50-255 PALISADES CPC

1994 ANNUAL REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

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ATTACHMENT

Consumers Power Company
Palisades Plant
Docket 50-255

1994 ANNUAL REPORT OF FACILITY CHANGES,
TESTS AND EXPERIMENTS

September 11, 1995

102 Pages

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT

FACILITY CHANGES, FUNCTIONAL EQUIVALENT SUBSTITUTIONS
SPECIFICATION CHANGES, TEMPORARY MODIFICATIONS
AND OTHER DOCUMENTS

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FACILITY CHANGES

FC-933-4
SE94-0902

PIP SYSTEM REPLACEMENT

FC-933-4, upgrades the Plant Information Processor (PIP). This modification is being done as part of a general upgrade to the Plant computer system.

The PIP will normally operate in conjunction with the host computer and the other components (e.g. monitors) on the ethernet highway. However, if the host computer or the ethernet highway were to be lost, the PIP is designed to operate in a stand alone mode.

Safety Evaluation Summary

Reference to the PIP and its functions are found in several sections in the Plant's Technical Specifications. These sections will not change with the new date processor. This PIP provides permissives for the bank withdrawal in the manual sequential mode. (Reference FSAR Figure 7-56). A single failure of a datalogger contact will not cause an uncontrolled rod or bank withdrawal. The PIP contacts in the new data processor will be of a proven design and will be as reliable as the contacts in the old processor. The probability of an uncontrolled rod or bank withdrawal is not increased.

The PIP is non-1E. It interfaces to 1E components through its analog inputs or AC power supply. Isolation devices consistent with IEEE 384 will be used to isolate the PIP from these 1E devices.

The Technical Specifications refer to a shutdown margin of 2% in reactivity when the reactor is in hot shutdown. The PIP annunciates an alarm when the rods are inserted greater than this limit. This limit will not be changed with the new design

FC-933-5
SE94-0901

SPI SYSTEM REPLACEMENT

FC-933-5, upgrades the Secondary Position Indication Processor (SPI). This modification is being done as part of a general upgrade to the Plant computer system.

The SPI computer will operate in conjunction with the host computer to provide information to the plant operators. The SPI-host system is a backup system to the PIP computer.

Safety Evaluation Summary

Reference to the SPI and its functions are found in several sections in the Plant's Technical Specifications. These sections will not change with the new date processor.

This SPI has no interfaces with any other Chapter 14 accident. The SPI is non-1E. It interfaces to 1E components through its analog inputs or AC power supply. Isolation devices consistent with IEEE 384 will be used to isolate the SPI from these 1E devices. The SPI computer operating with the host computer is designed to be as reliable than the existing SPI computer.

The Technical Specifications refer to a shutdown margin of 2% in reactivity when the reactor is in hot shutdown. The SPI-host annunciates an alarm when the rods are inserted greater than this limit. This limit will not be changed with the new design.

FC-942

SE94-1292

RELOAD P CYCLE 12

The Cycle 12 reload will include 56 new "P" assemblies, 60 "0" assemblies, 64 "N" assemblies (standard), one "N" (shield), 12 (modified) "M" assemblies, and 4 (modified) "N" assemblies.

Reload "P" continues to use the design changes initiated in cycle 10 to improve debris resistance. Cycle 12 is a low leakage design, which will minimize fluence on the critical reactor vessel axial welds and vessel base metal. A comprehensive description and evaluation is included in Siemens Power Report, "Palisades Cycle 12 Safety Analysis Report".

Safety Evaluation Summary

Refueling the reactor with fresh fuel and the core redesign is a routine activity that does not require changes to procedures.

The FSAR describes the use of mechanically fixed neutron absorber rods to allow for a reduction in the beginning-of-life boric acid concentration and a corresponding reduction in the positive moderator temperature coefficient. While Cycle 12 uses gadolinium as described in the FSAR for this purpose, the top and bottom 5.3 inches of these gadolinium rods will contain natural (un-enriched) uranium pellets without gadolinium. These "axial blankets" provide fuel cost savings with projected lower peak linear heat rate and no expected adverse effect on radial peaking factors.

The Reload "P" assemblies utilize an asymmetrical fuel rod assemble load scheme for the first time. Past practice has used asymmetric gadolinia rods (batch "N",and "O"), but not asymmetrical fuel rods. Reactivity benefits and improved fuel economy are a result of this design enhancement.

Twelve (12) modified "M" fuel assemblies (4th exposure in Cycle 12), four (4) modified "N" fuel assemblies (3rd exposure in Cycle 12), and eight shield "N" assemblies (third exposure in Cycle 12) are used in Cycle 12 to reduce neutron flux on the reactor vessel. Each "N" shield assembly consists of a normal cage assembly, and contains 160 rods are positioned in four rows, two rows on each of two opposing faces of the assembly. Each modified "M" and "N" fuel assembly consists of a normal cage assembly, 201 fuel rods, one instrument tube, and 7 stainless rods within the assembly, the 7 stainless rods being inserted for Cycle 12. The stainless rods used in the "M" and "N" assemblies are similar to those used in the Shield "N" assemblies.

The 56 fresh fuel assemblies of Reload "P" incorporate debris resistant design features that include longer solid zircaloy lower end caps, and a lowered bottom spacer grid designed to trap debris at a location in the bottom of the assembly where fretting would not affect the fission product barrier integrity.
Exterior dimensions and active fuel zones of the assemblies are not affected by the changes. These debris resistant features were first used in the cycle 10 Reload "N" assemblies.

FC-947

SE94-0208

OFFICE FACILITIES EXPANSION PROJECT - SERVICE BUILDING

A 33,000 square foot building addition was made to the west side of the existing Service Building. This addition consists of offices, conference rooms, a lunch room and associated used. The third floor and north end of the second floor of the existing Service Building were improved under this FC package. Minor changes were made to the first floor of the existing Building. In addition an emergency exit was constructed for the offices on the second floor of the Turbine Building. The Service Building Addition is sprinkled and has fire alarms. The building is serviced by the 10" fire water main located approximately 10' west of the existing Service Building. This main dead ends after serving the PMC&T Building, fire hydrant 9 & 10 and the existing Service Building. The only fire hydrant that FSAR section 9-6.7.1.4 requires to be operable in the vicinity of this modification is hydrant 3 which is located near the southwest corner of the existing Service Building. Its operability is not affected by this modification. Temporary fire water service was provided during construction to provide protection for the areas served by the existing fire main.

Safety Evaluation Summary

Office buildings in general were not evaluated as a concern for the safety of the plant in the FSAR, unless they served a purpose other than office space. None of the systems that were affected by this modification are discussed in the Technical Specifications. This facility change will have no effect on the FSAR other than changes figures.

FC-949

SE94-0248

CONTAINMENT SUMP PH CONTROL

This modification installs the Trisodium Phosphate (TSP) baskets in the containment which will ensure that a pH of 7.0 to 8.0 is achieved at the initiation of Recirculation Actuation Signal (RAS). The baskets containing TSP dodecahydrate will be placed on the floor or raised slightly above the floor (590' elevation) of the containment building. This system is a passive form of pH control for post LOCA containment spray and core cooling water and requires no operator action. The current NaOH system will be retired in place. Retirement of the current hydrazine system is also included in the scope of this modification.

An application for amendment to the Palisades Operating License is required. The change to the Technical Specifications to eliminate hydrazine injection has already been approved by the NRC.

Safety Evaluation Summary

The current revision of the Standard Review Plan Section 6.5.2 requires that containment sump pH be above 7.0 at the initiation of Recirculation Actuation Signal (RAS) to ensure iodine is retained in solution. Also the MHA analysis assumes a pH of 7.0 or greater at RAS. Additionally, Palisades hydrogen generation analysis requires a containment sump pH below 8.0. The current system controls containment sump pH by manual addition of sodium hydroxide (NaOH) from tank T-103. This could result in up to an eight hour time delay after RAS before a containment sump pH of 7.0 to 8.0 is reached.

High temperatures and low pH, which would be present after Loss of Coolant Accident (LOCA), tend to promote Stress Corrosion Cracking (SCC) which could lead to the failure of necessary safety systems or components. Installation of TSP baskets with the required quantity of TSP will ensure a containment sump pH of 7.0 to 8.0 before RAS and help inhibit SCC and reduce the probability of an accident. Since the TSP system is a passive system which performs its design function only after an accident, it will not affect normal operation of the plant.

As part of this modification, the NaOH tank T-103 will be retired after the installation of the TSP baskets.

The TSP baskets will be installed in the containment building at a floor elevation of 590'. The TSP basket frames will be fabricated out of stainless steel and will not be coated. The baskets will be raised about 6" from the floor to avoid loss of TSP due to any spillage or leakage. TSP will dissolve in water as a result of any accident which would cause flooding inside the containment and the water with dissolved phosphate will be drained into the containment sump through the five 16" and one 24" drain lines. There are three cases discussed in Chapter 14 of the FSAR which could cause flooding inside containment. The affect of TSP on the three cases are discussed below.

Case I. Loss of Coolant Accident (LOCA).

A large break LOCA results in initiation of the Safety Injection System. Borated water is pumped from the SIRW tank to the primary coolant system to provide core cooling. Borated water from the SIRW tank is also discharged through two heat exchanger to a dual set of spray headers and spray nozzles in the containment to limit the containment building pressure rise and reduce the potential for release of airborne radioactivity. The borated water will collect on the containment floor and will dissolve the TSP, as it drains into the containment sump. The containment sump pH level will be maintained between 7.0 and 8.0 before the start of recirculation of the sump.

The TSP baskets will have no effect on containment spray system or containment air cooler operation, and will not affect containment pressure following a LOCA. The TSP, once dissolved will prevent iodine re-evolution from the sump solution by maintaining a pH greater than 7.0 as assumed in the MHA analysis. Thus, TSP will not increase the radiological consequences of a LOCA. The TSP baskets are designed to maintain the sump solution pH less than 8.0 following recirculation. This is within the pH level used in the hydrogen generation analysis.

In case of a small break LOCA the recirculation phase may not occur at all or may set in few hours after LOCA. If recirculation is initiated in the effect of TSP would be the same as described above.

Case II. Main Steam Line Break inside containment (MSLB)

Following a postulated Main Steam Line Break inside the containment, the contents of the ruptured steam generator will be released to the containment. To limit the containment building pressure and the temperature rise the Containment Spray System will be activated and borated water will be sprayed inside the containment. The borated water from the containment spray and the water from the main steam line break will dissolve the TSP and drain into the containment sump making the sump water caustic. "There is no recirculation phase following a main steam line break," hence the caustic water will not be recirculated into the Primary Coolant System or

Engineered Safeguards System. The TSP baskets will not affect containment pressure following a MSLB.

Case III. Control Rod Ejection

In the event of a Control Rod Ejection a Loss of Coolant Accident is induced, the situation discussed in Case I above would result.

TSP is a passive form of containment sump pH control and requires no pumps, piping, heaters, etc. as employed by the current sodium hydroxide system. Retirement of T-03 and the corresponding installation of TSP baskets will decrease the number of components required to operate following a LOCA. In case of a LOCA or induced LOCA (Control Rod Ejection) TSP will control the sump pH before RAS to an approximately neutral level between 7.0 and 8.0. At the Recirculation Actuation Signal (RAS) water will be circulated inside the containment atmosphere. The safety equipment inside the containment are designed to operate at a pH near 7.0.

FC-957

SE94-1609

INSTALLATION OF CONTAINMENT BOOM CRANE

FC-957 installs a new 3 ton, pedestal mounted, hydraulic, telescopic boom crane inside the containment building at elevation 649 feet. The crane will be installed during the 1995 refueling outage to support the containment air cooler replacement activities and other various lifting activities which are normally handled by the polar crane. It will remain in the containment permanently and it will be used to move various equipment from the equipment hatch to elevation 590' through the access hatch. The crane must move heavy loads in a safe manner.

The new crane has an electric motor for its hydraulic system. The required minimum capacity is 3-tons at the maximum boom length of 25 feet. The crane will be equipped with a telescopic boom which can be rotated 360 degrees and raised, a remote control with 30 feet of cable for remote operation of the crane, a hook with a safety latch and a junction box on the crane for a convenient power hook-up. The required electric power will be supplied on a temporary basis during outages and no permanent wiring will be installed. The crane will meet the requirements of MIOSHA requirements,and ASME/ANSI requirements. The slings will meet ASME requirements, and be controlled by maintenance procedure.

The crane will be located on or near the primary pump service pit between the equipment hatch and the access hatch. The crane base will be mounted on the containment structure and its anchors will be seismically qualified. The load paths will be from the equipment hatch area to the access hatch toward the center of the containment. The crane may extend over the refueling pool. All potential loose parts

are tack welded, lockwired or secured with locktite to prevent parts from entering the areas of the reactor cavity.

The crane will use Mobile DTE 26 hydraulic fluid which is approved for use inside the containment building. The same hydraulic fluid is also used for the other boom crane L-6 inside the containment. The impact of adding the hydraulic fluid on the Fire Protection Program will be evaluated and documented.

Safety Evaluation Summary

The new crane will be operated according to an approved maintenance procedure which will be revised to include the new crane's operating path.

The new crane will not have functional interface with the existing equipment or systems. It will not have any physical interface with the existing equipment or systems other than the containment structure. The crane will be seismically mounted on the containment building floor and the boom will be secured with a boom rest which will also be seismically mounted in accordance with the CPCO approved design specifications. This will assure structural integrity and prevent any interference with the existing equipment, systems or structures during plant operation.

The crane will be coated with inorganic zinc primer in accordance with the approved specification and procedure FSAR Section 14.22 "Maximum Hypothetical Accident" analyzes a scenario of over-pressurization of the containment due to rapid generation of hydrogen following a LOCA. The zinc material is identified as one of the materials that causes hydrogen generation when subjected to the post-LOCA conditions expected inside the containment building. FSAR Table 14.22-4 "Material Inside Containment Subject to Corrosion by NaOH and Boric Acid Solution" identifies the total amount of zinc base paint inside the containment. The estimated zinc painted surface area of the new crane appears to be small. At this time due, to the lack of information on the exact amount of added zinc, it is assumed that the added amount of zinc is within the margin allowed in the design calculation and there is no change to the accident analysis.

There are no permanent electrical connections or other physical interfaces with the existing equipment or systems other than the structural interface with the containment structure. All structural interfaces will be seismically qualified to assure the structural integrity. The crane will not be used during plant operation and will be operated with the specified safe load paths and the plant conditions per the approved procedure. The administrative controls and the structural adequacy of the crane will ensure that a degradation of the new crane will not affect the safety related equipment or systems.

This modification will not degrade or prevent actions credited in the FSAR analysis of an accident or malfunction, or change an assumption made in the analysis. A degradation of the new crane will not affect the fuel cladding, the primary coolant system boundary or the containment boundary, each of which limit radiation releases to the general public or plant workers.

AUTOMATED WELDING PROCEDURE FOR MSB

The Ventilated Storage Cask (VSC) SER exemptions do not address the ASME Code requirements for post weld heat treatment for the MSB shield lid to shell weld and the three valve access cover welds. An exception to the code can be taken using the charpy V-notch impact testing performed for the welds. The impact testing showed that the lowest energy absorbed for any of the tests was 35 ft-lbs, which was a specimen removed from the unaffected base metal. The lowest energy absorbed by a specimen from either the weld or base metal heat affected zone was 43 ft-lbs.

The lack of a 200°F preheat for the shield lid to shell weld and the three valve access cover welds is not a problem due to the following:

1. A 516 Gr. 70 material is not significantly hardenable.
2. The three welds are multi-pass welds.
3. The impact properties for the weld procedures qualifications for FC-Lid and SM-Lid are adequate.

Safety Evaluation Summary

There will be no affect on any of the Chapter 11 accidents analyzed in the VSC SAR by the exemption to post weld heat treatment for the MSB lid welds. A hypothetical breach of the MSB lid welds is enveloped by the accident analysis.

The cracking of the MSB lid weld is similar to the accident analyzed in the VSC SAR. The VSC SAR describes a breach of the MSB and envelops the hypothetical accidental failure of the MSB lid welds. The release associated with a failure of the lid welds is less than the radioactive release described in the SAR. The resulting exposure due to a hypothetical crack in the MSB lid welds will not be significant as to increase occupational exposure.

The VSC SAR identifies that the worst accidental release would be one where there is a hypothetical failure of all the fuel pins within the MSB and a breach of the MSB containment. The probability of a failure of the welds due to no preheating is very small.

The environmental impact of a postulated breach of the MSB lid welds due to a crack is bounded by the environmental impact related to the accident described in the VSC SAR.

FUNCTIONAL EQUIVALENT SUBSTITUTIONS

REROUTING PIPING FOR WEST ENGINEERED SAFEGUARDS ROOM VENT MONITORING SAMPLE PUMP P-1811

Rerouting of the piping will include the following:

1) Relocating manual valves MV-VA121 and MV-VA125 to no longer interfere with the sampling skid removal for maintenance.

2) Replacing manual globe valves MV-VA121 and MV-VA125 with ball valves.

3) Replacing the existing piping and tubing between the new locations for MV-VA121 and MV-VA125 and the skid with hose suitable for this application.

4) Installing unions on the suction and discharge sides of P-1811 to ease removalof the pump during maintenance.

MV-VA121 and MV-VA125 are normally open valves. The purpose of these valves is to provide isolation between the Engineered Safeguards Room Ventilation System and the RE-1811 sampling skid for normal maintenance. The safety function of the piping and valves is to provide flow to the sampler. One radiation monitor in each engineered safeguard pump room provides room isolation signal upon high radioactivity levels in the applicable room. The automatic isolation allows maintenance of acceptable dose levels at the site boundaries. In the event of significant airborne contamination in the engineered safeguards rooms, the supply and exhaust dampers of those rooms are closed on a signal from the individual radiation monitor for each exhaust duct. The identified piping and valves assure flow to the sampler in case of a high airborne indication in this area. They do not contribute to the possibility of the event or provide any active function to mitigate the event.

The new valves and piping are a one-for-one replacement for the existing configuration and do not introduce any new or different operating parameters from the existing configuration.

The valve and piping functions are to 1) allow flow to the sampler, 2) maintain system pressure, and 3) provide isolation for maintenance.

<u>Safety Evaluation Summary</u>

The new piping and valves will have no affect on the consequences of a high airborne event in the engineered safeguards area as the replacement components will provide the required flow (one complete sample change through the system every 10 seconds per Specification M-217) as new configuration requirements (pressure retention and flow) are equal to or exceed the existing configuration requirements.

Operability and leak testing of the piping and sampler skid shall be performed prior to returning the sampler to service. This will provide assurance that the new configuration does not leak and will provide the required flow to the sampler. The valves and piping provide only a passive function and are not required to be operated and only provide flow.

The skid normally operates at a vacuum on the inlet side and any leakage would be drawn back into the ventilation system. The failure of the valves in such a way to block flow is no more credible for the new configuration than for the existing configuration.

FES-94-019

SE94-0656

REPLACEMENT OF LUBE OIL COOLER TEMPERATURE CONTROLLERS

The lube oil cooler temperature controllers TIC-0835 and TIC-0836 provide non-safety related automatic temperature control of the Turbine Lube Oil System to maintain 40°C by regulating CV-0835 and CV-0836. The original temperature controller had a capillary and bulb inserted into a thermowell to monitor Lube Oil temperature in the Lube Oil Coolers E-15A and E-15B. The replacement temperature controller will use an RTD inserted into the existing thermowell to monitor Lube Oil Temperature.

Safety Evaluation Summary

Temperature controllers TIC-0835 and TIC-0836 provide automatic temperature control of the Turbine Lube Oil Coolers to maintain adequate lube oil cooling. The replacement controllers will be performing the same function as the existing controllers. The type of temperature sensing element is being changed but will use the existing thermowell.

The two extreme failure modes of TIC-0835 and TIC-0836 would cause temperature control valves CV-0835 and CV-0836 respectively to fully open or fully close elimating the automatic temperature control. These two extreme failure modes are no different from those of the existing controller. The temperature sensing element is also being replaced but its failure is bounded by the two extreme failure modes of the controller.

FES-94-039

SE94-0176

REPLACE MV-ES3155A&B&C, MV-ES3130A&B&C, MV-ES3145A&B&C, MV-ES3160A&B&C WITH WHITEY VALVE MODEL SS-8RS8-G

Safety Injection Tank Pressure Transmitter and Pressure Sensor instrument isolation valves are being replaced. The original valves are 1/2" barstock body type 316 stainless steel manual regulating valves with a needle type disc. The valves are

specified as Palisades Class 2. Additionally, the valves will be installed with Type 316 SS tubing versus the Type 304 SS per the original specification.

Valve MV-ES3115C was previously replaced by specification change to a Model SS-8RS8 valve with teflon packing. The new valve will be a Model SS-8RS8-G which is the same valve except it uses Grafoil packing instead of the teflon packing.

Safety Evaluation Summary

The replacement valve is a manual needle type regulating valve whereas the original valve is a globe valve. The application for this valve is as a Safety Injection Tank Sensor Pressure Transmitter and Pressure Sensor isolation valve. As instrument isolation valves in this application they are the second isolation to the Safety Injection Tanks and are ASME Class 2. The replacement valves pressure and temperature ratings (4300 psig at 200°F, 850°F maximum Grafoil packing temperature rating) exceed that of the system requirements (250 psig and 200°F). The valve is a normally open manual valve located in the containment building.

The valve is located on 1/2" tubing, with swagelock fittings. The replacement type is SA213 Type 316SS and the GC Pipe Class calls out ASTM A-213 Type 304; since the specification calls out ASTM A-213 and since SA-213 Type 316SS is an A-213 material and since 316 SS is better in this application, it is acceptable for use.

None of the accidents in the FSAR are related to the isolation or failure of one of the Safety Injection Tank Pressure Transmitters and Pressure Sensor.

These valves are ASME Class 2 per Regulatory Guide 1.26. They do not function as a radiological barrier.

The replacement valves are manually operated normally open valves. They are fabricated to specification equal to or better than that of the original valve specification. The valves will be provided with Grafoil Packing which is the preferred packing for use in the containment building . The valves are not expected to fail under accident demands and environments as the valve and packing ratings exceed that of a post-DBA environment (283°F. 55 psig, and 100% relative humidity).

FES-94-102

SE94-0877

GOULD PUMP POWER END 3196MTX UPGRADES

Change power end of 20 Gould pumps from 3196mt to 3196mtx due to unavailability of replacement parts. (Gould no longer manufactures the 3196mt model)

Safety Evaluation Summary

Gould pumps has upgraded their 3196mt model power ends to the 3196mtx model which are nuclear grade quality. The form, fit and function of the new pump is the same as the original with the exception of a few added features to extend the life of the pump by eliminating the primary causes of bearing failure without compromising interchange ability of parts with the 3196mt model. Twenty pumps may be affected by this power end substitution. These pumps are not safety related, and belong to the RWS, DMW, CVC and MSS systems.

FES-94-299

SE94-1303

INSTALL SCREEN FOR VS-0224

The FES allows the use of a 100 * 100 mesh Monel screen in strainer VS-0224 since the mesh Monel screen per original design is no longer available. Also, the FES returns the strainer screen bolting to the configuration using studs and nuts as shown on the vendor drawing and replaces the gasket to stop leakage from VS-0224.

The purpose of the strainer is to remove particles from the Chemical and Volume Control System (CVCS) boric acid solution from the Boric Acid Batching Tank prior to entering the Concentrated Boric Acid Tanks. Although strainer YS-0224 is not safety related, the strainer should perform it's function without failure of the screen (which allows unstrained fluid to pass to the Concentrated Boric Acid Storage Tanks) and without clogging (which prevents the boric acid from reaching the Concentrated Boric Acid Storage Tanks). The probability of the 100 * 100 mesh screen failing is not expected to be greater than the probability of the 60 mesh screen failing since both screens are designed to withstand operating pressures within the pressure class rating stainer. Loss of the flow from the Boric Acid Batching Tank to the Concentrated Boric Acid Tanks would not prevent the CVCS from delivering sufficient boric acid concentration to bring the plant to a cold shutdown condition due to existing restrictions for CVCS operation. Technical Specifications item 3.2.3 requires at least one Concentrated Boric Acid Tank to be operable and contain a sufficient quantity of boric acid to bring the plant to a cold shutdown condition. Therefore, at least one Concentrated Boric Acid Tank would contain adequate inventory even if the strainer clogged.

Safety Evaluation Summary

No analyzed accidents are affected by the strainer. As noted above, failure of the stainer to pass flow will not prevent the CVCS from bringing the plant to a cold shutdown condition. Failure of the 100 * 100 mesh screen to strain will be more significant than failure of the 60 mesh screen to strain the boric acid prior to entering the Concentrated Boric Acid Tanks.

FES-94-369

SE94-1364

REPLACE VALVE MV-CVC2329A (VELAN GATE) WITH A CONVAL GLOBE VALVE

The existing 3/4" 150 lb Velan gate valve will be replaced with a 3/4" 150 lb Conval globe valve. The existing valve's seat leaks. The existing type of valve must be completely replaced when maintenance is required while the new valve can be repaired in place.

Valve MT-CVC2329A is used for periodic manual filling and makeup water for the Charging Pump Seal Lubrication Collection Tank.

Safety Evaluation Summary

The valve is not required to support any systems during the mitigation of the accidents identified in the FSAR. The failure modes of the valve are not changed by this FES.

A failure of valve MV-CVC2329A does not affect the operation of the Charging Pumps.

FES-94-412

SE94-1613

REPLACE MV-CD144

Replacement of MV-CD144, condensate pump discharge to pressure control valve. PCV-0764 from a gate valve to a ball valve.

Safety Evaluation Summary

The valve satisfies the applicable design requirements of the system. The valve is normally open. This valve does not normally contain radiologically active or contaminated fluid. Its operation does not introduce any new, or add to any existing, source terms used in the analyses of accidents.

This component is not considered "important to safety." This component does not affect the function of equipment important to safety. This change does not introduce any new failure modes for equipment, or functionally alter equipment or process routes.

The change results in a revision to an FSAR figure identifying the type of valve.

SPECIFICATION CHANGES

SC-92-014
SE94-0140

PROVIDE 120V AND 240V POWER FEED TO THE ANNEX FACILITY (OFFICE SPACE)

The 120v and 240v power feed to the Annex facility will be provided by connecting a 250 KVA, 2400/4160Y - 120/240v transformer to the existing 2400v overhead line which is connected to the plant's bus 1E.

Safety Evaluation Summary

The change does not effect safety related equipment. The plant's bus 1E was evaluated and determined to have enough capacity to support the load at the new DET facility.

The change results in a minor change to an FSAR figure.

SC-92-052
SE94-0581

UPGRADE AND REVISE SETPOINTS ON SERVICE WATER SYSTEM HIGH DIFFERENTIAL PRESSURE SWITCHES

The SWS high differential pressure switches (DPS-1319, 1321 and 1325) are being upgraded to supply the required overpressure protection. The existing isolation valves are being replaced with 5 valve manifolds to aid in instrument calibration. The switch set points will be revised per an engineering analysis.

Safety Evaluation Summary

This modification upgrades the pressure switches, adds five valve manifolds and revises the set points. These changes will improve system performance but do not change any of the equipment function.

The modification has no functional effect on components that mitigate an accident. This modification provides improved equipment for system operating. The revised set point will provide increased operating margin.

SC-92-113

SE94-0128

FEEDWATER PUMP DRAIN TRAP (PT-0546 AND 0547) REPLACEMENT AND Y-STRAINER ADDITION

The change is the addition of Y-Strainers to the inlet lines to the drain traps which drain T-26 A/B (Feed Pump Drain Tanks) to the main condenser. This is being done to protect the traps from the introduction of foreign material.

Safety Evaluation Summary

This change is to reduce the likelihood of a Loss of Condenser Vacuum due to trap failure. The traps and Y-strainers are non-safety-related and do not interact with any other safety-related system. The drain trap change is required due to changes on the feed pumps which resulted in decreased water flow to the drain tanks. This required drain traps to handle lower flow rates. The Y-Strainers were added to protect the new traps which have much smaller outlets than the original traps.

SC-92-219

SE94-0168

REPLACE P-47A AND P-47B

This specification change replaces the P-47, scale inhibitor metering pumps, in the Circulating Water System, re-routs the suction and discharge piping and lowers the pump foundation.

Safety Evaluation Summary

No analyzed accidents are affected by this change. The P-47 pumps are not identified in any analyzed accident scenario and are not required to operate in a post accident condition.

The P-47 pumps are non-safety related. The redesign of the piping system will not degrade the reliability of the pumps to operate.

SC-93-003

SE94-1596

REPLACE GATE VALVES IN DIESEL STARTING AIR SYSTEM DRAINS WITH WHITEY BALL VALVES

Manual valves provide drain paths for the diesel generator starting air tanks. The replacement ball valves will function in the same manner as the existing gate valves.

The pressure and temperature rating of the replacement valve is greater than that of the original.

Safety Evaluation Summary

The replacement valve type is better suited for the application and design characteristics of the replacement exceed those of the original. The Whitey ball valves have been found to be equivalent or superior than the existing gate valves in the individual design characteristics required for their positions in the DG starting air system.

The basic requirements of the starting air tank drain valves are full close, full open, and to retain rated pressure. All accidents concerning these requirements have been previously evaluated.

This change affects FSAR figures.

SC-93-041

SE94-0851

REPLACE LEFT CHANNEL HEAT TRACE SAMPLE LINE TEMPERATURE SWITCHES AND ADD THERMOCOUPLE FOR TEST AND CALIBRATION

Safety Evaluation Summary

The environmentally qualified life for the containment hydrogen monitoring heat trace temperature switches expired. The right channel switches are no longer considered in a harsh environment and not required to be environmentally qualified. The left channel switches require qualification.

Thermocouples TE-2401 - TE-2402 are being added to the Hydrogen Monitoring System to be used by maintenance during calibration of temperature switches. The H_2 monitoring system is a post accident monitoring system. The additional thermocouples will not cause malfunctions or other problems because they are only used during maintenance activities.

SC-93-056

SE94-1188

REPLACE pH AND CONDUCTIVITY INSTRUMENTATION IN THE C42 PANEL

Upgrade of the pH and conductivity instrumentation for steam generator sampling. The conductivity samples will be routed to the sink to simplify the taking of samples and increase reliability.

Safety Evaluation Summary

The change in routing of the samples affects P&ID drawings that are included in the FSAR. This change only affects the turbine sample panel which is remote from equipment important to safety. This change has no affect on safety equipment or any accident analysis.

SC-93-068

SE94-0689

SAFETY INJECTION, CONTAINMENT SPRAY AND SHUTDOWN COOLING SUPPORT MODIFICATIONS

To bring the piping and supports into compliance with FSAR design requirements based on the Safety Related Piping Reverification Program's as-built analysis.

Safety Evaluation Summary

This SC will add, remove, and modify existing ESS pipe supports. These modifications are required to bring the supports and piping into FSAR compliance. The interim operability criteria provides system operability prior to the modification. Hanger HC3-5151.1 will be working during the 1995 refueling outage while the system is declared inoperable.

There are no accidents involving any supports which will require modification on the Safety Injection, Containment Spray, and Shutdown Cooling Piping. The system is operable in accordance with the interim operability criteria.

Implementation of the Specification change will result in removal of some hydraulic shock suppressors identified in the FSAR.

SC-93-083
EA-SC-93-083-02
EDC-SC-93-083-05

SE94-0155
SE94-0405
SE94-0857

DESIGN CHANGES TO VSC - PHASE II

The changes reflect the lessons learned during the fabrication and loading of VSCs 1&2 and apply to the future VSC fabrication and installation in each of the categories evaluated as listed below:

- Painting Welds on MSB Structural Lids

- SNC Drawing Changes - MSB
- SNC Drawing Changes - VCC
- SNC Fabrication Specification Changes - MSB
- SNC Fabrication Specification Changes - VCC
- Rearrangement of VSCs on the Storage Pad
- Replace VDS Gauge
- MSB Dimensional Stack-up

Safety Evaluation Summary

Painting Welds on MSB Structural Lids

Carbo-zinc 11 primer is required on all carbon steel surfaces and is provided on the structural lids supplied with the MSB. Welding of the structural lid following loading results in an as welded surface. Primer was not specified to be reapplied on the structural lid weld surfaces after loading. This change restores the primer to the as welded surfaces of the structural lid.

SNC Drawing Changes - MSB

A description of changes made on the MSB drawings is grouped into three categories.

The first category is the drawing changes related to the structural lid bolt hole plug and the shield lid integral plate design changes which were previously evaluated.

The second category is the drawing changes made from clarity, editorial corrections, addition of tolerances for fit-up and inspection, and supplementary detail. The drawing changes in this category will not be further evaluated because they do not affect design of the MSB and governing design documents, SAR, SER and COC.

The third category is the drawing changes related to the weld change from V groove weld to a 1/2" double bevel weld between the shell and the bottom plate of MSB. This change was required to minimize the bow on the bottom plate and was not previously evaluated.

The changed weld is one of the MSB pressure boundary welds described in the SAR. The weld change does not reduce the integrity of the MSB because the weld is full thickness of the shell, full penetration and requires 100% radiography examination. The full penetration double bevel weld will not cause the consequences of the accident evaluated in the VSC SAR to be increased since the weld requirements including weld thickness and inspection requirements were not changed. The weld joint is not an active component and has no effect on the operation of the VSC. The weld joint is a passive item. The weld change will not affect the function of MSB and VSC. The weld material, welding requirements and examination requirements are not changed. The weld will maintain the MSB pressure boundary as originally designed. As indicated above, the weld change does not degrade the joint strength nor affect function of MSB and VSC. Weld joint strength is not reduced by the full penetration

double bevel weld. This welding will be done at fabricator's shop. There will be no increase in occupational exposure due to the weld change. Since the MSB boundary will be maintained, there will be no additional environmental impact as a result of the weld change.

SNC Drawing Changes - VCC

The drawing listed in the DCNs/DCRs related to the following items: Alignment plate size, air outlet assembly, air inlet assembly, shield rings, relaxed tolerances, and clarifications.

Item F, Clarifications, were made, editorial corrections, additions or tightening of tolerances for fit-up and inspection, etc, which will not affect VCC design nor the governing design documents, SAR, SER and COC.

The alignment plate size was changed from 40 in. to 70.5 in. and the number of Nelson studs per plate was reduced from four to two. The changes were to facilitate loading and fabrication of VCCs. Detailed engineering evaluation of this item is provided in Section 4.5.A of the Engineering Analysis. The alignment plates are not structural members, shielding components or part of the convection cooling function. Function of the alignment plates is to facilitate loading a MSB into a VCC. They are not important to safety of the VCC. The change does not affect the original function of the alignment plates or affect safety related items. Having better fit alignment plates will reduce loading time, which will also reduce radiation exposure time.

The changes made on the air inlet and outlet assemblies was evaluated in the Engineering Analysis. The changes were to facilitate fabrication and installation of the air inlet and outlet assemblies. The changes did not reduce air flow area or integrity of VCC. The changes do not reduce cooling capability or structural integrity of VCC. Function of the air inlet and outlet assemblies was not changed or reduced. The changes do not affect original function or reduce integrity of the VSC. Functions defined for the air inlet and outlet assemblies were not changed or reduced. The change made on configuration of the air inlet and outlet assemblies will not affect shielding. Plant procedures will provide guidelines and cautions for radiation protection without the shield blocks at the VCC air outlets.

Changes related to the shield rings evaluated in the Engineering Analysis attached. The changes were made to enhance installation of the shield ring in VCC. The changes do not reduce shielding capability of the rings. The changes on the shield ring does not affect a structural integrity or convection cooling capability of VSC. It is for radiation shielding for annulus area. Function of shield ring was not affected by the changes. Shielding capability and structural integrity of the shield ring were not affected by the changes. Configuration required for shielding was maintained.

The relaxed tolerances were evaluated in the Engineering Analysis attached. The relaxed tolerances did not impact on structural integrity or radiation shielding capability of the VCC. Reduction in flow area due to the relaxed tolerance is very small and not

at the critical flow area of cooling. Structural integrity and shielding & cooling capability of VSC were not reduced by the changes. Critical air flow area was not reduced. Shielding configuration was maintained as designed.

SNC Fabrication Specification Changes - MSB

The changes were evaluated in Section 4.6 of the Engineering Analysis. As indicated in the Engineering Analysis, the changes were to provide clarity to enhance the fabrication process and quality of the MSB. The changes do not affect design of MSBs.

SNC Fabrication Specification Change - VCC

The change provided in the DCNs/DCRs were evaluated in Section 4.8 of the Engineering Analysis, EA-SC-93-083-01. As indicated in the Engineering Analysis the changes were clarifications which do not affect design of VCC. The changes will enhance the fabrication process and quality of VCCs.

Rearrangement of VCC's on the Storage Pad

The change is needed to maximize the capability for storing and moving unloaded VCCs on the pad without interfering with loaded VSCs.

Replace VDS Gauge

The vacuum dry system gauge is used to measure vacuum pressure in the MSB during the drying process, helium backfill and leak test of the MSB. Detailed engineering evaluation of the VDS gauge replacement from digital to analog is provided in the Engineering Analysis.

MSB Dimensional Stack-up

Detailed engineering evaluation of this item is provided in the Engineering Analysis. The gap between top of the MSB and bottom of shield ring assembly may not be maintained as intended. The shield ring will be in contact with the MSB top or lifted by the MSB. Function of the shield ring is to provide radiation shield at the annulus between the MSB and VCC liner. The lifted shield ring assembly will not reduce the effectiveness of shielding, convection cooling capability nor will it affect structural integrity of the VCC liner bottom plate assembly.

The changes on MSBs and VCCs were evaluated and documented in EA-SC-93-038-01, Rev 2. Structural integrity and function of the MSB and VCC were not affected by the changes. The changes covered by EDC-SC-93-083-05 were found to be adequately addressed in the categories previously evaluated in 10CFR72.48 Safety Evaluation section of the Safety review, PS&L Log No. 94-0155.

EA-SC-93-083-02 documents a number of NRC accepted exceptions that are further documented in the SER and SAR. These include; changes in toughness requirements for Charpy testing, waiver of N stamp, full penetration weld for double seal welds, radiographic examination for final closure welds, and hydrostatic test. The EA also discussed the change to incorporated girth weld. It shows that the weld does not decrease strength, and therefore does not created an unreviewed safety question.

SC-93-094

SE94-0170

100 TON FUEL BUILDING CRANE L3 MODIFICATION

Replacement of fuel handling building crane L-3 control chief radio control panel with a new control chief panel, replace main hoist motor, current brake, relocate existing magnetic brakes and modify existing protective panel to eliminate dash pot over load relays associated with the main hoist power supply. Replacement of existing cables is necessary to facilitate modification.

Safety Evaluation Summary

In accordance with FSAR Section 9.11.4.3, the Fuel Building Crane is provided with positive means to prevent items normally held by gravity from being dislodged and falling on equipment or structures located below the crane.

Replacing the Main Hoist Motor with a larger horsepower motor and new motor controls along with new radio controls for the entire crane has no significant impact on the crane other than enhancing the overall performance and reliability.

Operating criteria, particularly lift capacity has not been changed, the consequences of a dropped load have not been changed.

The crane has no safety related function. It is safety related only in that it must not be dislodged or any items on the crane must not be dislodged and fall down on any safety related equipment and structures situated below it.

Engineering Analyses have demonstrated that the crane still satisfies this criteria with the implementation of this modification.

The impact of replacing the main hoist motor, motor controls and overall crane controls will have no effect on the functional characteristics of the crane. The crane will be tested to the same requirements as the crane meets currently. It will not directly or indirectly alter the design basis for any safety related or non-safety related equipment in either the containment building or auxiliary building addition.

MODIFY V-7 HEATING COIL ARRANGEMENT, DRAINS AND FILTER.

The function of the spent fuel pool area ventilation system is to maintain ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operations, anticipated operational transients and following postulated fuel handling accidents.

The change will improve condensate drainage, reducing the potential for coil freezing and will involve the following modifications:

- Removal of the air handling unit's existing roll-a-matic filter located between the pre-heat coil VHX-7A and reheat coil VHX-7B,
- Relocation of pre-heat coil VHX-7A and low temperature alarm probe to where the roll-a-matic filter was removed,
- Installation of fiberglass media throw-away filters (of similar efficiency and pressure drop) and holding frame,
- Installation of two condensate steam traps with integral vacuum breakers (one per coil), one condensate steam trap for the supply drip leg, one receiving tank/pump skid, and associated valves, and,
- Modification of steam piping to accommodate the new and relocated equipment.

Safety Evaluation Summary

V-7 is not relied upon for any analyzed accident. The corresponding exhaust fan is however. This modification maintains the same function of V-7; clean supply air heated to a setpoint. The filters will be changed manually rather than automatically. Worst case is if filters becomes plugged then supply air volume reduces which is in a conservative safety directions assuring negative pressure in the fuel pool area.

The modification adds a condensate tank and redundant pumps at the steam coil drains. The piping will meet plant pipe and weld specs. The tank capacity is approximately 20 gallons while the pump is 18 gpm. The failure of the tank to move condensate will bring in a hi-level indication at the control room. The tank will overflow to the floor at a flowrate of 3 to 6 gpm to a nearby floor drain.

The automatic filter will be removed and a preventive change - out will occur. Loss of air capacity is conservative. Should air flow drop on excessive dirt build-up on filter. The filter ΔP indication will remain.

This fan and heating coil are not safety related. The system provides fresh air to the fuel pool area. Upon an accident, operators manually trips V-7. This function will not change.

This modification enhances the function of the heating coils by removing condensate. The manual replacement of filters is equivalent to the automatic function as is being done on VF-10 and VF-33 which received a similar change. The piping change is built to CPCo Design Class 3 per ANSI B31.1 1973, Summer 1973 agenda.

Loss of V-7 heating and airflow is non-essential to fuel handling accident events.

SC-94-003

SE94-0351

THERMAL MARGIN MONITOR (TMM) ANNUNCIATOR TIME DELAY

Replace the existing capacitor with a capacitor of different value to cause a time delay of up to 8 seconds.

Safety Evaluation Summary

The Annunciator boards to be modified, EK-0603D, EK-0604D, EK-0607D, and EK-608D represent Channels A, B, C, & D respectively. The Annunciator for each Channel is a single alarm serving three purposes; 1) Nuclear - ΔT Power Deviation, 2) T-Inlet Off Normal, and 3) Calculator Trouble. The three alarms are initiated by each Thermal Margin Monitor. The capacitor replacement to the annunciator board will affect all three alarms.

The Annunciators are not Q-Listed and are isolated from the Thermal Margin Monitor (TMM). There are no automatic actions generated from any of the referenced annunciators.

With respect to Technical Specifications Table 3.17.4 Item 2 and Table 4.1.3 Item 9, the ΔT - Power Comparator refers to deviation meters DIA-0010A, B, C, & D, which are analog center scale meters located in the TMM panel (C-27). These meters provide a contact signal to the annunciators but the annunciators themselves are not a part of the Technical Specifications required equipment.

SC-94-004

SE94-1171

NITROGEN TO ELECTRICAL PENETRATIONS (NORTH ROOM)

Permanently install the nitrogen supply bottles for Nitrogen Station No.6 in the North Electrical Penetration Room that are used to maintain a nitrogen blanket on the internals of the north electrical containment penetrations. In addition to simply

permanently mounting the supply portion of the system, the Specification Change also ensures the nitrogen system meets the supply for the containment penetration boundary. The evaluation and support modifications provide the seismic mounting that is needed to meet the containment penetration boundary requirements. The system is located in the North Penetration Room, a Class 1 area providing the tornado protection.

Safety Evaluation Summary

Nitrogen Station No.6 already exists, although in a "temporary" condition consisting a chain for a bottle support and rubber hose for supply lines. The nitrogen station has existed this way since original construction.

This nitrogen system acts only passively following to an accident. As part of the modification, the distribution system is being evaluated and modified to ensure it is adequate as a containment pressure boundary.

A local leak rate test for the north electrical penetrations, will be used to periodically verify the leak tight integrity of the supply check valve.

The modification maintains all the functions of the existing "temporary" system while ensuring the ability of the system to meet the original design requirements for the electrical penetrations. The modified distribution system will meet the seismic requirements due to now being properly supported and the entire system will meet tornado requirements due to being located in the North Penetration Room.

SC-94-016
SC-94-018

SE94-0178
SE94-0179

WELD REPAIR OF CK-ES3166 AND CK-ES3181 CONTAINMENT SUMP RETURN CHECK VALVES

CK-ES3166 was determined to have through wall leak caused by intergranular Corrosion (IGC) assisted by a highly sensitized base metal. Examination of CK-ES3181 identified UT indications. There indications were treated as flaws. The repair method chosen is to use ASME Code Case N-504-1. This Code Case has not yet been generically approved for the use of the code case was submitted by the Nuclear Regulatory Commission (NRC) and approved by the NRC w/a letter dated.

Safety Evaluation Summary

The overlay is put on using standard welding techniques and will be performed with the plant in cold shutdown. The overlay will meet ASME Section XI stress requirements.

The overlay will restore the design margin of the valve. The valve remains fully capable of performing its function in mitigating accidents.

The overlay is located on the outer surface of the check valve away from the pivot shaft bearings. It will not affect the operation of the check valve in any way and will not affect any other equipment as it does not interfere with any other equipment. The overlay's weight is approximately 7% of the valves weight, thus it does not adversely affect the stress levels or seismic response of the piping system to which it is attached.

SC-94-038

SE94-0449

T-2 LOW LEVEL ALARM SETPOINT CHANGE

Condensate Storage Tank T-2 low level alarm setpoints on LIA-2021 and LIA-2022 will be changed from 50% to 68% to accommodate a revised inventory need, instrument uncertainty, and operating margin.

Safety Evaluation Summary

The configuration and operation of the T-2 low level alarms will not be altered, only the setpoint will be changed to reflect the revised inventory requirements. The low level alarm setpoint change will increase the operating margin since the alarm is activated to alert the control room of T-2 level potentially failing below the low inventory limit prior to reaching this limit.

This modification is to increase operations awareness of T-2 inventory when levels are decreasing towards the minimum acceptable limits. The increased awareness and operating margin should increase the probability of T-2 being capable of performing its designed function in support of AFW operation.

As discussed, this modification should increase operators ability to maximize the reliability of T-2 which increases the reliability of the AFW system.

The increased T-2 low level alarm setpoint will alert the control room to potential inventory problems in a more timely manner. The higher setpoint ensures less inventory may be lost before the control room alarm activates to identify the problem which more time for corrective actions to mitigate any malfunctions.

SC-94-044

SE94-0494

DISCONNECT QUALIFIED CETs FROM THE PIP DATALOGGER

The modification will isolate the 16 Qualified Core Exit Thermocouples (CETs) from the Primary Indicating Processor (PIP) datalogger. Each Qualified CET signal wire and the wire which routes it over to the temperature recorder will be lifted from the PIP terminal strip. These wires will be spliced. The PIP inputs for these 16 thermocouples will be jumpered plus to minus to prevent noise problems.

Safety Evaluation Summary

Core Exit Thermocouples are not used for any control or alarming function. CET temperatures are required for mitigating the consequences of accidents such as a Steam Generator Tube rupture. However, the PIP datalogger is not used for the indication for these temperatures. CET temperatures are available on the Critical Functions Monitor System (CFMS) and on the CET recorders. Emergency Operation Procedures either identify the CFMS as the indicator of choice or do not mention which indicator to obtain temperatures from.

This modification will take place in the CO6-2, PIP, cabinet. This cabinet is seismically qualified. This modification will provide isolation between the Qualified 1E CET circuits and non-1E PIP datalogger.

This modification does not affect the 1E or seismic qualifications of the Qualified CET circuit. Additionally, there are two indicating devices left in the circuit. Isolation from the non-1E PIP datalogger will therefore not degrade the Qualified CET system's operability during or after an accident.

SC-94-053

SE94-0702
SE94-0751

FLOOD BARRIER FOR FUEL OIL TRANSFER PUMPS

A flood barrier is required to be installed around the fuel oil transfer pumps P18A&B. This includes removal of the existing 1'-0" high concrete curb and construction a 4'-8" high concrete flood barrier to protect the pumps from maximum flood level of 593.5 feet.

Safety Evaluation Summary

The purpose of the flood barrier is to protect the pumps from external flooding due to seiche.

Addition of the flood barrier will only have impact on the two fuel oil transfer pumps inside the barrier. By erecting a flood barrier around these pumps, the potential is added to flood the pumps from an internal source. However, the source of the internal flood (spray from fire protection sprinklers or pipe rupture) would have already rendered the pump's motor inoperable regardless of whether or not a flood barrier existed.

The 4'-8" flood barrier will not cause the fire hazard consequences to change from those created by the existing 1'-0" curb around the fuel oil transfer pumps.

SC-94-064

SE94-1172

INSTALL NITROGEN CHECK VALVE IN SOUTH PENETRATION ROOM

The Specification Change is intended to ensure the nitrogen system meets the requirement for the containment penetration boundary. This is being done by making a small change in how the distribution system is supported, and by installing a boundary supply check valve. The supply check valve provides separation between the supply and distribution portions of the nitrogen system. This separation defines the barrier between the seismic and non-seismic portions of the nitrogen system.

The check valve is mounted and tornado protected boundary that will insure the integrity of the containment penetration second boundary. This modification ensures that the integrity of the electrical penetrations is maintained due to the nitrogen distribution system being properly protected from tornado effects and seismic events.

Safety Evaluation Summary

The nitrogen system to the south electrical penetration already exists, although it has not been evaluated for seismic adequacy and tornado.

As part of the modification, the distribution system is being evaluated and modified to ensure it is adequate as a containment pressure boundary. The local leak rate test for the south electrical penetrations, will be used to periodically verify the leak tight integrity of the supply check valve.

The modification maintains all the functions of the existing system while evaluating the ability of the system to meet the original design requirements for the electrical penetrations.

SC-94-065
EA-DPAL-94-150-01
SE94-0659
SE94-0621

CHANGE SHIELD COOLING SYSTEM FLOWRATE TO 134-154 GPM

The analysis is written to analyze the Shield Cooling System at a flowrate of 134 to 154 gpm. The required normal flowrate is 146 gpm and the Shield Cooling Heat Exchanger, E-64, is only designed for 125 gpm. The 154 gpm flowrate is the upper limit of flow due to the flow ranges given to balance each loop.

Safety Evaluation Summary

The Shield Cooling System is not safety related. The analysis is written to verify that the required system flowrate of 134-154 gpm is compatible with the components in the Shield Cooling System. Bechtel originally designed the Shield Cooling System for 125 gpm, however, during the construction phase, the floor coils were added. This resulted in additional flow requirements on the system. Bechtel re-sized the Shield Cooling Pumps, P-77A and B but the analysis of the system at the flowrate of 146 gpm could not be found.

The Design Basis Function of the Shield Cooling System is to maintain the biosheild concrete temperature below 165°F. The EA analyzes the system at a minimum flowrate of 38 gpm. Other analysis analyzed the system at a minimum flowrate through the wall coils of 30 gpm and this flowrate was found to be acceptable. Setting the minimum flow rates at 38 gpm through the wall coils maintains a margin of 8 gpm.

The analysis determined that the change in flowrate did not significantly effect the heat transfer coefficient.

SC-94-114
SE94-1462

UPGRADE C-42 SAMPLE LINE FILTERS

This modification will move the sample inlet filters from upstream of the sample rough coolers to upstream of the Sodium and Hydrazine analysis. This will be done to align the existing sampling system to the latest sampling system technology.

Safety Evaluation Summary

There is no accident evaluated in the FSAR for the EC-42 turbine plant sample panel. A malfunction of these filters will have no effect on safety related equipment.

These filters have no effect on any plant system other than the sample panel itself.

The change affects FSAR figures.

SC-94-510

SE94-0583

ADDITION OF ISOLATORS BETWEEN THE TMM AND THE TM/LP BI-STABLE TRIP UNIT

Install new isolators in the output circuit of the four TMM Variable High Power Trip Calculators located in panel C27. These isolators are to be mounted in the Calibration and Indication drawer/panel below each calculator. The FSAR Section states that the analog portion of the RPS is ungrounded. There is a 10K ohm resistor in the TMM's CRT display circuit that couples the circuit common to chassis ground. It was determined that this ground path conflicts with the FSAR. A second ground may result in a non-conservative signal being generated in some of the instruments on the Pressurizer Pressure instrument current loop.

Safety Evaluation Summary

Prior to this modification the TMM, the bistable trip TM/LP, and the Pressurizer Pressure loop all shared a signal common. By design the video circuit of the TMM has a ground. This signal common is carried though the TM/LP Bistable Trip to the Pressurizer Pressure (PP) instrument loop. This sets up a situation where, depending on where on the loop it is introduced, a ground on the PP loop can cause a nonconcervative signal being generated in some loop instruments.

Isolation is provided in the Action Instruments Model 4300, DC input isolating signal conditioner via a photo-isolator. It provides a DC input/output isolation of up to 600V DC or AC peak between input and output. There is no signal common or ground that is common between the input and output sides of the isolator.

This Specifications Change does not change any functionality of the TMM, the TM/LP Bistable Trip of the Pressurizer Pressure Loop. It does add another component (the isolator) between the output of each TMM and its associated TM/LP Bistable Trip unit. This isolator will ensure that a single ground in the pressurizer pressure instrument loop will not interact with the original design, 10KΩ, signal common to chassis ground path in the TMM.

TEMPORARY MODIFICATIONS

TM-94-002

SE94-0041

VHX-7A AND B CONDENSATE LINE

Currently, the outlet of VHX-7A and B, Fuel Handling area tempering coil, does not drain properly and could freeze. This TM allows VHX-7A and B to drain via the SFP Reheat Coil, VHX-66, drain line.

Safety Evaluation Summary

Spent Fuel Pool supply fans V-7 and V-69 are not described in the FSAR, although the corresponding exhaust fan V-8 is. This temporary modification maintains the function of V-7, clean supply air heated to a set point, but limits V-69 ability to supply heated air. This limitation is caused by VHX-66 being out of service, however, this should not be a concern as V-7 will supply improved heating. V-8A and B, SFP area exhausters, are not affected by this temporary modification.

This temporary modification adds a condensate return line that runs north away from the spent fuel pool then east to V-69 in the heating and ventilation room of the auxiliary building addition. The piping will be fabricated to meet plant pipe and weld specs, and flowrate is expected to be < 3 gpm and pressure < 1 psi. Therefore in the event of a pipe line break flooding will be controlled by existing floor drains, and jet impingement impact would not effect safety related equipment. V-7 fan and heating coils are not safety related. The system provides heat to the fuel pool area upon a fuel handling accident, operators manually trip V-7, this function will not be changed. This modification enhances the function of the VHX-7A, B heating coils by improving condensate removal. Loss of V-7 heating is non-essential to fuel handling accidents events.

TM-94-005

SE94-0049

INSTALLATION OF A TEMPORARY MODIFICATION TO SIMULATE STACK FLOW FOR RGEM DUE TO THE FAILURE OF FT-1818

This installation of a temporary modification will simulate stack flow for the RGEM (Radioactive Gaseous Effluent Monitor). This temporary modification will allow RGEM to become operable and therefore monitor the airborne radioactivity effluent present in the Stack Gas in both Normal and High Range modes.

Safety Evaluation Summary

This temporary modification involves artificially supplying a conservative stack flow indication which will result in a greater than required sample volume This system has no interconnections with the reactor or plant protection circuitry.

This system is intended to monitor radioactive material released via the stack for normal or accident conditions. This modification will allow the system to remain operational. The system does not initiate isolations or other trip signals. With regard to monitoring the magnitude of a release, a conservative stack flow value will be used. This will result in a larger sampling flow rate than is necessary for isokinetic sampling. For high range noble gases accurate concentrations will be estimated. For particulates the relatively large particles will be sampled accurately because their trajectories will not change. However, small particles will be over sampled because their trajectories will be changed. Essentially the small particles will be preferentially sucked in the sampling orifice while big ones will not. Particulate monitor readings will reflect this conservative over sampling.

The input of an artificially conservative stack flow rate signal (roughly 7% higher than average) will not result in any significant challenges to the system. The sample pump will not run harder, a flow orifice will simply be more open. The sampling pump has overload protection circuitry. This is the only equipment that will be affected as a result of this modification.

This system only monitors radioactive material that is flowing in the stack. It is unrelated and not interconnected to systems that could affect changes in the concentrations flowing in the stack.

TM-94-007

SE94-0100

INSTALL TEMPORARY VENT LINE FROM T-76 TO MV-WO6533

This TM will install a vent line from the Chem Lab Drain Tank T-76 to the exhaust Plenum. This will bypass a plugged section of the vent gas collection header (VGCH) and re-establish the vent path from T-76 and the associated vent header directly to the exhaust Plenum.

Safety Evaluation Summary

The T-76 tank collects drains from the NSSS and Radwaste sample sinks. The VGCH is then supposed to collect any gases that escape from these sample drains. The TM will not impact the operability of any safety related equipment or equipment required for plant operation.

This TM re-establishes the design flow path for the VGCH from the T-76 tank to the exhaust plenum.

TM-94-016

SE94-0156

REPLACEMENT OF SOLENOID VALVE SV 2410 WITH A MANUAL VALVE

Currently both waste gas decay tanks T68 and T101C can not be sampled because SV 2410 is out of service and can not be repaired or replaced in an acceptable period of time. In order to sample these tanks the existing 3 way solenoid valve will be replaced with a similar 3 way manual valve. These tanks must be sampled to determine if they contain an explosive mixture and also to determine their activity levels.

Safety Evaluation Summary

This sample valve is not safety related and will have no effect on any accident evaluated in the FSAR. It will allow a sample to be drawn from the Waste Gas Decay Tanks, T68C and T101C. This sample valve has no effect on other plant systems.

TM-94-025

SE94-0300

M-94-025

TM-94-025 will disconnect the upper and lower Power Range Nuclear Instrumentation (NI) inputs to the CFMS and Plant Datalogger which do not meet channel separation criteria. The inputs include Channel A, B, C, D signals to the Plant Datalogger and Channel C, D signals to the CFMS.

Safety Evaluation Summary

Both CMFS and Datalogger inputs are high impedance devices. Their removal from the circuit which also connects the NI signals to the TMM will have no electrical effect. The CFMS and Datalogger are monitoring systems with no control functions.

Two channels of upper and lower power will still be available to the CFMS for post-trip review. Along with wide-range and power range NI, there are adequate CFMS inputs for review purposes. There are no specific requirements for which inputs must be available on the CFMS or Datalogger for post-trip review.

The CFMS uses upper and lower NI inputs to calculate Axial Shape Index (ASI) value. This value is graphically trended on the CFMS monitor in the control room. This is a convenience to operators, not a requirement. It will still calculate an ASI value based on the two channel still input to the CFMS.

TM-94-043
TM-94-044
TM-95-045
SE94-0435
SE94-0436
SE94-0437

INSTALL D/P GAGE FOR BASKET STRAINER BS-1319

This temporary modification installs a differential pressure gage to obtain data on service water pump's basket strainers. The data will be used for a service water system flow analysis.

Safety Evaluation Summary

There will be no affect on plant accidents or equipment malfunctions previously evaluated. The gage is a small diameter instrument line and its failure would easily be made up by the SW pumps. The changes does not affect operation of the service water system. The differential pressure gage is not seismically qualified for operability but the modification meets the plant specifications for the tubing and therefore the seismic qualification of the system is maintained.

TM-94-046
SE94-0451

SHIELD COOLING FLOW BALANCE

The temporary modification installs a temporary pressure indicator in the piping at the shield cooling pumps P-77A and P-77B discharge.

Safety Evaluation Summary

The accident previously evaluated is loss of shield cooling with PCS temperature greater than 165°F, resulting in possible degradation of the Bioshield concrete. The probability of loss of shield cooling will not be increased by the temporary installation of a P-77A and B discharge pressure gauge. The PCS temperature will be less than 165°F and the shield cooling system will not be required to be operable. The TM will be removed prior to the need for system operability.

INSTALL PRIMING LINE FROM P-5 DISCHARGE TO COOLING TOWER PRIMING TANK T-34

The TM is installed to resolve a design problem with the warm water recirculation pump P-5 discharge piping. A temporary hose with isolation valves installed on each end in installed between the discharge of P-5 and the C-16 water box vacuum priming system. This allows the vacuum priming system to remove all the air in the discharge piping prior to operation.

Safety Evaluation Summary

No analyzed accidents will be affected by this change. The Warm Water Recirculation Pump is not in any analyzed accident scenario. The FSAR identifies the pump as being required if the intake crib collapses or some other event happens which causes a loss of inlet water to the Service Water Bay. The TM is installed to resolve a design problem with the P-5 discharge piping. There is no method to vent the air from the piping. This results in the pump becoming air bound on the discharge side, decreasing the pump flowrate and increasing the power requirements for the motor.

The TM will allow proper priming of the Warm Water Recirculation Pump discharge piping. This will allow the pump to operate normally and provide the required flow to the intake structure.

The TM will improve the reliability of the pump during normal operation and during a non-accident event by ensuring that all air is removed form the discharge pipe prior to operation.

The installation of the TM does not degrade or prevent actions specified in the FSAR. This TM will actually improve the performance of the pump by increasing the pump flowrate and decreasing the load on the motor.

The installation of the Temporary Modification will not cause a degradation to one or more fission product barriers or result in radiological risk to the general public in excess of the 10CFR100 limits. The Warm Water Recirculation Pump is required to operate during loss of water to the intake structure. Adding a priming system to this pump will not create any new types of accidents.

TM-94-058

SE94-0750

PROVIDE TEMPORARY CONTAINMENT VENTILATION PATH

A containment ventilation path will be provided by attaching a red rubber hose between vent gas collection leadeer drain valves MV-CRW543 and MV-WG533. Installed vent path is plugged. The vent path is from clean waste receiver tank, T-64D, to the exhaust plenum

Safety Evaluation Summary

The path that is plugged is downstream of the containment isolation valves. Automatic Isolation of containment will not be affected.

The hose bypasses the line blockage. The function of the vent path remains the same.

TM-94-060

SE94-0747

INSTALL TEMPORARY FLUSHING TO P-47A AND P-47B DISCHARGE LINE

Operation of the scale inhibitor metering pumps, P47A and P47B indicated that the discharge lines were plugged as no chemical was reaching the Circulating Water System. The most likely cause of this blockage is the hardening of old chemicals within the discharge lines. The hardened chemical can be removed by dissolving it in water. The temporary modification is to install a temporary water source to the discharge lines of P-47A and P-47B to allow the flushing to take place.

The source of the water will be from the seal water supply to main condensor vacuum pump, P-910. Specifically the TM will be attached to the Chicago fitting on MV-SW910 (wye strainer blowdown valve) and connected to the P-47B discharge vent valve, MV-CHM120. With the pump suction valves closed, the pump cross connect valve open and the pump discharge vent valves open, service water will flow out the discharge lines of the P-47 pumps.

No analyzed accidents will be affected by this TM. The P-47 pumps and the P-910 pump are not identified in any analyzed accident scenario or consequences from a previously evaluated accident in the FSAR.

Both the P-47 pumps and the P-910 pumps are non-Safety Related and are not designated as equipment important to safety. The service water is being taken from the non-critical service water header and will not affect the critical service water loads. In addition, the amount of water being used during the flush will be small. The maximum amount of flow that would be used is estimated at 10 gpm.

The flush of the P-47 pumps discharge lines using service water will not cause a degradation to fission product barriers or result in radiological risk to the general public in excess of 10CFR100 limits. No new accident scenarios are created by this flush.

TM-94-086

SE94-1149

DISABLE FIRE PROTECTION DELUGE TRIP OF COOLING TOWER FANS

To allow maintenance work on the Cooling Tower deluge system the nitrogen lines must be depressurized. This would cause pressure switches to sense a fire condition and trip the fans. To prevent loss of fans the affected contact will be jumpered out.

Safety Evaluation Summary

Cooling tower fans tripping on fire is not in any previous analyzed accident in the FSAR. The Cooling Tower fans are not equipment important to safety, therefore, whether they do or do not trip on a fire is not important to safety.

This change affects a FSAR figure.

TM-94-089

SE94-1264
SE94-1263
SE94-1262

TEMPORARILY CAP VENT GAS COLLECTION (VGCH) OFF CHARGING PUMP P-55C

The VGCH has been plugged with resins for several years. This portion of the header is designed to remove non-condensible gases from the charging pumps seal lubrication tank and cylinder block collection chamber. The collected gases are sent through the Rad Waste System for processing. The processing includes discharging through a high-efficiency filter to the suction side of the main vent exhaust fans, diluting with ventilation exhaust air, and discharging through the ventilation stack to the atmosphere. This TM provides for disconnecting the VGCH at the charging pump interfaces (two locations) and installing tube caps at the VGCH/charging pump locations. The VGCH will be flushed via the disconnected tubing locations per a separate work order. Upon completion of the flushing operation, the tubing caps will be removed and the tubing reconnected in its normal design configuration.

Safety Evaluation Summary

The charging pump has operated for past several years with essentially no VGCH in service due to the plugging of the header by resins. With the tube caps installed, the affect on the operation of the charging pump will be essentially the same as what

currently exists with the plugged header, that is, no removal of noncondensible gases from the pump collection chamber or seal lubrication tank. Therefore, the configuration of the VGCH required by this TM will not affect the operability of the charging pumps and is not anticipated to affect the pump operation for the short duration of this TM.

Any radioactive gases produced in the Auxiliary Building by a previously analyzed accident which might be pulled into the open tubing connections due to the slightly negative pressure in the VGCH would be processed through the Rad Waste System prior to being released.

The disconnecting and capping of the VGCH lines do not create scenarios which would increase the release of radioactive materials to the environment for any conceivable equipment malfunctions.

There are no new flow paths which are not processed by the Rad Waste System or new failure positions created by this temporary configuration.

MISCELLANEOUS

FSAR CH. 4
SE94-0010
SE94-0045

CORRECTION OF ASME CODE ADDENDA FOR PALISADES PRESSURIZER

The change to the facility is a change in the ASME Boiler and Pressure Vessel Code Addenda that is recorded as the design code for the Palisades Pressurizer.

Safety Evaluation Summary

During the preparation of the pressurizer temperature nozzle modification package during the 1993 refueling outage (SC-93-087) it was noted that the ASME Code addenda that the pressurizer vendor, ABB-Combustion Engineering, was using as an input to one of the modification calculations was different than the code addenda reported as the design code for the pressurizer in the FSAR. A review of the records for the Palisades pressurizer at ABB-CE revealed that the design code for the pressurizer was ASME Section III, 1965 with Winter 1966 Addenda rather than ASME Section III, 1965 with Winter 1965 Addenda as currently recorded in the FSAR. This fact required that the FSAR be corrected to properly reflect the pressurizer design code. The correction of this information does not constitute an unreviewed Safety Question because it merely corrects the FSAR to properly reflect the actual code to which the pressurizer was designed and does not change the actual physical facility in any way.

FSAR CH. 9
SE94-0033

CHANGE CONCERNING AIR COMPRESSOR COOLING

This FSAR change corrects an erroneous statement in Section 9.1.2.1 regarding cooling of air compressor C-2B. The original C-2B was cooled by service water. This compressor was replaced circa 1988 with a compressor that is air cooled. The FSAR change submitted at that time for the modification neglected to update Section 9.1.2.1 to indicate that the compressor is air cooled. This FSAR change makes this correction.

Safety Evaluation Summary

This change is merely editorial to reflect plant design; the replacement of the air compressor was reviewed and approved under 10CFR50.59 when the modification was performed.

PTC-94-02 Rev 6
SE94-0038

PROGRAM TEMPORARY CHANGE (PTC) TO NUCLEAR TRAINING PROGRAM 1

This program Temporary Change removes the requirement for successful completion of practical factors for the basis radiation worker (BRW) requalification and challenge training courses.

Safety Evaluation Summary

This item removes the requirement for successful completion of Practical Factors for the Requalification and challenge BRW courses. The program as changed does not require Practical Factors unless required by a worker's supervisor. The change is intended to enhance current performance levels of radiation workers in the field. This issue is administrative in nature. The item does not involve any equipment. An FSAR change was initiated to describe the modified requirements (FSAR section 12.2.1.2).

E-PAL-93-038D
SE94-0062

FAILURE OF ELECTRICAL CONTAINMENT PENETRATIONS TO MEET THE DESIGN FEATURES DESCRIBED IN TECH SPECS AND FSAR

As a result of this event corrective action was initiated to implement a modification to seismically qualify the nitrogen purge lines to the containment electrical penetrations in the north and southwest penetration rooms to comply with the plant design basis. Reference Palisades License Event Report 93-014.

Safety Evaluation Summary

The penetration's function of being a containment leakage barrier is not compromised. Only the level of redundance for selected events (ie. seismic or a tornado missile affecting the portion of the nitrogen system outside of SW penetration room) is in question. No loss of redundancy (ie. second barrier) is predicted for events that release radioactive material into containment (ie. LOCA, MSLB with SG tube leakage). The containment electrical penetrations have no effect on initiation of any FSAR described accident.

The selected events described above would only potentially affect the second barrier (ie. Nitrogen purge system). The penetration's first (ie. primary) barrier would not be jeopardized.

There is no advantage gained having two barriers. the single barrier meets IEEE 317 and Reg Guide 1.63. If a seismic event or tornado did occur, the plant would be

safely shut down. The nitrogen purge system would then be repaired if necessary and the affected electrical penetrations would be tested before the plant heated up. The inner barrier would remain intact in seismic or tornado events [only the selective events described in above would potentially affect the second barrier (ie. nitrogen purge system]. Only long term loss of the nitrogen purge would potentially affect the containment electrical penetrations by possible allowing moisture to build up inside the penetrations. Electrical type failures (eg. high currents, short circuits) would affect both barriers.

Only long term loss of the nitrogen purge system would potentially affect the electrical penetration. The consequence of a malfunction of electrical penetrations would not change.

The inner barrier would remain intact in seismic or tornado events [ie. selective events described in Item 1 would potentially affect the second barrier (ie. nitrogen purge system)]. Only long term failure of the nitrogen purge system could potentially cause electrical failure of the electrical penetrations by possible allowing moisture buildup in the penetrations and electrical type failures (eg. high currents, short circuits) would affect both barriers.

FSAR CH. 8

SE94-0132

ADDITIONAL ALARMS FOR LOCAL DG PANELS

Three alarms were not included in the FSAR description of local panels. They are: "Low Lube Oil Level", "Engine Trouble" and "High and Low Fuel Oil Level". This is an FSAR update only, no changes are being made to the facility.

Safety Evaluation Summary

No accidents are caused or mitigated by the inclusion of alarms in the description of the local alarm panel. The additional alarms were always present in the plant. The additional alarms allows better mitigation of the consequences of an EDG malfunction, also, they provide better information for mitigating the effects of malfunctions, but do not create any new failures. The addition of three alarms to the FSAR has no impact on the operability or reliability of the diesel generators.

MHACALC CODE AND CONDOSE CODE BENCHMARKING

The MHACALC Code was written to perform analysis of the Maximum Hypothetical Accident (MHA) using the up-to-date methodology. The code was written to include a radiological release path from post-recirculation dump water leakage to the SIRW Tank after discovery of this path. The previous method of analyzing the MHA was by using complicated LOTUS 123 spreadsheets that performed calculations very conservatively over extremely large time steps. The MHA code allows analysis of the MHA to be performed by simply making changes to an input deck and eliminates some excessive conservatism in calculations by using small time steps. With the exception of the modeling of the SIRW Tank releases, all of the methodology used in the MHACALC code is standard industry accepted practice or has been previously reviewed by the MHA analysis for the Palisades plant. The MHACALC code was also written to provide radiological release rate output for direct input to the CONDOSE code for control room habitability calculation.

The CONDOSE Code was written to perform control room habitability analyses for any radiological event. The CONDOSE code replaces the previously used CRHAB04 code which was found to have several inconsistencies and errors. The CONDOSE code was written very similar to the CONHAB code described in NUREG/CR-5659, with the exception that the CONDOSE code was written to be plant specific for Palisades.

Safety Evaluation Summary

The MHACALC and CONDOSE codes computer programs that will be used for evaluation the consequences of accidents. These codes do not affect plant processes or operations.

The MHACALC and CONDOSE will be used to predict consequences of accidents evaluated in the FSAR. These are analytical tools that incorporate conservative methodology for predicting radiological consequences. As long as proper inputs are appropriately specified, the codes will predict conservative results. Use of these codes will not change plant processes or operation.

Use of the MHACALC and CONDOSE codes will have no effect on equipment or equipment operability in the plant.

FSAR CH. 9

SE94-205

UPDATE OF FSAR TABLE 9-10 - AFW SUPPLY VALVES

This change primarily affects the description of the position of four auxiliary feedwater steam supply valves (CV-0521, CV-0512A, CV-0522A, and CV-0522B) after a loss of instrument air.

Safety Evaluation Summary

The correct position and operation of these valves is described in sections 7.4.1.4.1, 9.5.3.3, and 9.7.2.3 of the FSAR. Also, the operation of the valves can be deduced form P&ID M-205 sheet 2. This update is required to bring the description of the valves contained in Table 9-10 into alignment with the applicable FSAR text. No physical changes are being made to the plant.

C-173

SE94-0219

TECHNICAL REQUIREMENTS FOR THE ANALYSIS AND DESIGN OF PIPE SUPPORTS

The original version of the FSAR did not explicitly describe allowable loads and stresses for pipe supports. Work associated with IE Bulletin 79-14 required the detailed analysis of a large number of pipe supports. Therefore, explicit guidance was needed in project specifications and in the FSAR.

For standard steel shapes, the criteria were straight forward because the FSAR contained similar guidance on major steel structural members. However, hangers contain many vendor catalog components such as rods, clamps, bolts, pins, spring cans, struts etc. No guidance existed for the evaluation of such devices. For cases where vendor catalog allowables were provided on Load Capacity Data sheets (LCDs) for normal operating condition. Some vendors provided LCDs for faulted load conditions--Service Level D or SSE loading condition. Occasionally, the vendor would supply only component ultimate capacities based on test or analysis. It was not always clear as how to use such data.

An 1980 FSAR change provided explicit and implicit guidance on allowable for catalog components. The guidance was as follows:

Explicit Guidance:

1. For vendor catalog items where allowables are not provided, use 0.3 times the ultimate capacity of the components as determined by test or analysis for normal operating conditions.

2. For vendor catalog items where allowables are not provided, use 0.4 times the ultimate capacity of the component as determined by test or analysis for faulted operating conditions.

Implicit Guidance:

1. Use the vendor allowables as provided even though the vendor may not define safety factors for those allowables.

2. Should the vendor supply allowables for normal operating loads only, scaled up those allowables by 1.8 for the faulted operating condition.

The faulted conditions above clearly did not apply for concrete expansion anchor bolts. The 1.8 factor evolved from the ASME Subsection NF criteria ratio of .7Su/.6Su for carbon steel. The above guidance is still used at Palisades per subsequent revisions of C-173 (Q). Contact with vendors over the last several years has made it clear that the vendor will not state that LCD values reflect a safety factor of 5 for normal loads. However, vendors seem very comfortable stating that the safety factor is greater than 4. With a vendor safety factor of 4.5 for normal loads, the 1.8 factor for faulted loads would result in a faulted safety factor of 2.5 (using an allowable of 0.4 times ultimate capacity) and the explicit and implicit criteria would be the same for a component.

The C-173 (Q), Rev 1 change and the FSAR change with regard to vendor catalog items was directed toward clarifying existing practice. It attempted to make the implicit guidance explicit. It sought to use the information provided and to calculate the information not provided in order to provide consistent margin of safety.

<u>Safety Evaluation Summary</u>

The same or a more conservative design approach as stated in the FSAR is designated in the C-173 (Q) revision. The use of the ASME Code NF normal-to-faulted scaled factor for vendor catalog items is in accordance with current industry standards for pipe supports. Several failures postulated in Chapter 14 of the FSAR could be assumed to occur due to the failure of pipe supports.

The requirements stated in C-173 (Q), Rev 1 are the FSAR requirements, current C-173 requirements or current industry standards.

New and different margins are defined in the FSAR and in C-173 (Q), Rev 1 for items which are not included in current FSAR or C-173 (Q) guidance. These new margins may be different from those currently defined in the FSAR because data is not available from the vendor to establish such margins for certain catalog items. Therefore, vendor supplied capacities (and associated implied design margins) are used for such items. Thus, LCD references are used in the FSAR and C-173 (Q) apply. The revised C-173 (Q) and FSAR do not reduce the margins associated with the FSAR because the revised C-173 (Q) requirements or reflect current nuclear industry practice for the design of pipe supports.

JUSTIFICATION FOR CONTINUED OPERATION FOLLOWING REPAIRS TO CKES3166 AND CK-ES3181

The acceptability of continued plant operation following the discovery of defects in check valves CK-ES3166 and CK-ES3181 is being evaluated. These check valves are located between the containment sump and the engineered safeguards pumps. At issue is the integrity of other plant components that could be susceptible to the same type of failure and whether safe operation of the plant could be adversely affected.

A through-wall leak developed on check valve CKES3166. Ultrasonic examination of the sister check valve, CK-ES3181, indicated linear and axial indications but no through-wall leaks. Subsequent investigation attributed the defect on CV-ES3166 to intergranular corrosion caused by weld repairs performed on the valves during plant construction. Contributing factors were a high carbon contract and a defect in the inside diameter of the valve wall that could concentrate boric acid in a crevice.

During plant construction, the valves were installed and then cut out due to seat leakage. The valve weld ends were repair welded and then welded back in place. It is believed that welding during original construction sensitized the butt weld area of the welds, increasing susceptibility in this area to intergranular corrosion. While all stainless steel casting material is potentially susceptible to intergranular corrosion the rate of corrosion in these particular check valves was accelerated due to the reasons previously stated. The two check valves were subsequently repaired using a weld overlay technique described in ASME Code Case N-504-1, which was approved for use at Palisades by the NRC.

Intergranular corrosion is a mechanism that does not lead to a catastrophic piping failure. It attacks portions of the grain boundaries with the majority of the grain boundaries being left unaffected. The structural integrity of the pressure boundary is not significantly affected and the pressure boundary is capable of withstanding any design transients. Moreover, intergranular corrosion occurs slowly, so any through wall leaking would be readily apparent and repaired in a timely manner. It took about 25 years after the weld repairs of the check valves for the intergranular corrosion to cause through wall leakage.

There are several factors that provide assurance that the structural integrity of other plant components is not in doubt and that safe operation of the plant will not be compromised.

A VT-2 leak inspection has been completed on the 22 cast stainless steel valves that are located on the suction side of the HPSI, LPSI, and containment spray pumps in both east and west engineered safeguards rooms. No leaks were observed. A repair weld was observed on MO-3199 (shutdown cooling to LPSI pump P-67A) but no indications of past or present leakage was observed.

A document search did not uncover any other welded repairs to other valves CK-ES3166 and CK-ES3181 received excessive hear input prior to plant service and the majority of similarly susceptible valves would not have received this extensive sensitization.

Normal plant walkdowns and scheduled inservice pressure tests would identify leaks long before operability of equipment could be affected.

Safety Evaluation Summary

Intergranular corrosion is a mechanism that does not lead to catastrophic failure. The structural integrity of affected components is not compromised. Moreover, no leaks are expected because inspection of 20 other cast stainless steel valves did not reveal other leaks and evidence of weld repair was found on only one of these valves. In the highly unlikely event that a leak did occur due to intergranular corrosion, the leak would be visually detectable so repairs could be performed.

COLR

SE94-0266

CORE OPERATING LIMITS REPORT (COLR)

Limits and parameters that are related to fuel reloads are to be removed from the Technical Specifications and maintained in the core operating limits reports (COLR). The limits relocated to the COLR are developed using NRC-approved methodology, and the approved methodology documents are to be listed in the administrative reporting requirements section of the Technical Specifications. Therefore, calculational methodology and acceptance criteria are referenced in the Technical Specifications, but not the specific limits. This will permit specific values to be changed without prior NRC review and approval, provided they are developed in accordance with the specified methodologies. Requirements to maintain the plant within the appropriate bounds is to be retained in the Technical Specifications, with the specific values contained in the COLR. The COLR will be submitted to the NRC with each change.

Safety Evaluation Summary

Revision 0 of the COLR contains the same values and parameters that currently exist in the Technical Specifications. The wording has been copied directly from the Technical Specifications with no changes, with the exception of correcting a few typographical errors. With all limits and parameters being identical to current Technical Specifications, no changes in plant operation or allowable operations are being made. The current Technical Specifications, and hence, the values for the limits placed in the COLR, have been approved by the NRC. The safety consideration for these limits was previously addressed and approved by the NRC.

FSAR CH. 9

SE94-0280

CCW HEAT EXCHANGER OPERATION

Clarified existing wording regarding operation of the plant with only one CCW heat exchanger in service. The existing wording indicates that anytime a single CCW heat exchanger is in service it will always be subjected to high CCW flow rates. The new wording clarifies the original change's intent which was to describe the need for two CCW heat exchanger's in service when the plant is above Cold Shutdown.

Safety Evaluation Summary

Clarifying the design limitations of the CCW heat exchangers will help ensure that they are operated properly.

The change does not lessen the requirement for two CCW heat exchanger's being in service anytime the plant in operating above Cold Shutdown.

E-PAL-94-006

SE94-0288

PLANT OPERATION WITH CV-0913 AND CV-0950 IN THE OPEN POSITION

Justify plant operation with CCW valves CV-0950 and CV-0913 in the open position to provide continuous cooling to the engineered safeguards pumps. Operating with these valves open is desirable or to a single active failure that could isolate all cooling to the safeguards pumps during a LOCA.

These CCW valves are normally in a closed position and fail open on a loss of air. They open automatically on a safety injection signal (SIS) with a seal-in feature. The HPSI and containment spray pumps require cooling following a LOCA, when hot water is being recirculated through the pumps from the containment pump. Normally service water is available as a backup. The service water isolation valves are normally closed and require instrument air to open.

Operating with the CCW valves open, with no compensatory action, increases the probability of a cross-tie between the CCW and service water systems that could either drain the CCW to the lake or cause a flood in the Auxiliary Building.

Safety Evaluation Summary

The position of the valves has no effect on the probability of an accident previously evaluated in the FSAR.

The position of these valves has no effect on any radiological barriers. There is no effect on the severity of an accident or on its consequences.

Operation with the CCW valves open precludes the single failure scenario in which all cooling is lost to the safeguards pumps following a LOCA.

Since the PRA concluded that operator error is the major contributor to an inadvertent cross-tie between the CCW and service water systems, specific actions are being taken to prevent an operator error that opens the service water valves. These actions are hanging caution tags or place cards in the control room, notifying the operators on the need to prevent a cross-tie, and revising operations and surveillance procedures to assure that a cross-tie does not occur.

The effect of the diversion of CCW flow on the CCW system flow balance due to the CCW valves being left open is insignificant.

This proposal to operate with the CCW valves open is no different than the configuration that presently exists during routine safeguards pumps surveillance testing. These tests, which are performed at power operation, open the CCW valves in order to supply cooling to the pumps.

EA-WEN-94-01
EA-PAH-94-02
EA-C-PAL-94-0016A-01

SE94-0337
SE94-0338
SE94-0908
SE94-1546

CONTAINMENT FLOOD ANALYSIS

Following a LOCA, a large volume of water will be pumped into the containment building and the reactor vessel by the Safety Injection (SI) System in order to keep the core cool and containment pressure within its design value. A majority of this SI water will find its way to the Containment Building sump, where S.I. suction switches over to the sump from the SIRW tank on low tank level.

In the current analysis, values were checked and verified, and a walkdown of the containment 590' level was performed. Based on analysis the flood level is now set at its maximum level of 6' 11.5" to allow for a system leakrate of 4.6 gpm if necessary. A calculation was also done to incorporate the TSP baskets for pH control when they are installed during the 1995 refueling outage.

The environmentally qualified (EQ) equipment affected by the flood level is documented. EQ equipment predicted to be submerged are VOP-3007, VOP-3008

(cable only), VOP-3009, VOP-3010, VOP-3011, VOP-3013, VOP-3062, VOP-3064, VOP-3066, VOP-3068, SV-0861 (cable only), SV-0862 (cable only), and SV-0873 (cable only). These are the VOPs for HPSI and redundant HPSI injection header MOVs, VOPs for two of the LPSI injection header MOVs, the SVs that control air to the VHX-1 inlet and outlet service water valves and the SV that controls air to the VHX-3 outlet service water valve. Also, VOP-3012 and VOP-3014 will be submerged.

Safety Evaluation Summary

The containment building predicted flood level, and corresponding equipment submergence, result from a design basis accident (DBA).

The DBA that results in the maximum predicted containment flood level is the Loss-Of-Coolant Accident (LOCA). The reason a LOCA would result in the highest containment flood level is the addition of primary coolant system volume plus the contents of the SIRW Tank, and the Safety Injection Tanks. The containment flood elevation following a LOCA is predicted to submerge EQ equipment and cable that has not been qualified for submergence. The HPSI, RHPSI and LPSI injection header valves would be in the open position prior to VOP submergence since the valves open upon SIS (for the operable channel or channels) and the predicted flood level would not occur until after recirculation begins. Submergence of the energized VOPs could cause short circuiting and opening of the corresponding breakers. If this occurred, the VOPs would be electrically isolated to render the valves inoperable in the open position (for the operable SIS channel or channels). This situation has been previously addressed for VOP-3012 and VOP-3014 since both of these would have been submerged at the previously predicted maximum flood elevation.

With the HPSI, RHPSI and LPSI injection header MOVs in the open position, normal safety injection functioning to cover and cool the core following a LOCA would not be hindered. Hot leg injection for a large break LOCA would also not be hindered since the hot leg injection VOPs are not submerged. The ability to throttle SI flow with the injection header MOVs would be lost. However, the HPSI pumps could simply be cycled on and off once throttling criteria are met if pressure/temperature limits for cooldown are a concern. Also, LTOP would be in service at this point in the event to provide protection against pressure/temperature limit concerns.

At some point following the LOCA, shutdown cooling may be initiated. To start SDC, the HPSI injection header MOVs can remain in the open position after the HPSI pumps are shut off. Since the LPSI injection header MOVs would be open, SDC initiation would not be hindered. The ability to take PASM samples using a LPSI pump would not be hindered either. If LPSI throttling is required due to pump NPSH concerns, CV-3006 and CV-3025 could be throttled. Air is required for operation of CV-3006 and CV-3025, but by the time SDC could be initiated operators would have time to recover instrument air and Y01 power if lost during the event. Once through cooling could be continued until they are recovered if needed.

The SVs that would be submerged could result in loss of air to the corresponding control valves, if air is available to begin with. This would not be of a concern since the control valves are air to close and would fail in the open position, allowing service water flow through those air coolers for containment cooling. If the containment flood level is increasing due to service water leakage, step 7 of CA continuing actions in EOP 9.0 Rev. 5 directs operators to close the containment service water supply and return if the containment water level reaches 595' 0" and the air coolers are not needed. (This is 23.5" below the maximum predicted containment flood level, and the air coolers would likely still be needed at this time.) The containment service water supply and return valves, CV-0824 & CV-0847, also have backup nitrogen supply alleviating a concern over possible loss of instrument air. If service water to and from containment is isolated, operation of the service water valves for the individual air coolers would not be required. The TSC/EOF would be consulted at this point in an event if the containment air coolers are still needed for containment atmosphere control.

The increased containment flood level also submerges additional tubes in the bottom coils of the containment air coolers (CACs). The containment air coolers are identified as partially flooded following a LOCA in EA-D-PAL-93-272E-03, which assumed a flood height of 596' 11.5". The bottom two coils, out of eight coils, in each CAC may become close to completely submerged due to the increased flooding since their highest tube row is at 597'-1½". Per EA-D-PAL-93-272E-03, the bottom two coils are assumed to be completely submerged. The relative importance of CACs for heat removal following a LOCA was previously demonstrated in EA-D-PAL-93-110-04, which showed that a heat removal capacity equivalent to less than one CAC is required to remain within EQ temperature limits if a containment spray pump is operating. Although the LOCA containment analysis has been completely revised, the relative importance of CAC heat removal demonstrated in EA-D-PAL-93-110-04 is still valid. Therefore, the increased flooding in the containment air coolers would not challenge containment pressure/temperature limitations.

The EQ equipment affected by the increased containment flood level would fail, if failure occurs, in a safe condition by the time that they become submerged. It was also justified above that further operation of the equipment would not be required if the failure occurred. By the time a failure could occur, the failure would be in a safe position and accident mitigation could be safely performed without any further operation of the equipment. Since the failure of this EQ equipment is in a safe (open) position, neither safety injection flow nor containment cooling capability will be affected.

If the VOPs or SVs fail due to submergence, the associated valves will be in the safe (open) position for accident mitigation. Further repositioning of the valves associated with the subject VOPs and SVs would not be required to mitigate the accident or initiate shutdown cooling.

Containment flooding and corresponding VOP and SV submergence are the result of an accident. These VOPs and SVs could only be submerged if a DBA occurred. The worst possible accident that could occur has already been evaluated in the FSAR.

Submergence of EQ equipment in containment has been previously addressed for VOP-3012. Submergence of the subject VOPs and SVs due to the increased containment flood level could only result in failure of the subject valves to a safe position for mitigation of the accident in progress. This would not affect any other safety related equipment since the valves would fail in a safe position.

EA-FC-864-02 Rev 2

SE94-0420

10CRF72.48 SAFETY REVIEW DESIGN CHANGES TO VSC

The changes provided in the design change documents, i.e., DCNs, NCRs, etc., were reviewed and grouped into component categories to perform a evaluation. The Engineering Analysis provides a detailed evaluation of the changes by component category and identifies the source design change document.

SNC letter to the NRC dated November 25, 1991 identified changes between the generic design shown in the approved SAR and the drawings and specifications to be used for the fabrication of CPCo Ventilated Storage Casks.

NCRs are included under changes to reflect nonconformances with drawing requirements which occurred during fabrication. The nonconformances which are design changes were grouped and evaluated with the DCNs.

Safety Evaluation Summary

The weld prep detail for the joint between the MSB bottom plate and shell was changed to 60 degrees groove corner joint weld to provide the proper weld.

The weld change does not reduce the integrity of the MSB because the weld is full thickness of the shell, full penetration and requires 100% radiography examination.

Changes were made to enhance quality of MSB and to facilitate fabrication and installation of the Shield lid as indicated in the Engineering Analysis. The function of the shield lid is to provide confinement boundary and radiation shield. The changes addressed in this evaluation do not affect the design basis function of the shield lid. Two fuel support carbon steel plates, 1" x 3.5" x 8.80" were added to the bottom of the fuel storage sleeves to make the fuel sit straight in the storage sleeve. The Palisades fuel has two alignment pins not like other PWR fuel. This change was required to bring Palisades specific condition to SNC's generic design which was done with two flat support feet. The changes do not affect the integrity of the storage sleeve.

The out of tolerance conditions in overall VCC height by 0.9", VCC outside diameter by 0.3", shield ring outside diameter by 0.0875" are insignificant and do not affect the VCC integrity and shielding.

The revised air content range of 3 to 6 % complies with ACI-349 which is specified in Subsection 1.1 of the SAR but does not meet the SAR Table 1.2-5 requirement. Air content is not a significant parameter for VCC as long as the required concrete strength and density are satisfied. Average concrete density exceeded the required 145 PCF. Some loads showed tests with the density of 143.96 PCF which is 0.7% less than the required density. All concrete loads had compressive strength exceeded 5000 psi. The essential design feature of the VCC, per Section 12.2.4 of the SAR, is concrete density and concrete strength which have been maintained.

The material specification for the cask lid bolt was changed from ASTM A-325 to ASME SA-320, B8, Class 2. The change in material results in a yield strength reduction of 13%. Since there is sufficient margin in the SAR design to cover the reduction in yield strength, the change does not impact the intended function of the bolts.

Holes for lifting bolts were added to provide for rigging used to lift and remove the MTC lid. Added holes in the MTC lid does not affect the function of the MTC lid. The change facilitates the loading process. The function of the MTC lid and holddown bolts is to prevent inadvertent lifting of the MSB out of the MTC.

The MTC wall design was changed to provide for the use of lead bricks instead of poured lead as described in Subsection 5.2.3.2 of EA-FC-864-02. The change resulted in the removal of the middle shell. The lead and RX-277 regions were slightly increased to compensate for the removal of the middle shell.

The following changes were made to the cylinder strut and bracket: (1) revised hydraulic cylinder assembly and bottom strut detail, (2) changed bolt and nut material for hydro-line brackets, (3) nuts for securing brackets to the MTC outer shell are secured with a tack weld instead of a 1/4" fillet weld, (4) added Note 7 to permit use of equivalent material for angle block attachment to MTC outer shell.

The changes to the hydraulic cylinder assembly and the associated components maintain or improve the integrity and performance of the system. Tack welds are used to secure nuts during the construction process and are not required to serve any other function after the bolts are torqued. The tack welds were not bearing or pressure retaining elements of the MTC. Substitution of alternate materials/specifications for the angle block was permitted only when the replacement material was equal to or better than that originally specified.

Radius of shim rings was changed to 31.75" at outside face of shim. Material specification for shims and shim flanges was changed to ASTM A240 Type 304. Configuration of the weld between shim and flange was changed from square groove to bevel groove.

Changes made to the shim rings and flanges will maintain or improve the integrity and functionality of the interface between the MTC and MSB. The change which calls out the radius of the shim ring at the outside face provides a more precise approach for controlling the fabrication of the shim rings. The change of material specification to A240, Type 304 reflects the use of stainless steel which is an enhancement for decontamination purposes. The change to a bevel groove weld provides an enhanced weld configuration.

The trunnion was changed to a solid casting (A350 Gr. LF2) from a rolled pipe filled with lead and RX-277. The requirement for Charpy V-notch test value of 15 ft/lbs at °F was added to comply with NRC requirements. The trunnion length was changed to a maximum dimension of 14.96" from 15.1" The MTC trunnions were designed to be compatible with the Palisades fuel handling crane and met applicable requirements of NUREG 0612 for lifting heavy loads. Changing the trunnions to a solid casting will reduce stresses in the trunnion and will have no impact on stresses in the MTC wall.

The MTC shield door was fabricated 0.2" wider than permitted on the drawings. The height of the rail shield was found to be 0.12" greater than specified with tolerance. The rail assembly lower plate measures 0.07" wider than specified. The shield door overlap was measured to be 0.02" higher than specified. The inside diameter of the MTC inner shell was slightly larger than specified.

The change in shield door width was acceptable because the hydraulic door operating assembly could easily be adjusted to accommodate the slight increase in width. The small increase in rail shield height and the lower plate width were both acceptable because they do not impact shielding or operation of the shield door assembly. The small increase in the height of the shield door overlap was acceptable because the change did not impact the shielding function or the operation of the shield door assembly. The small increase in the inside diameter of the inner shell was acceptable because it did not impact the interface with the MSB because shims provide adequate flexibility to accommodate the increased gap between the MSB and MTC. The increase in diameter of the inner shell is negligible and there is negligible impact on shielding and strength of the MTC wall.

There are a number of site specific changes which were necessary in order to use the yoke design with the Fuel Pool Crane. The drawings, showed two different types of yokes which could be used at a plant site. The yoke pin material specification was clarified and the pin location, hook configuration and pin length were changed. The pin diameter was increased and a chamfer was added to the yoke hook. The design basis function of the yoke is to lift the MTC using the plant crane hook. The yoke hook and pin configuration were confirmed in stress calculation EA-FC-864-06. The calculation shows that the yoke meets all of the factor of safety requirements set by ANSI N14.6.

EA-SDW-94-001

SE94-0476

GENERATION OF AVAILABLE T-81 INVENTORY DATA ASSUMING GRAVITY FEED AND VARIOUS LEVELS IN T-2 AND T-81

An enhanced version of the RETRAN model developed in EA-E-PAL-94-019 to determine the quantity of water that could be transferred via gravity feed from T-81 to T-2. The 3" flowpath through CV-2008 and CV-2010 is used. Various combinations of initial tank levels are assumed. The results of this analysis will be used to generate a graph in the "PCS Cooldown Strategy" attachment in EOPs 2.0 through 9.0.

Safety Evaluation Summary

Nothing physical is being directly changed by this Item.

The restrictions on available condensate inventory calculated in the item have always existed at Palisades - nothing is changed from that aspect. The results of the analysis help ensured that operators do not overestimate the available condensate inventory and wind up in a situation where water inventory is depleted.

The ultimate backup for AFW suction supply, the Fire Protection and Service Water Systems, continue to be available. The procedural guidance on using these backups is sufficient to ensure their availability if required.

FSAR CH. 7

SE94-0477

CONTROL ROOM FIRE DESCRIPTION

The FSAR change distinguishes the control room proper from the north office and viewing gallery which have different types of ceilings. The control room proper has an open type ceiling and thus overhead cables for telephone, etc. are not considered concealed. The north office and viewing area have solid ceilings and cables above these are considered concealed.

Safety Evaluation Summary

The type of ceiling in the control room has no bearing on any accident or accident consequences.

The differences in ceiling types will not directly affect any safety related equipment. The concern is for fire protection purposes and the cables in question (even in the concealed ceiling) are low energy communications cables; thus of no risk as a source of fire.

This FSAR change only clarifies the limits of the description in the FSAR. Areas excluded by this clarification are separated from the control room proper by one hour fire walls. And the cables in question are low energy. The results of a control room fire are not changed by the reduced area of open ceiling material.

FSAR CH. 11

SE94-0547

FSAR CHAPTER 11 CHANGE REQUEST

Revise FSAR Section 11.6.6.7 Facility Contamination Control requirement for Portal Monitor use upon exit from the restricted area in the security building. This is to allow the portal monitors to be bypassed during SEP activations and drill to speed the egress of personnel form the plant protected area to their assembly areas and complete accountability with 30 minutes of SEP emergency classification initiation.

Safety Evaluation Summary

This item is one element of the Facility Contamination Control Portal, and is administrative in nature. This program relies on the PCM-16s in Access Control, and other locations, along with routine frisks and surveys of occupied areas to control contamination at its source. The PM-7s in the security building do not meet the same sensitivity requirements as the PCM-16s, and their function, as described, is as a final check in the event some element of the Contamination Control Program failed. Bypassing the PM-7s during SEP Assembly and Accountability would not significantly increase the probability that a contaminated person left site because there would be other indicators. These other indicators include habitability survey of the Security Building and Assembly Areas (including contamination surveys). During real SEP actuation, all personnel are required to be monitored prior to leaving site. During drills and exercises, most personnel would enter the restricted area and would be subsequently monitored upon their next exit. The few people who may not enter the restricted area after a practice drill for accountability are not likely to be contamianted because of the control program.

The change does not affect any equipment important to safety.

FSAR CH. 8

SE94-0568

DC SYSTEM COOLING FANS

Facility change 573 did not provide safety grade fans (although fans were seismically mounted) in the inverter and battery charger cabinets and justified not installing fans in the auxiliary feedwater cabinets. Therefore, a conflict exists between the FSAR statement and the as-built configuration of the plant. The purpose of this safety analysis is to justify acceptability of the as-built design.

Safety Evaluation Summary

Facility Change 573 was initiated to install the fans in the inverter, battery charger and auxiliary feedwater cabinets. As part of the evaluation for this FC, it was concluded that fans were not required in the auxiliary feedwater junction boxes, and that the cooling fans for the inverter and battery charger cabinets were an enhancement only and were not required for safety. These conclusions were reached based on the following:

> The auxiliary feedwater junction boxes do not contain temperature sensitive components which would be affected by ambient air temperature in the cable spreading room being at the design value of 104°F.
>
> Equipment Specification 5935-E-11 for the inverters and battery chargers specified that these devices operate at a design ambient temperature of 104°F using only natural circulation cooling. Testing of the inverters and chargers by the manufacturer were required to be performed to verify that the internal temperatures of these devices were not exceeded.
>
> Isolation of the non-safety related fans installed by this FC is provided by fuses in the inverter. There are no safety related loads connected to this breaker. These fuses were installed by Specification Change SC-94-042.

Loss of HVAC testing confirmed that the cable spreading room temperature would not exceed the design temperature, it was concluded that no additional forced-air cooling for these components was required. It was decided, however, to add non-safety related fans to the inverter and chargers to provide more temperature margins. These fans were seismically mounted to prevent them from becoming missiles and potentially doing internal damage to the cabinets.

FSAR CH. 8

SE94-0623

UPDATE UFSAR SUT 1-2 AND SGT 1-1 RESERVE CAPACITY AND REMOVE DISCUSSION OF SPT 1-2 LOAD SHED

This updates the transformer reserve capacity due to load shed and removes improperly implied use of Station Power Transformer, SPT, 1-2.

Safety Evaluation Summary

The change in Startup Transformer, SUT 1-2 and Safeguards Transformer, SGT 1-1 reserve capacity does not affect the transformers capability for supplying loads is (reserve capacity is a positive engineering margin). The administrative controls which limit SUT 1-2 loads based on grid voltage assure recovery of bus voltage above the second-level under voltage relay settings. The implied role of SPT-1-2 (immediate

power for accident mitigation) is inconsistent with other FSAR sections and the actual design. Removing the SPT 1-1 reserve capacity due to load shed does not affect SUT 1-2 and SGT 1-1.

The change is related to adequacy of offsite power supplies, for which a positive engineering margin exists. Administrative controls also are used to assure adequate in-plant voltages when supplied from SUT 1-2.

D-PAL-94-171

SE94-0630

SERVICE WATER PIPING (HB-23) CV-0823 AND CV-0826

SFC 82-155 replaced cast steel valves with cast iron valves. The change in material is justified in a pipe stress calculation. There are no procedures, tests or experiments affected. The function of the system remains unchanged.

Safety Evaluation Summary

There are no accidents involving the service water system, valves CV-0823 and CV-0826 or the component cooling water heat exchangers.

The valves were replaced due to valve failure. The function of the valves are the same as the original valves. The valves are rated for the same temperature and pressure as the original valves. The valve material used has been justified in a pipe stress calculation. The valves have the same flow rates.

FSAR Table 9-2 states that all valves in the service water system are to be cast steel. In 1982, SFC 82-155 replaced the valves the cast iron valves. The valves have been identified to be cast iron and found acceptable. The valves are in the process of being replaced back to cast steel valves as part of the service water replacement project.

FSAR CH. 7

SE94-0639

FSAR CONFLICTS WITH ACTUAL PLANT CONFIGURATION FOR RPS

This change removes the paragraph from 7.2.9.2 of the FSAR that states that the start-up rate of change signal is fed to the Plant Information Processor (PIP).

Safety Evaluation Summary

The lack of start-up rate signal to the PIP does not remove any operator information of automatic actuations from the plant instrumentation. High rate signals are annunciated and displayed directly from the source/wide range instruments and do not require the operation of the PIP for response either manually or automatically.

The information that would be provided to the PIP is available elsewhere. The PIP provided no alarming or response function to high rate signals. The effect on plant operations is insignificant.

FSAR CH. 8

SE94-0643

ADD DISCUSSION OF AC BUSES ON THE BYPASS REGULATOR

The sequencers have a 10 second delay on start up, operation of preferred AC buses on the bypass will introduce an unanalysed time delay on loss of offsite power due to sequencer start up. Present accident analysis do not allow for any extra time delays in safety equipment actuations. Therefore, preferred AC buses are considered inoperable when on bypass.

Safety Evaluation Summary

The preferred AC buses are required for the response to an accident.

The Technical Specifications allow an 8 hour LCO for the preferred AC buses. The AC buses have never operated on bypass longer than the allowed LCO.

The restriction on operability is a result of timing problems.

The extra time injected by the sequencers is not analyzed, but is avoided by the inoperability determination when on the bypass regulator.

EA-D-PAL-93-272E-03

SE94-0648

LOCA CONTAINMENT RESPONSE ANALYSIS WITH DEGRADED HEAT REMOVAL SYSTEMS USING THE CONTEMPT EI-28A COMPUTER CODE

Deviation Report, D-PAL-93-272, was written to address service water system concerns at Palisades. Part of the response involved determining SWS flows under degraded conditions. This engineering analysis used the previous LOCA analysis of record and incorporated degraded service water flow or well as degraded CCW flow, higher SW temperature, and a higher containment flood level. The containment peak pressure and temperature are unchanged from the previous LOCA containment

response analysis. The overall containment temperature responses continues to be bounded by the environmental qualification (EQ) temperature curve. The initial part of the EQ curve is always less than the containment temperature response. This is caused by physical limitation in EQ test chambers (i.e. they cannot ramp temperature and pressure as fast as a LBLOCA). In EQ analysis, this initial part of the curve is accepted based on thermal lag arguments, passive equipment response considerations, or as an artifact of the testing process.

Safety Evaluation Summary

This item is a reanalysis of a previously analyzed accident. Nothing physical or procedural is altered by this item.

The containment temperature response is below the EQ temperature curve. The peak temperature and pressure are unchanged from the previous LOCA containment analysis. Containment integrity is assured. Equipment important to safety is designed and rated for service the harsh environment in containment. Additionally, the containment design pressure is not challenged by the LOCA.

This analysis used the worst single failure of an active component (failure of right or left channel of SIS). The results of this analysis are acceptable.

FSAR CH 5&6

SE-94-0668

FSAR CHANGE TO ADD DESCRIPTION OF PENETRATION 65 COMPLIANCE TO GDC 56

This FSAR change request makes two additions to the FSAR. One, it adds a description of the NRC's acceptance of the design of Penetration 65 (instrument air to containment). This penetration does not explicitly comply with General Design Criteria 56. This issue was evaluated by the NRC under SEP Topic VI-4 and the NRC concluded that, although the penetration does not explicitly comply with the valve configuration specified in the GDC, the penetration does not have to be backfit because, in part, a probabilistic risk study concluded that the probability of penetration failure with two valves outside containment was the same as having one valve inside containment and one valve outside.

The isolation capability of Penetration 65 was also reviewed by the NRC under NUREG 0737 Topic II.E.4.2 and was found to be acceptable. The review concluded that isolation is provided in a way that is equivalent to automatic isolation on a containment isolation signal.

The second addition being made to the FSAR is a justification of the classification of Penetration 65 as a Class C2 but no justification is provided. Such a justification

should be added to the FSAR because the basis for the classification of this penetration as Class C2 may not be readily apparent.

Safety Evaluation Summary

The change has no effect on any plant equipment or on any accident analyses.

This FSAR change is administrative and has no effect on plant equipment. It merely adds license basis information to the FSAR with regard to GDC 56 and adds justification for the classification of a penetration.

FSAR CH. 11

SE94-0672

FSAR CHANGE REQUEST - RADIOACTIVE WASTE MANAGEMENT AND RADIATION PROTECTION

Corrections and clarifications of numerous descrepancies in Chapter 11.

Safety Evaluation Summary

The plant discontinued use of both dirty waste and fuel pool filters because of ALARA concerns. In spite of modifications, the filters could not be changed without high dose pickup. The dirty waste filters and are not needed after the modification. The fuel pool system was evaluated and it was determined that the crud from systems would not adversely affect the operation of the Fuel Pool Demineralizer (increased resin use or channeling) and the resin system is better designed to handle high levels of radiation. The filter elements were removed and the filter has not been used for many years.

The Laundry system onsite has not been used since 1989 except for minor volumes of building decon fluids. There is no longer a special Technical Specification for laundry effluents. The minor amounts of building and equipment decontamination and cleaning are greatly diluted and processed in the miscellaneous waste system.

Criteria was added for a new storage building outside protected area to agree with original Zone I criteria.

Added a reference to the decontamination unit which is installed in North Fuel Pool area. The unit does not increase radioactive effluents. Significant radioactive material from Big Rock Point or other Consumers Power licensees will be returned to originator.

Other miscellaneous changes were made to clarify or correct information in the FSAR.

FSAR CH. 4
SE94-0695

EDITORIAL CHANGE OF FSAR SECTION 4.5.6

FSAR Section 4.5.6 is being revised to clarify what is meant by relatively frequent periodic intervals in 4.5.6(1). The section is being revised to show that interval scheduling is in accordance with ASME B&PV Code, Section XI.

Safety Evaluation Summary

This change clarifies the fact that ASME B&PV Code, Section XI controls the interval examination requirements for ASME class 1, 2, and 3 components. The ISI program for each ten year interval is updated and submitted to the NRC for their review and approval as identified in FSAR 6.9. Scheduling is in accordance with ASME B&PV Code, Section XI as required by T.S. 4.0.5 and FSAR 6.9.1.

SOP 3
SE94-0724

SAFETY INJECTION AND SHUTDOWN COOLING SYSTEM

Change the minimum flow value of the HPSI pump from ≥ 30 gpm to ≥ 25 gpm. This change is necessary because the orifice is sized for 30 gpm. With the present limit of ≥ 30 gpm, no allowance is given for minor fluctuations in flow from the designed value. After discussion with the vendor and engineering data taken during performance of surveillance testing it was determined that a lower minimum flow would be acceptable. If flow is below 28 gpm additional monitoring of the pump will be necessary to ensure no damage will result to the pump. It is conservative to monitor the pump and provide limits when the pump should be secured. It should be noted that the orifice size is the same size as it always has been. Therefore, historically the pump has been running with the same minimum flow with no detrimental damage to the pump.

Safety Evaluation Summary

The probability of an accident previously evaluated in the FSAR is not increased. The orifice used in the pump is the same size as in the past. The orifice size is designed to pass a nominal 30 gpm. The FSAR lists in Table 6-3 that the minimum flow for the HPSI pump is 30 gpm. No allowance is given for pump degradation or instrument inaccuracies. This change allows for a minimum valve lower than listed in the FSAR. The system engineer consulted with the manufacturer. It was determined that a lower flow rate would be acceptable. The procedure was modified to allow a lower flow to 25 gpm. If flow drops to less than 28 gpm in non-accident conditions then operations will monitor the equipment for abnormal operations and secure it if required.

FSAR CH. 7

SE94-0729

INCORES

The change identifies the proper number of qualified incores, describe properly the use of incore data in relation to ex-cores, and show heat rate as continuously monitored.

Safety Evaluation Summary

The incores provide no direct input to safety functions. The use of incore data for calibrating safety systems is not a change in operations and is supplemented by heat balance calculations. The validity of ex-core calibrations remains unchanged.

The change to the number of core exit thermocouples does not represent added equipment but rather environmental qualification applied to existing equipment.

The use of incore data for calibration input to the ex-cores reflects is part of a calibration process that is unchanged.

EA-A-PAL-93-036-01

SE94-0815

REACTOR VESSEL SURVEILLANCE CAPSULE W-110 RESULTS AND INCORPORATION OF RESULTS INTO STATUS [REACTOR SURVEILLANCE PROGRAM]

Surveillance capsule W-110 was removed from the reactor vessel on June 21, 1993. The 10CFR50 Appendix H required report of test results was submitted to the NRC on June 21, 1994. This engineering analysis supports the information supplied in that submittal. Section 4.5.3 of the FSAR describing the reactor vessel surveillance program required a revision to incorporate the information.

Safety Evaluation Summary

The information in this engineering analysis is required in the FSAR. The results of the analysis show that the current position on the docket with regard to pressurized thermal shock need not be revised as a result of the described tests. The status of the Palisades reactor vessel integrity remains unchanged.

EA-SP-20034-01

SE94-0835

REVIEW OF THE SAFETY RELATED PIPING REVERIFICATION PROGRAMS SRPRP PIPING BOUNDARIES

A review of all large bore piping was performed to ensure all safety related piping (Reg Guide 1.26) was evaluated in the SRPRP program. The analysis resulted in some non-safety piping being removed from the SRPRP scope.

Safety Evaluation Summary

The calculation is a review to ensure all large bore (2-1/2" in diameter and larger) safety related piping was included in the SRPRP. As result of this review several items emerged: Nonclass piping was removed from SRPRP and piping instrument diagrams were revised to eliminate out of date and unused class and seismic symbols.

No physical changes to piping and supports will be made. Existing FSAR compliance for safety related piping will be maintained.

FSAR CH. 4

SE94-0850

REVISE FSAR TABLE 4-23

This FSAR change changes footnote (a) to refer to FSAR figure 4-11. Another note is added to remind anyone using this table that deviating from the surveillance capsule removal schedule as given in the FSAR table requires NRC approval. This NRC approval requirement comes from 10CFR50 Appendix H section II.B.3.

Safety Evaluation Summary

The change involves no physical changes to the plant equipment. It corrects information and specifies the requirements for surveillance capsule removal.

FSAR CH. 8

SE94-0870

UPDATE FSAR TO INCLUDE "AS-LICENSED" ELECTRICAL DESIGN OF FUEL OIL TRANSFER SYSTEM

Enhance the description in the FSAR for electrical "as-licensed" design of the fuel oil transfer system, to show the existing separation of transfer pumps.

Safety Evaluation Summary

No aspect of the fuel oil transfer system electrical or controls design can initiate an accident- either directly or indirectly.

Redundant means of fuel transfer via the transfer pumps does not affect day tank makeup. Also unaffected is the capability of re-filling the day tanks via oil tank trucks.

Capability to transfer fuel or refill day tanks from a truck is not affected. Therefore, the emergency power supplies are not affected.

The electrical design of the fuel oil transfer system is not related to any safety margins. The FSAR update is written to capture the licensing basis with respect to the electrical design.

FSAR CH. 9

SE94-0880

TESTING AND INSPECTION

Clarification of "Duplicate Components" in FSAR Section 9.10.5.

Safety Evaluation Summary

The FSAR Section 9.10.5 Rev 15 uses the term "Duplicate Components" which has caused confusion, a review of the FSAR was conducted to verify the term "Duplicate" did not have a special meaning within this section of the FSAR, or any other section of the FSAR. This was done in response to the evaluation of a Design Basis Document Discrepancy Report. An editorial change is being made to the FSAR to minimize confusion.

EA-FC-864-49

SE94-0881

SOIL EROSION FROM DUNES AT THE SPENT FUEL STORAGE INSTALLATION

The NRC requested the additional sedimentation evaluation be performed for the Palisades ISFSI pad area. This evaluation has been performed to show that the sedimentation, which is postulated, will not impact the operation of the VSC-24 system casks. The evaluation shows that the cask inlet vent may be covered with sand, but there is adequate time to clear the sand.

Safety Evaluation Summary

The EA does not direct any work to be performed at the ISFSI pad. The calculation has been written to demonstrate continued compliance with the intent of 10CFR72

Subpart K Section 72.212(b)(2). Specifically the calculation demonstrates that the existing procedural controls for the ISFSI pad VSC-24 system casks are adequate in terms of ensuring that any forcible blockage of the VSC air inlet vents will be detected and cleared in a short time period. Although DWO-1 basis document discusses the 30 hr time limit to reach 350°F, the VCC can withstand this temperature for an additional 24 hrs. Also there is justification for extending the time limit based on the VCC air outlet to ambient temperature difference remaining below 110°F.

EA-ELEC-LDTAB-009

SE94-0886

BATTERY SIZING FOR THE PALISADES CLASS IE STATION BATTERIES ED-01 AND ED-02

This new Engineering Analysis (EA) supersedes the previous EA which was based on a lotus model. The new EA is based on a EDSA software model. The results of the EA are the same as the previous EA, therefore, there is no impact or change to the plant by the new model.

Safety Evaluation Summary

The EDSA software model produces the same results as the Lotus spreadsheet. The new software will allow development of a more detailed battery model. This change results in a minor FSAR change to reference the new EA.

FSAR CH. 8

SE94-0891
SE94-1035
SE94-1310

UPDATE FSAR DISCUSSION OF EDG FUEL OIL TRANSFER SYSTEM

Revised FSAR sections 8.4.1.2 and 8.4.1.3 discussion on Emergency Diesel Generator (EDG) fuel oil transfer system to ensure safety-related status. FSAR section 8.4.1.3 also is changed to correct the EDG jacket water surge tank location. Additionally, Figure 8-29 is being added.

Safety Evaluation Summary

The FSAR change does not reflect actual change in the facility. These changes update the FSAR description of the EDG fuel oil transfer system to its "as-built" and "as-licensed" condition. The change in description will not increase the probability or consequences of an accident. The "as-built", "as-licensed" transfer system is already used in FSAR chapter 14 analysis.

The changes reflect the re-classifying of the EDG fuel oil transfer system as safety-related which will subject the system to stricter quality standards, overview, and maintenance and operating requirements.

The surge tank location or size are not identified or implied in any FSAR analysis. The location has not changed since the original plant design.

FSAR CH. 5

SE94-0900

UPDATE FSAR DISCUSSION OF EDG OIL TRANSFER SYSTEM

This FSAR change updates the discussion of the emergency diesel generator fuel oil transfer system. FSAR section 5.1.2.2

Safety Evaluation Summary

The loss of fuel oil from T-10 to the EDG does not result in an increase in the probability of an accident occurring.

The NRC has formally accepted the increased risk associated with this issue by letter date June 7, 1994, in response to Palisades letters for justification for continued operation dated May 23, June 3, and June 6, 1994.

FSAR CH. 12

SE94-0971

PALISADES FSAR

The NOD reorganization has moved responsibility for emergency preparedness and the Jackson Dosimetry Laboratory from the Radiological Services Manager to the Director Nuclear (NOD) Services.

Safety Evaluation Summary

The proposed changes are administrative in nature. All of the current listed program responsibilities will not be changed or deleted. They have only been reassigned. Therefore, the current EP and Dosimetry Lab Programs will not change. The TLD Laboratory provides a dose measurement device, and has no interface with plant equipment.

REPLACEMENT PRESSURE-TEMPERATURE CURVES

The revision of the Primary Coolant Pump starting limits, PCS P-T curves, and the PORV setting limits would not cause any changes to the capability or operation of plant systems. These revisions simply update the existing requirements to account for additional reactor vessel fluence.

Safety Evaluation Summary

The reduction of the allowable pressurizer heatup rate would have no effect on operation of the plant. The current limit is physically unobtainable with installed equipment. The proposed change better aligns the Technical Specifications limits with the design analysis.

Substitution of a limit on the availability of automatic HPSI pump starting and valve opening signals for a limit on the operability of the HPSI pumps provides a similar level of protection against an inadvertent injection, while allowing the HPSI pumps to be available for makeup.

The revised specifications, PCP starting limits, PCS P-T limits, pressurizer heatup rate, PORV setting limits, and HPSI pump restrictions, all are directly related to, and intended to prevent, a previously analyzed event, failure of the Reactor Coolant Pressure Boundary.

The revised PCP starting limits, PCS P-T limits, and PORV setting limits are calculated using a similar methodology as the limits which they replace.

The revised pressurizer heatup rate reduces the currently allowable limit which is in the direction of increased margin of safety. Since there is no equipment installed which would cause reaching either the existing or the proposed limit, there will be no change on the operation of the plant equipment.

Substitution of a restriction on automatic HPSI signal for a restriction on HPSI pump operability provides a similar level of assurance against inadvertent mass injection into the PCS and provides the additional benefit of allowing immediate HPSI pump availability for PCS make up.

EA-A-PAL-92-095-02

SE94-0983

DETERMINATION OF THE REACTOR BELTLINE TO PRESSURIZER PRESSURE TAP DIFFERENTIAL HEAD

A Technical Specifications Amendment for the revised Pressure-Temperature Curves and LTOP setpoint curve will place additional restrictions of primary coolant pump operation below 300°F. The basis for the new restrictions are developed in this analysis. The analysis takes advantage of plant restrictions to have only two primary coolant pumps operating below 300°F. The present Pressure-Temperature curves assume four pumps are operating as a conservative assumption. By assuming only two pumps are operating, the analysis can assume a lower differential pressure through the PCS so the allowed PCS pressure can be slightly higher allowing a larger operating band for the operator to control pressure in.

Safety Evaluation Summary

The reactor vessel pressure will still be within the 10CFR50 Appendix G limits. The amendment will have to be approved by the NRC prior to use.

FSAR CH. 11

SE94-0998

DILUTION FLOW DESCRIPTION

Under normal operating conditions, the expected release to the environment from the clean and dirty liquid waste sections is zero. All clean and dirty wastes are recycled or accumulated and prepared for shipment in accordance with applicable rules, regulations and orders of governmental authorities and turned over to a carrier or carriers licensed by governmental authorities have jurisdiction for shipment to an authorized disposal area or areas.

Safety Evaluation Summary

Discharged wastes are released to the discharge mixing basin after proper monitoring. These wastes are mixed and diluted and then overflowed from the mixing basin into Lake Michigan.

During normal operation, the only radioactive liquid released will be processed miscellaneous waste. During abnormal operation, some processed clean or miscellaneous waste might also be released. As liquids are released from the plant to Lake Michigan, they will be diluted in the discharge from the Circulating Water System.

Under normal operating conditions, such as (but not limited to) steam generator tube leakages, fire, or pipe breaking, clean and dirty wastes will be discharged to the environment after processing to reduce and discharged radioactivity to levels which are as low as practicable, and in any event, in accordance with the objectives of 10 CFR 50, Appendix I.

To access the potential impact of these releases, the maximum individual doses from liquid effluent were calculated assuming 100% release of the liquid effluent from the clean and dirty waste system.

Maximum individual doses from liquid effluent were calculated by the NRC LADTAP computer code, using models given in Regulatory Guide 1.109 (March 1976) (see Reference 5). Dose factors, bioaccumulation factors and the shore-width factor is given in Regulatory Guide 1.109 and in the LADTAP code were used, as were use factors for water and fish ingestion and for water-related activities.

FSAR CH. 6

SE94-1012

FSAR CHANGE - REACTOR CAVITY FLOODING SYSTEM

A discussion to FSAR Section 6.8.1.2 describing the purpose of the flooding operation and stating that no chapter 14 event takes credit for this system has been added.

Safety Evaluation Summary

The description of the Reactor Cavity Flooding System was contained in response to question 6.3 in Amendment 14 of the Palisades Application for Construction Permit and Operating License. Section 6.8 is verbatim quote of our response and Figure 6-7 is a reproduction of Bechtel drawing SK-M-224 Revision B, System for Flooding Reactor Cavity. Amendment 14 response discussion for flooding operation only describes how the system should function. This FSAR update adds a paragraph describing the purpose of the flooding operation and states that none of the events analyzed in FSAR Chapter 14 relay on the operation of this system.

Bechtel drawing SK-M-224 shows the prominent features of the cavity flooding system, but it does not correctly show other details in the reactor vessel cavity. Changes made to FSAR Figure 6-7 do not change the description of the flooding system contained in the text of Section 6.8, Reactor Cavity Flooding system. The following plant drawings provide the basis for the changes made to Figure 6-7.

1. M-70 Revision 5, Piping Drawing Nuclear Detector Wells
2. M-74 Revision 8, Underground Piping Reactor Building
3. M-76 Revision 2, Piping Drawing Reactor Cavity Floor Cooling Coils

4. M-81 Revision 6, Plumbing & Drainage Area 1 Plans at EL. 590'-0"
5. M-82 Revision 2, Plumbing & Drainage Area 1 Plans at EL. 625'-0" and EL. 6649-0"

SO-54

SE94-1021

SUPPLEMENTAL EQUIPMENT OPERATING AND TESTING REQUIREMENTS

The added requirements to standing order 54 are provided to ensure that there are restrictions on movement of heavy loads that have the potential to impact spent fuel and on the handling of irradiated fuel in the Spent Fuel Pool when the control room emergency air cleanup system is inoperable. The requirements are consistent with those added to General Operating Procedure GOP-11. The new requirements do not add to the probability of an accident occurring. They do not ensure that if an accident were to occur resulting in operator dose that the control room emergency air cleanup system would be operable.

Safety Evaluation Summary

The requirements are to ensure that the dose to the operator, following an event resulting in a potential radioactive release, is within the 10CFR50 Appendix A criterion 19 limits of 5 rem whole body or its equivalent for the duration of the accident.

The requirements are to minimize the probability of operator dose by requiring that if the control room emergency air cleanup equipment is inoperable, that activities subject to radioactive release from handling fuel or heavy loads are ceased.

These requirements assure that the plant licensing basis is being met by requiring that the equipment required to reduce operator dose is operable when moving irradiated fuel in the spent fuel pool and when moving heavy loads which could impact spent fuel. Clarification of these requirements in the Standing Order - 54 provides consistency with the GOP-11 and the FSAR. It further ensures that the potential operator dose is within the 10CFR50 Appendix A, Criterion 19, 5-REM limit.

EA-C-PAL-429-1

SE94-1068

ADDENDUM TO AUXILIARY FEEDWATER ANALYTICAL REPORT FOR 32°F FEEDWATER TEMPERATURE

This Engineering Analysis identifies the design documents affected by 32°F auxiliary feedwater that could be fed to the steam generators. The steam generators were previously analyzed with 40°F auxiliary feedwater. The engineering analysis also evaluated the structural integrity on affected components for 32°F auxiliary feedwater. The analysis determined that the only steam generator components affected were the

recirculation nozzle and the auxiliary feedwater nozzles. The recirculation nozzle had previously been analyzed for 32°F. The analysis demonstrated the structural integrity for the AFW nozzle.

The analysis also noted the only accident analysis affected by the lower auxiliary feedwater temperature was the Main Steam Line Break (MSLB) accident, which had previously used 32°F as the bounding auxiliary feedwater temperature. Therefore, it was not affected and did not need to be reanalyzed.

Safety Evaluation Summary

The applicable accident (MSLB) had previously been evaluated for 32°F auxiliary feedwater. The component affected by the lower auxiliary feedwater temperature was evaluated. The design temperature differential for the AFW nozzle of 500°F was increased by 8°F which had little impact on the design margin.

EA-FC-864-50

SE94-1072

DRY FUEL STORAGE PROJECT (FC-864) SUPPORT ANALYSIS-MSB 4 STRUCTURAL INTEGRITY ASSESSMENT

Two crack indications at locations 52 and 57 on MSB 4 weld LS-01 are not in compliance with the 1986 ASME Section III code called out in Section 2.0 and 3.0 of the SAR. All of the weld radiographs for MSB 4 were inspected by a CPCo level three NDE inspector to check for ASME B&PV Code Section III acceptance. All of radiographs inspected were found acceptable with the exception of two indications located at 52" & 57 " from the top of the structural lid and a slag indication at location 115.5" which are not in compliance with the ASME Section III NC code allowables. The indication at location 52 is a 0.75" in vertical length and the indication at location 57 is 0.375" transverse crack. Other analysis determined the minimum wall thickness required to meet ASME code requirements using the 0.75" length crack as bounding. This analysis has determined that significant propagation of the cracks in the MSB will not take place over the next 50 years.

The crack found at MSB 4 radiograph location 52 on the 1" thick LS-01 weld, 50 inches from the top of the MSB was most probably caused by a cracked tack weld which was not completely removed during backgouging of the weld groove during fabrication. The weld in question is a double-V groove weld. Based on this theory of the characterization of the crack, the indication would be 1/8" to 3/16" in thickness.

Previous analysis were performed to calculate the minimum MSB shell wall thickness required by the ASME code. EA-FC-864-50 shows that a shell thickness of 0.31" is required for normal operating and transient loading conditions. For accident conditions, absent accidental drop loading considerations, the minimum required MSB shell less than the 1" normal shell thickness and no propagation is expected within the

next 50 yrs. The values used for minimum required wall thickness are also bounding for the crack characterized depth dimensions of 0.125" to 0.1875" given the 1" shell thickness.

The cracks in the MSB that have been identified on the radiography will not increase the probability of any of the accidents evaluated in the VSC-24 SAR. The MSB will not be moved until this analysis and Structural Integrity Associates' independent calculation determined that MSB 4 can withstand all loading conditions described in the VSC-SAR.

These cracks have no effect on the radiation consequences of a breach of the MSB as analyzed in Chapter 11 of the SAR.

The impact on the functional capabilities of the MSB due to the crack indications found will be negligible. The EA concludes that MSB 4 is still capable of performing its design function.

The occurrence of a through wall crack is bounded by the analysis of a postulated failure of all the fuel pins within the MSB and a subsequent ground level breach of the MSB as described in section 11.2.1 of the SAR.

The calculation shows that the minimum required wall thickness to handle the SAR loading conditions is 0.31: and the depth of the crack is characterized as 0.125" to 0.1875".

The possibility of a through-wall propagation of the crack is bounded by the accident analysis as described in section 11.2.1 of the SAR. This analysis shows that an extreme release from the MSB would still be within the 10CFR72.104&106 limits for radiological exposure. There has been no increase in occupational exposure since accelerated monitoring of MSB 4 has been in place.

FSAR CH. 14

SE94-1116

POSTULATED CASK DROP ACCIDENTS

Prior to the loading of MSB3 two issues were raised concerning handling of DFS heavy loads in the track alley. One issue as whether or not the empty MTC required an impact limiting pad during the movements back and forth from the washdown pit to the track alley in between cask loadings for decontamination purposes. A safety evaluation was written to address the concern (Ref. PS&L log # 94-0130) but did not reference the basis calculation EA-FC-864-09. This calculation was the basis for the change to the VSC equipment preparation procedure, FHS-M-33. The conclusion of the evaluation states that FHS-M023, is acceptable for handling the MTC in the track alley since the probability of a load drop is extremely small and the MTC is not carrying the loaded MSB which would require load drop considerations.

The second issue is EA-FC-864-11 which was revised to address a change in the postulated cask drop of the loaded MTC/MSB in the track alley. FHS-M-33 currently calls for the VCC to be centered ± 6" below the Track Alley hatch. With the VCC now centered beneath the spent fuel pool-track alley hatch, it is no longer possible for the loaded MTC/MSB to fall on the Load Distribution System (LDS) from the 649' evevation.

EA-FC-864-11 analysis draws the following conclusions:

1. The loaded MTC will survive the postulated 70 inch drop on to the VCC intact. The MTC will come to rest against the concrete slab at the 649' elevation after impacting the VCC.

2. The concrete and steel framing at the 649' elevation is capable of resisting the impact of the MTC. The 6" concrete curb around the hatch opening will be crushed but the slab will not fail.

3. The loaded MTC will not drop below the top of the VCC when the VCC is centered +/- 6" below the center of the track alley hatch opening.

Safety Evaluation Summary

The probability of an accident has been reduced due to the relocation of the VCC centered beneath the track alley hatch. There is no change in the probability of a drop due to the revision to EA-FC-864-09 to allow movement of the empty MTC without the decon pit impact limiting pad.

The consequences of an accident have been reduced due to the centering of the VCC below the track alley hatch. There will be no longer be a possibility of damage to the Load Distribution System(LDS) due to a hypothetical drop of a loaded MTC/MSB on to the bridge. The possibility of a drop of the empty MTC on the LDS is still extremely small.

The resultant damage to the loaded MTC/MSB due to the drop onto the VCC has been reduced since the MTC/MSB will no longer be postulated to fall through the track alley hatch.

EA-FC-864-11 analyzes that accidental drop of the loaded MTC/MSB on to the VCC with respect to the damage incurred by the MTC and the concrete slab at the 649' elevation. This accident is no different than the previous FSAR analysis with respect to the integrity of the MTC and the track alley structure.

The accident scenario analyzed in EA-FC-864-11 looks at the possibility of the malfunction of the loaded MTC and the track alley structure which is in line with the malfunction methodology considered in the FSAR.

The margin of safety has been increased by centering the VCC below the track alley equipment hatch.

FSAR CH. 4

SE94-1166

FSAR CHANGE TO CH 4 ON STEAM GENERATOR BLOWDOWN FLOW

FSAR Table 4-4 did not reflect what the actual blowdown flows were and also conflicted with other sections of the FSAR. FSAR section 10.2.1.5 describes the blowdown flow as being a minimum of 5,000 lb/hr per steam generator (10,000 total) for effective steam generator chemistry control and the blowdown system is designed for continuous operation up to 30,000 lb/hr per S.G (60,000 total).

Safety Evaluation Summary

There is no accident caused by blowdown flow.

The consequences of an accident (Steam Generator Tube Rupture) could be increased by increasing the blowdown flow by allowing more radioactivity to escape containment. It is not increased by this FSAR change because 1) the blowdown has radiation monitors that will isolate the blowdown and 2) in the analysis in FSAR 14.15. the radioactivity goes out the ADVs and safety valves due to a loss of off-site power.

The increased flow is within the blowdown system design as stated in FSAR section 10.2.

SO-62

SE94-1185

TECHNICAL SPECIFICATION INTERPRETATIONS/GUIDANCE

Revision 25 to Standing Order 62 is intended to correct a long-standing misapplication of Technical Specification 3.2.3.d. The present S.O. 62 wording invoking TS 3.0.3 if both heat trace channels are inoperable for a particular point has no technical or licensing basis as discussed by that evaluation. The proposed wording makes clear that the intended definition of a heat trace channel is the system of heat tracing, as opposed to a particular section of that heat tracing. Individual sections of heat tracing are addressed by the LCO of the boric acid feed path which they support, i.e. TS 3.2.3.c. As discussed by the definition of boric acid feed path found in the existing S.O. 62, heat tracing is an integral component of the feed path.

Safety Evaluation Summary

Concentrated Boric Acid is not required to mitigate FSAR chapter 14 analyses. The worst consequence of a failed section of heat tracing is loss of the associated boric acid feed path, which is addressed by the LCO of TS 3.2.3.c.

Temperature limits for regarding the heat tracing associated with a particular boric acid feed path as inoperable ensure that the 25 degree F margin is maintained above the maximum permittable boron concentration and the correct LCO (TS 3.2.3.c) is applied. Failure of all heat tracing will still result in invoking TS 3.0.3 as a condition in excess of those addressed by technical specifications.

FSAR 6.2.3.2

SE94-1228

CSS HEAT REMOVAL CAPACITY

The second paragraph of FSAR Section 6.2.3.2 is being revised to reference FSAR Section 14.18 as the source for the required containment spray system heat removal capacity. FSAR Section 14.18 documents the correct LOCA/MSLB containment response analysis of record. The containment spray system heat removal capacity is taken from the results of special tests T-216 and T-223 and is factored into the analysis.

Safety Evaluation Summary

The proposed change makes reference to another FSAR section for the required data and does not change procedures or facilities. The proposed change does not affect the flow margin of availability of the containment spray system to mitigate an accident. The change to FSAR Section 6.2.3.2 is proposed to make FSAR Section 14.18 the source of this data and eliminate duplication of information.

FSAR CH. 6

SE94-1233

FSAR SECTION 6.2.3.3

FSAR Section 6.2.3.3 describes a shop thermal transient test from 100°F to 360°F. The shop test report shows actual test temperatures that have been 101°F to 350°F. The FSAR is updated to show the actual test values.

Safety Evaluation Summary

The tested temperatures impose a greater temperature differential and hence a greater stress upon the pump, than is imposed under evaluated accident conditions.

Changing the containment spray pump shop test temperatures will not increase the consequences of a malfunction of equipment important to safety because they do not change of influence the outcome of a equipment malfunction.

No design requirements or conclusions have changed.

The FSAR design requirements for the pump remain unchanged and the tested values are more severe then the accident conditions.

EA-FC-864-30

SE94-1234

SPENT FUEL CASK LIFTING DEVICE

FSAR Change Request "Spent Fuel Cask Lifting Device" changes FSAR standard for Cask Lifting Device for ANSI N14.6-1987 to ANSI N14.6-1986 or 1987.

Safety Evaluation Summary

A reconcilation was done between the two revisions of the ANSI standard concluding there are no design differences. The DPS storage system used ANSI N14.6-1986.

Since there are no design differences there is no affect on equipment or plant operation.

FSAR CH. 8

SE94-1246

FSAR TABLE 8-4 ELECTRICAL DATA

The proposed changes to the FSAR correctly identify the actual vendor's published data for Load Centers 15, 16, 90, 91, and Motor Control Center 99 and pose no impact on safety or the associated equipment's ability to perform its function. These changes are not a result of any modification to the existing equipment, but are in fact a more accurate description of their design bases. The information identified in these changes have been verified by vendor documents received at the time of equipment purchase or published product literature.

Safety Evaluation Summary

The values identified in the FSAR are less than the actual vendor's published data, and therefore conservative. The proposed changes only correct the FSAR Table 8-4 information to the actual vendor's published data. These changes pose no impact on safety or the associated equipment's ability to perform its function.

FSAR CH. 9

SE94-1275

SWS FLOW REQUIREMENTS

Changes to FSAR Table 9-1 are the result of resolved discrepancies discovered during the research and preparation of Design Basis Document, DBD 1.02. These changes correct the FSAR's insufficiency to correctly tabulate all of the major Service Water System (SWS) flow requirements. The changes to the FSAR Table involve normal and shutdown operation flow demands that are documented as part of the SWS Design Bases, and are reflected in the FSAR Chapter 14 analyses. The changes primarily involved the non-critical Service Water member component list. One change is proposed in the Critical Service Water member component list to indicate the actual continuous service water flow through the Engineered Safeguards Room Coolers.

Safety Evaluation Summary

These changes do not add newly identified cooling demands and therefore, will not affect the original engineered design basis of the SWS or its function.

The SWS is used to mitigate the consequences of an accident. The changes to FSAR Table 9-1 correct the major SWS flow requirements but do not add any newly identified cooling demands. The changes involve non-critical service water flows and do not affect the flow margin or availability of critical service water flows.

FSAR CH. 2

SE94-1278

TABLE 2.19 REVISION

Update Table 2.19 to use the 5 year annual average X/Q agreeing with the ODCM Table 1.3 (1-1-88 to 12-31-92). Add clarification to better explain use of meterology values.

Safety Evaluation Summary

Most of the accident analyses have used the short term values in tables 2.17 and 2.18. This code has not been run on updated values. However, the highest annual average X/Q at the site boundary with the 1978 data was 1.5 E-06 sec/m. Palisades has gone to a 5 year average annual X/Q to provide stable values for effluent dose projections. If the annual average stays $\pm$25% of original values it it reasonable assurance that the start term X/Q has not changed significantly and accident analysis is valid.

The current 5 year annual average value at the site boundary is 1.7 E-06.

FSAR CH. 5
SE94-1291

CLARIFICATION OF CONTAINMENT AIR COOLER MOTORS SAFETY CLASSIFICATION

FSAR Tables 5.2-4 and 5.2-5 are being revised to clargy the safety classification of the containment air cooler (CAC) motors and the instrumentation and control associated with these motors and coolers.

Safety Evaluation Summary

The CACs are not an initiator of an accident evaluated in Chapter 14. CAC fans V-1A, V-2A, and V-3A are relied upon to mitigate the consequences of an accident. CAC V-4A is not relied upon in an accident.

The proposed changes would not impact equipment important to safety.

Failure of the motor would not cause degradation of fission product barrier.

FSAR CH. 8
SE94-1306

EDG TROUBLE ALARM

The change to FSAR 8.4.1.3 description of EDG Jacket Water Surge Tank makeup corrects FSAR to the as-built condition and as-used condition.

Safety Evaluation Summary

No change to the facility has occurred as a result of this FSAR change. It does not impact any accident previously evaluated. Surge tank makeup is not required by any Chapter 14 analysis and does not impact malfunction of equipment. Manual makeup increased reliability because automatic makeup system leaked. Tank low level is alarmed locally and repeated in control room at "EDG Trouble" alarm. Tank makeup is not identified in the Tech Specs and is not required by an FSAR Chapter 14 analysis.

FSAR CH. 6
SE94-1307

FSAR CHANGE TO SECTIONS 6.2.1 AND 6.3.1

Deletion of the statements of 6.2.1 and 6.3.1 which state that the Containment Spray System and Containment Air Recirculation and Cooling System are redundant.

Current Chapter 14 analysis supports the configuration one Containment Spray Pump and three Air Coolers being required to maintain desired Containment Pressure. This configuration is given in Table 14.18.1-2 of the FSAR.

Safety Evlauation Summary

The change clarifies the stated requirements of current FSAR Table 14.18.1-2.

FSAR Table 14.18.1-2 requires 1 spray pump and 3 CACs for maintaining Containment pressure following a LOCA. The proposed change eliminates incorrect statements in 6.2.1 and 6.3.1 which indicate that the CSS and Containment Air Coolers are totally redundant systems.

FSAR CH. 6

SE94-1308

FSAR CHANGE TO FIGURE 6-8

Removal of the Capacity/Pressure curve below 5 psi in Figure 6-8 and revision of the minimal flow note to coincide with the Manufacturer-supplied curve which illustrates a minimum flow of 5.5 GPM.

Safety Evaluation Summary

The proposed change has no effect on any accident previously evaluated in the FSAR. A previous FSAR change which added the figure 6-8 and flowrate note refers to the previous flowrate of Table 6-7 and 15.2 GPM for 40 psi pressure drop. This point coincides with Figure 6-8 which is being changed to its original form. The new minimal flowrate also agrees with Figure 6-8.

No equipment is being modified. The flowrates illustrated by Figure 6-8 are those provided by the nozzle manufacturer.

No change in nozzle function, or flow to the nozzles is being made. Revision to Figure 6-8 is providing manufacturer supplied flow capacity information which was used in original plant accident analysis.

FSAR CH. 6

SE94-1309

MARGIN OF CAPACITY

Change 6.2.3.2 which reads: "Minimum required spray flow to achieve the design basis is 2,836 gpm. The minimum spray flow delivered is 3,150 gpm." To read: "The minimum spray flow delivered is 3,150 gpm. See Section 14.18 for spray flow rates required to achieve the design basis."

Safety Evaluation Summary

No FSAR analyzed accident is affected by this proposed change. A redundant flowrate is being removed with a referral being added to the apporiate FSAR section to find the values for flowrates.

No equipment is being affected by the proposed change.

EA-ELEC LDTAB-005

SE94-1317

EMERGENCY DIESEL GENERATORS 1-1 & 1-2 STEADY STATE LOADINGS

The Engineering analysis tabulates EDG 1-1 and 1-2 loadings, under a worst case scenario of a large break LOCA. This calculation incorporated error for uncertainties associated with obtaining real plus reactive power values.

Safety Evaluation Summary

The EDGs would not initiate a Large Break LOCA, and they are within the licensing basis as stated in the FSAR. The EDGs are used to mitigate an accident.

The calculated EDG loadings do not exceed their, 3840BHP/2750 KW, two hour output rating.

FSAR CH. 12

SE94-1318

CONDUCT OF OPERATIONS

The change to the FSAR is to the Corporate and Site Organizations. It identifies the reporting relationships, responsibilities, and qualifications of individuals in the organizations. The changes stem from reorganization of the corporate organization that will be effective in mid-November, and the organizational changes to the Plant, NECO, NPAD, NSD, and PS&L departments to better align responsibilities. These changes are administrative in nature.

Safety Evaluation Summary

The Operating Shift Crew Composition has been removed from the FSAR. The minimum crew staffing is a Technical Specifications Requirement and through Administrative Procedure AP-4.00 the staffing requirements are delineated to ensure the Technical Specifications requirements and commitments made in the site emergency plan are met. This change removes a redundant requirement which is not necessary in the FSAR. Qualification requirements described in the CPC Quality Program topical report CPC-2A and the qualification commitments (ANSI N18.1, 1971)

for the Plant and NECO staffs continue to be met. None of the changes alter the responsibility to safely operate, maintain, and modify the plant and train the personnel.

EA-PAH-94-05

SE94-1336

UPDATED JUSTIFICATION FOR CONTINUED OPERATION FOR THE MHA AND RELATED ISSUES

This analysis updates the justification for continued operation until resolution of all MHA analysis related issues have been resolved. This analysis supersedes the previous JCO. New information incorporated in this update includes the leakage rates for sump water paths to the SIRW Tank which have all been found tested, atmospheric dispersion factors for control room air intakes based on wind tunnel test of a scale mode of the plant, and instrument inaccuracies for all measured control room HVAC system air flow rates.

Safety Evaluation Summary

This analysis justifies continued operation for the MHA related issues. The analysis does not change operation of the plant.

The justification demonstrates that offsite dose limits would not be exceeded in the event of a design basis accident. The justification also demonstrates that the allowable unfiltered air leakage into the control room is high enough to prevent exceeding control room dose limits in the event of a design basis accident.

PTS

SE94-1365
SE94-1407

PTS SCREENING CRITERIA MARGIN BASES ON PRELIMINARY RETIRED STEAM GENERATOR CHEMISTRY AREA

The date at which the Palisades plant pressure vessel will reach the PTS screening criteria will change if preliminary chemistry data from the retired steam generator welds becomes final.

Safety Evaluation Summary

The PTS screening criteria is a limit set in 10 CFR 50.61. Accidents evaluated in the FSAR assume this limit has not been passed. EA-RDS-94-02 shows the Palisades reactor vessel has not reached this limit, therefore the FSAR accident analyses are still valid. This chemistry data does not affect the Technical Specifications Pressure - Temperature limits because the weld in question is not the limiting weld in the basis of these calculations.

The screening criteria has not been reached, therefore all accident analyses remain unchanged. What has changed, if the preliminary chemistry results become final, is the date at which these analyses must be readdressed.

Because the pressure vessel has not reached the screening criteria no adjustments must be made in its failure probability.

EDC-SC-083-06

SE94-1507

ENGINEERING DESIGN CHANGE

This engineering design change evaluates and dispositions thirteen non-conformance reports and two drawing changes initiated during manufacture of the Multi-Assembly Storage Baskets (MSB) for the Palisades Independent Spent Fuel Storage facility. The non-conformances involved material lamination, dimensions out of specification, a gouge on the MSB shell, a weld deposit, chipped paint, coating applied over debris, poor welds, and weld repair to repair tears in material. The non-conformance reports were either dispositioned by repairing MSB (for defects) or dispositioned to use as-is if the non-conformance had no effect on the design or function of the MSB.

Safety Evaluation Summary

These items do not adversely impact the ability of the MSB to perform its design functions. The pressure boundary is not affected by the disposition of the non-conformances. The functional capability of the MSB is also not affected.

The occupational exposure will not be increased since the disposition of the non-conformances will not affect the design basis dose.

The disposition of the non-conformances do not have any environmental impact.

FSAR CH. 7

SE94-1510

CLARIFY LTOP DESIGN BASIS

This change corrects some errors and rephrases the description to more clearly show that 435 degrees F is a capability of the system. 435°F was selected to be sufficiently larger than the Technical Specifications (TS) limit to ensure that the system would be able to perform in full compliance with TS 3.1.8.

Safety Evaluation Summary

This change does not affect any analyzed FSAR accidents. The change shows that the number is a design limit rather that an operating limit.

Because this is a clarification, no actual changes to any safety system will occur. The system will operate the same.

Set points are still controlled by the curve in FSAR Figure 4-17.

EA-RGB-94-02

SE94-1512

ENGINEERING ANALYSIS FOR QUALIFIED LIFE OF ROSEMOUNT TRANSMITTERS IN CONTAINMENT AIR ROOM

This analysis revises the qualified life of Rosemount transmitters to reflect a year of temperature data. Various instruments were evaluated with no instrument having a qualified life of less than 23 years. This EA documents the basis for the ambient temperature input to the qualified life calculation and calculated the qualified life for Rosemount transmitters located in the Containment Air Room at elevation of 602' and below.

Safety Evaluation Summary

The peak temperatures recorded during plant operation was 87°F during winter and 108°F during summer. Qualified life calculations bases on the Rosemount thermal aging parameters and the Containment Air Room temperature conservatively assumed to be 108°F for 2/3 of the year and 87°F for 1/3 of the year. The transmitter electronic circuit boards are qualified for 23 years and the complete transmitter is qualified for 34 years.

This analysis is input to the environmental qualification program for electrical equipment program. The EQ program is designed to ensure that no unreviewed safety questions exist due to environmental concerns.

FSAR Table 9-20

SE94-1523

OPERATING PARAMETERS ION EXCHANGERS

Revise the following Operating Parameters for the Ion Exchangers - T51A, T51B, and T52: Maximum flowrate from 160 gpm to 120 gpm and Retention Screen from 140 US Mesh to 80 US Mesh.

Safety Evaluation Summary

The ION exchangers are not involved with any accident scenario. The new operating parameters are according to the Engineering Specification Data Sheets for these units.

The function of the Exchangers remains the same and the normal flow rate of 40 gpm is well below the maximum flow rate for the exchangers. The transient changes in flow of 40 gpm to 80 gpm to 120 gpm remains the same.

FSAR Table 9-20

SE94-1524

OPERATING PARAMETERS LETDOWN ORIFICE - RO2003, RO2004, AND RO2005

Revise the Letdown Orifice Operating Parameters for RO2003, RO2004, and RO2005 as follows: Design Temperature - 650 degrees F, Design Pressure 2,485 psia, Normal Temperature of Fluid - 250 degrees F, Maximum Temperature of Fluid - 450 degrees F, Normal Down Stream Pressure - 470 psia, and Normal Upstream Pressure - 1,970 psia.

Safety Evaluation Summary

The valves either match or are very close to those of the original Engineering Specification for the Letdown orifices. These orifices are installed parallel with a stop and trip valve to reduce the pressure of a portion of the primary coolant water for purification and volume control purposes. The changes are small or in most cases more conservative in direction from the FSAR valves.

The orifices work in conjunction with trim valves to reduce the pressure of the primary coolant system water for purification. The new parameters are equal to or similar to the original orifice parameters.

FSAR Table 9-20

SE94-1525

CONSTANT SPEED CHARGING PUMPS - P55B AND P55C

Revise the FSAR Table 9-20, Sheet 8, 1.11 "Maximum Pressure Pump Starts Against" from 2,485 psig to read 2,500 psig.

Safety Evaluation Summary

The parameter for pump starting is being changed - in a more conservative direction. The maximum pressure the pump can start against is affected and this value is 15 psig in a more conservative direction. The new value is based upon manufacturer's pump data.

FSAR Table 9-20

SE94-1526
SE94-1527
SE94-1561

OPERATING PARAMETERS - REGENERATIVE HEAT EXCHANGER - LETDOWN HEAT EXCHANGER

Replacement of operating parameters for the Regenerative Heat Exchanger and Letdown Heat Exchanger with those of the specifications.

Safety Evaluation Summary

For the Regenerative Heat Exchanger the shell side flow changes of 1 gpm for normal operation and when on maximum purification are insignificant. The tube side outlet temperature changes vary from 0.2% to 11.6%, however, they are still below the required 450°F temperature which ensures that the letdown heat exchanger maximum inlet temperature is not exceeded. During normal operation the temperature is reduced, but only by 2%, which is insignificant. The heat transfer changes are proportionate to the temperature changes.

The parameters being changed are still within design requirements. The flow parameter changes are insignificant and the heat added to the charging side of the exchanger is enhanced according to manufacturer design parameters.

For the Letdown Heat Exchanger, the inlet temperature changes still maintain a temperature which is below the maximum outlet temperature of the heat exchanger. The shell side flowrate changes are still less than the maximum allowable. The tubeside outlet temperature change is still less than the maximum allowable for the purification system operation. The design pressure is being revised from 600 psig to 650 psig. All values are according to the manufacturer's technical manual. All changes are within the maximum allowable flowrate and temperature parameters of the down stream equipment and the affected heat exchanger.

FSAR CH. 6

SE94-1545

COMPONENT DESCRIPTION

Revise "All portions of the system are fabricated of stainless steel for corrosion protection" to read: "All portions of the system in contact with the Primary Coolant System or borated water are fabricated of stainless steel or other corrosion-resistant material."

Safety Evaluation Summary

The change clarifies information within the FSAR. It does not revise or address any accident scenario.

No equipment or system is being affected by this change, only a clarification is being made.

The margin of safety is not affected by this change because no equipment or system is being modified-only a clarification made.

FSAR Table 9-20

SE94-1556

OPERATING PARAMETERS-BORIC ACID BATCHING TANK - T-77

Revise the internal volume of the Boric Acid Batching Tank from 569 gal to 580 gal and the useful volume from 470 gal to 457.4 gal.

Safety Evaluation Summary

The Boric Acid Batching Tank is not addressed by any accident scenario.

The T-77 tank is not quality related, and has no systematic effect on quality, safety related, equipment.

FSAR Table 9-20

SE94-1557

OPERATING PARAMETERS - VOLUME CONTROL TANK - T54

Revise the "Internal Volume, Minimum" value of "4,165 gal" to read "Internal Volume, Maximum - 4,170 gal."

Remove "Liquid Volume, Maximum - 3,068 gal" due to the variability of this value as a result of plant operations.

Safety Evaluation Summary

The value of 4,170 gallons is only 5 gallons different than the 4,165 gal value. The difference is negligible. Removal of the maximum liquid volume value does not impact any accident scenarios.