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CPSES-200202892
Log # TXX-02146
File # 10010

August 1, 2002

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
10CFR50.59 EVALUATION SUMMARY REPORT 0010 AND
COMMITMENT MATERIAL CHANGE EVALUATION
REPORT 0005

Gentlemen:

Please find attached the following periodic reports pertaining to CPSES Unit 1 and Unit 2.

(A) Attachment 1 is the report required by 10CFR50.59(b)(2) for activities since August 2, 2000, at CPSES Units 1 and 2. This report contains a brief description of the changes, tests and experiments implemented or performed pursuant to 10CFR50.59(a), including a summary of the safety evaluations for each. Items in this report are referenced by their 10CFR50.59 Evaluation Numbers. Please note that the change in numbering for the 50.59 evaluations corresponds to the implementation of the revised 50.59 rule on May 30, 2001. This report includes those activities which were completed or partially completed between August 2, 2000, and February 1, 2002, and which were not reported to the NRC in a previous submittal. This report also includes certain activities completed or partially completed after February 1, 2002.

(B) Attachment 2 is the CPSES Units 1 and 2 report (Commitment Material Change Evaluation Report 0004) per the recommendations of NRC document SECY-95-300, "Guidelines for Managing NRC Commitments." The tracking document for this process at CPSES is the "Commitment Material Change Evaluation (CMCE)" which identifies the affected commitments and origin, original criteria, proposed changes and the justifications for the changes. This

IE47

report pertains to commitment material changes (in docketed correspondences) which require reporting between August 2, 2000 and February 1, 2002, which were not addressed in the 10CFR50.59 evaluations.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

Sincerely,

TXU Generation Company LP
By: TXU Generation Management Company LLC,
Its General Partner

C. L. Terry
Senior Vice President and Principal Nuclear Officer

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

JDS/js

Attachments

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Resident Inspectors, CPSES

ATTACHMENT 1 to TXX-02146

SE-99-039	Rev. 0	59EV-1999-002168-01-01
SE-00-003	Rev. 0	59EV-2001-000426-01-00
SE-00-015	Rev. 1	59EV-2001-001870-01-00
SE-00-017	Rev. 0	59EV-2001-000751-01-00
SE-00-028	Rev. 0	59EV-2002-000587-01-00
SE-00-031	Rev. 0	59EV-2002-001634-01-00
SE-00-033	Rev. 0	
SE-01-002	Rev. 0	
SE-01-006	Rev. 0	
SE-01-007	Rev. 0	
SE-01-009	Rev. 0	
SE-01-010	Rev. 0	
SE-01-011	Rev. 0	
SE-01-012	Rev. 0	
SE-01-013	Rev. 0	
SE-01-014	Rev. 0	

Evaluation Number: SE- 99-039 **Units: 1 and 2**
Revision 0

Activity Title:
ADDITION OF AN ISOLATION VALVE AT THE INLET TO THE CCW DRAIN TANK TO PROVIDE PERSONNEL PROTECTION DURING WORK ACTIVITIES

Description of Change(s):
Personnel are occasionally required to enter the Component Cooling Water Drain Tank for maintenance for cleaning activities. There are no means of isolating the inlet line to the drain tank. This activity involves the addition of a narrow profile knife gate type valve with a ratchet style manual gear operator in the inlet section of drain piping to the Unit 1 and Unit 2 Component Cooling Water Drain Tank. This allows isolation of the tank from the drain lines when work is being performed inside the tank. This has been identified as a personnel safety concern and allows isolation of the tank to protect personnel inside the tank. The valve shall be normally open. When the valve is closed due to ongoing work inside the tank, provisions must be made to direct the drainage flow through an alternate drain path. This can be accomplished by hooking up a drain line to the 4 inch clean out connection that is located at the intersection of the 4 inch drain lines. A hose can be connected to this drain point and routed to a local sump to provide the required alternate drain path while the tank inlet connection is isolated for work inside the tank.

Summary of Evaluation:
This activity does not involve an unreviewed safety question because it does not introduce any new failure modes, does not increase the probability of a flooding event or its consequences, and complies with existing design requirements. The addition of the inlet isolation valve does not adversely affect the operation of the plant. This activity is an enhancement provided to improve personnel safety.

Evaluation Number: SE- 00-003 **Units: 1 and 2**
Revision 0

Activity Title:
RELOCATE & CHANGE POWER SUPPLY OF ONE OF THREE TSC PLANT COMPUTER SYSTEM DATA DISPLAYS. FDA-99-3409-1; LDCR SA-2000-005

Description of Change(s):
One of three plant computer data display units in the Technical Support Center (TSC) was relocated from a hallway to a room where the TSC Engineering Team conducts most of its emergency response activities. The relocation involved changing the display unit's power supply as described in FSAR Section III.A.1.2 from a Non-1E, battery backed, Uninterruptable Power Supply (UPS) System (off plant Inverter IV1C6) to a Non-1E, 120 VAC common supply.

Summary of Evaluation:
This change was implemented in order to place the subject data display unit in a more useful location within the TSC. The new location is closer to the work area where the TSC Engineering Team conducts most of its emergency response functions/activities. This relocation enhances the TSC Engineering Team's emergency response function.

Guidance from NUREG 0696 does not require that all TSC data displays units be powered from a UPS System. Two of three TSC data display units remain powered from the UPS source, and the continued availability of multiple (redundant) data display units within the TSC maintains the high reliability for being able to display plant data and meets the NUREG guidance with respect to emergency response facility design criteria.

The TSC display units display plant data independent of action in the Control Room and without degradation or interfering with Control Room and plant functions. This equipment does assist the TSC Engineering Team in mitigating the consequences of an accident; however, the equipment itself performs no safety-related functions with respect to CPSES operations.

Evaluation Number: SE- 00-015 **Units: 1 and 2**
Revision 1

Activity Title:
SERVICE WATER INTAKE STRUCTURE (SWIS) CHLORINATION BUILDING REFURBISHING - DMA 1999-1929 (SMF1999-1929)

Description of Change(s):
In the SWIS Chlorination building, there are two tanks (one 500 gallon tank filled with sodium bromide and one 1500 gallon tank filled with sodium hypo-chlorite) solutions. These tanks are sized such that there is a new truck load of chemicals required to fill both tanks once a week or more often in the summer time when more chlorination is required. This activity replaced the existing tanks and pump skid with 6 larger (1900 gallons each) tanks with a much larger capacity, and a new pump skid with 4 new pumps. This activity was worked in conjunction with DMA 1998-2070 which improves the distribution system for the chemicals into the SWIS intake bay. The new tanks reduce the number of fill times required annually, thereby reducing the potential for an inadvertent chemical spill with respective accident ramifications.

Summary of Evaluation:
This activity replaces high-maintenance, unreliable components with new pump skids and greater chemical storage capacity. The new pump skids will increase the reliability of the system by reducing the probability of fouling of the Service Water side of the CCW heat exchangers from the loss of chemical injection. Reducing the probability of heat exchanger fouling will reduce the probability of a loss in heat transfer and subsequent reduction in CCW performance. Increasing the quantity of chemicals stored on site will decrease the probability of a transfer spill. The Control Room Habitability Analysis is not adversely impacted by the increased amount of chemicals and therefore the continued safe operation of the plant is assured. The injection points and injection quantities are not affected by this mod nor have the chemicals changed from what is currently in use. As before, the new components are non-safety related and non-seismic, and their failure will not adversely affect the safe operation or shutdown of the plant, nor will their failure prevent safety systems from performing their design functions. No regulatory commitments are affected by this activity.

Evaluation Number: SE- 00-017 **Unit: 1**
Revision 0

Activity Title:
Unit 1 UPGRADE REFUEL MACHINE CONTROLS MODIFICATION

Description of Change(s):
This modification replaces the existing Refuel Machine Motor Control Center (MCC) and Control Console (CC) with new equipment. Additional system changes are also made by this activity to enhance system reliability. The upgrades provided by this modification utilize equipment that is an acceptable substitute and is functionally interchangeable with the original equipment.

Summary of Evaluation:
This activity replaces and upgrades the existing Refueling Machine control system. Although this is a replacement of relay controls with a digital (PLC) type control system, the installation of this modification will not change the function of the refueling machine. Failure modes of the equipment are evaluated as being no more severe than the original system. The weight of the MCC and CC has been evaluated and remains within the allowable limits. The existing circuit breaker provides adequate protection for this application. Replacement of those components by this activity does not impact the systems' functionality, operation or qualification.

Evaluation Number: SE- 00-028 **Units: 1 and 2**
Revision 0

Activity Title:
PROVIDE SPECIFIC RELATIONSHIP BETWEEN 480V SWITCHGEAR PENETRATION CONDUCTOR PROTECTION PRIMARY BREAKERS AND THEIR ASSOC. RELAYS

Description of Change(s):
Technical Requirement Manual Tables 13.8.32-1a and 13.8.32.1b are revised to provide specific relationship between 480V switchgear penetration conductor protection primary breakers and their associated relays, and backup breakers and their associated relays.

Summary of Evaluation:
1. Tripping of the primary breakers is actuated by the breaker Amptector Unit. TRM is revised to list breaker Amptector Units as an associated relay for the primary breakers.

2. Long time and instantaneous 50-51 relays and time delay relays 62-1 are used in the tripping circuit of both backup breakers, i.e., bus incoming breaker and bus tie breaker. TRM is revised to list these relays as associated relays for both the backup breakers.

3. Relays 62-1X are used in the control schematics of primary breakers to trip the breaker if it is not already tripped by the amptector. This function is a defense in depth function and no credit is taken for this trip to provide penetration conductor primary protection. As such these relays are deleted from the TRM table.

Evaluation Number: SE- 00-031 **Units: 1 and 2**
Revision 0

Activity Title:
INSTALLATION OF PARTIAL DISCHARGE MONITOR BUS COUPLER ON SSW PUMP MOTORS CP1 & CP2-SWAPSW-01M, -02M

Description of Change(s):
Installation of Partial Discharge Monitor Bus Couplers on Units 1 and 2 Station Service Water Pump Motors. Updates to FSAR are to be incorporated per LDCR SA-2001-027.

Summary of Evaluation:
This system is used for diagnostic purpose only. Partial discharge measurements provides information on the degradation of SSW Pump motor stator insulation (CP1 & CP2-SWAPSW-01M, -02M).

The partial discharge monitor system has been procured and installed as safety related equipment Failure of the 15 KV Jumper cable and the interface is extremely remote. Failure of signal cable and Diagnostic Equipment is also remote and even if they fail, will not affect the operation of the motors. This system has been used in the Industry for more than 8 years without any failure.

It is concluded that installation of the partial discharge monitor bus coupler and diagnostic equipment will not adversely affect the safety related Station Service Water Pump Motor(s). This addition presents no new credible potential failure modes or accidents for the plant or any systems.

Evaluation Number: SE- 00-033 **Unit: 1**
Revision 0

Activity Title:
REMODEL AND/OR RECLASSIFY RADIATION ZONE OF SELECTED ROOMS IN THE UNIT 1 TURBINE BUILDING

Description of Change(s):
This activity remodeled selected rooms within the Unit 1 Turbine Building and revised the CPSES design basis documents and FSAR figures to reflect current room use, description and floor plan. Also, selected rooms (1-018, 1-019, 1-020, 1-021 and 1-033) were reclassified from Radiation Zone II to Radiation Zone I. The affected Unit 1 Turbine Building rooms and current use/description and Radiation Zone designation are as follows: 1-018 (Corridor- Zone I); 1-019 (Toilet room- Zone I); 1-020 (Woman's Locker Room - Zone I); 1-021 (RP and Chemistry Briefing Room - Zone I); 1-022 (Chemistry Supervisor's Office - Zone II); 1-026 (Chemistry Storage Closet - Zone II); 1-027 (Chemistry Instrument Lab - Zone II); 1-027A (Chemistry Storage Room - Zone II); and 1-033 (Steam Generator Remote Monitoring Facility - Zone I).

Summary of Evaluation:
The room remodeling occurred in the Turbine Building which is a non-safety related, non-seismic building. The physical changes to affected rooms were non-structural in nature; the overall remodeling activities were architectural and affected room layout and associated doors, lighting, ingress/egress. These activities did not affect the structure of the Turbine Building or have any impact on safety related systems, structures or components. The Turbine Building rooms that were reclassified to Radiation Zone I are not expected to contain any transient or permanent radiation sources. These rooms were removed as part of the CPSES Radiologically Controlled Area (RCA); they have uncontrolled, unlimited access from the Control Building El. 810' North-South Corridor, Room 1-106. The intended use and function of these rooms complies with the required dose rate limits of Radiation Zone I (less than or equal to 0.25 mrem/hour) as defined in the CPSES design basis documents and FSAR Section 12.3.1.3. Ventilation air flow was not modified as part of the room remodeling activities and Radiation Zone reclassifications. The existing airflow path continues to comply with the considerations of Reg. Guide 8.8 as the air flows from areas of low potential airborne contamination to areas of higher potential contamination (i.e. low radiation zones to high radiation zones). The rooms reclassified to Radiation Zone I were originally classified as Radiation Zoned II based on the expectation that the areas could potentially contain radioactive or radiologically contaminated equipment or components. The radiological classification change reduces the size of the RCA at CPSES. This reduction in size of the RCA does not increase the probability of any accident or increase the consequences of any accident as it is not an initiator of any licensing basis accident or used to mitigate the consequences of those accidents. The change in the room Radiation Zone classification is an administrative or "document change only" and is consistent with the radiation zoning specified in the CPSES design basis documents and FSAR. These changes do not affect any systems or components that could contribute to the initiation of an accident or malfunction of equipment important to safety. Similarly, the Radiation Zone reclassifications limit the area of CPSES that can contain radioactive material and hence do not contribute to any increase in the spread of radiological material.

Evaluation Number: SE- 01-002 **Units: 1 and 2**
Revision 0

Activity Title:
REVISION TO FSAR SECTION 9.3.1.1 (INSTRUMENT AIR) TO CHANGE THE AIR QUALITY REQUIREMENTS TO MEET ANSI ISA S7.0.01-1996 INSTEAD OF S7.3-1975

Description of Change(s):
Change the air quality requirements for the Instrument Air System to meet ANSI ISA S7.0.01-1996 instead of ANSI ISA S7.3-1975.

Summary of Evaluation:
ANSI ISA S7.3-1975, "Quality Standard for Instrument Air" has been superseded by ANSI ISA S7.0.01-1996. This activity will change the Air Quality Standard to a standard which supersedes the existing Regulatory Commitment of meeting ANSI ISA S7.3-1975, made by Comanche Peak Steam Electric Station (CPSES). The change in the specification that will impact CPSES is the particulate limit. The particulate limit for an acceptable quality of air will change from 3um to 40um or the particulate limit of the end user device, whichever is smaller. Both specifications meet the air quality requirement imposed by Generic Letter 88-14 and NUREG-1275, Volume 2 (page 58), which is to ensure air quality is consistent with the equipment manufacturer's specification through monitoring and testing. CPSES has evaluated all end-users of the instrument air system. Any user that has a particulate limit of <40um has additional point-of-use filters to ensure equipment is protected. This activity does not impact dew point or hydrocarbon requirements for CPSES. ANSI ISA S7.0.01-1996 requirements for dew point and hydrocarbon remain the same as ANSI ISA S7.3-1975. Adopting the new air quality standard for Instrument Air will have no adverse affect on the ability of the Instrument Air system to perform its function.

Evaluation Number: SE- 01-006 **Unit: 1**
Revision 0

Activity Title:
REVISE UNIT 1 TURBINE DISK INSPECTION BASIS (LDCR SA-01-00018)

Description of Change(s):
Low Pressure (LP) Turbine disk inspection interval of 50,000 hours is based upon vendor evaluations referenced in the FSAR. Original vendor reports for the licensing basis for Unit 1 LP turbine rotor disk inspection intervals are no longer applicable to both Unit 1 and 2 rotors as implied by the FSAR. These reports were developed from specific metallurgical samples from the Unit 1 turbines. The new LP turbine rotor for Unit 1 contains metallurgical samples that exhibit different attributes. Ref. FSAR 10.2.3.6 for Inservice Inspection, ER-8402. The new vendor report concludes that the probability of a turbine disk burst is still below the CPSES licensing basis. This activity provides for the revision of the FSAR to document the Unit 1, LP 2 disk inspection basis. It will differentiate between the Unit 1 and Unit 2 disk inspection basis.

Summary of Evaluation:
The LP turbine disk inspection is discussed in the FSAR sections 3.5.1 and 10.2.3.6 for turbine missile generation and inservice inspections of the turbine generator. The LP turbine disk inspection interval is derived from the vendors analysis of the turbine properties and estimated probability of disk burst failure. The vendors estimated probability of failure is an input to the CPSES missile analysis. The missile analysis was performed to ensure the safety and integrity of systems, structures and components. The original CPSES turbine missile analysis used the vendor engineering reports to determine the probability of missile generation from LP turbine disk burst. The vendor has submitted a new engineering report which uses improved calculation techniques and actual inspection data from previous CPSES turbine inspections. This evaluation reviews the data available to determine that the new Unit 1 LP turbine basis is unique and that the inspection intervals are still valid. The normal inspection intervals of 50,000 hours between inspections are not affected by this evaluation. Inspection results indicate that the Unit 1 LP turbines can continue to be operated until the next scheduled disk inspection.

Evaluation Number: SE- 01-007 **Unit: 1**
Revision 0

Activity Title:
Unit 1, Cycle 9 Core Configuration

Description of Change(s):
During the next refueling outage for Unit 1 (1RFO8), prior to operation of Cycle 9, 92 fresh Region 11 fuel assemblies manufactured by Siemens Power Corporation (SPC), along with 1 Unit 2 Region 2 partially burned Westinghouse (W) optimized fuel assembly (OFA) assembly, will replace 12 Region 9B, 64 Region 9A, 12 Region 8, 4 Region 7 SPC assemblies and 1 Unit 2 Region 2 partially burned Westinghouse optimized fuel assembly (OFA) assembly. The partially burned assembly is from the spent fuel pool and was discharged from Unit 2 at the end of its Cycle 1. For the Unit 1 Cycle 9 core configuration, 92 fresh SPC fuel assemblies will be co-resident with 100 partially burned SPC fuel assemblies and 1 partially burned OFA manufactured by Westinghouse.

Summary of Evaluation:
The CPSES U1C9 mixed core configuration has been evaluated for mechanical and thermal-hydraulic compatibility between the different SPC and W fuel assemblies. All applicable design criteria were determined to be satisfied at the current power levels. The neutronic characteristics of the Cycle 9 core configuration have been evaluated for their effect on the accident analyses. In all cases, it was determined that the applicable event acceptance criteria are satisfied. Because all mechanical design criteria continue to be satisfied, there is no reduction in any failure point introduced by the Cycle 9 core configuration. All acceptance criteria of the accident analyses continued to be satisfied; therefore, there is no increase in the consequences of any accident previously analyzed. Based on the foregoing, it is concluded that the Unit 1 Cycle 9 core configuration does not reduce any margin of safety as defined by the plant Technical Specifications; therefore, the proposed change does not involve any Unreviewed Safety Question.

Evaluation Number: SE- 01-009 **Unit: 2**
Revision 0

Activity Title:
EVALUATE PLACING PRESSURIZER SPRAY VALVE 2-PCV-0455C IN A BACKSEATED CONDITION IN ORDER TO STOP LEAKAGE FROM THE VALVES PACKING

Description of Change(s):
Due to identified packing leakage, Pressurizer Spray Control Valve 2-PCV-0455C has been backseated in the closed position. This condition renders the valve non-functional and Loop 4 Pressurizer Spray line as non-functional. This 10CFR50.59 evaluates the compensatory measures taken to control RCS leakage from this component. Pressurizer Spray Control Valve 2-PCV-0455C is described in the FSAR. Two separate, automatically controlled spray valves, 2-PCV-0455B and 2-PCV-0455C, with remote manual overrides are used to initiate pressurizer spray.

Summary of Evaluation:
This Safety Evaluation has determined that the described activity of placing Pressurizer Spray Valve 2-PCV-0455C in an non-functional condition in order to stop leakage from the valve packing does not involve an Unreviewed Safety Question or require an amendment to the Technical Specifications. The activity does not increase the probability or consequences of accidents evaluated in the Licensing Basis Documents, or create the potential for an accident no previously analyzed in the Licensing Basis Documents, or reduce any safety margins existing in Tech Spec Bases. This activity does not affect any system used for accident mitigation, and will not affect plant impact or response to a system failure. This evaluation concludes that no credit is taken for the proper operation of the Pressurizer Spray Control Valves during an accident or plant transient and that no credit is taken for the valve in the Fire Safe Shutdown Analysis. Therefore the compensatory action of placing the valve in an non-functional condition does not create an Unreviewed Safety Question.

Evaluation Number: SE- 01-010 **Units: 1 and 2**
Revision 0

Activity Title:
USE HIGH PRESSURE SODIUM PORTABLE UNDERWATER LIGHTS FOR FUEL HANDLING OPERATIONS

Description of Change(s):
DBD-ME-080 and FSAR Sections 9.1.4. and 17 are being changed to describe the use of portable underwater lights and to invoke augmented quality requirements on their use. This change allows the use of Portable High Pressure Sodium Vapor Lights for fuel handling activities.

Summary of Evaluation:
The use of mercury containing lights are subject to inspections, precautions and limitations that assure that nuclear safety is not affected. These include augmented quality inspections prior to use, precautions to minimize the likelihood of damage to the lights and evaluations that demonstrate that nuclear safety would not be compromised in the event mercury is lost. The evaluation concludes that no unreviewed safety question exists and a license amendment is not required. Based on the results of this evaluation, implementation of the proposed activity does not involve an Unreviewed Safety Question.

Evaluation Number: SE- 01-011 **Units: 1 and 2**
Revision 0

Activity Title:
LDCR TB-2000-14 AND LDCR TR 2001-003, USE OF GAMMA-METRICS NEUTRON DETECTORS TO SATISFY TS 3.9.3 AND TR 13.3.32.

Description of Change(s):
The activity changed the relevant licensing basis documents to allow the use of the Gamma-Metrics neutron monitors in place of the two Westinghouse-supplied BF3 source range neutron flux detectors during operations in Modes 3, 4, 5, or 6 when the source range reactor trip function is not required to be Operable.

Summary of Evaluation:
TS 3.9.3 requires two source range neutron flux monitors to be operable during Mode 6 operations. TR 13.3.32 requires 2 channels of Source Range Neutron Flux function to be operable during Modes 3, 4, and 5 when the source range neutron flux reactor trip function is not required. The Westinghouse-supplied source range neutron flux detectors have typically been used to provide these functions. However, the Gamma-Metrics Neutron Flux Monitoring System is functionally equivalent and can be used to provide the same function. The proposed use of the alternate set of neutron detectors to monitor the core reactivity during refueling and other operations during Modes 3, 4, 5 and 6 does not affect the capabilities of any SSCs required to mitigate a licensing basis accident. The accident analyses presented in FSAR Chapter 15 are unaffected by the proposed activity. Therefore, the radiological consequences of the accident analyses are unaffected. Because all accident analyses remain valid and are unaffected by the proposed activity, the bases for all Technical Specifications remain valid. In addition, because the accident analyses are unaffected and because the proposed activity does not affect the failure point of any fission product barrier, the margin of safety is unaffected by the proposed activity.

Evaluation Number: SE- 01-012 **Units: 1 and 2**
Revision 0

Activity Title:
EVALUATE THE ADDITION OF A DYE WITH A TRADEMARK NAME OF AQUASHADE TO THE SSI TO CONTROL ALGAE

Description of Change(s):
This activity involves the addition of a blend of blue and yellow dye (trademark name Aquashade) to the SSI to act as a filter of wavelengths of sunlight to control algae.

Summary of Evaluation:
This Safety Evaluation evaluates the effects of the addition of the AQUASHADE dye to the SSI. The growth of algae during warm weather has caused instances where cleaning of Service Water System strainers has become excessive. As a means of controlling algae growth, the use of a dye to limit sunlight penetration into the Safe Shutdown Impoundment water has been recommended. The dye has no adverse effects on aquatic life, other animals, or plant systems, structures, and components. The dye will only limit algae growth. This evaluation shows that during the time period when the AQUASHADE dye is added to the SSI, the SSI will continue to meet safety functions and licensing basis requirements. This activity does not involve an Unreviewed Safety Question.

Evaluation Number: SE- 01-013 **Units: 1 and 2**
Revision 0

Activity Title:
ADDITION OF MCC'S 1EB2-3 AND 2EB2-3 TO THE LIST OF EQUIPMENT THAT CANNOT BE TESTED AT FULL POWER (DURING RX OPERATION)

Description of Change(s):
Adding MCCs 1EB2-3 and 2EB2-3 to the list of equipment that cannot be tested at full power (during reactor power). This equipment is not tested to prevent damage or upset plant operation as described in FSAR Section 7.1.2.5.

Summary of Evaluation:
Testing of the load shed feature for MCCs 1EB2-3 and 2EB2-3 using an overlap testing methodology has been determined to be an acceptable alternative test method. The probability that the protection system will fail to initiate the final actuated device (MCCs 1EB2-3 and 2EB2-3) is also evaluated and has been found to be acceptably low even though the MCCs are not trip actuated during reactor operation.

Evaluation Number: SE- 01-014 **Unit: 2**
Revision 0

Activity Title:
Re-route of CCW cooling water supply piping to Instrument Air Compressor 2-01

Description of Change(s):
The Component Cooling Water (CCW) supply piping to Instrument Air Compressor (IAC) 2-01 is being re-routed to a configuration similar to the Unit 1 configuration in order to provide supplemental cooling of the CCW water (using the existing trim cooler CP2-CCHXCH-01) to instrument air compressor 2-01. One energize to close, direct acting, solenoid bypass valve and associated piping is being added to the CCW cooling water line to each instrument air compressor, CP2-CICACO-01 and CP2-CICACO-02. Additional vent and drain valves are added to provide adequate fill and venting of each compressor cooling system after maintenance activities. Manual isolation valves are also provided for IAC 2-02 to allow isolation for maintenance. Inlet and outlet temperature indicators are provided for IAC 2-01 for operator convenience.

Summary of Evaluation:
The Instrument Air System is in operation during all normal modes of plant operation. The system performs no safety-related function but is necessary for plant operation. Plant recovery following an emergency is also facilitated by instrument air availability. Unit compressors can be powered from Class 1E bus during UPSET conditions, but will be automatically tripped during a Safety Injection Actuation Signal (SIAS). The Common compressors are available, during normal plant operations, as a backup to the unit compressors when a unit compressor is down for maintenance. Each Unit's Instrument Air System is comprised of two 100% capacity trains. Each train is backed up, during normal operations, by a 100% capacity Common train.

In the exceptional case where all electrical power is interrupted, the air compressors stop operating. Air-operated valves throughout the plant are arranged for safe failure in the absence of air. In this case, the valves are positioned to preserve the safety of plant and personnel. Certain valves are provided with local individual air accumulators. Accumulators are designed and sized to provide the required air quantity for a short period of time after loss of the main instrument air system.

Implementation of the proposed changes will only affect the Unit 2 Instrument Air Compressors (IAC) and will not reduce the margin of safety as defined in the basis of any Technical Specification or safety analysis.

Based on the results of the evaluation, implementation of the proposed activity does not involve an Unreviewed Safety Question.

50.59 Evaluation No. 59EV-1999-002168-01-01 **Units: 1 and 2**
Date Completed: 12/04/2001

Activity Description:
A tracer gas will be used to determine the unfiltered in-leakage into the Control Room when the Control Room HVAC System is in the emergency recirculation mode.

The activity involves the following test procedure.
1. PPT-TP-01C-007: Tracer Gas Test for CR with Train A in operation
2. PPT-TP-01C-008: Tracer Gas Test for CR with Train B in operation
3. PPT-TP-01C-009: Tracer Gas Test for ductwork penetrating the CR Pressure Boundary
4. PPT-TP-01C-010: Tracer Gas Test for damper leakage in the CR with Train A in operation
5. PPT-TP-01C-011: Tracer Gas Test for damper leakage in the CR with Train B in operation

The activity is:

(a) The use of tracer gas equipment in determining unfiltered air in-leakage through various components by utilizing different CR HVAC configurations (alignments).

(b) The use of the tracer gas, Sulfur hexafluoride (SF6), in the control room environment.

(c) A procedure to support the component testing of the ductwork penetrating the CR pressure boundary; PPT-TP-01C-009: A procedure to support the component testing of the dampers leaking with Train A in operation, PPT-TP-01C-010: A procedure to support the component testing of the dampers leaking with Train B in operation; PPT-TP-01C-011: A spatial uniformity test on the Control Room to determine in-leakage with Train A in operation, PPT-TP-01C-007, and a similar test with Train B in operation, PPT-TP-01C-008.

Summary of Evaluation:
The proposed activity is a test to determine the unfiltered in-leakage into the Control Room when the Control Room HVAC System is in the emergency recirculation mode. During the test, each train of the Control Room HVAC System was placed in the emergency recirculation mode and the amount of unfiltered in-leakage was measured using a tracer gas. With the exception of the use of the tracer gas, the test configurations are the same as those used to perform routine surveillance tests to satisfy requirements of the Technical Specifications. Throughout the test, the Control Room HVAC System was either in its emergency mode, or in its normal standby mode, ready to respond as designed to an Engineered Safety Features Actuation Signal.

The use of tracer gas testing methodology in determining control room unfiltered in-leakage will not adversely affect the health of the control room operators. The amount of the inert tracer gas (sodium hexaflouride, SF6) is well below the limits for personnel protection. The tracer gas has also been evaluated for potential adverse effects on control room equipment and the efficiency of the control room filtration system charcoal beds, prefilters, and HEPA filters, and found to have no negative impact.

50.59 Evaluation No. 59EV-1999-002168-01-01

Summary of Evaluation (continued):
Because the Control Room HVAC System was not be used in a condition outside the reference bounds of its operation, and because the tracer gas used to determine the unfiltered in-leakage has no adverse effects on equipment and instrumentation in the control room and does not degrade the performance of the Emergency Filtration System, it is concluded that prior NRC approval of this test is not required.

50.59 Evaluation No. - Rev No. 59EV-2001-000426-01-00 **Units: 1 and 2**
Date Completed: 07/12/2001

Activity Description:
The Subcompartment Pressurization Analysis as described in UFSAR Section 6.2 was performed using the RELAP 4 MOD 5 computer code. A detailed description of the code as well as the input decks is included in the UFSAR. Because of the unavailability of the RELAP 4 MOD 5 code, the COMPARE MOD 1 computer code is being considered as the tool to perform these subcompartment analyses. This activity involves the use of the COMPARE MOD 1 code in Subcompartment Pressurization Analysis as described in section 6.2 of the UFSAR.

Summary of Evaluation:
COMPARE MOD 1 has been used in similar applications at CPSES. The COMPARE MOD 1 computer code is the same code used in the subcompartment environmental/pressurization analyses as described in the CPSES FSAR Section 3.6B.1.2.3 and in the Tornado venting subcompartment pressurization analyses as described in FSAR Section 3.3.2.3. The application of COMPARE is similar to the applications currently presented in the FSAR in that it is used to determine time dependent pressures and temperatures in models involving air volumes and restricted flowpaths. These applications include subcompartment environmental analyses involving pipebreaks outside of the containment and the tornado venting analyses. This application of the COMPARE MOD 1 computer code is well within its capabilities and normal intended use. The COMPARE computer code has also been used in this and similar applications by the NRC in its evaluation of the original CPSES design and by at least one other utility.

Detailed input data depicted in the CPSES UFSAR for containment subcompartment analysis was executed using the COMPARE MOD 1 computer code. The output of these runs was compared with the original RELAP 4 MOD 5 analyses output. The results of these benchmark analyses compare reasonably well.

The 10CFR50.59 evaluation concludes that this activity does not result in a departure from the method of evaluation described in section 6.2 of the UFSAR. Because this activity does not result in a departure from a method, a license amendment is not required.

50.59 Evaluation No. 59EV-2001-001870-01-00 **Unit: 2**
Date Completed: 08/21/2001

Activity Description:
Unit 2 Power Operated Relief Valve (PORV) 2-PCV-455A has been observed to have minimal seat leakage as evidenced by increased tailpipe temperatures which resulted in it being declared INOPERABLE and isolated in accordance with TS 3.4.11. No other operational impacts from this slight seat leakage have been experienced. PORV Block Valve 2-8000A is being maintained closed to prevent further seat degradation on the associated PORV. The proposed clearance would be based on a conclusion that the existing minimal seat leakage is not excessive and does not render the PORV inoperable. Since the FSAR and TS do not address isolating a degraded but Operable PORV with a closed block valve, a 50.59 screen on the proposed compensatory action (Shift Manager Clearance tracking Closed Block Valve) taken in response to the degraded condition (seat leakage greater than zero) was performed. It concluded this would be a change to the procedures as described in the Updated FSAR.

It is assumed that the clearance would be in effect until the next refueling outage during Modes 1 ,2 and 3. Keeping the block valves closed is necessary to prevent further degradation of the valve seat which would ultimately become excessive.

Summary of Evaluation:
The only impact of operating with a single Operable PORV block valve closed and isolating an Operable PORV is an increase in the likelihood of challenging a Pressurizer Safety Valve. The probability of a safety valve sticking open if challenged is not affected. The probability of being challenged is increased with the block valve closed. Because the other PORV block valve is open and its PORV is capable of automatic operation, one-half of the design function (the other PORV) is available to automatically prevent challenges to the safety valves. Because the block valve is also Operable and the operator can open it in response to any transient, 2-PCV-455A should also still be available after a short delay. Therefore, it can be concluded that the increase in the likelihood of challenging a safety valve is less than a factor of two. In accordance with Section 6.2.2 of the 10 CFR 50.59 Resource Manual, this increase is minimal. Therefore, a License Amendment is not required.

50.59 Evaluation No. 59EV-2001-000751-01-00 **Units: 1 and 2**
Date Completed: 02/07/2002

Activity Description:
Revise DBD-ME-013 Attachment 1, FSAR Table 6.2.4-2, TRM Table 13.6.3-1, and IST Table 18 and 19 to remove Type C Leak rate testing from MOVs 8701A and 8701B (RHR Pump Hot Leg Recirc (OMB) Isolation Valves) and RHR Hot Leg Suction relief valves 8708A and 8708B.

Summary of Evaluation:
These valves do not provide a direct connection between the inside and outside atmospheres of the primary containment during normal operation. They are not required to close automatically upon receipt of a containment isolation signal. They are not required to operate intermittently post-accident. These are the criteria for Type C tests for PWRs per 10CFR50, Appendix J, II.H.

The technical basis for the change is that the containment isolation function would continue to be performed even if there were leakage through the subject containment isolation valves. The design of the penetration is for the closed system outside containment to absorb any leakage and ensure that radiological consequences are within acceptance limits. An effective fluid seal on these penetrations is provided by the suction sources to the residual heat removal pumps during and following an accident. The RHR is a closed system outside containment post-LOCA which is monitored in accordance with TMI Section III.D.1.1. This seal is assured even in the event of a single active failure. Therefore, the change may have some limited effect on the Containment Isolation Valves containment function (since testing is relaxed) but will not significantly affect the Containment Isolation Design Function.

The design provides the water barrier which prevents leakage of the containment atmosphere. Therefore, there is no need to test the valves for air leakage. The Nuclear Safety Function of Containment isolation is performed without reliance on local leak rate testing and there is no effect on the off-site radiological dose analysis. Any water leakage is more than bounded by the TMI III.D.1.1 monitoring which is unchanged. The Appendix J acceptance criteria which is based on the radiological safety analyses is not affected by the change. Therefore, this change does not increase the consequences of an accident previously evaluated in the FSAR and any radiological consequences resulting from this change would be negligible.

Therefore, a License Amendment is not required and this change may be implemented without NRC approval.

50.59 Evaluation No. 59EV-2002-000587-01-00 **Units: 1 and 2**
Date Completed: 03/27/2002

Activity Description:
Five NRC-approved methods of evaluation were used by Westinghouse to support their ZIRLO-clad fuel assembly designs to be used at CPSES. These methods replace similar methods of evaluation previously approved for use at CPSES. Three of these methods of evaluation were referenced as the basis for Technical Specification Amendment 95 dated March 26, 2002. The other two approved methods of evaluation directly support the ZIRLO-clad fuel assembly design to be used at CPSES. The application of these two methods to the CPSES reload fuel of Westinghouse design (beginning with Unit 2 Cycle 7) and the change to UFSAR Section 4.2 and Apppendix 4B to reflect these new methods are the subject of this evaluation.

Summary of Evaluation:
Two of the NRC-approved methods of evaluation used by Westinghouse to support their ZIRLO-clad fuel assembly designs at CPSES are different from those methods described in the CPSES UFSAR. These methods are: "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, March 1995; and, "Westinghouse Improved Fuel Performance Analysis and Design Models (PAD 4.0)," WCAP-15063-P-A, Revision 1, with Errata, July 2000. Both of the above referenced Westinghouse methodology reports have been approved by the NRC for use by the fuel vendor for the Westinghouse fuel designs used at CPSES. The revised Westinghouse methods as presented in the WCAPs are not a departure from the methods described in the UFSAR because they have been approved by the NRC for the intended application. Therefore, the application of these methods to CPSES fuel of Westinghouse design does not represent a "departure from a method of evaluation," and prior NRC approval is not required.

50.59 Evaluation No. 59EV-2002-001634-01-00 **Units: 1 and 2**
Date Completed: 04/26/2002

Activity Description:
Four Thermal Relief Valves (namely 1/2CT-0005 & 1/2CT-0056) are to be temporarily gagged during plant operation as a compensatory action for a degraded and non-conforming condition. The basic function for these valves is an ASME CODE requirement to protect the tube side of heat exchangers CP1/2-CTAHCS-01 & CP1/2-CTAHCS-02 and associated piping from overpressurization due to thermal expansion. This could occur during maintenance activities and where isolation of the heat exchangers may be required. The basic Nuclear Safety Function of these valves is to maintain the system pressure boundary during and after an accident and it is this function which is degraded and requires compensatory action. The CT system will perform its Nuclear Safety Functions with the valves gagged. Gagging has no effect on system performance. Gagging these valves will reduce the likelihood of a loss of pressure boundary integrity in the CT system due to a stuck open relief valve and ensure the consequences of an accident or malfunction are not increased. Therefore, this function does not require evaluation under 10CFR50.59.

The design function of the relief valves to protect the tube side of heat exchangers from overpressurization, is a requirement per ASME Code Section III Subsection NC/ND 7000, RG 1.67, "Installation of Over Pressure Protection Devices", and ASME Code Case 1569. This function may be called upon during maintenance activities where an unlikely event may be initiated for thermal and pressure transients to occur in the system. In such a scenario, the relief valves may exceed their set pressure and consequent valve opening would prevent overpressurization. Gagging the valves would be adverse and is the subject of this evaluation. Therefore compensatory action to gag subject thermal relief valves is closely controlled by issuance and implementation of design change documents with appropriate notes added to affected vital station drawings. Plant operation and maintenance personnel working to these vital station drawings are required to remove the gags from the relief valves when maintenance is being performed on the heat exchanger or the heat exchanger is to be isolated for any reason.

FDA-2002-002189-01-00 will revise affected vital station drawings (flow diagrams) to identify the gagged valves and to provide a precaution.

Summary of Evaluation:
Compensatory action to gag thermal relief valves in the containment spray system for both units will be in place due to the recent failure of two of the valves to reseat following lifting during the recirculation of the RWST. Historical and plant computer data shows that when the valves lift during the recirculation of the RWST, pressure is not restored to levels low enough that would allow the thermal relief valves to reseat and thereby allowing flow from the Containment Spray (CT) system into Vents and Drains [See QTE-2002-002189-01-01]

50.59 Evaluation No. 59EV-2002-001634-01-00

Summary of Evaluation (continued):
The thermal relief valves, 1/2CT-0005 and 1/2CT-0056, have two design functions:

1. In accordance with ASME III and RG 1.29, the relief valves maintain system pressure boundary during operation of the containment spray system. [FSAR Section 6.2.2.1 and 6.2.4.1.5]
2. In accordance with ASME III thermal overpressure protection for the containment spray heat exchanger when it is isolated (valved out) for maintenance. [FSAR 6.2.2.2.1 and ASME III]

Gagging the subject thermal relief valves to maintain system pressure boundary during and after an accident is the compensatory action. The effect on ASME Code overpressure protection is the only other design function affected.

Although, gagging the valves would increase the likelihood of an inadvertent overpressure if the component were improperly isolated, the chance of this happening is negligible and would not result in more than a minimal increase for the component because the sequence of necessary events would be highly unlikely. For this to occur, not only would a gross mistake have to occur, it would have to go undetected and it would have to have been so severe as to bring the component to the point of failure such that it would not be detected until after it were placed in service. Such a sequence of unlikely events could not cause a more than minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety.

It is concluded that these compensatory actions may be implemented in accordance with site procedures without NRC approval.

Attachment 2 to TXX-02146
CMCE's: 01-01
01-03

CMCE: 01-01
Commitment Number: 14848
Change Type: Deletion

Source Document:

NRC INSRPT-445/8514
TXX-4779

Original Commitment Description:

Chemistry section forms have been evaluated. The following improvements were identified and will be incorporated:

* All data sheets associated with the CHM-500 series procedures will have the same general lay-out;

* Parameter limits will be highlighted in red ink; (Note: Per conversation with Senior NRC Resident Inspector, Bill Jones, it was concluded that a letter to the NRC is not required since a 50.59 review was performed for chemistry procedure CHM-109, and parameter limits do not have to be highlighted in red ink.)

* Sample data and time columns will be clearly delineated;

* Where applicable, a chemical addition column will be included; and

* Each form will include the following note, "Circle out-of-specification parameters."

Chemistry data sheets (forms) associated with the following procedures are being revised:
CHM-501 CHM-502 CHM-503 CHM-505 CHM-506
CHM-508 CHM-509 CHM-510 CHM-511 CHM-517 and CHM-519

Revised Commitment Description:

Commitment is deleted no description revision required.

Justification for Change:

The commitment was deleted based upon system data management maturity. The commitment description captured near-term-operation and operation requirements using manual graphs, charts, procedures and other paper/pen derived data capture methods. The current paperless, electronic format will maintain the intent of the commitment, but does not either rely on the same capture techniques or display formats.

CMCE: 01-03
Commitment Number: 26528
Change Type: Revision

Source Document:

SSER-26
TXX-01026
SECY-00-045
NEI-99-04

Original Commitment Description:

The residual heat removal (RHR) system piping on the discharge side of RHR pumps 1 and 2 contains a total of approximately 250 welds on 8-inch and 10-inch NPS Schedule 40 piping with wall thicknesses of 0.322 inch and 0.365 inch, respectively. The PSI examinations performed by the applicant were on the suction side of the RHR pumps where the 12-inch and 16-inch NPS pipe wall thicknesses are equal to or greater than 0.375 inch. When developing the inservice inspection plan, the applicant should consider redistributing the 7.5 percent sample to include volumetric examination of welds on the discharge side of the RHR pumps.

Revised Commitment Description:

In lieu of selecting 7.5% of the welds from the "thin wall" (0.375") discharge piping for the RHR and CT pumps, the selection will be based on the EPRI R1-151 methodology. When developing the inservice inspection plan, the applicant should consider redistributing the 7.5% sample to include volumetric examination of welds on the discharge side of the RHR pumps.

Justification for Change:

The commitment identified in NRC SSER-26 was not a mandatory commitment (reference SECY-00-045 and NEI-99-04). The selection of welds will be based on the EPRI R1-151 methodology.