Illinois Power Company
Clinton Power Station
P.O. Box 678
Clinton, IL 61727
Tel 217 935-5623
Fax 217 935-4632

John G. Cook
Vice President

ILLINOIS POWER

U-602500
L48-95(10-18)LP
1A.120
JGC-435-95
October 18, 1995
10CFR50.71(e)

Docket No. 50-461

Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station Submittal of the Updated Safety Analysis Report, Revision 6

Dear Sir:

Pursuant to 10CFR50.71(e), Illinois Power (IP) hereby submits one signed original and ten copies of Revision 6 to the Clinton Power Station (CPS) Updated Safety Analysis Report (USAR).

In accordance with 10CFR50.71(e)(2)(ii), the attached Clinton Power Station 10CFR50.59 Report (Attachment 2) identifies all safety evaluations for implemented changes affecting the USAR for the period of December 10, 1993 through April 29, 1995. These safety evaluations were performed by IP under the provisions of 10CFR50.59.

The revised USAR pages (Attachment 3) are annotated with Revision 6 and revision bars. Subsequent revisions of the USAR will be submitted within six months of the completion of a CPS refueling outage and will reflect all changes up to a maximum of six months prior to the date of filing.

As required by 10CFR50.71(e)(2)(i), the information given in the attached USAR accurately presents changes made since the previous submittal and analyses submitted to the Commission or prepared pursuant to Commission requirement.

Sincerely yours,

250002

J. G. Cook
Vice President

GBS/csm

A053 1/11

9510250398 951018
PDR ADOCK 05000461
K PDR

Attachments

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety

Attachment 1
to U-602500

J. G. Cook, being first duly sworn, deposes and says: That he is Vice President of the Nuclear Program at Illinois Power; that the Clinton Power Station Updated Safety Analysis Report, Revision 6, has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said letter and the facts contained therein are true and correct.

Date: This 18 day of October 1995.

Signed: J G Cook
J. G. Cook

STATE OF ILLINOIS } SS.
DeWitt COUNTY }

" OFFICIAL SEAL "
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/97

Subscribed and sworn to before me this 18th day of October 1995.

Jacqueline S. Matthias
(Notary Public)

Attachment 2
to U-602500

CLINTON POWER STATION 10CFR50.59 REPORT

Log No. 91-0119

Modification WX-F010, Level Instrumentation for the Solid Radwaste System Tanks

Modification WX-F010 replaces the continuous ultrasonic level indicating system with an on-demand (plumb bob level) system and a fixed high level probe for the concentrate tank (0WX01TA) and the waste sludge tank (0WX02TA). This modification prevents the tanks from overflowing. USAR Figure 9.3-2 is being revised to reflect this change. Also, USAR section 7.7.1.12.3.1 is being revised to reflect that the tanks will alarm on high water level only. This change does not impact the probability or consequences of the Radwaste Tank rupture event.

Log No. 92-0034

Modification DG-F047, Install Vents for Diesel Generator Heat Exchanger Chemical Injection

Modification DG-F047 installs vents on the diesel generator heat exchangers, using existing spare connections on the tube side of the shutdown service water system heat exchangers. These vents allow injection of chemicals to control microbiologically-induced corrosion of the heat exchanger tubes when the heat exchangers are in the standby mode, with no flow passing through them. The vents are designed to the same design requirements as the existing shutdown service water system piping. Therefore, this change does not increase the probability of a failure of the diesel engine jacket water cooler as identified in USAR Table 9.5-8.

Log No. 92-0046

Modification MS-F013, Moisture Separator Reheater Relief Valve Setpoint Change

Modification MS-F013 changes the main steam (MS) system moisture separator reheater (MSR) relief valve setpoints to match the as-built configuration in the plant. During installation of the MS relief valves, 1B21F309A,B,C,D, the valves were switched between MSRs #1 and 2 per designer recommendations. However, the affected design documents were not revised. The MSR relief valves are installed on the two MSRs for overpressure protection. This change revises the pressure setpoints and capacities to match the as-built details. No new equipment failures are created by the change, since the equipment will operate as originally designed.

Log No. 92-0092

Modification SS-F017, Security System Modification

Modification SS-F017 installs a security system upgrade to the perimeter intrusion detection and assessment equipment as well as portions of portions of the roadway and perimeter fence. This change improves the security system reliability to reduce the false and nuisance alarm rate. Installation of new equipment reduces operating and maintenance costs. No new failure modes have been determined to be created by the change. The change will not reduce the effectiveness of the security plan under 10CFR50.54 (p).

Log No. 92-0125

Tagout 91-0618, Removal of Laboratory Humidification Unit from Service

Tagout 91-0618 removes from service indefinitely the laboratory heating and ventilating system humidification unit (0VL013B). Condition Report 1-91-05-012 identified the potential for a rupture of high-pressure steam piping which could result in temperatures in the immediate area to exceed the design basis limits for temperature and humidity. Removing the unit from service prevents this potential failure. Portable humidifiers will be used in the chemistry lab to add humidity to the area during low humidity periods.

Log No. 93-0001

Modification SA-F009, Service Air Upgrade for Moisture Removal

Modification SA-F009 installs two new stainless steel volume tanks between the service air compressor and the air dryer skids. This modification prevents moisture buildup and degradation of the air dryers. The current configuration is not performing adequately and the existing drain traps require frequent maintenance and replacement. The change in material to stainless steel help eliminates buildup and carryover of rust into the prefilters and dryers. No new failures are created by the change. The new tanks are passive in nature and the new drain traps improve the draining capability of the collected moisture and improve the reliability of the service air system dryers. The probability of the loss of instrument air is not increased by this change.

Log No. 93-0005

Modification A-157, Installation of Women's Rest Room in Radwaste Building

Modification A-157 installs a women's rest room in the radwaste building. USAR section 9.4.9.2 and Figure 9.4-16 were revised to reflect the inputs to the machine shop ventilation system. No new failure modes have been created by this change.

Log No. 93-0043

Modification WS-020, Replace Service Water System Flow Control Valve Bodies

Modification WS-020 replaces the existing service water system flow control valves with severe service valve bodies. The change prevents cavitation damage to the valves. Valves replaced are 1WS079B, 1WS079C, 2WS079A, and 2WS079C. The flow control valves control the flow rate through the plant chilled water system chillers. Only the valve bodies are being changed; the valve controllers, actuators, and air supplies remain unchanged. The change has no impact on any safety related equipment. No new failures have been created by the change.

Log No. 93-0066/94-0029

Modification VP-024, Change Actuator Gears and Motor Operator Spring Pack on Drywell Cooling System Containment Isolation Valves

Modification VP-024 changes the actuator gears for the following drywell cooling system (VP) containment isolation valves: 1VP004A, 1VP004B, 1VP005A, 1VP005B, 1VP014A, 1VP014B, 1VP015A and 1VP015B. The gear ratio has been changed from 36.5:1 to 62.5:1 that result in a corresponding change in valve closure time. The new calculated closing time is 86 seconds and the new in-service inspection (ISI) time is 129 seconds. The valve spring packs are being replaced by a newer type. These changes result from NRC Generic Letter 89-10 requirements. Supplement 1 to VP-024 changed the gear ratio from 62.5:1 to 52:1. The resulting closing time changed from 86 seconds to 74 seconds. The new ISI limit is 84 seconds. These times will be reflected n a change to USAR Table 6.2-47. The increased valve stroke time is acceptable because the closing times for these valves are not an input to the off-site dose calculations following an accident. The closure times also meet the intent of Regulatory Guide 1.141 and ANSI Standard N271-1976. Therefore, this change does not increase the consequences of an accident.

Log No. 93-0079

Modification CC-F007, Component Cooling Water System Filter Demineralizer Unit

Modification CC-F007 installs two prefilter assemblies, one demineralizer unit and miscellaneous monitoring instrumentation in the component cooling water (CC) system. Temporary Modifications 87-018 and 90-028 are eliminated by this modification, which will maintain CC system water quality within system specifications by reducing chloride concentrations and conductivity. Flooding, fire loading and resin intrusion were evaluated and no new failures were identified.

Log No. 94-0001

Modification VC-F015, Control Room Ventilation During Station Blackout

Modification VC-F015 provides main control room cooling during a station blackout using a gas-operated 5000 cubic feet per minute portable fan and associated duct work. In the event of a station blackout, hot air from the main control room will be exhausted by setting up the portable fan and duct work. In addition, doors leading to the main control room will be opened to provide a flow path for cooler outside air. Station off-normal procedures will be revised to provide specific actions for setting up the fan and ductwork. No other accidents are assumed to occur during the event; therefore, no increased exposure to the main control room operators is assumed to occur. The fan will be stored in a fire-proof cabinet away from the main control room and any other equipment important to safety to minimize the potential for a fire hazard.

Log No. 94-0004

Temporary Modification 94-04, Install Temporary Piping in Fire Protection System

Temporary Modification 94-04 removes fire protection (FP) system valve, 0FP015, and installs a temporary spool piece in its place. Also, two other valves, 0FP073 and 0FP271, are being replaced with temporary blind flanges. The change is necessary to restore the FP system following a break in the underground portion of the system. The FP system will remain in this configuration until replacement valves can be obtained and installed. The areas affected by the change include the makeup water pump house, the operations gate house, the NRC Resident Inspectors' office, and the service building. Compensatory measures have been taken to ensure that a fire in one of these areas can be extinguished. The only areas that contain safety related equipment or important to safety items are located in the service building. The record storage vault and the security alarm station are equipped with independent halon fire suppression systems. This temporary modification will not affect the safe shutdown analysis.

Log No. 94-0005

Temporary Modification 94-01, Insert Blind Flange and Install Electrical Jumpers on the Continuous Containment Purge System

Temporary Modification 94-01 inserts a blind flange in the suction of continuous containment purge (VR) system fan, 1VR06CB, and electrical jumpers in the fan start circuitry to allow repair of the fan. The blind flange (galvanized sheet metal) will be inserted in suction line common to 1VR06CA and 1VR06CB. This will allow the running fan, 1VR06CA, to continue to operate during the repairs. This temporary modification reduces the redundancy of the containment purge system supply fans, since there will not be an automatic start of the standby fan. Testing performed in 1988 showed that airborne contamination levels in the containment remained below the levels required for radiological posting, thereby maintaining habitability and access to the containment.

Log No. 94-0007

Modification RI-040, Change Motor Operator for Valve 1E51-F045

Modification RI-040 replaces the motor operator on the steam supply valve, 1E51-F045, to the reactor core isolation cooling (RCIC) system turbine. The new motor provides higher starting torque (25 ft-lb. compared to 15 ft-lb.) than the existing motor. The increased starting torque results from more stringent requirements from NRC Generic Letter 89-10. The increased motor size affects the total electrical load for the Division I DC system; however, the increased load is still within the equipment design rating. The RCIC system capabilities are unaffected by the change.

Log No. 94-0009

Install Inlet Pressure Indicators for Steam Jet Air Ejectors

This change installs new pressure gages on the steam supply to the offgas system steam jet air ejectors. The gages will provide local indication to the operators to assist in putting the air ejectors into service. This change does not introduce any new type of failure and does not increase the probability of a loss of condenser vacuum.

Log No. 94-0010

Remove Continuous Loading Columns from DC Load Tables in USAR

This change removes the continuous DC load columns from the USAR associated with amperage requirements for the first four hours following a design-basis accident. USAR Tables 8.3-8, 8.3-9, 8.3-10 and 8.3-11, identify the load requirements for all four divisional batteries. A second change increases to the load value for emergency lighting. This change is corrective action to condition report 1-93-02-011. A third change updates the recharge time for the Division 1, 2, and 4 batteries from 10 to 12 hours. This change is consistent with IEEE Standard 946. These changes do not impact the capability of the DC system to perform its safety function in the first four hours following a design-basis accident.

Log No. 94-0011

Modification FP-088, Fire Pump Idle Speed Capability

Modification FP-088 installs the capability for the operator to manually reduce the speed of the diesel driven fire pumps, 0FP01PA and 0FP01PB, from rated speed to idle speed following testing. This capability will extend the life of the diesels and increase their reliability. Operability testing of the diesel fire pumps is demonstrated weekly by automatically starting the diesel. The shutdown procedure requires a manual shutdown at 80 percent load and 100 percent speed. This modification allows for a gradual shutdown of the diesel. While operating in this mode, the automatic start function of the diesel will not be available. Should a fire occur while operating in this mode, the operator will be capable of returning the engine to rated speed, if required. These changes will not adversely affect the ability to achieve and maintain safe shutdown of the plant, should a design-basis fire occur. Administrative controls will ensure that the throttle valve is not inadvertently mispositioned.

Log No. 94-0013

Manual Closure of VR/VQ Containment Isolation Valves

This change allows containment ventilation and purge (VR/VQ) system containment isolation valves to be manually assisted closed as long as they are not opened during conditions when containment integrity is required by technical specifications. Once closed, the valve will remain closed and tagged shut. Should any of the valves be reopened, the valve would be declared inoperable and would require the appropriate technical specification action to be followed. The valves affected by this change are 1VR001A, 1VR001B, 1VR004A, and 1VR004B.

These valves are not opened during operating modes 1, 2 or 3. These valves may be considered manual valves when the valve is maintained closed under administrative control. No new failure modes will be created by this change, since the safety function of these valves has not been affected.

Log No. 94-0014

Change Valve Stroke Times for RCIC Valves

This change increases the valve stroke times for the reactor core isolation cooling (RCIC) system turbine steam inlet and steam bypass valves, 1E51-F045 and 1E51-F095. The opening time for 1E51-F095 has been increased from 10 to 13 seconds. Both valves required closing time has been changed to 60 seconds. The changes are a result of analyses performed in response to NRC Generic Letter 89-10. The change in the opening time is acceptable since the design basis requirement of the RCIC system to be able to inject water to the reactor at a flow rate of 600 gpm within 30 seconds. The 60-second closing time is to provide a basis for in-service inspection trending purposes. No new failures are assumed to occur since the safety function of the RCIC system has not been changed.

Log No. 94-0015

Modification DG-F047, Install Vents on Division 1 Diesel Generator Heat Exchanger

Modification DG-F047 was originally prepared to install vent valves on all three emergency diesel generator heat exchangers to allow injection of chemicals to control microbiologically-induced corrosion when the heat exchangers are in the standby mode. The modification was installed only on Division 1 in 1992 and subsequently it has been determined that chemical injection without system flow is not advantageous. Thus, the modification will not be installed on Division 2 and 3. This safety evaluation supersedes the previous safety evaluation performed in 1992. The conclusion is that installation of the vent valves on only one division does not create any new failures.

Log No. 94-0016

Modification SY-011, Addition of Third Terminal for Latham 345KV Line

Modification SY-011 enhances the capability of the Latham 345kV transmission line for the CPS offsite power system. This modification prevents unacceptably low voltages throughout Decatur during 1994 summer loads during first contingency outages. Also, it was installed to prevent Decatur area electrical system blackouts during various double contingency outages. In addition, the transmission system modification provides assurance of acceptable electrical supply to Illinois Power's largest industrial customers. The modification consists of adding a third terminal to the existing two terminal 345kV transmission lines. The modification also changes the transmission protective relays and their associated communication equipment. USAR Figures 8.2-4, 8.2-5 and 8.2-8 through 8.2-18 and Tables 8.2-1 through 8.2-4 are revised; however, future changes to these tables and figures will only be made as major system changes warrant. This modification enhances the reliability of the offsite power system.

Log No. 94-0017

Delete Ampacity Requirements for Control Cables

This change deletes control cables from being required to be included in thermal ampacity calculations. Deleting control cables from these calculations is acceptable because analysis has shown that the heat generated from control cables is negligible compared to power cables. This change allows control cables to be added to cable trays without having to re-perform thermal ampacity calculations, while only having to evaluate for dead weight and seismic acceptability. No new failures are created by this change since the heat generated from control cables is not high enough to generate a fire.

Log No. 94-0019

Modification TE-F006, Auxiliary Building Ventilation System Cooling Coil Drains

Modification TE-F006 provides a drainage path for condensate from plant cooling coils to be connected directly to the sewage treatment system. The cooling coils affected are the containment building ventilation air supply, the auxiliary building ventilation air supply, and control building ventilation area cooler. This change permits the condensate to be treated as non-contaminated and reduces the input to liquid radwaste treatment systems. The drainage has been evaluated to having no potential radioactivity and can be treated as being non-contaminated. No new release pathway has been created by this change.

Log No. 94-0023

Design Basis Requirements of the Drywell and Containment Hydrogen/Oxygen Monitoring System

This change revises the USAR description of the minimum design basis requirements for the drywell and containment hydrogen/oxygen monitoring system operation. The change defines the minimum number of sample points required to be maintained to ensure the minimum requirements for plant technical specifications are being met. This change does not detract from the system's capability to monitor the drywell and containment following a design basis accident. This change does not reduce CPS's commitment to NUREG-0737, "Post TMI Requirements."

Log No. 94-0024

Modification IP-F004, Modification to NSPS Inverters

Modification IP-F004 makes three changes to the nuclear system protection system inverters. The first change improves the system's capability to withstand DC system voltage transients by increasing the capacitance of the input filter capacitor and adding a blocking diode on the input bus. The second change qualifies the system to operate at a low input voltage (less than 100 vdc) compared to the present trip setpoint (102 vdc). The third change installs a maintenance switch to allow the Division I and II inverters to be removed from service for inverter calibration and maintenance. This change improves the overall reliability of the NSPS power supply (including inverter fuses). The changes ensure that the system will function as designed during all analyzed events.

Log No. 94-0025

CPS Emergency Plan Revision

Advance Change Notice 10/3 of the CPS Emergency Plan implements administrative changes as a result of the 1993 review of the emergency preparedness program. The scope of the changes includes updates to letters of agreement with outside agencies, organizational changes, revision of the definition of "emergency action level" for consistency with other emergency plan implementing procedures, and training requirements for respiratory protection. The changes have been determined to not decrease the effectiveness of the CPS Emergency Plan per 10CFR50.54 (q).

Log No. 94-0026

Modification M-073, Snubber Reduction Program (PS Subsystem)

Modification M-073 removes snubbers from the process sampling (PS) system. The piping subsystem affected by the change is the PS-10 subsystem. The lines are liquid reactor sample lines which form part of the post-accident sampling system. These lines may also be used to obtain reactor samples during normal operation. Three snubbers and associated hardware are being removed and portions of the snubber attachments are being retired in place. The operability of the PS system remains unaffected after the change. The piping stresses remain within ASME code allowable values, therefore, no new failures have been created.

Log No. 94-0027

Modification M-073, Snubber Reduction Program (MS Subsystem)

Modification M-073 removes snubbers from the main steam (MS) system. The six piping subsystems affected by the change are the following: 1H22P015A, 1H22P025B, 1MS84, 1MS85, 1MS86, and 1MS87. The lines are instrument sensing lines or are instrument air lines that supply the main steam isolation valve air actuator assemblies. The modification eliminates snubbers and associated hardware, and portions of the snubber attachments are being retired in place. The operability of the MS system remains unaffected after the change. Since the instrument sensing lines affected by the change are smaller than one-inch in diameter, they are exempt from pipe break analysis. The instrument air lines are not considered high-energy lines. Since, the piping design stresses remain within ASME code allowable values, no new failures have been created.

Log No. 94-0028

Revision to Turbine Valve Testing Requirements

This change revises the frequency of testing the main steam turbine stop valves, combined intercept valves, and the turbine control valves from their current frequency (weekly or monthly) to as recommended by the manufacturer. The reason for this change is to provide greater flexibility in determining the testing frequency based upon plant experience and maintenance history. The testing frequency had been removed from the plant technical specifications in License Amendment 60 due to the turbine's favorable orientation for potential missile hazards. This change does not increase the probability or consequences of an equipment malfunction important to safety

Log No. 94-0030

Modification RD-025, New Control Rod Blade Design

Modification RD-025 allows for the replacement of existing control rod blades with control rod blades of a newer design. The new type of control rod blade is Duralife-230, which has been used in boiling water reactors (BWRs) since 1987 and in BWR-6 reactors since 1992. The NRC topical report that approved the design was issued in May 1988. The overall rod weight is unchanged, but the velocity limiter is lighter because of the additional weight resulting from the additional hafnium absorber plate installed in the top six inches of the blade. This new design increases the life of the blade and resists irradiation-assisted stress corrosion cracking. Since the overall weight is unchanged, the rod drop velocity acceptance limit of 3.11 ft/sec is unchanged. Therefore, there is no reduction in the margin of safety.

Log No. 94-0031

Modification RR-030, Replace Position Transducers and Eliminate Automatic Load Following

Modification RR-030 consists of two basic changes to the reactor recirculation (RR) system. The first change involves the replacement of the flow control valve position transducer from a linear variable differential type to a rotational variable type. The second change involves the elimination of automatic load following which is a means of controlling reactor power through turbine generator demand. The change in the flow control valve position transducers results in more reliable operation of the flow control valve since rotational variable transducers are directly coupled with the valve stem. The automatic load following circuitry has never been utilized and this change eliminates this capability. Failures of the flow control valve are not affected by this change. The changes do not affect the maximum stroke rate allowed by technical specifications.

Log No. 94-0032

Modification AS-017, Install New Relief Valve Discharge Line to Main Condenser

Modification AS-017 installs a new pressure relief valve in the auxiliary steam (AS) system deaerating steam line, an equalization line around the auxiliary steam isolation valve to the turbine building, and upgrades the main steam system piping for moisture separator reheater steam blanketing. The changes will not increase the potential for equipment failure since the piping design codes for other piping systems connecting to the main condenser are used. The potential for additional air in-leakage to the main condenser, and the potential effects of a loss of condenser vacuum have also been evaluated.

Log No. 94-0033

Change Frequency of Fire Detector Channel Functional Testing

CPS Procedure No. 9337.81, "Fire Detector Channel Functional," is being revised to change the frequency of fire detection system testing from every six months to every two years. While no change to the USAR is required, the National Fire Protection Association code conformance evaluation is being revised to reflect the change in frequency. A review of performance records of the fire detection system reliability indicates an extremely high reliability rate. This change will eliminate unnecessary testing of the system. This change will not degrade the fire protection system, nor will it impact the safe shutdown analysis.

Log No. 94-0034

Change Frequency of Fire Protection Flow Alarm Testing

CPS Procedure No. 3822.01, "Fire Protection Water System Flow Alarm Test," is being revised to change the frequency for sprinkler flow alarm testing from every two months to once annually. An evaluation was performed of the testing results over the past six years and it was determined that the five types of sprinkler systems, automatic wet pipe, automatic preaction, manual preaction, automatic deluge, and manual deluge, have a highly reliable history of operation and testing. The change to the testing frequency will not affect the ability to safely shutdown the plant in the event of a fire.

Log No. 94-0035

Planning and Scheduling Organization Responsibilities

This is an organizational change dealing with the description of the Planning and Scheduling of CPS. The changes are entirely administrative in nature and involve title changes and more clearly define group responsibilities.

Log No. 94-0036

Modification OG-F016, Extend Nitrogen Connection Outside HEPA Filter Room

Modification OG-F016 extends the nitrogen purge piping for the offgas treatment (OG) system HEPA filter room from inside the room to an existing connection outside the room. The modification is being installed to increase accessibility of nitrogen purge connection in the event of an OG charcoal adsorber fire. This change also reduces radiation exposure to plant personnel. The change will not increase the probability of an offgas system failure.

Log No. 94-0037

Modification RH-042, Change Actuator Motor Size for Containment Spray Valves

Modification RH-042 changes the motor operator size for residual heat removal (RH) system containment spray isolation valves, 1E12F028A and 1E12F028B, from 10 ft-lb. to 25 ft-lb. The change is necessary for the valves to meet more conservative factors resulting from NRC Generic Letter 89-10. The piping system has been evaluated for the increased weight due to the change in motor and valve yoke size. The change in valve stroke time has been determined to be insignificant. The change does not impact the operability of the RH containment spray system.

Log No. 94-0038

Change Operator Actions for Abnormal Reactor Coolant Flow

CPS Off-Normal Procedure No. 4008.01, "Abnormal Reactor Coolant Flow," is being revised to include BWR Owners Group guidelines regarding thermohydraulic stability, provide additional instructions for determining reactor core flow during single loop operation, and during conditions when the reactor recirculation flow control valve may be drifting open. The first change is acceptable since the BWR Owners Group recommendations do not violate technical specifications for operation within the restricted zone of the power-to-flow map. The second change allows the operator to determine core flow using a computer point that measures core plate differential pressure. The third change provides conservative instructions to the operator in response to a drifting open flow control valve. The changes do not decrease any margins to fuel thermal limits; therefore, there is no reduction to any safety margins.

Log No. 94-0039

Completion of Protective Action Once Initiated

USAR Section 7.3.2.4.3.1.16 discusses the containment spray system compliance to IEEE 279 paragraph 4.16, "Completion of Protective Action Once it is Initiated." The USAR is being revised as part of corrective action to condition report 1-93-10-012 to delete the statement that the control system resets itself following an interruption of AC power. It was found during a design review that all of the safety related 4160 volt circuit breakers will not lock up the anti-pump mechanism and prevent reclosure following an undervoltage condition, except for the A and B residual heat removal (RHR) system pumps when the RHR system is operating in the containment spray mode. The evaluation concluded that the event that causes a lockup of the RHR pumps is beyond the design basis of the plant. Therefore, this change does not create a new type of accident or equipment malfunction since the containment spray function will perform its intended design basis function.

Log No. 94-0040

Change Frequency for Fire Protection Standpipe Hose Visual Inspections

CPS Procedure No. 9601.11, "Fire Protection Stand-Pipe Hose Visual Inspection," is being revised to extend the frequency for visual inspections of stand-pipe hose stations from monthly to once every 18 months. This extension is based on an evaluation of the monthly inspections performed for the last five and one-half years. The frequency of inspection meets the intent of NFPA Standard 14 that states that stand-pipe stations should be inspected frequently to ensure the presence of required hardware and proper racking of hoses. The new inspection frequency provides an adequate level of assurance that the changes will not degrade the fire protection program.

Log No. 94-0041

Modification HP-032, Eliminate High Pressure Core Spray Instrumentation Inaccuracies

Modification HP-032 revises the high pressure core spray (HP) system to eliminate considering flow measuring instrument inaccuracies when determining required flow to the reactor vessel during performance of technical specification surveillance testing. Condition Report 1-91-09-028 was written to document that instrument inaccuracies were not taken into account when determining emergency core cooling system (ECCS) pump flows as recommended by GE design specification data sheets. Initial corrective action included installation of high accuracy flow gages to determine required flow. However, due to significant performance margins in the ECCS pumps, it is not necessary to include instrument inaccuracies in the acceptance criteria. This change eliminates the need for installing high accuracy flow measuring devices during testing and allows the use of installed instruments for ensuring required flows are met. This change does not impact any of the technical specification acceptance criteria and does not reduce any margin of safety.

Log No. 94-0042

Revise Instrument Air System to Match As-Built Configuration

This change revises the instrument air (IA) system to match the as-built configuration of the plant. The changes include the addition of valves that were not previously shown on design drawings. The valves were installed to aid maintainability of the system; however, design drawings were never updated to reflect the changes. No new failures or equipment malfunctions have been identified.

Log No. 94-0043

Change Frequency of Fire Protection Valve Line-Up Verification

CPS Procedure No. 9071.19, "Monthly Fire Protection Valve Line-Up," is being changed to require valve line-up verifications to be performed annually rather than monthly. This change is being made based on operating experience that the system has been lined up correctly for the past six and one-half years. An evaluation was performed to determine that the current frequency is too frequent. All of the more critical valves have locking devices to prevent inadvertent valve mispositioning. Other effective administrative controls are in place to ensure that valve mispositioning is detected. The change does not affect the ability to safely shutdown the plant in the event of a fire.

Log No. 94-0044

Replace Valve Actuator Spring Packs and Revise Gear Ratios for Valves 1E12F037A, 1E12F037B

This design change revises the actuator gearing and spring packs for residual heat removal (RHR) system upper containment pool cooling valves 1E12F037A and 1E12F037B due to NRC Generic Letter 89-10 requirements. The valves also serve as containment isolation valves. The gearing change is required to provide more torque when opening and closing the valves. The resulting gearing change requires a change in the valve stroke times from 95 to 120 seconds. The change in stroke time is acceptable since these valves are normally closed and containment isolation is provided. Also, the valve stroke times are not an input to the offsite dose analyses. The valves are opened only during plant shutdown when fuel assemblies are stored in the upper containment pool. The changes do not affect any of the accident analyses assumptions or results.

Log No. 94-0045

Interim Low Level Radioactive Waste Storage

CPS Procedure No. 1913.02, "Radioactive Waste Storage and Inventory," is being revised to support the on-site storage of wet waste in the Radwaste Shielded Storage Area and dry active waste on the 702' and 720' elevations of the radwaste building. The changes are necessary due to the lack of access to an off-site disposal site. A floor loading analysis has been performed to allow the storage of the containers. Also, the long-term storage of the containers has been evaluated for fire and radiological effects. The changes have been compared to other events involving radwaste system failures and do not result in any increase in consequences or new equipment malfunctions important to safety.

Log No. 94-0046

Upgrade Fire Protection Containment Isolation Valves

This change revises the stroke time for fire protection (FP) system containment isolation valves 1FP050 and 1FP092 from 48 seconds to 58 seconds due to the installation of new actuator gears, spring packs and limiter plates. The changes are a result of more conservative NRC Generic Letter 89-10 requirements. These valves are normally open and have an active safety function to close on a containment isolation signal. The change is acceptable since it meets the guidance criteria in Regulatory Guide 1.141 and ANSI Standard N271-1976, which state that containment isolation valves on lines greater than 12 inches in diameter should generally close in less than one minute. Since the closure time meets the guidance criteria, there is no increase in the consequences of an accident.

Log No. 94-0047

Modification RI-046, RCIC Storage Tank Low Level Alarm Setpoint Change

Modification RI-046, changes the setpoint for when the reactor core isolation cooling (RCIC) system and the high pressure core spray (HPCS) system switch suction paths from the RCIC tank to the suppression pool when a low tank level occurs. This change results from a condition report, 1-94-02-020, which was written during a safety system functional assessment of the RCIC system. The original setpoint calculation did not account for vortexing, and requires the low level setpoint to be raised two and one-half inches from its existing value. The change may be implemented since the new setpoint remains within the technical specification allowable values, however, the technical specifications will be revised to reflect the actual setpoint. The change is not an unreviewed safety question since the change allows the RCIC and HPCS systems to function as designed and the capacity of the RCIC tank has not been reduced.

Log No. 94-0049

Revise Stroke Times for Instrument Air Containment Isolation Valves

This change revises the stroke time for instrument air (IA) system containment isolation valves 1IA012B and 1IA013B from 13 and 14 seconds, respectively, to 25 seconds due to the installation of new actuator gears, spring packs and limiter plates. The changes are a result of more conservative NRC Generic Letter 89-10 requirements. These valves are normally open and have an active safety function to close on a containment isolation signal. The change is acceptable since it meets the guidance criteria in Regulatory Guide 1.141 and ANSI Standard N271-1976, which state that containment isolation valves on lines less than three inches in diameter should generally close in less than 15 seconds. Since the closure time does not meet the general guidance criteria, the closure time for a one-inch line was compared to the potential release of a 12-inch line with a 60-second closure. The potential for release of a one-inch line is well within the bounds of a 12-inch line, therefore, the valve closure time is still within the intent of the regulatory guidance and there is no increase in the consequences of an accident.

Log No. 94-0050

Modification FP-089, Install Smoke Detector in Division 3 Battery Room

Modification FP-089 installs an ionization smoke detector in the Division 3 battery room as corrective action for condition report 3-91-10-040. During the review of fire load calculations, it was found that non-conservative assumptions were made in determining the quantities of combustible materials. To comply with 10CFR50, Appendix R and the CPS safe shutdown analysis, an automatic fire detection system is necessary because of the increased fire loading. The detector is installed in a non-safety related circuit; therefore, no new loads are placed on a safety related circuits. The detector and associated electrical conduits are seismically mounted to prevent any damage to safety related equipment. No new failures have been created by this change.

Log No. 94-0051

Changes to Nuclear Station Engineering Department, CPS Organization Structure

This change involves the reorganization of the Nuclear Station Engineering Department (NSED) and CPS-Plant Technical groups. The change deletes the position of Supervisor-Plant Testing and reassigns the responsibilities of the Plant Technical group to NSED. Also, the title of Supervisor-Nuclear Engineering on the Facility Review Group membership is changed to Senior Engineer-Nuclear. These are administrative changes only and do not affect qualification or experience levels to occupy those positions.

Log No. 94-0052

Modification M-071, Snubber Reduction Program (RH Piping Subsystem)

Modification M-071 reduces the number of snubbers required to maintain acceptable design margins in the following residual heat removal (RH) system piping subsystems: 1H22P004B, 1RH-02, 1RH-06, 1RH-63, 1RH-64, 1RH-70, and 1RH-71. These subsystems contain piping and instrumentation that perform the following functions: line break detection and isolation, low pressure coolant injection and containment spray, vent non-condensable gases during steam condensing mode operation (which has been deleted). The changes increase the stresses in the piping systems; however, the increases are within design and code allowable values. Hence, no new failures are created as a result of the elimination of the snubbers.

Log No. 94-0053

Modification M-071, Snubber Reduction Program (LP Piping Subsystem)

Modification M-071 reduces the number of snubbers required to maintain acceptable design margins in the following low pressure core spray (LP) system subsystem, 1H22P004B. The change affects the line break detection instrumentation line on the LP injection line. Portions of the snubber assembly are being retired in place. Although piping stresses are increased as a result of the changes, the stresses are within piping and code allowable values. Hence, no new failures are created as a result of the changes.

Log No. 94-0054

Modification M-071, Snubber Reduction Program (RCIC Piping Subsystem)

Modification M-071 reduces the snubbers required to maintain acceptable design margins in the following reactor core isolation cooling (RCIC) system piping subsystems: 1RI-10, 1H22-P004B, and 1H22P015B. These subsystems perform the functions: RCIC injection and high steam flow detection and isolation. Portions of the snubber assembly are being retired in place. Although piping stresses are increased as a result of the changes, the stresses are within piping and code allowable values. Hence, no new failures are created as a result of the changes.

Log No. 94-0055

Modification FP-090, Dedicated Air Supply for Fire Pump Diesel Engines

Modification FP-090 installs a dedicated air supply to the diesel driven fire pumps, 0FP01PA and 0FP01PB, located at the screen house. The change is a result of a condition report, 1-93-03-004, which identified a condition that may prevent sufficient combustion air if the dampers are closed during cold weather conditions. The dedicated air supply consists of an eight-inch diameter air intake pipe designed to provide 1000 cfm of combustion air. The air intakes are designed to prevent clogging from natural hazards. This change also eliminates the minimum open position of the screen house ventilation louvers. The changes satisfy National Fire Protection Association and vendor recommendations. No new failures are created by this change.

Log No. 94-0056

Evaluation of Thermo-Lag in Fire Zone A-1a

This evaluation concerns the acceptability of Thermo-Lag fire wrap material that is installed to protect Division 2 power, control and instrumentation cables in Fire Zone A-1A, which is the general access corridor at the 707' elevation in the auxiliary building. The purpose of the evaluation is to accept the fire wrap as is even though the material does not fully provide a one-hour fire rating. This 10CFR50 Appendix R deviation from a one-hour rating is justified based on several design and programmatic fire protection features to ensure safe shutdown capability is maintained. A defense-in-depth approach was considered in determining the acceptability of the Thermo-Lag installation. The acceptability was based upon the following factors: (1) administrative controls, (2) physical layout of the fire zone, (3) fire modeling using EPRI guidance, (4) fire endurance capability, (5) probabilistic risk assessment of the safety benefit of the Thermo-Lag material, and (6) operator response capabilities in the event of a fire. It has been concluded that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0057

Emergency Core Cooling System Response Time Evaluation

This change corrects USAR tables and text to be consistent with emergency core cooling system (ECCS) response times reflected in design documents. The specific changes are to ECCS injection valve opening times and ECCS pump start times. The reason for the changes is to eliminate confusion and misinterpretation of the response times. No physical changes to equipment, control logic or setpoints result from this change. The changes reflect the design basis parameters included in the ECCS analyses and do not change any of the ECCS acceptance limits.

Log No. 94-0059

Change Gear Ratios for Component Cooling System Containment Isolation Valves

This change revises the stroke time for component cooling water (CC) system containment isolation valves 1CC060 and 1CC127 from 61 to 64 seconds due to the installation of new actuator gears, spring packs and limiter plates. The changes are a result of more conservative NRC Generic Letter 89-10 requirements. These valves are normally open and have an active safety function to close on a containment isolation signal. The change is acceptable since it meets the guidance criteria in Regulatory Guide 1.141 and ANSI Standard N271-1976, which state that containment isolation valves on lines greater than three and one-half inches in diameter should generally close in less than one minute (60 seconds). Since the closure time does not meet the general guidance criteria, the closure time for a one-inch line was compared to the potential release of a 12-inch line with a 60-second closure. The potential for release of an eight-inch line with a 64-second closure time is well within the bounds of a 12-inch line with a 60-second closure time; therefore, the valve closure time is still within the intent of the regulatory guidance and there is no increase in the consequences of an accident.

Log No. 94-0060

Change Gear Ratio for RCIC Valve, 1E51F076

This change revises the stroke time for reactor core isolation cooling (RCIC) system containment isolation valve, 1E51F076, from 8 seconds to 14 seconds due to the installation of new actuator gears, spring pack and limiter plate. The changes are a result of more conservative NRC Generic Letter 89-10 requirements. This valve is normally closed, but when it is open, it has an active safety function to close on a containment isolation signal. The change is acceptable since it meets the guidance criteria in Regulatory Guide 1.141 and ANSI Standard N271-1976, which state that containment isolation valves on lines less than three inches in diameter should generally close in less than 15 seconds. Since the closure time meets the general guidance criteria, there is no increase in the consequences of an accident.

Log No. 94-0061

Change Gear Ratio for Component Cooling System Containment Isolation Valve

This change revises the stroke time for component cooling water (CC) system containment isolation valves 1CC049 and 1CC054 from 66 and 89 seconds, respectively, to 84 seconds, due to the installation of new actuator gears, spring packs and limiter plates. The changes are a result of more conservative NRC Generic Letter 89-10 requirements. These valves are normally open and have an active safety function to close on a containment isolation signal. The change is acceptable since it meets the guidance criteria in Regulatory Guide 1.141 and ANSI Standard N271-1976, which state that containment isolation valves on lines greater than three and one-half inches in diameter should generally close in less than one minute (60 seconds). Since the closure time does not meet the general guidance criteria, the closure time for a one-inch line was compared to the potential release of a 12-inch line with a 60-second closure. The potential for release of a 10-inch line with an 84-second closure time is within the bounds of a

12-inch line with a 60-second closure time; therefore, the valve closure time is still within the intent of the regulatory guidance and there is no increase in the consequences of an accident.

Log No. 94-0064

Change CPS Commitment to Regulatory Guide 8.6

This change revises CPS commitment to Regulatory Guide 8.6, "Standard Test Procedure for Geiger-Mueller Counters," for source check frequency. The current reference specifies performing source checks prior to each use, during intermittent use conditions and several times a day during continuous use. The frequency is being changed to source check each unit at least each day the instrument is in use. This change reflects industry-accepted practices and complies with 10CFR20 requirements to ensure that instruments and equipment used for quantitative radiation measurements are calibrated periodically for the radiation measured.

Log No. 94-0065

Temporary Modification 94-027, Cross-Tie DC Buses, 1F and 1B

Temporary Modification 94-027 installs a temporary cross-tie between the 1F and 1B DC motor control centers (MCCs) during troubleshooting and repair of the 1B battery charger during plant shutdown (maintenance outage MO-6). This cross-tie is necessary to ensure adequate power supply to the safety related 1B bus while the 1B battery charger is out of service. The change is acceptable since only one division (Division 1) is required to be operable during cold shutdown. The cross-tie connects a temporary cable from a spare breaker in DC MCC 1F to a disconnect switch located in a test breaker in DC MCC 1B. Administrative controls such as equipment tag outs, fire watches and security watches will be established during the installation. The Division 2 battery charger will be restored to an operable status prior to restarting the plant. Therefore, the change will not increase the probability of an equipment malfunction important to safety.

Log No. 94-0066

Delete Transmission Factors for Non-Noble Gas Measurements

This change deletes the requirement for applying a transmission factor to non-noble gas measurements when identifying relative increases in airborne radioactivity. Continuous air monitors (CAMs) are used to detect airborne radioactivity in selected exhaust ventilation ducts. Transmission factors are used when the CAMs are used for quantitative measurements; however, since the

CAMs are only used for qualitative measurements, changes in magnitude can be observed whether a transmission factor is applied or not. This change will not affect the operator's response to relative increases in airborne activity.

Log No. 94-0067

Replacement Chart Recorders for Main Control Room

This change replaces several chart recorders in the main control room with a different model recorder. The parameters measured by the recorders being replaced are average power range monitor (APRM) flux, intermediate range monitor (IRM) flux, source range monitor (SRM) flux, reactor water cleanup system conductivity, reactor recirculation/reactor water cleanup/control rod drive oxygen content, reactor core differential pressure, feedwater turbidity, feedwater steam flow, narrow/upset range reactor water level, turbine steam flow and reactor pressure. During replacement of individual recorders, the operator relies on computer displays, using the process/display computer systems. The recorders have no automatic function or alarm function associated with them. The recorders are not used to mitigate the consequences of any accident.

Log No. 94-0068

Change CPS Commitment to Regulatory Guide 1.88

This change revises CPS' commitment to Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," that records, such as radiographs, photographs, negatives, microfilm, and magnetic media will be stored to prevent damage from excessive light, stacking, electromagnetic fields, temperature and humidity as appropriate to the record type. Also, computer and electronically-stored data will be maintained in accordance with NIRMA Standard TG15-1993, "Management of Electronic Records." The change is not a reduction in commitment of the quality assurance program per 10CFR50.54 (a).

Log No. 94-0069

Extended Operation with Reactor Recirculation Flow Control Valve Motion Inhibited

This change evaluates full power operation with the "A" reactor recirculation (RR) system flow control valve (FCV), 1B33F060A, motion inhibited through the end of cycle 5. Extended operation with the FCV hydraulically locked in the open position is necessary to prevent excessive FCV oscillations, resulting from problems with valve position control instrumentation. FCV oscillations cause fluctuations in reactor core flow and reactor power. Inhibiting valve motion prevents the 'A' FCV from automatically closing in the event of a feedwater pump trip with a concurrent low reactor water level (Level 4). The FCV runback is a reactor scram avoidance feature should the above two conditions exist. The failures associated with this configuration were compared with the accident analysis events associated with FCV failures, increasing and decreasing.

Log No. 94-0070

CPS Tagging Authority Responsibility Changes

CPS Procedure No. 1014.01, "Safety Tagging," is being revised to implement the Fix It Now (FIN) process, which is a multi-disciplined work process to repair minor plant deficiencies under streamlined controls. Because of the reduced controls on the FIN team, not all maintenance items are within the scope of the team. The FIN Team Leader, an active Senior Reactor Operator (SRO), will select, authorize, and control the work of the FIN team. This makes the FIN Team Leader an additional tagging authority to the Shift and Assistant Shift Supervisors. This makes the level of qualifications for the FIN Team Leader comparable to the Shift and Assistant Shift Supervisor.

Log No. 94-0071

Revision to CPS Emergency Plan Implementing Procedure, EC-02, Emergency Classifications

The CPS Emergency Plan Implementing Procedure, EC-02, "Emergency Classifications," is being revised to include numerical values for determining an Unusual Event and Alert due to liquid radiological releases. Previously, entry action levels were based only on a valid alarm. The values for the liquid radiological effluent monitor alarm setpoints are more realistic and calculated from actual chemistry data. The changes do not reduce the effectiveness of the CPS Emergency Plan per 10CFR50.54 (q).

Log No. 94-0072

Modification M-082, Pressure Locking Prevention, Low Pressure ECCS Injection Valves

Modification M-082 modifies low pressure emergency core cooling system (ECCS) injection valves, 1E12F042A, 1E12F042B, 1E12F042C, and 1E21F005, by drilling a hole in the valve disk. This modification prevents pressure locking and thermal binding as identified in GE Service Information Letter 368, NRC Circular 77-05, and NRC Information Notices 81-31 and 92-26. Valves 1E12F042A, 1E12F042C and 1E12F005 were modified in the fifth refueling outage, RF-5. 1E12F042B will be completed in RF-6. No new failures have been created by the change. The change eliminates a potential failure of disabling one or more of the injection valves.

Log No. 94-0073

SX Pipe Tunnel Hatch Cover Removal

This change allows for the personnel hatch leading from the shutdown service water (SX) system pipe tunnel to the Division 2 SX pump cubicle to be eliminated as a flood barrier. The flooding analysis was reviewed and a calculation was performed to determine the flooding effects of the SX pump rooms due to leaving off the hatch cover. It was determined that it would take 755 hours to flood the SX pipe tunnel to the elevation that would prevent operation of the SX system. However, the duration of the probable maximum flood is only 358 hours; therefore, there is no common mode failure with permanently removing the hatch cover and the operability of the SX system is not affected by this change.

Log No. 94-0074

Evaluation of Thermo-Lag in Fire Zone CB-1e

This evaluation concerns the acceptability of Thermo-Lag fire wrap material that is installed to protect Division 2 power, control and instrumentation cable in Fire Zone CB-1e, which is the general access corridor between the 737 and 751 elevations in the control building. The purpose of the evaluation is to accept the fire wrap as is even though the material does not fully provide a one-hour fire rating. This 10CFR50 Appendix R deviation from a one-hour rating is justified based on several design and programmatic fire protection features to ensure safe shutdown capability is maintained. A defense-in-depth approach was considered in determining the acceptability of the Thermo-Lag installation. The acceptability was based upon the following factors: (1) administrative controls, (2) physical layout of the fire zone, (3) fire modeling using EPRI guidance, (4) fire endurance capability, (5) probabilistic risk assessment of the safety benefit of the Thermo-Lag material, and (6) operator response capabilities in the event of a fire. It was concluded that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0076

Evaluation of Thermo-Lag in Fire Zone C-2

This evaluation concerns the acceptability of Thermo-Lag firebreak material that is installed to protect Division 2 power, control and instrumentation cable trays in the south side of Fire Zone C-2, which is the containment building outside the drywell at elevation 803'-3". The purpose of the evaluation is to accept the firebreak as is even though the material does not fully provide a one-hour fire rating. This 10CFR50 Appendix R deviation from a one-hour rating is justified based on several design and programmatic fire protection features to ensure safe shutdown capability is maintained. A defense-in-depth approach was considered in determining the acceptability of the Thermo-Lag installation. The acceptability was based upon the following factors: (1) administrative controls, (2) physical layout of the fire zone, (3) fire modeling using EPRI guidance, (4) fire endurance capability, (5) probabilistic risk assessment of the safety benefit of the Thermo-Lag material, and (6) operator response capabilities in the event of a fire. It was concluded that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0077

Change Frequency for Inspecting Fire Rated Assemblies and Penetration Seal Devices

CPS Procedure No. 9601.01, "Fire Rated Assemblies and Penetration Sealing Devices," is being revised to eliminate the inspection of internal conduit seals, and change the frequency of fire damper and other fire barrier inspections from every 18 months to every 36 months. The bases for these changes are that the internal inspections have not shown any discrepancies in the last four years, and the inspections of fire dampers have been 100 percent acceptable for the last five inspections, and the inspections of fire barriers has shown only one minor discrepancy in the last four years. The changes indicate a highly reliable fire barrier integrity, and the changes will not reduce the capability to safely shutdown the plant in the event of a fire.

Log No. 94-0079

Evaluation of Thermo-Lag in Fire Zone CB-1f

This evaluation concerns the acceptability of Thermo-Lag fire wrap material that is installed to protect Division 2 power, control and instrumentation cable in Fire Zone CB-1f, which is the general access and equipment area at the 762 ' elevation in the control building. The purpose of the evaluation is to accept the fire wrap as is even though the material does not fully provide a three-hour fire rating. This 10CFR50 Appendix R deviation from a three-hour rating is justified based on several design and programmatic fire protection features to ensure safe shutdown capability is maintained. A defense-in-depth approach was considered in determining the acceptability of the Thermo-Lag installation. The acceptability was based upon the following factors: (1) administrative controls, (2) physical layout of the fire zone, (3) fire modeling using EPRI guidance, (4) fire endurance capability, (5) probabilistic risk assessment of the safety benefit of the Thermo-Lag material, and (6) operator response capabilities in the event of a fire. The conclusion of the evaluation is that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0080

Main Condenser Scale Inhibitor Chemical Treatment

This change converts Temporary Modification 93-017 (Reference Log 93-0078 reported with USAR Revision 5) which provides an injection system to add a scale-inhibiting chemical to the circulating water (CW) system, into a permanent installation. The purpose of the chemical treatment is to prevent further scale build up on the main condenser tubes. The chemical feed equipment is located in the circulating water pump house and injects chemicals into the discharge flow of the "A" or "C" CW pump. The chemical treatment does not require changes to the National Pollution Discharge Elimination System permit or approval by the Environmental Protection Agency. No new failures have been introduced by the change. The additional fire load has been evaluated and does not affect the safe shutdown analysis.

Log No. 94-0081

Alternate Shutdown Cooling Using Fuel Pool Cooling in Mode 5 (Refueling)

This change allows the fuel pool cooling (FC) system to be utilized as an alternate shutdown cooling method during refueling (Mode 5). This method relies on natural circulation within the core region to provide coolant circulation. The FC system heat exchangers would then remove the warmer water. This method will only be used with the vessel disassembled and the reactor cavity pool flooded. No new failures have been created. This alternate shutdown cooling method enables technical specification requirements to be met. The FC system is a safety related system and has sufficient decay heat removal capacity (4-5 days after shutdown) and is designed to meet seismic and single failure criteria.

Log No. 94-0082

Evaluation of Thermo-Lag in Fire Zone F-1p

This evaluation concerns the acceptability of Thermo-Lag fire wrap material that is installed to protect the piping penetration through the fuel building wall in Fire Zone F-1p. The piping which passes through the fuel building wall is a 16-inch main condenser vacuum pump discharge line which connects to the common station ventilation stack. The purpose of the evaluation is to accept the fire wrap as is even though the material does not fully provide a three-hour fire rated penetration seal as required by 10CFR50 Appendix R. It has been determined that the construction of the piping penetration assembly is substantial enough to provide protection equivalent to a fire damper. The conclusion of the evaluation is that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0083

Evaluation of Thermo-Lag in Fire Zone D-8

This evaluation concerns the acceptability of Thermo-Lag fire wrap material that is installed to protect Division 2 main power feed cables in Fire Zone D-8, which is the Division 1 diesel generator ventilation fan room and breathing air filter train and compressor room at the 762' elevation in the diesel generator building. The purpose of the evaluation is to accept the fire wrap as is even though the material does not fully provide a one-hour fire rating. This 10CFR50 Appendix R deviation from a one-hour rating is justified based on several design and programmatic fire protection features to ensure safe shutdown capability is maintained. A defense-in-depth approach was considered in determining the acceptability of the Thermo-Lag installation. The acceptability was based upon the following factors: (1) administrative controls, (2) physical layout of the fire zone, (3) fire modeling using EPRI guidance, (4) fire endurance capability, (5) probabilistic risk assessment of the safety benefit of the Thermo-Lag material, and (6) operator response capabilities in the event of a fire. The conclusion of the evaluation is that the Thermo-Lag installation will not increase the probability or consequences of an accident; and, the ability to safety shut down the plant in the event of a fire has not been reduced.

Log No. 94-0084

Containment Penetration Overcurrent Protection

This change revises the actions necessary to comply with Operational Requirements Manual (ORM) Section 2.5.1, Containment Penetration Conductor Overcurrent Protective Devices. As a result of License Amendment 95, former CPS Technical Specification requirement 3.8.4.1 was relocated in its entirety to ORM Section 2.5.1. During the relocation process, it was found that portions of it were technically incorrect. The terminology associated with 6.9 kV circuit breakers and lower voltage circuit breakers were interchanged to make the terminology consistent with the installed equipment. Also, for lower voltage circuit breakers, it was decided to take additional administrative controls when a device is found to be inoperable. An additional change was made to add an action statement that is unique to the upper containment polar crane containment penetration conductor overcurrent protective device. No new equipment malfunctions are created since the actions ensure that the equipment and containment penetrations are protected from a potential overcurrent condition.

Log No. 94-0085

Revise Electrical Containment Penetration Conductor Overcurrent Protective Devices Testing Requirements

This change revises the testing frequency of containment penetration conductor overcurrent protective devices from testing and inspecting a 10 percent population of breakers every 18 months to 100 percent every 72 months. The possibility exists under the current testing program that a breaker may go as long as 180 months without being tested and inspected. The change relaxes the requirement to test an additional 10 percent of like protective devices until no additional failures are found. This change will not increase the probability of an equipment failure since operating experience has indicated a low-failure rate (2.2 percent).

Log No. 94-0086

Install Permanent Gagging Devices on Steam Jet Air Ejector Control Valves

This change installs a permanent mechanical gagging device on steam jet air ejector (SJAE) control valves, 1B21F437A and 1B21F437B, to the valves closed at all times during SJAE operations. These valves are designed to be used during SJAE startup at low operating pressures (less than 185 psig). Since the SJAE's are normally put into operation at pressures greater than 185 psig, the valves do not perform any useful function. Mechanically locking the valves closed will prevent the valves from inadvertently opening following a loss of air or actuator failure, thus preventing an unplanned shutdown. No new failure modes are created by this change.

Log No. 94-0088

Revise Control Rod Scram Accumulators Action Statement

This change revises the Operational Requirements Manual (ORM) Section 3.1.2, Control Rod Scram Accumulators, to permit an alternate means of verifying adequate pressure in the control rod scram accumulators in lieu of declaring the accumulator(s) inoperable when an alarm condition occurs. The change allows timely visual or manual accumulator pressure and level verification, by verifying pressure locally every 24 hours and/or verifying that the accumulator is drained once per week. This ensures that the accumulators have adequate pressure and are not leaking. Thus, the accumulators will perform their functions as described in the USAR. This change will also permit the calibration of scram accumulator leak and pressure detection in Modes 1 and 2.

Log No. 94-0089

Reactor Operation with Increased Reactor Pressure Setpoint

CPS Procedure No. 2800.63, "Pressure Setpoint Increase During End of Cycle 5 Coastdown," provides instructions for maintaining reactor pressure as close to full power steam dome pressure as possible during the end of cycle (EOC) coastdown evolution. This procedure is intended to maximize plant electrical energy output during the coastdown by increasing reactor thermal energy output to a level slightly above that normally expected. Reactor steam dome pressure will be maintained significantly below the technical specification requirement and turbine throttle pressure will be kept below its rated performance limit. No changes to instrument setpoints are required. The effects of increased dome and core pressure were evaluated against the change in minimum critical power ratio (MCPR) due to the reduced power at the end of cycle. The results of the evaluation were that the MCPR would stay above the safety limit during coastdown operations at increased dome pressure at the end of cycle.

Log No. 94-0090

Radiation Protection Organization and Facilities

This change administrative and organizational changes to reflect current CPS Radiological Protection Program policies and practices. The title changes includes the renaming of the Radiological Engineering group to the Health Physics Analysis group. The change corrects descriptions of the primary and back-up facilities for decontaminating personnel. The change also correctly depicts the number of egress points, and clarifies the definitions of High Radiation and Restrictive High Radiation Areas to be more consistent with 10 CFR 20.

Log No. 94-0091

Temporary Modification 94-033, Mechanical Blocking of Heater Drain Valve, 1HD030B

Temporary Modification 94-033 mechanically locks open the 4B feedwater heater normal drain valve, 1HD030B, at approximately 75 to 80 percent open, and allows the other normal drain valve, 1HD066B, to control heater level. The reason for the change is to gain better level control over the 4B heater and prevent the emergency drain valves from opening unnecessarily, causing a loss of efficiency and generation capability during normal plant operation. The actuator for the 1HD030B valve has an air leak and cannot be repaired until a plant shutdown. No new failures are created by this change. The main turbine water induction protection will still be provided in the event of a major feedwater heater tube leak.

Log No. 95-002

Modification FH-030, Inclined Fuel Transfer Tube Testing Assembly

Modification FH-030 revises the fuel handling (FH) system by installing a bellows testing assembly and a two-valve test connection to containment penetration sleeve (1MC-4) which forms a part of the inclined fuel transfer tube. The new assembly will provide a means of applying a static test pressure to the fuel transfer bellows to ensure containment integrity will be maintained in accordance with 10CFR50 Appendix J. No new failures have been created since the changes are being made in accordance with ASME code requirements.

Log No. 95-003

Temporary Modification 95-01, Defeat Interlock Between 'B' Feedwater Heater String and Heater String Bypass Valve

This temporary modification defeats the electrical interlocks between low pressure feedwater heater bypass valve, 1CB007, and feedwater heater isolation valves, 1CB006B and 1CB008B, such that isolation valves can be closed without the bypass valve automatically opening. This temporary modification allows maintenance (tube plugging) to be performed on feedwater heaters 2B, 3B, 4B or 5B, on-line as required. The change also disables the seal-in function of 1CB006B to enable throttling flow through the 'B' feedwater heater string once work on the heaters has been completed. This change does not increase the frequency of a loss of feedwater heating transient, nor does it make a loss of feedwater heating transient more severe.

Log No. 95-004

Modification RH-043, Revise Low Pressure Permissive Interlock for Shutdown Cooling Containment Isolation Valves

Modification RH-043 revises the residual heat removal (RH) system shutdown cooling containment isolation setpoint and reset point for high reactor vessel pressure (RH shutdown cooling cut-in permissive). This change is being made as a result of NRC Generic Letter 89-10 recommendations that are more conservative than those used in the original design. Changing the setpoint for the RH shutdown cooling cut-in permissive lowers the maximum differential pressure the RH system isolation valves, 1E12-F008 and 1E12-F009, will be required to close against. The change also ensures that the lower pressure RH system piping is protected against increasing reactor pressure. The isolation setpoint is being changed from 135 psig to 104 psig. The reset point is being changed from 90 psig to 96.5 psig. The change will not impact the probability of a malfunction of equipment important to safety in that the low pressure RH system piping will not be subject to higher than allowable pressures.

Log No. 95-005

Modification M-080, Revision to Equipment Qualification Design Criteria

Modification M-080 provides corrective action for condition report 1-90-02-048 by revising the equipment qualification design criteria, DC-ME-09-CP, to incorporate the 45-second time delay that was not considered in the reactor water cleanup (RT) system high energy line break analysis. This time delay results in a longer blowdown time, and subsequent higher area temperatures and pressures (in six areas), than that originally calculated. The results of the new analysis are relatively insignificant, resulting in only minor changes to area temperatures and pressures. In general, the RT high energy break analysis is bounded by the main steam line break analysis. Therefore, the change does not increase the probability of equipment malfunctions to that previously evaluated.

Log No. 95-006

CPS Emergency Plan Revision

The CPS Emergency Plan is being revised to change the minimum required shift complement of the chemistry group on back shifts and weekends. This change is a result of recommendations to the chemistry technicians rotation schedule. The new schedule does not require chemistry technicians to be routinely on site during second and third shifts on Saturdays and Sundays. To respond to emerging plant conditions, a chemistry technician will be able to respond to the plant within 60 minutes by telephone or by pager. This 60-minute response time is the same as that for emergency response organization personnel. Other CPS Emergency Plan changes include the declaration of an unusual event or alert based on chemistry analysis of reactor coolant samples for dose equivalent iodine. The dose levels cited in the emergency plan are significantly higher than those required by plant technical specifications. The criteria for declaring an unusual event and alert has been revised to allow declaration based upon offgas monitor indications at the recombiner effluent. The position of emergency medical coordinator has been eliminated and the responsibilities are being assumed by security force personnel. None of these changes decrease the effectiveness of the CPS Emergency Plan per 10CFR50.54 (q).

Log No. 95-007

Change Safety Function for Fire Protection Containment Isolation Valves

This change revises the safety function for fire protection (FP) system containment isolation valves, 1FP078 and 1FP079, from active to passive. These valves are being changed from normally open to normally closed due to applying more conservative factors from NRC Generic Letter 89-10 recommendations. By maintaining these valves in a closed position, the safety function of these valves has been achieved. The fire protection in the drywell is not an automatic system, and these valves are not required to be open during normal operations. Also, during normal operations, the amount of combustible material in the drywell is low and access to the drywell is restricted. If a fire is detected in the drywell, the main control room operator would be able to open these valves should conditions warrant. No new failure modes are created by this change.

Log No. 95-008

Alternate Auxiliary Building Roof Hatch Installation During Plant Shutdown

This change allows the capability to install an alternate hatch in the auxiliary building roof during cold shutdown or refueling (Modes 4 or 5), when secondary containment integrity is required. The change allows the permanent concrete plug to be not installed and a lighter weight hatch to be used to maintain secondary containment during plant outages. The design of the alternate hatch will meet security, tornado wind pressure, and weather protection (winds, rain, snow, ice) requirements. The hatch will not meet tornado missile protection requirements, since the secondary containment boundary is not tornado missile-resistant. No new failures are created and the consequences of an accident have not been increased.

Log No. 95-009

Low Pressure Turbine Warming

CPS Procedure No. 2800.57, "Low Pressure Turbine Warming," was developed to determine the most efficient methods to perform low pressure turbine warming to help mitigate turbine rotor cracking. Warming is achieved by increasing main condenser pressure (reduce condenser vacuum) during startup and low power operation. The increased saturation temperature will ensure the turbine rotor is above the fracture appearance toughness temperature, avoiding brittle fracture failure, and will extend the life and inspection frequencies of the turbine rotor. The change will not increase the consequences of an accident, since the main steam isolation valve containment isolation trip will not be bypassed during the performance of this test and this test will performed only during low power operations.

Log No. 95-010

Remove Area Radiation Monitoring from Control Rod Drive Service Area

This change eliminates the need for area radiation monitor, 1RIX-AR018, and its associated sensor from the control rod drive (CRD) rebuild (service) room, located at the 737' elevation of the fuel building. The change also removes from the room, the CRD flush and ultrasonic tanks, which are being replaced. Elimination of the area radiation monitor eliminates unnecessary radiation exposure when performing preventive maintenance on the monitor. Should work be required in the room, continuous radiation protection coverage would be provided for work activities that could result in radiation level changes. The equipment is not used to mitigate the consequences of an accident or equipment malfunction.

Log No. 95-011

Control Rod Blade Storage in Upper Containment Pool

This change allows four spent control rod blades to be stored in the upper containment pool until the next refueling outage (RF-6) when four additional control rod blades will be removed and transferred to the spent fuel pool. This change allows the inclined fuel transfer control rod blade fixture to be installed only once for the two refuelings. The control rod blades are stored in the equipment storage rack located in the steam dryer section of the upper containment pool. This section of the pool does not contribute to the suppression pool makeup volume and is maintained with at least 23 feet of water above the top of the blades during normal operation. No new type of accident or equipment malfunction has been created.

Log No. 95-012

Source Range Monitoring Rod Block Operability During Refueling

This change revises the frequency of performing source range monitoring (SRM) system rod block channel functional testing from every seven days to 31 days. This change to the Operational Requirements Manual (ORM) section 4.2.2.1 is being made to make the ORM testing frequency consistent with the technical specification surveillance frequency for SRM instrumentation. The change is also being made to prevent interruption of refueling activities when these instruments have proven to be highly reliable. The SRM rod block function ensures control rod movement is performed only when sufficient neutron monitoring is available.

Log No. 95-013

Average Power Range Monitoring and Intermediate Range Monitoring Rod Block Instrumentation

This change revises the plant conditions during which rod block instrumentation for the average power range monitoring (APRM) and intermediate range monitoring (IRM) systems are required to be operable during refueling operations. The change allows the APRMs (Inoperative, Neutron Flux-Upscale) and IRMs (Detector-not full in, Upscale, Inoperative, and Downscale) to be operable in Mode 5 only when a control rod is withdrawn from a core cell containing one or more fuel assemblies. This change eliminates the performance of operability tests required by the Operational Requirements Manual when all control rods in core cells containing fuel are fully inserted. Inadvertent criticality is precluded by the refueling one-rod-out interlock and shutdown margin.

Log No. 95-015

Modification IA-019 Functional Verification Test

CPS Procedure No. 2800.56, "Plant Modification IA-019 Functional Verification," tests the logic associated with isolation of instrument air (IA) system containment isolation valves, 1IA005, 1IA006, 1IA007, and 1IA008. The modification changes the logic from a one-out-of-two to a two-out-of-two logic when a low reactor water level, level 1 signal is experienced. The testing ensures proper operation of the valves and the logic cards. The modification enables the isolation logic of the IA containment isolation valves to be consistent with other plant systems' isolation logic. The modification allows single channel testing without initiating the isolation logic, and also prevents inadvertent isolation resulting from a single channel malfunction. The testing will be conducted during modes 4 or 5 (cold shutdown or refueling), therefore technical specifications will be complied with and no new failures are created.

Log No. 95-019

Fire Load Classification Revisions

This change revises the load classifications for four fire zones in the control building and auxiliary building. Three of the fire zones are in the control building, and the remaining fire zone is in the auxiliary building. The changes are necessary due to a fire load re-calculation effort as a result of corrective action for condition report (CR) 3-91-10-040. The fire loads are higher primarily due to the addition of offices and other combustible materials not previously considered in the safe shutdown analysis. The evaluation also considers the fire loads due to the addition of equipment cages in the control building and radwaste building that were identified in CR 1-93-12-034. The changes will not prevent the capability of the plant to achieve and maintain cold shutdown of the plant in the event of a fire.

Log No. 95-020

Revision to Inclined Fuel Transfer System Testing Frequency

This change revises the Operational Requirements Manual (ORM) Section 4.6.6.2 testing requirement for the inclined fuel transfer system (IFTS) carriage position and liquid level sensor interlocks from once every 12 hours to once per seven days. The new frequency is based on the frequency of other fuel handling equipment surveillances are performed. The IFTS control system uses two sets of sensors for liquid level and carriage position indication. If a mismatch occurs between the two sets, an alarm is provided. The operator then manually selects which set is correct and resumes the transfer sequence. A review of the surveillance history for CPS Procedure No. 9092.01, "Inclined Fuel Transfer System Interlocks Functional," covering the last three refueling outages identified no surveillance failures for the last 84 performances. Therefore, there is no increased probability of equipment failure due to the change.

Log No. 95-021

Revision to Inclined Fuel Transfer System Testing Frequency

This change revises Operational Requirements Manual (ORM) Sections 4.6.6.2 through 4.6.6.5 to require Inclined Fuel Transfer System (IFTS) equipment testing to be performed within seven days prior to use rather than the current requirement of four hours. This change removes the requirement for unnecessary testing, when it has been performed within its normal frequency. The change does not increase the consequences of an accident or equipment malfunction since the IFTS will maintain exposure levels below 10CFR20 for personnel during normal operation.

Log No. 95-023

Change Frequency for CO_2 Valve Position Verification

CPS Procedure No. 9071.08, "Fire Protection CO_2 System Valve Position Check," is being revised to extend the surveillance frequency from monthly to annually. The change is based on the last 78 performances over the last six and one-half years which have been 100 percent correct. There are no specific requirements to perform the position verification on any specific frequency. The change does not impair the capability to safely shut down the plant in the event of a fire.

Log No. 95-024

Change Frequency for CO_2 Actuation Testing

CPS Procedure 9071.09, "Fire Protection CO_2 System Auto Actuation Test," is being change to extend the frequency for testing the operability of the carbon dioxide system for the diesel generator rooms from 12 months to 18 months. This test also allows the full flow "puff" test optional based on the results of other portions of the surveillance. The basis for the change is the testing has been performed successfully for the last eight years. There are no specific requirements to perform this test on any specific frequency. The changes do not impair the capability to safety shut down the plant in the event of a fire.

Log No. 95-026

Jet Pump Beam Storage in the Upper Containment Pool

This change allows the storage of 20 jet pump beams to be stored in the equipment storage rack of the upper containment pool during the next operating cycle. The jet pump beams and bolts will be stored in two containers, fabricated from stainless steel. The containers will be placed on the control rod blade hangers in the spent fuel storage pool until the sixth refueling outage (RF-6). At that time the containers will be removed and transferred to the spent fuel storage pool, along with control rod blades removed during RF-6. This change allows the inclined fuel transfer system control rod blade fixture to be installed only once. No new failures are created by this change. The containers were evaluated for their total weight, seismic, radiation and corrosion effects.

Log No. 95-027

Change Frequency for Yard Hydrant Flow Checks

CPS Procedure No. 9071.22, "Fire Protection Flow Check of Fire Hydrants," is being revised to extend the inspection frequency from every six months to once per year. An evaluation determined that reducing the frequency of performing flow checks will not impact the reliability of the fire protection system. Surveillances for the previous six and one-half years were fully acceptable. The changes will not degrade the fire protection system to perform its function during a design basis fire.

Log No. 95-028

Change Frequency for Yard Hose House Inspections

CPS Procedure No. 9071.18, "Fire Protection Hose House Equipment Visual Inspection," is being changed to extend the normal inspection frequency from monthly to annually. The changes are the result of an evaluation of the last 80 performances of the inspection. The inspections were all fully acceptable. The changes meet the guidelines of NFPA 24, Outdoor Protection, and will not degrade the fire protection system to perform its function during a design basis fire.

Log No. 95-029

Change Frequency for Halon Storage Tank Weight and Pressure Verification Tests

CPS Procedure No. 9476.03, "Halon Storage Tank Weight and Pressure Verification," is being revised to extend the normal testing frequency from every six months to every 24 months. The main control room (MCR) halon suppression system consists of 24 tanks that are weighed periodically to ensure they can provide a minimum concentration to suppress a fire in the cables below the electrical panels. To ensure that sufficient halon capacity is present in the tanks, CPS Procedure No. 9071.26, "MCR Halon System Monthly Valve Position Verification," (Reference Log No. 95-030) is being revised to increase the minimum tank pressure to 330 psig. The changes will not prevent the halon fire suppression system from performing its function in the event of a fire.

Log No. 95-030

Change Frequency for Main Control Room Monthly Valve Position Verifications

CPS Procedure No. 9071.26, "Main Control Room Halon Monthly Valve Position Verification," is being revised to increase the minimum tank pressure to ensure adequate halon capacity exists. This change is coupled with the increased interval between surveillances performed by CPS No. 9476.03, "Halon Storage Tank Weight and Pressure Verification," (Reference Log No. 95-029). The increased minimum ensures that the halon has not depleted below the point that the halon fire suppression system can perform its function in the event of a fire.

Log No. 95-031

Change Frequency for Main Control Room 18-Month Operability Testing

CPS Procedure No. 9071.11, "Main Control Room Halon System 18-Month Operability Test," is being changed to be performed on a 24-month frequency. This change is coupled with other changes associated with extending frequencies of halon fire suppression system testing (Reference Log Nos. 95-029 and 95-030). The changes do not degrade the design basis function of the fire protection systems and will not prevent the plant from safely shutting down following a design basis fire.

Log No. 95-032

Change Frequency for Fire Damper Drop Testing

CPS Procedure No. 9601.13, "Fire Damper Drop Test," is being changed to eliminate the requirement to perform fire damper drop testing every 18 months. In lieu of fire damper drop testing, visual inspections will be performed every 36 months. Fire damper drop testing will only be performed as post-maintenance or post-modification testing. This change is acceptable due to the visual inspections performed under CPS Procedure No. 9601.01, "Fire Rated Assemblies and Penetration Sealing Devices," that confirm hardware configuration, duct and damper slide cleanliness, and freedom from obstructions. The change will not degrade the capability of the dampers from performing their function in the event of a fire.

Log No. 95-034

Training for Escorted Radiation Workers and Tourist Escorts

This change revises the training requirements for escorted radiation workers and tourist escorts as described in USAR Section 13.2.1.1.1. The change consists of not requiring individuals to be trained, who are classified as tourists (infrequent visitors) and who are not be required to perform work in the plant. Training is required for escorted radiation workers expected to be on-site for a limited visit and having the potential to receive occupational radiation dose. Access requirements and exposure limits are defined by Radiation Protection supervision for each visit. The changes meet the requirements of 10CFR19.12.

Log No. 95-035

Temporary Modification 95-020, Alternate Air Supply to VR/VQ Dampers

Temporary Modification 95-020 provides a temporary air supply from the service air (SA) system to operate continuous containment purge dampers of the containment building ventilation (VR/VQ) system during the instrument air system outage. The dampers affected by this temporary modification are 1VQ02Y, 1VQ17Y, 1VR18Y, 1VR55Y, 1VR006A, 1VR006B, 1VR007A, 1VR007B, 1VQ003, and 1VQ020. The alternate air supply has the same reliability as the instrument air system. The SA system is also supplied at the same pressure and dew point. The plant will be in cold shutdown, with no core alterations, movement of irradiated fuel, or evolutions that could drain the reactor vessel. The affected isolation dampers are not required to be operable during the installation of the temporary modification.

Log No. 95-037

Temporary Modification 95-024, Temporary Power Supply for Control and Indication of VR Dampers

Temporary Modification 95-024 provides temporary power for control and indication for containment building ventilation (VR) system dampers 1VR18Y, 1VR17Y and 1VR55Y during the fifth refueling outage (RF-5). The temporary change is necessary to maintain partial containment ventilation during a bus outage. No new type of malfunction of equipment important to safety has been created since the VR system and the containment penetration conductor overcurrent protective device associated with this temporary modification are not required to be operable.

Log No. 95-038

Temporary Modification 95-021, Cross-Tie DC Motor Control Centers 1F and 1B

Temporary Modification 95-021 installs a cross-tie between DC motor control center (MCC) 1F to DC MCC 1B, while the Division 2 battery charger is out of service for maintenance. The cross-tie maintains a float charge on the battery so that battery parameters meet technical specification limits. Although the Division 2 battery will be on a float charge from a non-1E power source, the operability of the Division 2 battery will not be impacted provided the battery parameters meet the technical specification requirements. The connection to the non-1E power source is through a breaker that will shunt trip upon receipt of a loss of coolant accident signal (high drywell pressure or low reactor water level). This arrangement will not compromise the integrity of the 1E power source and conforms with Regulatory Guide 1.75, "Physical Independence of Electric Systems." Administrative controls will be in place, such as fire watches and security watches, during the installation of the temporary cable. No new failures are created since the 1F electrical bus will not be loaded in excess of its design ratings during the installation of the temporary modification.

Log No. 95-041

Temporary Modification 95-028, Defeat Travel Safety Computer on Main Refueling Bridge Crane

Temporary Modification 95-028 installs an electrical jumper that overrides the travel safety control computer interlocks which prevent bridge and trolley movement when the refueling crane bridge reaches preset limits in the X (trolley movement) and Y (bridge movement) directions. The computer override pushbutton switch is being simulated to be depressed by placing a jumper across its contact. Display indications for position are not affected. The change is needed due to the unreliability that has been experienced with the computer and the trolley encoder during the fifth refueling outage (RF-5). This temporary modification will be removed prior to plant startup. The change eliminates travel safety function which is a backup method to operator manual actions (The override system is strictly a manual system.) Additionally, the override function provides only a non-safety related function. The safety related refueling interlocks are not being affected by this temporary modification. A permanent modification is being planned after the completion of RF-5 to replace the entire control system.

Log No. 95-042

Change Relief Valve Setpoints

This change revises the setpoint for service water (WS) system relief valves, 1WS166A, 1WS166B, 1WS153A, and 1WS153B, to be consistent with the maximum allowable working pressure for the turbine oil cooler and generator stator cooling water heat exchangers. The relief valves will still provide thermal relief protection for the heat exchangers. The changes are corrective actions for condition report 1-94-09-062. No new failures are expected since the system will function as designed.

Log No. 95-043

Temporary Modification 95-030, Cross-Tie 1F Battery Charger to Division 1 DC Bus

Temporary Modification 95-030 installs a cross-tie from DC motor control center (MCC) 1F to DC MCC 1A while the Division 1 battery charger is out of service for maintenance. The cross-tie maintains a float charge on the battery so that battery parameters meet technical specification limits. Although the Division 1 battery will be on a float charge from a non-1E power source, the operability of the Division 1 battery will not be impacted provided the battery parameters meet the technical specification requirements. The connection to the non-1E power source will be through a breaker that shunt trips upon receipt of a loss of coolant accident signal (high drywell pressure or low reactor water level). This arrangement will not compromise the integrity of the 1E power source and conforms with Regulatory Guide 1.75, "Physical Independence of Electric Systems." Administrative controls will be in place, such as fire watches and security watches, during the installation of the temporary cable. No new failures are created since the 1F electrical bus will not be loaded in excess of its design ratings during the installation of the temporary modification.

Log No. 95-044

Temporary Modification 95-031, Cross-Tie 1F Battery Charger to Division 2 DC Bus

This temporary modification is identical to temporary modification 95-021 (reference Log No. 95-038) and is necessary to support Division 2 troubleshooting activities. This temporary modification installs a cross-tie between DC motor control center (MCC) 1F to DC MCC 1B, while the Division 2 battery charger is out of service for maintenance. The cross-tie maintains a float charge on the battery so that battery parameters meet technical specification limits. Although the Division 2 battery will be on a float charge from a non-1E power source, the operability of the Division 2 battery will not be impacted provided the battery parameters meet the technical specification requirements. The connection to the non-1E power source will be through a breaker that shunt trips upon receipt of a loss of coolant accident signal (high drywell pressure or low reactor water level). This arrangement will not compromise the integrity of the 1E power source and conforms with Regulatory Guide 1.75, "Physical Independence of Electric Systems." Administrative controls will be in place, such as fire watches and security watches, during the installation of the temporary cable. No new failures are created since the 1F electrical bus will not be loaded in excess of its design ratings during the installation of the temporary modification.

Log No. 95-045

Technical Specification Bases Change 95-007

This change revises the technical specification bases for surveillance requirement (SR) 3.4.4.3, SR 3.5.1.7, and SR 3.6.1.6.1 to clarify that there is no technical specification requirement to periodically exercise (open) the main steam safety relief valves (SRVs) with steam. The change identifies that alternate means may be used to open the SRVs provided actions are taken to preclude valve damage upon reclosure. The changes do not prevent the SRVs from performing their function. Also, the change does not increase the probability of the SRVs becoming blocked.

Log No. 95-046

RPV Water Level Mismatch During Power Operation

The purpose of this safety evaluation is to determine the effects of continued operation with the Division 2 reactor water level instruments reading approximately 5.5 inches lower than the other three divisions of reactor water level instrumentation. This safety evaluation is identical to previous safety evaluations for previous operating cycles (Reference Log Nos. 92-0106 and 93-0128). The results of the analysis performed indicate that the level differences are acceptable for the next operating cycle. The impact of the water level differences has a minimal impact on peak cladding temperature (less than three degrees), and the safety limit of 2200 degrees is not exceeded. The change does not require any change to any of the technical specification setpoints, nor does it change any of the analytical limits. This safety evaluation reviews the known differences between the reactor pressure vessel narrow range and wide range water level instrumentation. Disagreements of indicated water level have existed since startup testing in 1985. The Division 2 level instruments have consistently indicated lower than the other three divisions. The analyses indicated that there are no impacts on the technical specification analytical limits nor nominal trip setpoints. All trip functions will occur as designed. The change does not affect the consequences of a design basis loss-of-coolant accident since fuel cladding limits have not been exceeded.

Log No. 95-047

Technical Specification Bases Change 95-008

This change revises the technical specification bases Section 3.7.5, "Main Condenser Offgas," to correct the listing of noble gas isotopes used for the "sum of the six" noble gases calculations. The change includes krypton-85m (Kr-85m) rather than krypton-85 (Kr-85), and adds the "Station Nuclear Engineering" manual as a reference for the isotopic listing. The use of Kr-85m is more appropriate to be used in source term calculations since it has a larger contribution to the total source term than Kr-85. This change will ensure that technical specification limits for noble gases are not exceeded.

Log No. 95-048

Technical Specification Bases Change 95-009

This change revises the definition of "equilibrium" in the technical specification bases Section 3.1.2, "Reactivity Anomalies," following a reactor restart from a refueling outage. Due to xenon transients, reactor core flow is changed to compensate adjust total core power. These transients may last for up to 50 hours. This change triggers the performance of Surveillance Requirement (SR) 3.1.2.1, to verify core reactivity difference between monitored and predicted rod density within 24 hours after equilibrium conditions are reached. Equilibrium has been defined to be reached when steady state operations (with no control rod movement) at greater than 80% reactor thermal power have been obtained for at least 24 hours. Verifying the reactivity difference between the monitored and predicted rod density is within the limits provides further assurance that plant operation is maintained within the assumptions of the accident and transient analyses.

Log No. 95-049

Evaluation of Foreign Material in Reactor Vessel

This safety evaluation allows continued reactor operation with an unrecoverable cotter pin in the reactor annulus. Condition Report 1-95-03-068 was written to document that the pin could not be retrieved and that there would be no detrimental effect on plant operations with the pin left in the reactor vessel. It was determined that the material is compatible with the other materials in the vessel, and the probability of a malfunction of equipment important to safety has not been increased because of the pin's location in a low flow region.

Log No. 95-050

SRV Setpoint Determination for Acoustic Monitors

This change revises the main steam system safety relief valve (SRV) acoustic monitor setpoint determination methodology. The change allows the acoustic monitors to be tested and adjusted to alarm at a point above background noise levels. The former methodology required establishing normal plant operating conditions at less than 90 percent power prior to opening the SRV. The actual setpoint was then determined to be 25 percent of the value with the SRV full open. Setting the alarm using this methodology caused spurious nuisance alarms. The new method will allow the alarm to be set with the SRV closed. The change will allow the SRV to alarm as required by Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

Log No. 95-051

Supplemental Reload Licensing Report for Cycle 6

This safety evaluation documents the results of the reload licensing analyses for reload 5, cycle 6. Included in this analysis are the plant conditions assumed, identified fuel bundle types used and their proposed locations, applicable margin improvement and operating flexibility options, and core and transient analysis results. Some changes in critical power ratio, peak cladding temperature, reactor pressure and other parameters result from the changes in the nuclear design to meet cycle 6 energy requirements. However, there is no decrease in any margins of safety.

Log No. 95-052

Revise Diesel Generator Failure Reporting

This change eliminates Operational Requirements Manual (ORM) requirement 6.9.2.1 for reporting diesel generator failures. This change is consistent with the requirements in NUREG-1434, "Improved BWR-6 Technical Specifications," dated September 1992. A special report will only be required when a diesel generator has experienced 4 or more valid failures in the last 25 demands, which is consistent with the criteria for accelerated testing of the diesel generator in Technical Specifications (TS) 3.8.1 and 3.8.2. This change does not modify the TS testing requirements for the diesel generators or the reporting requirements of 10CFR50.72 or 10CFR50.73.

Log No. 95-053

Containment Weir Box Flow Indication Removal

This change removes from operation the containment floor and equipment drain weir box flow measuring devices and takes credit for alternate methods of leak detection. The weir box flow measuring devices (v-notch plates) are located in containment and are high maintenance devices caused by clogging. The clogging has caused high flow indications and false alarms. Alternate methods to determine leakage are monitoring containment equipment and floor drain sump fill and run timers and flow totalizers. The alternate methods will assure compliance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Log No. 95-056

Modification LD-027, Alternate Drywell Floor Drain Flow Monitoring

Modification LD-027 installs a leak detection (LD) system flow element on the existing drywell floor drain system sump pump discharge line. The new element will provide a flow signal to a flow totalizer and will aid in determining the amount of unidentified operational leakage. This new flow monitoring equipment will provide an alternate method for determining unidentified reactor coolant boundary leakage. The new equipment will provide a backup to the existing leak detection equipment and may prevent an unnecessary plant shutdown should the leak detection equipment become inoperable. This safety evaluation covers the first phase of installation prior to releasing the equipment for operation. No new failures are created by the change since the new equipment is passive in nature and the piping has been evaluated for the increased weight.

Log No. 95-057

Reactor Recirculation Pump Motor Bearing Oil Reservoir Installation

This change adds a constant level oiler device the lower bearing lubrication reservoir on each of the reactor recirculation (RR) system pump motors. The new oil reservoir ensures bearing coverage for the lower motor bearings, which have been found to be losing oil level during pump operation. The configuration has been evaluated for increased fire loading, seismic effects, and the potential for a gross failure of the reservoir. No new failures were identified.

Attachment 3
to U-602500

Attachment 3 contains the following information:

1) Filing information to aid in updating the Clinton Power Station Updated Final Safety Analysis Report (USAR). This information is not required in the binders.

2) Effective Page Listing for Revision 6 of the Clinton Power Station USAR.

3) Revised pages of Revision 6 to the USAR.