liquid Waste Processing System (WP). LDCR SA-91-071.

Description of Change(s):

Remove the spent resin sample point to eliminate potential resin traps.

Summary of Evaluation:

Removal of the spent resin sample point and existing piping will eliminate potential resin traps. This sample point is no longer required as the spent resins are transferred to vendor supplied processing equipment. Since this change does not create any new accidents not previously evaluated nor modify any previously evaluated accident nor impact the margin ao safety of any previously evaluated event, it does not result in an unreviewed safety question.

Evaluation Number
SE-91-062

Activity Title:

Radioactive material handling and staging in areas outside the plant.

Description of Change(s):

An area outside existing plant buildings (east of Unit 1 and Unit 2 and within the Protected Area) was established to provide additional space for handling and staging radioactive material and/or radwaste generated by the plant. This change was necessary because of insufficient space in the plant to adequately address this activity.

Summary of Evaluation:

The subject area was established by paving approximately 10,000 square feet and erecting a perimeter fence. Combustible bulk staging will not take place within 50 feet of exterior safety related structures, walls, tanks or equipment; therefore there are no structures, equipment or system parameters that could be affected by implementation of this activity. Since the activity does not affect equipment important to safety, there is no impact on previously identified accidents and malfunctions evluated in the licensing basis documents. Additionally, the activity is not associated with the Technical Specifications; there is no impact on the margin of safety.

There are no credible failure modes for structures, systems or components; however, the most serious credible radiological failure for this activity was postulated to be a dropped High Integrity Container (HIC) and subsequent spillage of high activity resin in the area outside the plant. This accident is not specifically described in the licensing basis documents; however, it was determined to be similar to an accident described in the ATCOR Topical Report (which is referred to in SAR Section 11.4) i.e., overflow of evaporator concentrates in the Fuel Building. After technical evaluation, it was determined that the consequences of the ATCOR accident would result in offsite doses well within the limits of 10CFR100 criteria. The consequences of the HIC accident were determined to be less significant.

Another accident considered was a mishap involving dry active waste (DAW) (e.g., contaminated trash in a sea-land van). If this mishap occurred during inclement whether, there could be radioactivity removed from the DAW which might end up in Squaw Creek Reservior; however, the curie content typically found in a DAW container is well below the curie content used in the Liquid Waste Holdup Tank failure accident calculation described in SAR section 2.4.12 and 15.7.2; therefore, this mishap is bounded by the liquid release accident.

Station procedures require venting HICs inside the Fuel Building to minimize the possibility of an unmonitored release. Also, the area is controlled as a Radiological Controlled Area (RCA).

Evaluation Number
SE-91-063

Activity Title:

Failure to hydrostatically test all type MV penetrations.

Description of Change(s):

This FSAR change, LDCR-SA-91-074, adds a note to page 3.1-45, Section 3.1.5.2 "Discussion" and to page 3.9B-28, Section 3.9B.3 "ASME CODE CLASS 2 AND 3 COMPONENTS AND COMPONENT SUPPORTS" stating that some of the type MV containment penetration welds were not visually inspected during the required ASME B&PVC pressure testing. The note is as follows: "Inspection of attachment welds during hydrostatic testing, as required by the ASME B&PVC was not performed on all type MV containment penetrations for Unit 1."

Summary of Evaluation:

The piping associated with the penetrations recieved ASME B&PVC Section III pressure tests, although the subject attachment welds to the type MV penetrations were not observed during the pressure testing. The attachment welds did recieve magnetic particle examinations. Local leak rate testing has been performed on the penetrations and throughout pre-operational testing, start-up, and plant operations there has been no documented evidence of leakage at the penetrations. A review of the calculated primary stresses on a sample of the penetrations indicates considerable available stress margin existing to ensure piping integrity. Therefore, the neither the Containment integrity or a system has been compromised.

Evaluation Number
SE-91-064

Activity Title:

Turbine-Driven Auxiliary Feedwater Pump (TDAFP) Response Time

Description of Change(s):

The response time for the TDAFP has been changed from sixty (60) to eighty-five (85) seconds. The TDAFP is an Engineered Safety Feature and the response time is used in the plant safety analyses.

Summary of Evaluation:

Increasing the TDAFP response time is a change to an analysis parameter, hence, is not in itself an initiating event and does not create a credible accident or malfunction. Accidents which rely on the auxiliary feedwater system (AFW) for event mitigation were evaluated to assess the effect of increasing the TDAFP response time:

1. For Loss of Nonemergency AC Power to the Station Auxiliaries and Main Feedline Break, the analyses assume that the most limiting single failure is the TDAFP. Hence, increasing the TDAFP response time has no effect.

2. For Main Feedline Break (FLB), the analysis demonstrated that no hot leg boiling occurs at any time during the event. Hence, FLB is not adversely affected by the increase in response time.

3. For Steam Generator Tube Rupture (SGTR), the acceptance criterion is isolation of the affected steam generator prior to it being completely filled with liquid. Increasing the TDAFP response time delays the delivery of AFW to the ruptured steam generator, which increases the time to steam generator overfill. Hence, SGTR is not adversely affected by the increase in response time.

4. For Small Break Loss of Coolant Accident, the original analysis assumed a total AFW flow of 1225.5 gpm. The actual measured flow is 1290 gpm, which is consistent with Technical Specification requirements. This increased flow offsets the increased response time such that there is no change in the calculated Peak Cladding Temperature.

5. Main Steam Line Break - Outside Containment Accident was evaluated for effects on equipment qualification and radiological dose. With the increased response time, the environmental parameters are consistent with the previous environments used for equipment qualification. The radiological consequences are not adversely affected because the increased response time reduces the amount of AFW delivered to the steam generator, which reduces the total mass released by the break.

The margins of safety associated with the Technical Specifications are unaffected because all event acceptance criteria were satisfied for the accidents considered.

Evaluation Number
SE-91-065

Activity Title:

LP2 Turbine repairs- TM 91-048

Description of Change(s):

This activity involves temporary repairs to the Unit 1 low pressure turbine No. 2 (LP2) to allow continued operation until permanent repairs are made. Repairs include machining rotor seal lands and cutting approximately 45% of the original row 5 blade lengths on both the generator and turbine end. In addition repairs to the casing include removing row 5 staionary blade shrouds and shaving 2mm from the blade lengths and then reinstalling the shroud.

Summary of Evaluation:

Potential impacts on turbine missile analyses, turbine overs, ·d and turbine trips were reviewed. Based on analysis of the modifications being performed and the results of non-destructive examinations performed on the rotor and blades which verified the repair condition of the components to be free from cracks, it was determined that the integrity of the turbine to withstand the of continued operation is maintained. Thus the assumptions and conclusions of existing analyses related to turbine missiles, overspeed and trips remain valid.

Evaluation Number
SE-91-066

Activity Title:

Modification of FSAR section 3.11B to allow class 1E elec. and certain RG 1.97 equip. to be qualified to mild environment. SA-91-069

Description of Change(s):

The FSAR change allows Class 1E electrical and certain accident monitoring equipment/instrumentation which experience 100% relative humidity only to be qualified using the CPSES mild environment criteria. However, the mild criteria only applies if it can be demonstrated by test, design, application, or analysis that the 100% relative humidity will not degrade the equipments' function.

Summary of Evaluation:

The FSAR change takes advantage of information available in industry which supports that equipment subjected to 100% relative humidity only (i.e. harsh for no other parameters, e.e. pressure, temperature, radiation, etc.) will perform its safety function as demonstrated by test, analyses, design and/or application. It has been determined that for these instances, it is not reasonable to apply a full-blown harsh environment qualification program. Thus, when an evaluation shows that 100% relative humidity will not impact the equipment's performance, then that equipment is effectively located in a mild environment.

Evaluation Number
SE-91-067

Activity Title:

Replacement of carpet in Control Room, Simulator and Technical Support Center.

Description of Change(s):

The existing carpet in the Control Room is worn and stained and needs to be replaced. Since the Simulator should match the actual Control Room, the carpet in the Simulator needs to be replaced also. Carpet is also being added to the Technical Support Center (Room 149A), and the office area (Room 148A) at elevation 840'-6'.
The existing carpet conforms to ASTM E-84 ratings of flamespread 30, fuel contribution 30 and smoke development 100. However, carpet conforming to this standard is no longer available. The new carpet has been tested to ASTM E-648, which is a more applicable test method for carpet installed in the Control Room. The new carpet qualifies to Class I interior floor finish per NFPA 101.

Summary of Evaluation:

TU Electric has reviewed the fire test stipulations required by NFPA 101 standards against the fire test stipulation criteria required by ASTM E-84 and determined that for the application at CPSES, equivalent fire protection is maintained assuming the new carpet, manufactured in accordance with ASTM E-648, is installed. Since the new carpet has combustible characteristics comparable to the existing carpet, there are no structures, systems or components that can be adversely affected by the implementation of this activity. The installation of the new carpet does not change the credible failure modes of any structure, system or component since the new carpet has equivalent flammability and weight characteristics as compared to the existing carpet. The installation of this new carpet does not adversely affect the ability of CPSES to achieve and maintain safe shutdown in the event of a fire.

Evaluation Number
SE-91-068

Activity Title:

Separation of NSSS snd Steam Generator Blowdown spent resin transfer lines. (LDCR SA-91-086.)

Description of Change(s):

Piping and valves are being removed from the Liquid Waste Processing System (LWPS)/Spent Resin Subsystem to separate NSSS and Steam Generator Blowdown (SGBD) spent resin transfer lines. This will avoid contamination fo SGBD spent resin lines with radioactive NSSS spent resin.

Summary of Evaluation:

Fourteen (14) valves and approximately 100 feet of piping are being removed form the LWPS to separate the SGBD and NSSS spent resin transfer lines. Two (2) valves and about 65 feet of piping will be installed to aid in this separation as a bypass line. The implementation of this modification does not change the process only the flow paths to enhance blowdown, back flushing, and prevent system cross contamination. As this modification does not create nor modify the possibility of accidents previously identified, nor effect equipment important to safety different than previously e aluated, nor impact on the margin of safety, implementation of this change does not constitute an unreviewed safety question.

Evaluation Number
SE-91-069

Activity Title:

Switchyard modification to the 345kV Parker line

Description of Change(s):

The design modification relocates the off-site power feed to transformers, XST2, 1ST and XST1/2 from the Parker transmissin line to a bus tie feed in the 345 kV switchyard. The switchyard bus tie feed is a new double breaker scheme which will allow the transformers to be fed from either the east or west 345kV switchyard bus.

Summary of Evaluation:

Prior to the design modification, XST2, 1ST and XST1/2 were fed directly from the 345 kV Parker transmission line. Thus, disturbances and/or a loss of the Parker line would result in tripping of the 6.9 kV motors, starting of the diesel generator and a slow bus transfer to the alternate power supply. On several occasions prior to the modification, the Parker line experienced external disturbances which resulted in the above electrical actuations. To preclude unnecessary challenges to the plant electrical system and to enhance the availability of XST2, 1ST and XST1/2, the design modification was implemented such that the transformers are fed from the east or west 345 kV buses via a new dual bus tie breaker scheme. This change allows the plant electrical power supply to be less susceptable to disturbances on the Parker transmission line.

Evaluation Number
SE-91-070

Activity Title:

Removal of jumper in the voltage to pulse converter to suppress output to the Containment sumps totalizer in the low flow range.

Description of Change(s):

This modification involves removal of jumper in the voltage-to pulse converter 1-FY-5159B to activate its "low pulse cut-off" circuit which is adjusted and set at 1.25 VDC (equivalent to 12.5% of span or 40 GPM). This will suppress the converter's output to the totalizer 1-FQI-5159 when the input is < 1.25 VDC (40 GPM) to suppress the instrument uncertainty in the low range and provide a meaningful indication. The change will eliminate totalizer readings at lower region in the flow indications when no sump pump is operating.

This system is required to detect and monitor leakage from the RCS pressure boundary. It is consistent woth the recommendations fo Regulatory Guide 1.45. Readings off this loop are taken periodically for trending and early detection of leakage and when required to confirm the presence of some leakage and in locating the posssible source. It is important to note that these measurements are not used to quantify the leakage for comparison with the Technical Specification limits.

Summary of Evaluation:

This change will not affect the probability of occurrence of a LOCA because the elimination of the totalizer reading in the lower range of the flow measurement only improves on the ability to provide a better trending data for detection of excessive leakage which is a precursor to a LOCA.

This change could not affect the radiological consequences of a LOCA because this loop is rendered inactive during a LOCA by a Containment Isolation signal which isolates the valves upstream of the flow sensor. Erroneous readings or loss of this loop completely attributed to the potential failure of the cut-off point failing in such a way that the cutoff voltage will neither be driven to an extreme (10 VDC) or low (0 VDC) region will be evident by the execution of periodic surveillance readings. Also, early detection is available by other methods in the leak detection system. This instrument is not used in accident mitigation, it does not have a control function and is solely used for indication.

Evaluation Number
SE-91-071

Activity Title:

Documentation of the proper connections of relays 27-2A, B.

Description of Change(s):

This change revises the drawing to correct the wiring connections of undervoltage relays 27-2A, B/1EA1, 2. These relays trip the 6.9kV motors, start the diesel generator and remove one permissive from the slow transfer to alternate supply circuitry.

Summary of Evaluation:

The drawing change documents the separation of the protection for the undervoltage relays from the Solid State Safeguard Sequencer (SSSS) relays. The drawing correction was necessary to ensure that the design reflects the as-built configuration of the plant due to pre-licensing work associated with the Fire Safe Shutdown Analysis (FSSA). The pre-licensing work was performed to ensure that if a fault occurred in the SSSS relay circuit due to a fire in the control room or cable spread room, the fault would be isolated to the SSSS circuitry without causing a false undervoltage signal.
This undervoltage signal, if actuated, would trip the 6.9kv motors, start the diesel generator, and remove one permissive from the slow bus transfer to the alternate power supply.

Evaluation Number
SE-91-072

Activity Title:

Temporary power to air compressor 1-02 TM-91-1-053.

Description of Change(s):

Implemented a temporary modification to power instrument air compressor, CP1-CICACO-02 from "C" Train motor control center, (MCC) XB1-6, until a bad shunt trip relay in the original power supply, 1EAB4-1, which was identified during testing, could be replaced.

Summary of Evaluation:

With the temporary modification installed, instrument air compressor, CP1-CICACO-02, is powered from a non-safety motor control center. The instrument air compressor is non-safety equipment which relied on the shunt trip relay to isolate it from the safety bus on a S signal, as required by RG 1.75. Using Train C power as an alternate power supply ensures that the air compressor does not affect the safety bus in the event of an S signal. In the event of an S signal, the nonsafety related bus will remain available to feed the compressor. The MCC does not have access to the diesel. Therefore, in the event of a station blackout with the temporary modification in place, the instrument air compressor could not be manually loaded onto the diesel generator. However, only one out of four air compressor is required for 100% air supply capacity, and air compressor, CP1-CICACO-01 would be available. Loss of instrument air has been evaluated as and determined to be not required for safe shutdown of the plant.

Evaluation Number
SE-91-073

Activity Title:

Installation of 4" diameter temporary piping and supports required for flushing Unit 2 piping in Unit 1 areas (REF: TM-91-1-050, Rev 0).

Description of Change(s):

This activity involves the temporary installation of 4 inches diameter flushing header with connections at various elevations in the Auxilliary Building for the purpose of flushing/testing Unit 2 equipment located in common areas.

Summary of Evaluation:

The above activity does not involve any safety related systems. The above activity has been analyzed for flooding, negative pressure boundary and unmonitored effluent releases. Based on this evaluation, the above activity does not impact any analyzed accident/ malfunction of equipment or create the potential for a new accident/ malfunction of equipment or safety margin.

Evaluation Number
SE-91-077

Activity Title:

Addition of hydrogen gas dryer to turbine generator hydrogen gas system DM-90-269

Description of Change(s):

This activity replaces the existing single tower turbine generator hydrogen gas dryer with a dryer having a dual tower design. The new tower increases the reliability and improves the maintainability of the hydrogen dryer system.

Summary of Evaluation:

The systems affected by this modification are not safety related, nor will the modification introduce any potential failure modes not previously analyzed. The potential for increasing turbine trips was reviewed and found to be unaffected by this modification.

Evaluation Number
SE-91-078

Activity Title:

Update the ODCM to add bromine isotope and change control sample location for food products. (Ref LDCRs OD-91-001 & OD-91-002.)

Description of Change(s):

Revise the ODCM to add Br-82 to list of isotopes in Table 1.1 which contribute to the ingestion dose from liquid effluents. Change the control sample location for food products listed on Table 3.1 and illustrated on Figure 3.1.

Summary of Evaluation:

Theses changes do not alter the sampling process described in the ODCM nor do they affect any equipment, system, or component related to the safe operation of the plant; therefore, these changes do not involve an unreviewed safety question.

Evaluation Number
SE-91-079

Activity Title:

Auxiliary Feedwater Temperature Instrumentation
LDCR SA-91-075

Description of Change(s):

The modification consists of relocating four temperature elements and adding four new temperature elements to facilitate the detection of AFW check valve back leakage. The AFW check valves between the S/G and the AFW pumps can potentially leak and cause steam binding of the AFW pumps. Previously, there was one temperature element in each of the four 4" AFW headers just upstream of the tie into the preheater bypass line. In this configuration the existing temperature elements did not accurately reflect check valve temperature due to the temperature elements close proximity to the preheater bypass line (which is at approx. 450 degrees) whereas the normal AFW check valve temperature is approximately 140 degrees. Relocating the existing temperature elements to just upstream of the check valves in the electric motor driven AFW pump supply lines and adding four additional elements upstream of the check valves in the turbine driven AFW pump supply lines, enables the operators to more accurately assess the extent and location of any check valve back leakage.

Summary of Evaluation:

The thermowells and asssociated temperature instrumentation for this modification are passive (indication function only) and provided no control or interlock function. Therefore their failure will not prevent AFW system from performing its intended function. The purpose of the temperature elements is to alert the operators of potential AFW pump steam binding due to check valve back leakage. The new installation improves the accuracy of such a determination.

There are no failure modes for this modification which could increase the probabilty or the consequences of any previously analyzed accident or create the potential for a new accident.

Evaluation Number
SE-91-080

Activity Title:

Provide filter micron rating range for the flexibility in filter cartridge selection, LDCR SA-91-097

Description of Change(s):

This activity involves revising the acceptable micron ratings of filters for the Boric Acid Filter, Reactor Coolant Filter, Seal Water Injection Filter and Seal Water Return filte. This change allows the flexibility for the replacement of filter cartridges with a lower micron rated filter cartridge using a different filter medium and manufacturer while maintaining required design characteristic.

Summary of Evaluation:

Potential failure modes for the filter changes all relate to faster clogging due to the lower micron rating of the filters.

Two potential failure modes associated with Seal Water Injection filters were identified related to rapid clogging of the filters which could impact RCP seals and ECCS performance. These failure modes were previously evaluated and a summary submitted to the NRC in 1990 (SE-90-204) for the installation of the Temporary Modification which preceded this permanent change. The results of that evaluation indicated that the modification would not adversely impact RCP seals or ECCS performance.

The Seal Water Return filter is protected with a relief valve so that clogging of the filter will not result in overpressurization.

The Reactor coolant filter filters resin fines and particulate from coolant leaving the purification system demineralizers or the discharge of the letdown heat exchangers when the demineralizers are bypassed. The coolant leaving the filter is directed to the VCT. A reduction in the amount of flow to the VCT due to filter plugging is offset by other sources of makeup or by bypassing the filter. This change will not adversely affect system performance.

Plugging of Boric Acid Filters is easily detected by existing flow and differential pressure indicators and alarms. Bypasses around the filters are used during filter change out.

For all the filter as described above, pressure indicators are provided to monitor the differential pressure across the filter cartridge which indicates cartridge plugging. The flow capacities of the filters are greater than the maximum design flow rates. The filters can be bypassed during changeout. The differential pressure used to indicate the need to change the filter of 20 psi is much less than the maximum allowed of 75 psi. During initial use, the replacement cartridge is expected to reach the CPSES replacement differential pressure limit of 20 psi at a more rapid rate. As the average particle size in the CVCS diminishes and cartridge performance history is developed, replacement intervals will increase.

Evaluation Number
SE-91-080

In addition, the increased particle removal ability will reduce the probability of reactor coolant pump seal degradation and reduce particulate matter wit'ⁱⁿ the CVCS thereby reducing radiation fields.

Implementation of this activity has basically the same effect on structures, systems, or components and/or system parameters as the use of the current filter cartridges. None of these changes introduce any credible potential failure modes.

Evaluation Number
SE-91-081

Activity Title:

Emergency Diesel Generator (EDG) turbocharger bolt replacement from 92 hours to 260 hours of the EDG operation.

Description of Change(s):

Emergency Diesel Generator (EDG) turbocharger bolts would be replaced periodically after approximately 92 hours of operation, and that prior to completion of the first refueling outage of Unit 1, additional testing may be performed to more accurately determine the loads in these bolts and that bolt life might be increased based upon this additional testing (Reference TXX-89826).
Additional testing on these bolts has been completed. This acivity uses the results of these tests to increase the replacement interval for these bolts from 92 hours to 260 hours of EDG operation.

Summary of Evaluation:

This activity extends the replacement interval of the turbocharger holddown bolts based upon bolt load data obtained while the diesel generator was running. This load data was used to calculate a new replacement interval using ASME III design methodology which is consistent with the Diesel Generator design. In addition, these bolts will still be visually inspected after each running of the diesel generator and if an extended diesel generator run is required (greater tha 48 hours) these bolts will be visually inspected on a daily basis after 48 hours of run time.

The turbocharger bolts failure results in the loss of combustion air to the EDG. Based on this evaluation, The above activity does not impact any analyzed accident/malfunction of equipment or create the potential for a new accident/malfunction of equipment or safety margin.

Evaluation Number
SE-91-082

Activity Title:

Meteorological report processor computer upgrade.

Description of Change(s):

Replaces the existing Report Processor computer (General Atomics model RM-21) with the Meteorological Report Processor computer (Met computer). The purpose and function of the system (i.e., gathering meteorological data from the Meteorological Monitoring System, averaging the data, and printing reports on demand in a format which satisfies regulatory guidelines) will remain unchanged while system performance and reliability will be enhanced.

Summary of Evaluation:

Meteorological data will be transmitted to the new Met computer which replaces the Report Processor computer (RM-21). The existing equipment racks and meteorological system signal cables were retained. The new Met Computer will perform the same function of the RM-21 computer in a similar manner except that meteorological tower data input will no longer be supplied to the Radiation Monitoring System (RMS). The removal of the communication link for meteorological data to the RMS will not impact CPSES in generating reports since the radio-chemistry data which are manually inputed for the reports is derived from multi-channel analysis. The Met computer is a non-interactive, non safety related system that does not perform any mitigation function of any postulated accident, nor is used by operators to make decisions regarding the operation of equipment that may affect the radiological consequences of any particular accident and is not part of any equipment that controls plant operating systems or equiptment.

Evaluation Number
SE-91-084
Revision 1

Activity Title:

Replace time delays, update documentation on Turbine lube oil purification.

Description of Change(s):

This activity replaces existing time delay relays in turbine lube oil purifier skids 1 and 2 with time delay relays of a different manufacturer. The replacement was necessary because the previous relays were not sized adequately to allow the lube oil purifiers to be used to heat cold oil.

This activity also involves incorporating new vendor recommended instrument setpoints for process control and alarms related to flow, pressure and temperature.

Summary of Evaluation:

This activity was reviewed for its potential for impacting previously analyzed accidents/malfunctions, specifically turbine trip, and was found not to increase the probabilty or severity of that accident.

Evaluation Number
SE-91-085

Activity Title:

Permanent installation of thermal flow switch for 1RE-5100 and other minor changes to Turbine Bldg sump #2.

Description of Change(s):

The activities described by this safety evaluation are:

a design modification that makes perman... two plant modifications perviously installed u... the temporary modifications program. The design modification makes permanent the addition of an Omega thermal flow switch, originally installed via Temp Mod TM-90-1-048, and the removal of checkvalve 1VD-0908 accomplished via TM 90-1-060.

The addition of a temperature controller to protect the sample pump from excessive liquid temperatures.

The addition of a screen on top of the strainer basket.

The activity was performed to improve the availbility of sample flow to the radiation monitor.

Summary of Evaluation:

The implementation of the design modification will only affect the supply of sample flow to the radiation monitor. Failure of any on the newly installed components will only result in loss of flow to the radiation monitor which will cause the radiation monitor to be declared inoperable, and re-direct any flow being pumped from the turbine building pump to the co-current waste treatment system. Failure or inoperability of the rad monitor would cause the sampling in accordance with the Offsite Dose Calculation Manual.

Implementation of the DM affects only the sample flow to the rad monitor and thus does not increase the probability of an accident previously evaluated, increase the consequences of an accident previously evaluated, increase the possibility of a new accident not previously evaluated nor does it impact the radiological consequences of any radiological accident.

Evaluation Number
SE-91-086

Activity Title:

Redefinition of RCS reduced inventory from 3 ft to 5 ft below the RV flange.

Description of Change(s):

This activity re-defines a "reduced inventory condition" in the Reactor Coolant System (RCS) as five feet below the reactor vessel flange rather than the three feet below the flange provided in Generic Letter 88-17. The activity is intended to limit the instances when the significant procedural restrictions of reduced inventory operations must be imposed.

Summary of Evaluation:

This activity will not increase the probability of occurence or consequences of an accident previously evaluated in the licensing basis documents since it is limited to RCS and RHR system operation in Modes 5 and 6 only; the ECCS (accident mitigation) function of the RHR system is only required in Modes 1 through 4.

The potential impact of the activity on the decay heat removal function of the RHR system was evaluated as a potential malfunction of equipment important to safety previously evaluated in the licensing basis documents. In-plant vortex testing had been performed prior to Unit 1 fuel load to characterize the RCS level versus flow limits based on excessive air intrainment. A detailed examination of the reduced inventory integrated operating procedure as revised to implement the activity demonstrated that redefining reduced inventory at the lower level would not affect the probabilty of failure of the RHR system due to excessive air intrainment. Thus the activity will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety previouslyn evaluated in the licensing basis documents.

The potential impact of this activity is on decay heat removal by the RHR system during reduced inventory operations. This function is documented in the licensing basis documents. Therefore the activity does not create the possibility of an accident or malfunction of equipment important to safety different from any already evaluated in the licensing basis documents.

The activity does not reduce the margin of safety as defined in the basis for any Technical Specification since it will not affect the probability of failure of the RHR system to perform its decay heat removal function in Modes 5 and 6.

Evaluation Number
SE-91-088

Activity Title:

Re-route process flows from the Evaporation Pond to the Low Volume Waste (LVW) Pond (removing the Evaporation Pond)

Description of Change(s):

Design Mod 91-070 rev 0 re-routes process flows which currently discharge to the Evaporation Pond (Total Retention Pond) to the Low Volume Waste (LVW) Pond. The principal systems involved are the Condensate Polishing Unit Phase Separator and the Reverse Osmosis Reject. This change alters the composition of the process flows which discharge to the LVW pond. The Offsite Dose Calculation Manual (ODCM) has also been revised to modify the sampling and analysis of process streams flowing into the LVW Pond and add sampling and analysis requirements to effluents from the LVW Pond (See SE-91-106).

Summary of Evaluation:

There are no new secondary effluent streams being created by this change. The existing Condensate Polishing Unit Phase Separator and Reverse Osmosis Reject discharges are re-routed from the Evaporation Pond to the LVW Pond and the sampling and analysis processes which were applicable to the evaporation pond are now applicable to the LVW Pond; consequently, there are no potential accidents or malfunctions of equipment important to safety affected by implementation of this modification.

The implementation of this design modification will not affect the consequences of any identified design basis accident. However, the modification could result in an uncontrolled discharge of powdex resins to Squaw Creek Reservoir (SCR). This failure was discussed previously in the CPSES ODCM basis for section 3/4.11.1.4. The limitations on powdex resin discharges to the evap pond given in the ODCM are based on limiting the consequences of an uncontrolled release of the pond's inventory by limiting the radioactive inventory to 10CFR20 limits. These limits and conditions are being set by a revision to the ODCM. As the ODCM bases discussed this type of accident, the implementation of this change does not create any new unanalyzed event.

This change does not have any impact on the margin of safety as there is no impact on the CPSES Unit 1 Technical Specifications.

Evaluation Number
SE-91-090

Activity Title:

Install filter cards with a 1 second lag time constant to S/G level instrumentation (excluding WR loops)

Description of Change(s):

The activity adds a filter card with a 1 second lag time constant to the Steam Generator Level Narrow range Instrumentation. The activity modifies the narrow range level channels in protection cabinets 1,2,3, and 4.

Summary of Evaluation:

The lag time constant will filter out short duration signals generated by sensing the pressure pulse that results from rapid movement of turbine control valves, opening of atmospheric relief valves (ARV), or opening of steam dump control valves. The filter card will prevent the short duration pulse from causing unnecessary reactor trips and ESF actuations at power levels under 50%. The SE concluded, by reviewing tests performed to determine worst case response time, that a 1.0 sec lag time would still have a response time less than assumed in the FSAR accident analysis. Further, the SE concluded that the consequences of the accidents previously evaluated are not increased by the addition of the circuit boards. Since the addition of the new cards does not affect the protective system trip points nor invalidate the assumptions made in the accident analysis in regard to response time the activity does not affect the margin of safety.

The SE reviewed six accidents and steam line break mass energy releases discussed in the FSAR, as well as the failure modes of the new circuit boards to evaluate the effect that a failure of the new cards would have on the accidents considered in the FSAR. The SE concluded that failure of the card would not increase the probability of accidents evaluated in the Accident analyses, does not create the possibility of a new accident not previously evaluated and does not impact acceptance limits or margin of safety.

Evaluation Number
SE-91-091

Activity Title:

Blind flange CCW U1/U2 cross-connects.

Description of Change(s):

This activity involves the removal the CCW Unit 1 - Unit 2 cross-connect valves and capping each end with blank flanges. This is temporary modification to prevent CCW leakage from Unit 1 to the depressurized Unit 2 during construction of Unit 2. The cross-connect butterfly valves will be evaluated and restored prior to Unit 2 becoming operational.

Summary of Evaluation:

The function of the CCW cross-connects is to provided some additional measure of redundancy when both Unit 1 and Unit 2 CCW systems are operative. During the construction of Unit 2 the function of the cross-connect valves is to isolate the operating CCW system from the CCW system under construction. The cross connect valves have been observed to leak (to the lower pressure Unit 2 CCW system) as evidenced by the changing CCW surge tank levels. Removal of the subject valves and replacing them with blank flanges provides a more reliable means of preventing cross-unit leakage.

The activity does not impact any existing accident/malfunction analysis since the current Unit 1 licensing basis does not take credit for the operation of Unit 2 CCW (while Unit is under construction).

Evaluation Number
SE-91-093

Activity Title:

Removal of Feedwater pump suction strainers and associated differential pressure instrumentation.

Description of Change(s):

This activity removes the Feedwater Pump suction strainers and their associated differential pressure switches. The strainers are prone to leakage following plant trips. The relatively large mesh size strainers were originally installed to prevent possible construction debris from entering the pumps or the S/Gs. It is common industry practice to remove these strainers once the system has been cleaned up. Inspection of these strainers have found them to be clear of debris.

Summary of Evaluation:

All components affected by this change are non-safety related. Since the Feedwater system is clean, removal of the strainers does not affect the system performance (except to reduce condensate leakage following plant trips). It is not expected that this modification will have any impact on the frequency or consequences of any analyzed accident/malfunction or create the potential of a new accident/malfunction.

Evaluation Number
SE-91-094

Activity Title:

Additional drain valves in the safety injection piping to expedite draining for penetration leak rate testing.

Description of Change(s):

A modification was made to install normally closed 3/4 inch valves on the horizontal pipe upstream of the cold leg injection header isolation valves 1-8809 A & B. The drain valve installation will expedite draining of the Safety Injection System piping for penetration leak rate testing.

Summary of Evaluation:

The valves are normally closed and will not adversely affect system operation or design basis. The valves are installed as part of ASME Code Class 2, Seismic Cat I piping. The effects on piping and supports, including valve seismic qualification have been evaluated. These changes do not introduce any credible potential failure modes.

Evaluation Number
SE-91-095

Activity Title:

Installation of connections to the CCW return from the Spent Fuel Pool for a future filter demineralizer for Unit CCW

Description of Change(s):

This activity adds two 2" connections to the 12" non-safeguards CCW return from the No. 2 Spent Fuel Pool heat exchanger. The connections include root valves which are closed and capped. The connections are being added for the future installation of a Unit 2 filter demineralizer which will provide fluid chemistry control for the Unit 2 CCW system.

Summary of Evaluation:

Since these capped connections provide pressure tight integrity equivalent to the original CCW pressure boundary, there are no new failure mechanism associated with this change. Neither will the change effect the function or operation of the CCW system or the spent fuel pool cooling and cleanup system.

Evaluation Number
SE-91-101

Activity Title:

Revision 3 to STA-742, "Snubber Surveillance Program"

Description of Change(s):

Revision 3 to STA-724, "Snubber Surveillance Program," included deletion of the reject line from Figure 7.1 and revisions to the visual examination frequency table. These changes necessitated a change to the Technical Requirements Manual, Technical Requirement 3.1, "Snubber Inservice Inspection Program".

Summary of Evaluation:

Safety Evaluation SE-91-015, prepared for the associated TRM change, completely encompass the changes made to procedure STA-742 described above. See SE-91-015 for additional information.

Evaluation Number
SE-91-103

Activity Title:

Temporary modification to allow the use of a Containment penetration for steam generator maintenance.

Description of Change(s):

A temporary modification was made to allow penetration MIII-30 to be used for providing access for hoses and cables during steam generator maintenance. The permanent, as designed, penetration was restored prior to the plant entering MODE 4. Also, hoses and lines were run from the steam generator bowl areas and sludge lance ports to penetration MIII-30 and from room 88 to Door S27 prior to entering the yard.

Summary of Evaluation:

Penetration MIII-30 was not allowed to be opened until the plant had been shutdown for greater than 100 hours in MODES 5 or 6. This penetration is not required to be a pressure boundary in MODES 5 or 6 and is only required to prevent a direct flow path from Containment to the atmosphere.

A direct access flow path from the Containment to the atmosphere did not exist because the sludge lance suction hose nozzle was always under-water and the discharge hose was valved. The free space in the penetration was foamed to prevent direct access path to the atmosphere. Air lines going from atmosphere to Containment were not allowed to blow free.

The use of plastic to cover the opening at Door S27 allowed the negative pressure in the Safeguards Building to be maintained.

Evaluation Number
SE-91-104

Activity Title:

Addition of airlock equalization and hydraulic valves as Containment Isolation Valves to the Technical Requirements Manual.

Description of Change(s):

The list of Containment Isolation Valves contained in Table 2.1.1 of the Technical Requirements Manual was revised to add the Personnel and Emergency airlock equalizing valves and the personnel airlock hydraulic system valves. All of these valves have been identified to be allowed to be opened under administrative controls. Also, three new notes have been added to:
a- Clarify the applicable surveillance requirement to be used for testing the equalization valves to satisfy 10 CFR 50 Appendix J.
b- Clarify that the equalization valves are also subject to the controls of Specification 3.6.1.3 and are associated with their respective airlock door to ensure a single Containment boundary is maintained to satisfy the LCO for airlock OPERABILITY.
c- Clarify which equalization valves are interlocked with an airlock door operating mechanism to satisfy the locking requirements of GDC 56 and Specification 3.6.1.3.

Summary of Evaluation:

The addition of these valves provides added assurance that CONTAINMENT INTEGRITY is maintained and thus cannot affect the radiological consequences of a LOCA. The addition of these valves to the TRM does not create the possibility of human error but rather makes it less probable. The valves satisfy Technical Specification 3.6.1.3 because they are kept closed except when the air lock is being used. The equalization valves must be opened and closed in conjuction with their respective doors to satisfy the LCO for airlock OPERABILITY. Therefore, manual valves opened under these administrative controls are OPERABLE.

Evaluation Number
SE-91-106

Activity Title:

Revise the liquid waste sampling and analysis program described in the Offsite Dose Calculation Manual (ODCM)

Description of Change(s):

Revise ODCM Table 4.11-1 "Radioactive Liquid Waste Sampling and Analysis Program" to add sampling and analysis requirements for the Low Volume Waste (LVW) Pond discharges to Squaw Creek Reserv ir.This change is necessary for purposes of radioactive materials accountability and offsite dose calculations. Additionally, the change modifies the sampling/anlaysis requirements for inputs to the LVW Pond and revises the affected calculational methodlogy in Part II of the ODCM to reflect the revised discharge pathways and sampling/analysis requirements.

This revision is administrative in nature and is related to the activities of DM 91-070 and DCN 2973 (see SE-91-088) which involved re-routing the discharge lines of the Condensate Polishing Unit Phase Seperator and the Reverse Osmosis Reject from the Evaporation Pond (Total Retention Pond) to the LVW Pond. This change affects the requirements for sampling and analyzing secondary waste streams for radioactive materials resulting from primary-to-secondary leakage and changes the criteria for requiring diversion of secondary waste streams from the LVW Pond to the Waste Water Holdup Tanks.

Summary of Evaluation:

The ODCM revision is made to identify new discharges to the LVW Pond and to ensure proper sampling and analysis requirements and appropiate controls on effluent discharge are in place. The change does not directly affect plant structures, systems, components or system parameters since the change involves administrative reqirements for sampling and analyzing secondary waste streams. The change will not affect the concentration of radioactive materials in discharges and therefore could not create the possibility of an accident different from any already evaluated in licensing basis documents. The change will not affect the acceptance limits for release of radioactive effluents as described in the Technical Specification Administrative Controls. These acceptance limits remain as required per 10 CFR 20; therefore, this change does not impact the margin of safety.

Evaluation Number
SE-91-107

Activity Title:

Increase of the upper Lithium limit in the Reactor Coolant System to reduce corrosion and achieve lower radiation fields.

Description of Change(s):

The upper limit of the pH control band, listed in FSAR Table 5.2-5, was revised to indicate an operating range at constant Lithium concentration of 2.2 plus or minus 0.15 ppm with boron concentrations upto 1500 ppm. Lithium is added to the Reactor Coolant system for pH control as boric acid concentration is changed for reactivity control. Primary coolant pH influences corrosion product release, transport, activation and deposition onout-of-core surfaces. Industry experience indicates that lower radiation fields can be achieved with end-of cycle pH in the 7.1-7.4 pH range (generally lower fields than with a pH 6.9). This modified program of the pH control is denoted as principle 3 in EPRI NP-7077, PWR Primary Water Chemistry Guidelines, Revison 2.

Summary of Evaluation:

Increased values of Lithium in the Reactor Coolant system could not affect the radiological consequences of equipment malfunctions because Lithium hydroxide introduced into the Reactor Coolant is in the form of Li-7 which can not be activated in the core.

Test data on Alloy 600 reverse U-bend specimens does not indicate that operation with Lithium at a maximum of 2.35 ppm would increase the time to initiation of Primary Water Stress Cracking Corrosion. No exacerbation of the corrosion rate of the fuel cladding can be attributed to elevated lithium operation based on studies conducted.

There is no indication that increasing the Lithium concentration to a maximum of 2.35 ppm will increase the probability of corrosion for systems in contact with Reactor Coolant.

The modified lithium/boron correlation includes boron concentration upto 1500 ppm. At elevated boron concentrations RCS pH can be maintained to a minimum value of 6.9.

Evaluation Number
SE-91-109

Activity Title:

Rod Cluster Control Change Tool

Description of Change(s):

This activity deletes the FSAR reference to an incore shuffle and adds a description of the Rod Cluster Control(RCC) Change tool. Full core offloads reduce grid interaction associated with standard Westinghouse fuel. During full core offloads insert component shuffles are performed in the Fuel Building. The Rod Cluster Control (RCC) Change Tool will be used to shuffle Rod Cluster Control Assemblies (RCCA) in the Fuel Building.

Summary of Evaluation:

The evaluation considered the impact of this change on the previously analyzed design basis fuel handling accident (dropping of a spent fuel assembly resulting in the rupture of all fuel rods). It was concluded that the revision to the fuel shuffling process and the use of the Rod Cluster Control (RCC) Change Tool would not adversely impact the probability or consequences of this accident nor would it create the possibility of a new type of accident/malfunction.

Evaluation Number
SE-91-110

Activity Title:

Change in interval for testing ionization and thermal smoke detectors to be consistent with NFPA requirements.

Description of Change(s):

This change updated the test frequency to that currently recommended by the National Fire Protection Association (NFPA) in NFPA 72E-1987 and NFPA 72H-1988 for ionization and thermal detectors. The NFPA recommendations reflect the reliability of ionization and thermal detectors. Plant testing of these detectors by TU Electric support the recommendations of the revised NFPA recommended practice.

Summary of Evaluation:

Over an 18 month period (since system turnover), 11 failures out of 1298 ionization detectors have occurred at CPSES Unit 1. Over the same 18 month period, only one thermal detector was replaced in the non-safety related turbine area detection system. No other failures have occurred.

The operating environment is important particularly in the case of dust and dirt, which can cause failure of ionization detectors. These type failures normally result in the detector to fail in the alarmed condition. Even though a relatively clean environment is maintained at CPSES, these type of failures are more typical. Ambient temperature can also result in the failure in ionization detector; however, no failures attributed to this cause have been detected.

There are no new failure modes associated with the implementation of this change. Fire detection is used in fire safe shutdown procedures as indications of a fire situation that may cause shutdown, but do not determine or influence the outcome of the fire safe shutdown procedure implementation. The change of test frequency does not affect the ability to shutdown the plant in case of a fire situation; therefore, no radiological consequences are associated with this change. A single failed detector (or even several detectors) does not render a fire zone inoperable. The probability of a large number of random detector failures occurring between the new test intervals is remote and not considered likely to occur. A single detector, or even several detector failures will not preclude an eventual alarm or actuation of the Halon system. The change in test frequency does not affect the ability to achieve and maintain safe shutdown in the event of a fire.

Evaluation Number
SE-91-111

Activity Title:

Adds new condenser air inleakage monitoring equipment

Description of Change(s):

This activity involves the addition of new condenser air inleakage monitoring equipment installed in the vacuum exhaust piping. The new equipment simplifies leakage monitoring and improves the measurement accuracy.

Summary of Evaluation:

All equipment involved with this modification are non-nuclear non-safety related located in the turbine building. No accidents or malfunctions as described in the FSAR are affected by this change.

Evaluation Number
SE-91-114

Activity Title:

Revise Relief Request 13.1 and delete Relief Request 13.2 in the Inservice Testing Program Plan

Description of Change(s):

Delete Relief Request 13.2 and add the associated valves from Relief Request 13.2(Valves 1CT-148 and 1CT-149) into Relief Request 13.1. The four valves are: the two Containment Spray Pump Suction Check Valves from the Containment Recirculation Sumps (1CT-148 and 1CT-149) and the two Injection Header Check Valves(1CT-142 and 1CT-145).

Summary of Evaluation:

Currently each relief request(RR-13.1 and 13.2), addresses two valves and requires one of the two valves be dissassembled and exercised each refueling outage. This results in two of the four valves being disassembled and tested during a refueling outage. This change combines all four valves into one relief request and proposes one of the four valves be disassembled and tested during each refueling outage due to the similarities of the valves. This change is in accordance with Generic Letter 89-04, Attachment 1, Position 2. In addition, SSER 23 provides interim approval for the entire Unit 1 IST Program Plan including RR 13.1 and RR 13.2.

The safety evaluation concludes that there is no unreviewed safety question involved with the change in frequency for disassembly and testing.

Evaluation Number
SE-91-120

Activity Title:

Relaxation of latching requirements for doors to Main Steam and Feedwater penetration areas while personnel is in the area for safety.

Description of Change(s):

The requirement to fully latch doors S1-38A and S1-40B at all times creates a safety concern for personnel working in the Main Steam and Feedwater Penetration areas. In the event of a steam leak with personnel in the areas, the prompt evacuation of the area would be impeded by the closed and fully latched water-tight doors.

The closure criteria for door S1-38A above has been relaxed from closed (fully latched) to closed on one latch for short periods (<15 minutes) while personnel are in the room. Furthermore, door S1-40B can be open for extended periods as long as it is free to close and a person is stationed at the door to close it in the event of an accident.

Summary of Evaluation:

The probability of all postulated piping failures in the main steam and feedwater areas concident with occupancy during short operational periods is estimated to be less than 1.0E-04 per year. Water-tight doors have latches designed to seal in the event of flooding on either side of the door. However doors S1-38A and S1-40B both open into their respective piping areas.

There is no sensitive essential equipment located adjacent to door S1-40B on the auxiliary building side. Door S1-40B could perform its intended safety function even if it were not latched since it opens into the room and any significant flow would tend to close it and seal it tight.

Although it is still recommended to close door S1-38A and dog 1 latch to protect essential equipment, it is not essential to dog even one latch. A differential pressure as little as 1 inch of water, as a result of a small leak, would hold the door shut since the door opens into the piping area.

It was concluded, with the above changes in the latching requirements, that doors S1-38A and S1-40B continue to perform their intended safety function and at the same time increase personnel safety.

Consideration of the probabilities of failures affecting the Main Steam and Feedwater Penetration areas, the orientation and configuration of the subject water-tight doors, and the potential impact of the change on equipment important to safety leads to the conclusion that relaxation of the latching requirements for S1-38A and S1-40B is acceptable to address the concern for the safety of individuals working in the subject areas.

Evaluation Number
SE-91-121

Activity Title:

Unit 1 Cycle 2 reload safety evaluation.

Description of Change(s):

Following the end of Cycle 1 of CPSES Unit 1, a fresh batch of fuel replaced the discharged Region 1 fuel assemblies. All 193 fuel assemblies were offloaded and placed in the spent fuel pools. Once-burned Region 2 and 3 fuel assemblies were re-inserted into the core, along with Region 4 fresh fuel and placed in a low leakage arrangement. The loading configuration meets the energy requirements for CPSES Unit 1 Cycle 2 operation. The Cycle 2 reload design utilizes 56 Westinghouse standard fuel assemblies similar in design to the Cycle 1 fuel, 36 with an initial enrichment of 3.0 w/o and 20 with an enrichment of 3.4 w/o. Three hundred and sixty-eight Wet Annular Burnable Absorber (WABA) rods were also used. There are 16 3.0 w/o assemblies with 8 WABAs and 20 3.0 w/o assemblies with 12 WABAs.

Summary of Evaluation:

Because the replacement fuel assemblies are completely compatible with the existing fuel assemblies from a mechanical and thermal-hydraulic standpoint, no equipment, important to safety or otherwise, is affected by this cycle reload design.

Each of the accident analyses presented in FSAR Chapter 3.6B, 6, and 15, as supplemented or superseded by the Positive Moderator Temperature Coefficient and boron dilution reanalyses (Amendments 5 and 6 to the Operating License) remain valid. There are no changes in the conclusions of the accident analyses required to support this cycle reload. Failure values as a result of the mechanical fuel design conform to all applicable design criteria; hence there is no reduction in failure values for the new fuel. Because neither the accident analysis event acceptance criteria nor the failure values changed as a result of this reload, the margin of safety remains unchanged.

Evaluation Number
SE-91-124

Activity Title:

Removal of smoke detectors in computer room.

Description of Change(s):

This activity involves removal of two smoke detectors mounted on the Unit 2 computer room floor and their associated mountings, conduits and cables. These detectors were mounted below the Emergency Response Facilities Computer (ERFC) raised floor. The ERFC and the raised floor have been removed due to the installation of a new plant computer. These detectors are no longer required per the requirements of NFPA 72E and had become a tripping hazard.

Summary of Evaluation:

Removal of the subject smoke detectors do not impact the safe operation of any systems, component or structure. This activity does not affect accidents and malfunctions evaluated in License Bases Documents and does not effect the ability to achieve and maintain safe shutdown in the event of a fire.

Evaluation Number
SE-91-125

Activity Title:

Addition of new generator diagnostics and deletion of tritium detector and associated generator trip.

Description of Change(s):

This activity adds the following turbine generator diagnostic equipment:

1. Turbine Supervisory instrumentation - trend analysis of turbine and generator vibration/expansion.
2. Radio frequency monitoring system - detects partial discharges and arcing across winding insulation.
3. Generator flux probe - detects shorted windings.
4. A second generator core monitor - detects overheated windings.
5. Generator gas hygrometer - detects water or air inleakage (also includes an alarm in the control room).

In addition the existing tritium monitoring system and its associated generator trip are deleted. The tritium monitoring system was originally installed to provide a extremely sensitive method of detecting very low levels of primary water inleakage, which could lead to a failure of the rotor retaining rings due to stress corrosion cracking. With the replacement of of the rings with a material not susceptible to stress corrosion cracking, that level of sensistivity to leakage is not required. The deletion of this system eliminates the need to add tritium to the primary water system. The new generator gas hygrometer provides adequate detection for moisture in leakage.

Summary of Evaluation:

None of the equipment installed or removed is safety related and does not affect equipment important to safety. The installation of the diagnostic equipment has no effect on plant function or operation since the instrumentation is passive. The implementation of this activity reduces the likelihood of a spurious plant trip due to the elimination of the tritium detector generator trip.

There are no credible potential failure modes which could increase the probability or the consequences of any previously analyzed accident/malfunctions nor create the potential of new accident/malfunction.

Evaluation Number
SE-91-130

Activity Title:

Correction of Safety System Inoperable Indication (SSII) voltage and current values.

Description of Change(s):

The FSAR change corrects discrepancies regarding the SSII field input circuit current value and the field contact minimum voltage and current ratings. No circuits external to the SSII panel are affected.

Summary of Evaluation:

FSAR Section 8.3.1.2.1, Item 7f, provides an analysis of the SSII circuits to justify the lack of electrical separation between the non-safety related SSII circuits and the safety-related input circuits to the SSII. The 0.767 amp value specified as the field contact current limited value at the logic card is instead 0.767 mA by means of a series resistor. In addition, the field contact rating of 250 Vdc and 5 amps, is not reflective of the various SSII field device ratings. The ratings of these devices vary with a minimum rating of 125 Vdc and 0.5 amp. The changes in the SSII circuit ratings have been determined to be more conservative than the previous analysis results. Therefore, the existng separation is considered to be adequate to protect the safety-related portions of the circuit from faults in the non-safety related portions.

Evaluation Number
SE-91-134

Activity Title:

Comanche Peak Unit 1 Cycle-2 Core configuration.

Description of Change(s):

During offload of the CPSES Unit 1 Cycle 1 core, two fuel assemblies (A-34 and C-30) with leaking fuel pins were identified, thus necessitating the reconfiguration of the Cycle 2 core design. The revised core configuration superseded the previously approved core loading plan.

The Region 1 fuel assembly A-34 was originally intended to be placed in the H-8 location for Cycle 2 operation. The Region 1 fuel assembly A-37 was originally intended to be discharged from the core. In the revised core configuration, fuel assembly A-37 was found to be an acceptable replacement for A-34. Fuel assembly A-37 is structurally identical to A-34 and has a similar burnup history. Furthermore, no abnormalities were detected during the post-offload inspection of A-37.

The second fuel assembly C-3C, is a Region 3 fuel assembly intended to be placed in the J-3 location. A second revision to the core configuration was necessitated. The revised loading plan discharged C-30 and its three cyclic partners (C-55, C-01, and C-61) and replaced them with four Region 1 fuel assemblies (A-54, A-28, A-30, and A-25).

Summary of Evaluation:

To address the impact of these changes, the Reload Safety Evaluation (RSE) was revised which formed much of the bases for Safety Evaluation SE-91-121. The RSE was revised to address the final core configuration. Because of the similarity of the burnup history for fuel assemblies A-34 and A-37, this change was considered to be essentially a one-for-one replacement. However, because of the replacement of four Region 3 fuel assemblies, Westinghouse was required to redesign the core configuration and perform additional evaluations to ensure that the original RSE remained valid. The evaluation of the nuclear design concluded that no reactor physics parameters used in the accident analyses for Cycle 2 operation are adversely affected. For all safety evaluations, the applicable design and safety limits continue to be satisfied. The conclusions reached in the original RSE remained valid.

It was concluded that the bases for the safety evalation SE-91-121 are not adversely affected by the revised core configuration. Hence, safety evaluation SE-91-121 is still directly applicable to the final core configuration and the conclusions reached are still applicable.

Evaluation Number
SE-91-135

Activity Title:

Removal of restricting flow orifice from instrument air line supplying containment.(REF temp mod. TM-91-1-84)

Description of Change(s):

Removal of the flow limiting orifice CP1-C1ORPR-01 from instrument air line 3-CI-017-151-2 will allow for greater capacity of air supplied for breathing air and leak rate tests inside contaiment during Unit 1 RFO-1. Current dsign limits supply to only 50 SCFM.

Summary of Evaluation:

The above activity does not involve any safety related systems or functions. The failure mode includes the break in the 3 inch instrument air line, which increase the pressure in the containment and causes the flooding in the fuel building. The above activity is analyzed and does not impact any analyzed accident/ malfunction of equipment or create the potential for a new accident/ malfunction of equipment or decrease in the margin of safety.

Evaluation Number
SE-91-136

Activity Title:

Evaluation for operating Unit 1 with one Reactor Vessel Stud (Stud #6) detensioned.

Description of Change(s):

This evaluation allows one of the 54 Reactor Vessel Studs to be in a partially withdrawn position and detensioned during operation. During removal of the closure studs from the Unit 1 Reactor Vessel, following detensioning at the beginning of the First refueling, difficulty was encountered in turning several studs out of the Vessel flange stud holes. Stud #6 encountered so much resistance that it could not be turned out of its stud hole. As a result, efforts to remove Stud #6 were aborted with the stud turned out approximately 3 inches.

Summary of Evaluation:

During the outage, the stud was capped and sealed shut with a stainless steel can to protect it from the borated refueling water while the cavity was flooded.

The results of the calculation performed indicate that the stresses in the stud and flange hole threads adjacent to the non-functional stud remain within the appropriate ASME Code limits. The evaluation of the o-ring gaskets considered the maximum allowable stud spacing without leakage, and the effect of the closure head flange rotation due to increased operating loads on the studs adjacent to the non-functional stud. The results of the calculations demonstrate that the function of the o-ring gaskets would be maintained.

Evaluation Number
SE-91-137

Activity Title:

Containment isolation valve position indication, RG 1.97 Rev. 2

Description of Change(s):

Revised FSAR to reflect the CPSES position regarding accident monitoring position indication requirements for containment isolation valves. This change affects remote manual containment isolation position indication and ERF computer displays in the control room, TSC and EOF.

Summary of Evaluation:

Position indication on the ERF computer for remote manual containment isolation valves (CIVs) is considered not required to meet the intent of Reg Guide 1.97, Revision 2. The remote manual CIVs do not receive an automatic containment isolation signal. These valves are event driven and controlled by the operator based on post accident conditions. CIVs which operate automatically upon receipt of a containment isolation signal have ERF displays in the control room, TSC and EOF. For remote manual CIVs, direct and immediate trend or transient information is considered not essential for operator information or action short term. Long term, the operator maintains awareness of the valve position via control board displays (i.e monitor light boxes and/or lights on the control switches). The TSC and EOF staffs would be aware of valve manipulations via the control room interface as required as part of the Emergency Reponse organization. Thus, ERF computer position indication for remote manual CIVs is not required for essential functions of the TSC and EOF.

Evaluation Number
SE-91-138

Activity Title:

Safety impact of unrecovered loose parts in the Reactor Coolant System
SORC approved in meeting 91-101 11/25/91

Description of Change(s):

Assess the impact of operation of Comanche Peak Unit 1 with unrecovered loose parts in the Reactor Coolant System (RCS). The loose parts specifically addressed in this evaluation were one segment of a fuel rodlet plenum spring and one nut (Standard 1/4" x 20). The evaluation specifically addressed the impact of the unrecovered loose parts on the following:

- Materials
- Reactor Core
- Reactor Vessel and Internals
- Steam Generator Structural Integrity
- Pressurizer
- Reactor Coolant Pumps
- Piping
- Chemical and Volume Control System
- Residual Heat Removal System
- Safety Injection System
- Safety Related Taps and Instrumentation
- LOCA Analyses.

Summary of Evaluation:

The unrecovered loose parts do not increase the probability or consequences of an accident previously evaluated in the FSAR. The objects do not impact the various RCS components and auxiliary systems such that their safety-related functions are challenged. Further, the loose parts do not adversely impact any safety-related instrumentation associated with RCS flow, pressure, level, and temperature measurements.

The unrecovered loose parts do not increase the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. The RCS components and auxiliary systems have been reviewed and it has been determined that there is no additional probability of malfunction for RCS components deemed important to safety.

Operation with the presence of loose parts in the RCS does not require a change in the Technical Specifications nor does the subject loose parts prevent inspections required by Technical Specifications.

It was concluded that operation of CPSES Unit 1 with the presence of the subject loose objects does not adversely affect the existing mechanical components, plant systems and safety-related instrumentation.

Evaluation Number
SE-91-141

Activity Title:

Revise station procedure which assigns responsibilities for implementing the Radiological Effluent Controls of the ODCM

Description of Change(s):

Revise procedure STA-603, "Control of Station Radioactive Effluents" in order to implement changes to the Offsite Dose Calculation Manual (ODCM) as approved in LDCR-OD-91-003, "Addition of Low Volume Waste Pond Discharge Sampling/Analysis Requirements" and LDCR-OD-91-006, "Revision of Secondary Resin Discharge Inventory Limit". This procedure revision also incorporates administrative changes to streamline and improve the effluent release permitting process.

Summary of Evaluation:

The procedure revision is directly related to the implementation of approved ODCM changes which were evaluated under the scope of SE-91-088 and SE-91-106: therefore, see evaluation summaries for SE-91-088 and SE-91-106 included in this report.

Evaluation Number
SE-91-144

Activity Title:

REVISE IST PROGRAM PLAN

Description of Change(s):

Delete IST Program Plan Relief Requests P-10, 6.1, 12.1, 15.19, and 16.1. Revise Relief Request 10.1. Revise Appendix A and B to reflect the deletion and revision to the Relief Requests.

Summary of Evaluation:

Eleven IST Program Plan Relief Requests were identified in LER-91-003-01 as needing further review in order to assess whether the justifications presented in them were valid. The Relief Requests were for deviating from the ASME B&PVC Section XI requirements for test frequency. Six of the eleven Relief Requests have been determined to be in need of change as a result of this review. Those that were changed can be tested in accordance with the Section XI frequency requirements. The effect of this change is to withdraw the relief request P-10 for pumps TBX-CSAPCH-01, 02; and TBX-CSAPBA-01, 02. The effect of this review is also to withdraw Relief Requests 6.1 for valve 1-8046; 10.1 for valve 1CA-016; 12.1 for valve 1-CH-024; 15.19 for valve 1SI-8968; and 16.1 for valve 1-8381.

The safety evaluation determined that there are no unreviewed safety questions associated with this change.

Evaluation Number
SE-92-036

Activity Title:

Reduction in monitoring frequency of the surface alignment monuments installed in the SSI and Squaw Creek Dams.

Description of Change(s):

This safety evaluation documents the change in commitment for reducing the frequency of monitoring the surface alignment monuments installed in the SSI and Squaw Creek Dam from every six months to annually.

FSAR Section 2.5.8.1, SSI Instrumentation, states that "periodically" during impoundment and operation, the horizontal and vertical coordinates of the alignment monuments will be determined and compared with the original coordinates to determine the vertical and horizontal movements of the crest of the SSI Dam. NRC NUREG-0797, Supplement 22, Section 2.5.6.7, Instrumentation, states "the staff understands, on the basis of discussions with the applicant, that periodic determinations for both the surface alignment and piezometers will occur at least every six months and the semiannual inspections of the piezometers and the surface alignment monuments for the SSI Dam will be administratively controlled by means of a CPSES surveillance procedure. The staff considers that these inspections satisfy the applicants commitment to perform these inspections at least annually as recommended in RG 1.127."

Summary of Evaluation:

Surveys of the horizontal and vertical alignment monuments installed on the surface of both the SSI and Squaw Creek Dam have been conducted approximately every six months since 1977 as a maintenance activity to trend any movements of the dams. Based on the structural integrity which has been demonstrated by the surveys for the last 15 years, a reduction in monitoring frequency from every six months to annually has been determined to be acceptable and will still allow CPSES to satisfy the RG 1.127 recommendation for annual inspections.

The reduction to an annual monitoring frequency will still provide an early detection of possible dam degradation and will have no impact on the failure of the dam to be able to maintain a water level assumed to be available in the LOCA analysis for the Ultimate Heat Sink.

Evaluation Number
SE-92-037

Activity Title:

Installation of Hot Tool Room in Containment Building

Description of Change(s):

A Hot Tool Room was installed (DM 90-489) in the Unit 1 Containment Building at elevation 808'-0'' in order to facilitate the control and issue of tools during a plant outage. This activity will allow work to be completed more efficiently and will control the possible spread of contamination from carrying "hot" tools in and out of the Containment Building.

Summary of Evaluation:

During normal operation the Hot Tool Room frame structure will remain in the Containment Building. If any tools or equipment are left in the tool room a Safe Zone will be designated in accordance with approved station procedures for storing non-plant equipment inside Seismic Category I structures. Since the Hot Tool Room framing is designed to withstand Safe Shutdown Earthquake (SSE) loads, interaction with other structures, systems or components is limited to the possibility of the expanded metal being detached during a LOCA and ending up against the screen of the Containment Sump. The Hot Tool Room frame material is constructed of carbon steel and therefore there exists no potential for contribution to hydrogen production in the Containment Building.

The only effect that this activity can have on the Containment Sump would be reduction or restriction in flow through the screen, provided that this material can travel to the screen location. Since this metal is welded to the tool room frame and is flexible, the possibility of becoming detached is unlikely. If it did become loose this material will not float and, therefore, would not travel to the screen and affect the Containment Spray System.

Because it was concluded that no structures or safety systems would be affected, there is no impact on previously identified accidents or potential for creation of a new accident. No Technical Specificaton is associated with this activity; therefore, there is no impact on any margin of safety.