AmerGen

An Exelon/British Energy Company

Clinton Power Station
P.O. Box 678
Clinton, IL 61727
Phone: 217 935-8881

10CFR50.71(e)
10CFR50.59

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Submittal of the Updated Safety Analysis Report, Revision 9

In accordance with 10CFR50.71, "Maintenance of Records, Making of Reports," paragraph (e), AmerGen Energy Company, LLC (i.e. AmerGen) hereby submits one original and 10 copies of Revision 9 to the Clinton Power Station (CPS) Updated Safety Analysis Report (USAR). The submittal includes changes from July 1, 1999 through November 12, 2000.

In accordance with 10CFR50.59, "Changes, Tests and Experiments," paragraph (d)(2), a 10CFR50.59 report is attached that describes all changes implemented per this regulation. This report includes a description of changes affecting the USAR as required by 10CFR50.71, paragraph (e)(2).

This submittal is subdivided as follows:

1. Attachment A provides the 10CFR50.59 report for all changes during this reporting period.

2. Attachment B provides a summary of changes in commitments that CPS has evaluated and revised under administrative controls.

3. Attachment C provides a summary of the changes made to the CPS Operational Requirements Manual since the submittal of the CPS USAR Revision 8 to the NRC.

4. Attachment D provides a summary of the deletions made from the USAR since the submittal of the CPS USAR Revision 8 to the NRC, through November 12, 2000.

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5. Attachment E provides a copy of the revised USAR pages associated with Revision 9. Filing instructions are included in this attachment.

As required by 10CFR50.71, the information given in the attachments accurately presents changes made since the previous submittal and analyses submitted to the Commission or prepared pursuant to Commission requirement. Known USAR discrepancies are evaluated, tracked, and corrected via the CPS corrective action program. A Current Licensing Basis project is in progress at CPS.

Respectfully,

J. M. Heffley
Vice President
Clinton Power Station

EET/blf

Attachments:

	Affidavit
Attachment A:	10CFR50.59 Report
Attachment B:	Summary of Changes in NRC Commitments
Attachment C:	Operational Requirements Manual Changes
Attachment D:	USAR Deletions
Attachment E:	Revised USAR Pages

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DEWITT	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	**Docket Number**
CLINTON POWER STATION, UNIT 1	)	**50-461**

SUBJECT: Submittal of the Updated Safety Analysis Report, Revision 9

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

J. M. Heffley
Vice President
Clinton Power Station

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 4th day of

May, 2001.

" OFFICIAL SEAL "
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Jacqueline S. Matthias
Notary Public

Re-reported 50.59 Summaries

REACTOR RECIRCULATION PUMP VIBRATION MONITORING

Activity Evaluated: Modification RRF015 Log Number: 93-045 R/1

Modification RRF015 makes permanent two temporary modifications (TMs) previously installed. TM 90-031 reported for Updated Safety Analysis Report (USAR) Revision 3 installed a vibration monitoring system on the Reactor Recirculation (RR) system pumps. The monitoring system consists of two accelerometers and two velocity transducers for each pump. The accelerometers were mounted on the pumps' stuffing box lifting lugs using welded support blocks. The accelerometers measure pump casing vibration. The velocity transducers are mounted on support blocks welded to the top of the motor casing and measure lateral motor casing vibration. TM 92-007 later installed two horizontal proximity probes on each pump. These are also to monitor pump vibration. RRF015 changes these two TMs to permanent status and installs a personal computer in the control room computer room to provide continuous monitoring and analysis of vibration data for the RR pumps and motors. This modification changes USAR Appendix E and F fire protection information. The installation was evaluated for seismic loads, divisional separation, and fire loading and was found acceptable. This modification provides a monitoring function only; no trips or interlock permissives are associated with the operation of the vibration monitors. No new failures or accident types are created by the change since the new equipment is passive in nature and seismic and fire protection concerns have been evaluated. As reported in USAR Revision 8, this change had not been fully installed as of the cut-off date for USAR Revision 8 but has been completed as of the cutoff date for reporting in Revision 9.

ACTUATOR AND SUPPORTING HARDWARE REMOVAL FROM ECCS INJECTION CHECK VALVES

Activity Evaluated: Modification M-079 Log Number: 93-114

Modification M-079 removes the actuator hardware from the Emergency Core Cooling System (ECCS) testable check valves. This change prevents the remote operations of valves 1E12F041A, 1E12F041B, 1E12F041C, 1E21F006, and 1E22F005 during power operation. This remote operation capability was to satisfy a portion of the ASME Code, Section XI. An ASME Code, Section XI, relief request (2014) was submitted and approved by the NRC on September 13, 1993, to allow full stroke exercising of these valves only during refueling operations. This will be done by the lever which will remain installed on the split shaft and will allow the valve to be tested locally. Therefore, the valves will still function as testable check valves. These valves are no associated with any assumed accident initiators. They are associated with accident mitigation equipment; however, the design basis function and operating characteristics are not changed by this modification. Therefore, the probability and consequences of an accident or malfunction of equipment important to safety have not increased. This modification can not be postulated to initiate an accident or malfunction or equipment important to safety of a type different than that previously evaluated. This modification does not impact the Technical Specification requirements, nor does it impact the assumptions and bases used to establish those requirements. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification. As reported in USAR Revision 8, this change had not been fully installed as of the cut-off date for USAR Revision 8 but has been completed as of the cutoff date for reporting in Revision 9. See also Log Number 2000-068 summary in this revision.

ELIMINATION OF THE STEAM CONDENSING MODE OF RHR OPERATION

Activity Evaluated: Modification RH-033 Log Number: 93-115

The steam condensing mode of the Residual Heat Removal (RHR) System was intended to be used to increase plant availability by maintaining a hot standby condition until the plant could be restarted. This would be accomplished by directing main steam through a Reactor Core Isolation Cooling (RI) System line into one of the two RHR heat exchangers. The startup testing of this mode of operation was deleted by Safety Evaluation 87-1353 and reported in Updated Safety Analysis Report (USAR) Revision 0. As such, removing this as a startup test requirement and as a mode of operation at Clinton Power Station did not decrease the margin of safety or increase the consequences of an accident. In addition, Temporary Modification (TM) 90-04 removed the piping spool piece between the RI system piping and the RHR heat exchangers. This TM was reported in USAR Revision 2. Modification RH-033 makes this change permanent. This change will allow seven pipe hangers to be changed from snubbers to struts. The change provides for the de-termination of twelve Motor Operated Valves (MOV) at the Motor Control Center (MCC) and de-terminating of cables entering the Main Control Room (MCR). RH-033 includes extensive USAR changes to eliminate reference to this mode of RHR operation. The use of the steam condensing mode including the capability to vent non-condensibles is not considered in any design basis accident. The isolated nature of the system can not function as the initiator of any evaluated accident or malfunction or any equipment important to safety. Nor can this isolated system be considered as a credible initiator of an accident or malfunction of equipment important to safety of a different type than those previously evaluated. The Technical Specifications were issued predicated on the absence of this cooling mode. Therefore, documentation of the abandonment of the steam condensing mode of RHR operation does not result in a reduction of a margin of safety defined in the basis of any Technical Specification.

As reported in USAR Revision 8, this change has not been fully complete, and is still not fully complete as of the cutoff date for Revision 9. However, portions have been installed. USAR text and figures are revised as portions of this modification are released for operations.

MODIFICATION TO NSPS INVERTERS

Activity Evaluated: Modification IPF004 Log Number: 94-024

Modification IPF004 makes three changes to the Nuclear System Protection System (NSPS) inverters. The first change improves the system's capability to withstand Direct Current (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 of 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.

This change was reported in USAR Revision 6, when the change was partially implemented. This change has not yet been fully complete as of the cutoff date for USAR Revision 9.

CONDENSATE FILTRATION SYSTEM MODIFICATION

Activity Evaluated: Modification CP-020 Log Number: 94-075 R/1

Modification CP-020 changes the Condensate Polisher (CP) System by introducing condensate filtration. The Condensate Filters (CF) are placed upstream of the condensate demineralizers. One filter is located upstream of each demineralizer internal to the existing demineralizer cubicle. The CFs remove non-soluble impurities, primarily insoluble iron, from the condensate prior to the deep bed condensate demineralizers. This modification does not impact plant operation because the CP can operate using eight of the nine cubicles to maintain adequate condensate flow and pressure. As reported in USAR Revision 8, this modification was not complete as of the reporting cutoff date. Presently, there are four condensate filters released for operations and in service with one additional filter still under construction as of the cutoff date of this revision. The status of this modification will be reported in Revision 10 to the USAR.

MANUAL COMPENSATORY ACTION FOR DEGRADED VOLTAGE RELAYS

Activity Evaluated: OD 1-97-01-276, Action #1 Log Number: 97-054

This evaluation was performed for manual Compensatory Action Number 1 established as part of the Operability Determination (OD) for Condition Report (CR) 1-92-04-031. This CR identified that the voltage setpoints of the Division 1, 2, and 3 degraded voltage relays were set too low and were not sufficient to ensure proper low voltage protection for all class 1E equipment. As a result of that CR, Clinton Power Station (CPS) installed an annunciator alarm for the Divisions 1, 2, and 3 4160 volt buses and established operator actions in order to ensure adequate voltage for proper operation of all class 1E equipment. CPS annunciator procedure 5008.05, "4kV Bus Low Voltage," states: "IF 4.16 kV Class 1E bus voltage falls below 3890 VAC, THEN Start the Diesel Generators (DGs) and transfer the buses to the DGs...".

The operator action does not affect equipment malfunctions that are described in the Updated Safety Analysis Report (USAR). Therefore, the operator action will not increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. The USAR does not evaluate an accident concurrent with degraded voltage. The method of isolating from a degraded source is changed, but not the probability of needing to isolate. Therefore, this change does not increase the probability or consequences of an accident previously evaluated in the USAR. This change does not involve any manipulation of plant equipment not previously discussed in the USAR. As a result, operation of this equipment can not initiate an accident not previously analyzed. However, because of this change, the operator has to respond for all three divisions simultaneously. If the operator does not respond properly or quickly enough, there is a possibility of creating a malfunction of equipment important to safety. Therefore, this change is deemed to involve an unreviewed safety question. In accordance with 10CFR50.59, a license amendment has been submitted to the NRC seeking review and approval of the proposed change (reference Letter U-602794, dated July 20, 1997). In addition, plant modifications are in progress to permanently resolve the degraded voltage issue (reference Modifications AP-37 and AP-38). As an update from this Revision 8 summary, the following has since been completed before the cutoff date for Revision 9; Modifications AP-37 and AP-38 have been completed, released for operations, and the equipment placed in-service. All system modifications have been completed to eliminate the low voltage condition as of the cutoff date for reporting in Revision 9.

CONFORMANCE WITH GDC-17 – ELECTRIC POWER SYSTEMS

Activity Evaluated: Tech Spec Bases Change Log Number: 97-151

The purpose of this change is to revise the basis for how Clinton Power Station (CPS) complies with General Design Criteria (GDC) 17. A section of GDC-17 requires that each of the two independent offsite power sources shall be designed to be available in sufficient time following a loss of all onsite Alternating Current (AC) power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. In the event of a loss of onsite AC electric power, both the 138 kV and 345 kV offsite sources are considered to be available immediately. During certain grid conditions, it has been predicted that the voltage of either offsite source may be too low to be an immediate source. However, evaluations indicate that there is sufficient recovery time to ensure that fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

The Updated Safety Analysis Report (USAR) does not evaluate the consequences of accidents or equipment malfunctions concurrent with having only one of the two offsite sources available. Therefore, the probability of an accident or a malfunction of equipment important to safety may be slightly increased, by operator actions specified in the bases change will help ensure that GDC-17 will be met. Thus, these actions will help offset any potential increased probability or consequences. No new type of accident has been created, because the loss of offsite power is the bounding analyzed event. No possibility of an equipment malfunction important to safety has been created that has not been previously evaluated, because no new failure modes have been created. It has been interpreted that the Nuclear Regulatory Commission's (NRC) "acceptance limit" is "two immediate sources," therefore, this change is a deviation from what was previously "accepted." This change is a reduction in the margin of safety as defined in the basis for the Technical Specifications.

In accordance with 10CFR50.59, a license amendment has been submitted to the NRC seeking review and approval of the proposed change (reference letter U-602794, dated July 20, 1997). In addition, plant modifications are in progress to permanently resolve the degraded voltage issue (reference Modifications AP-37 and AP-38). As an update from this Revision 8 summary, the following has since been completed before the cutoff date for Revision 9; Modifications AP-37 and AP-38 have been completed, released for operations, and the equipment placed in-service. All system modifications have been completed to eliminate the low voltage condition as of the cutoff date for reporting in Revision 9.

INSTALLATION OF STATIC VAR COMPENSATOR FOR RESERVE AUXILIARY TRANSFORMER

Activity Evaluated: Modification AP-37, ECNs 30526, 30527, 30528, and USAR change 8-205 Log Number: 98-087

Engineering Change Notices (ECN) 30526, 30527, and 30528 provide for installation of the Reserve Auxiliary Transformer (RAT) Static VAR Compensator (SVC). The SVC is designed to quickly and efficiently change the total impedance of the Offsite Power System to match the capacitance and inductance. The use of the SVC to match the capacitance and inductance reduces the reactive component of the current to a negligible value and leaves only the real current component flowing through the RAT, thus lowering the voltage drop and raising the voltage at the 4 kV terminals of the RAT. The SVC can accomplish this task very quickly and accurately by use of reactors, capacitors, and thyristors. The equipment and control building for the RAT SVC are installed on foundations which were designed and installed under Modification AP-036. The power for the RAT SVC is supplied from two sources; one from a 4 kV-480 volt transformer in the SVC yard and the other from a 480 volt Balance of Plant (BOP) source which is normally fed from the Unit Auxiliary Transformer (UAT) or the RAT.

The work authorized by this modification is the mounting and wiring for SVC switches and meters in the Main Control Room (MCR), the additional wiring for the auxiliary contacts from the safety related and non-safety related switchgear, and the power supply for station power. The new design for the wiring and meters mounted on MCR panel P870 meets the same standards as the rest of the circuitry. The installation of these components could introduce some risk of human error causing a Loss of Offsite Power (LOOP) since both RAT and Emergency Reserve Auxiliary Transformer (ERAT) controls are located on P870. No connections are being made from RAT to ERAT circuits and the new wiring uses existing terminal blocks and wiring. The installation of the RAT SVC will be performed in accordance with plant procedures. The installation of the additional wiring for the auxiliary contacts from the safety related and non-safety related switchgear will be done both outside of and during divisional outages, but will have little or no interaction with a RAT or ERAT outage. The station power design brings power from the plant to the SVC. No connection will be made between the plant and the SVC until Nuclear Regulatory Commission (NRC) approval has been received for requested Technical Specification changes. This activity does not affect any safety-related systems' ability to perform its intended function, nor unnecessarily challenge the system during installation. The scope of work involving wiring from divisional components to the optical isolators will be performed within the scope of divisional outages with appropriate Technical Specification (TS) Limiting Condition of Operation (LCO) and with risk assessments as defined by plant procedures. Therefore, this activity does not increase the probability or consequences of an accident or malfunction or equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). Since the wiring, cabling, meters and switches meet existing design standards (in particular, for electrical separation), this activity does not introduce any new failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The installation of the RAT SVC has no effect on the Diesel Generator (DG) nor onsite power systems' ability to perform TS required functions, and thus, have no effect on any acceptance values as long as no connections are made between the plant and the RAT SVC. Therefore, this activity does not reduce an margin of safety as defined in the basis for any TS.

This completes the re-reporting of 50.59 summaries from USAR Revision 8.

New 50.59 Summaries

RELOCATION OF FIRE EXTINGUISHERS

Activity Evaluated: ECN 30738, USAR Change 8-104 Log Number: 98-029 R/1

Engineering Change Notice (ECN) 30738 relocates the fire extinguishers in Fire Zone CB-6b at elevation 800'-0" and Fire Zone CB-1e at elevation 737'-0". Relocating the extinguishers makes them more accessible, maintains the extinguishers within the same fire zones, and maintains compliance with National Fire Protection Association (NFPA) requirements. These new locations were incorporated in Updated Safety Analysis Report (USAR) Appendix E Figures FP-10b and FP-14b. USAR Chapter 15 does not discuss any accident analyses initiated by a fire extinguisher. The fire extinguishers are used to mitigate the consequences of a design basis fire and will continue to serve its intended function in the new location. The portable fire extinguishers can not cause an equipment malfunction except to fall on it, which is precluded by the extinguisher mounting being designed for seismic loads. Since these extinguishers meet the current NFPA specifications and are to be staged in the same global area of the plant, the potential for missile hazards have not been increased. A review of the USAR Figures FP-10b and FP-14b indicate that there is no equipment important to safety in the immediate area of the extinguishers. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The relocation makes the extinguishers easily accessible and provides for a more efficient use by the fire brigade; it is not probable that the extinguishers would initiate any type of accident. Also, there is no equipment important to safety in the immediate area of the extinguishers. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. All acceptance values or design limitations to the Clinton Power Station Fire Protection System have been previously deleted from the Technical Specifications and reflected in USAR Appendix E and USAR Appendix F. Therefore, this change does not reduce a margin of safety as defined in the basis for any Technical Specification.

REMOVAL OF AUTOMATIC CLOSING DEVICES FROM ROLLING STEEL FIRE DOORS

Activity Evaluated: ECNs 30641 through 30648 and USAR Change 8-125 Log Number: 98-030

The subject Engineering Change Notices (ECNs) remove the automatic closing device on the rolling steel doors. The automatic closing device design of the rolling steel fire doors are a personnel safety hazard; removing the automatic closing device will remove this safety hazard. This activity does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, because the compensatory measures taken, whenever these doors are open, are the same as measures taken whenever these doors are taken out of service or for repair. Clinton Power Station Procedure 1893.01, "Fire Protection Impairment Reporting," already states that when the doors are open, they will have continuous fire watches posted at the doors and will be inspected every 24 hours to assure that the doors are in a closed position. When closed, the function of the fire door is to act as a barrier and ensure that the fire does not spread. The doors will still function as fire rated doors; this activity does not affect the structural integrity or fire resistivity of the doors. NUREG-0123 Revision 3 states that for non-functional fire doors, in fire zone boundaries protecting safety related areas, a continuous fire watch shall be established on at least one side of the affected penetration or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. The compensatory measurements meet this requirement. There are no specific regulatory requirements that fire doors need to be automatic closing. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). Since the fire walls retain their integrity and fire resistivity, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. All acceptance values or design limitations to the Clinton Power Station Fire Protection System have been previously deleted from the Technical Specifications and reflected in USAR Appendix E and USAR Appendix F. Therefore, this change does not reduce a margin of safety as defined in the basis for any Technical Specification.

DIVISION 2 NSPS INVERTER MONITORING

Activity Evaluated: Procedure 2813.02 R/0 Log Number: 98-039

This activity is to monitor the Division 2 Nuclear System Protection System (NSPS) loads and effect on the associated inverter. The NSPS bus loads will be monitored using a high speed recorder to record current wave forms at the NSPS bus breakers. This procedure is to be performed in Mode 4 (Cold Shutdown) only and is to determine the cause of the inverter transferring from the normal power source to the alternate power source. The test equipment used is considered passive, consisting of clamp-on current detectors and high impedance voltage probes. The intent of the procedure is to operate the NSPS inverter & bus in the same manner as normal operating conditions but have monitoring instrumentation in place to gather data on the transfer logic to help identify the cause of the inverter transfer. Since the monitoring is being performed in Operational Condition (Mode) 4 – Cold Shutdown, and the inverter and bus are declared INOPERABLE per Technical Specifications (TS), this activity will not increase the probability or consequences of an accident previously evaluated in the USAR. For the same reasons, this activity will not increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. Also, this activity will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than evaluated previously in the USAR. Lastly, for the same reasons (Mode 4 and INOPERABLE), this activity will not reduce the margin of safety as defined in the basis for any technical specification.

HVAC TESTABILITY DESCRIPTION REVISIONS

Activity Evaluated: USAR Change 8-130 Log Number: 98-061

Updated Safety Analysis Report (USAR) Change 8-130 revises the testability description for specified Heating, Ventilating, and Air-Conditioning (HVAC) system control logic circuitry. The intent of this activity is to describe the actual testing methodology on installed plant equipment, to reflect how Clinton Power Station meets the Regulatory Guide and the Institute of Electrical and Electronic Engineers Standard commitments, and to standardize the testability descriptions among the affected HVAC systems. This activity does not impact any design basis accidents discussed in Chapter 15 of the USAR. This activity does not change the design, material, and construction standards applicable to this system. Since there are no physical changes associated with this USAR revision, system response characteristics, operational design limits, and interaction with other systems remain unaffected. This activity does not affect any environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not modify or add any structures, systems or components, nor does it establish any new system interactions. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The USAR change does not revise any Technical Specification acceptance criteria, safety limit, or Bases. Also, this activity does not impact any of the specific testing requirements delineated in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INOPERABLE FUEL POOL COOLING CONDUCTIVITY MONITORS

Activity Evaluated: USAR Change 8-131 and Modification FC-027 — Log Number: 98-064

The Fuel Pool Cooling (FC) conductivity monitors 1CTFC09, 10, 11, 2CTFC010, and 011 were declared inoperable because they did not meet their operability criteria due to insufficient operating range. Plant Modification FC-027 upgrades the conductivity monitors to have the correct operating range. One of the design basis functions of the fuel pool filter demineralizer is to maintain fuel pool clarity. The conductivity monitors provide information on the effectiveness of the filter demineralizers in performing their purification function. The monitors can also provide indication of chemical intrusion. In the absence of the monitors, grab samples can also provide this information. Without the high conductivity monitor alarm, the fuel pool conductivity could exceed the limit during the grab sample interval. However, because corrosion causes slow cumulative damage, a conductivity exceeding the limit for this period would not significantly increase the total corrosion damage. The monitors are not associated with the postulated initiators of a fuel handling accident. Neither the monitors, nor the fuel pool filter demineralizers are credited as providing any mitigatory function for a fuel handling accident. Therefore, this change does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The effects of higher fission product concentrations and degraded water purity have previously been evaluated for their potential effect on analyzed accidents and equipment malfunctions. There are no known malfunctions of equipment important to safety or accidents of a different type than previously evaluated in the USAR associated with this activity. The Technical Specifications do not discuss the conductivity of the spent fuel pool. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REMODELING OF NRC/IDNS/OCC OFFICE SPACE

Activity Evaluated: ECNs 30823, 30824 and 30825; USAR Changes 8-140 and 8-142 — Log Number: 98-072

Engineering Change Notices 30823, 30824, and 30825 relocate the Nuclear Regulatory Commission (NRC) and Illinois Department of Nuclear Safety (IDNS) from the Northeast corner of the second floor of the Service Building to the Southwest corner of the first floor of the Service

Building. The perimeter walls of the new NRC space are designed such that they provide visual and acoustical privacy. Also, two additional offices will be added adjacent to the NRC and IDNS space and the Outage Control Center in the Service Building basement has been changed to split the conference room layout into two offices.

Updated Safety Analysis Report (USAR) Chapter 15 does not discuss any design basis accident that would be affected by this activity. The activity involves changes to non-safety, non-quality related structures, systems and components which are not required for safe shutdown of the plant or mitigation of the affects of design basis accidents. The remodeling activities meet the applicable design, material, and construction standard requirements of the original construction. This activity does not affect overall system performance such that it changes system response characteristics, causes system operation outside of its design limits, causes operational transients in the system, or causes interaction with other systems. Also, this activity does not affect the environmental, seismic, or separation criteria of any system required to operate during a design basis accident. In addition, this activity does not change, degrade, or prevent actions described or assumed in the accident analysis for mitigating the effects of any accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Failure modes of systems, structures, and components added by this activity include structural failure of components, leakage to the compressed air system in the Service Building, and shorts in the electrical system. Structural or design barriers exist for each mode of failure added by this activity, which preclude these failures from affecting safety systems, structures, and components. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. None of the systems or structures affected by this change are governed by the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE FIXED TAP ERAT TRANSFORMER WITH LOAD TAP CHANGER TYPE

Activity Evaluated: Modification AP-40, Supplement 6
ECN 30897, USAR change 8-154
Log Number: 98-081 R/1

The Emergency Reserve Auxiliary Transformer (ERAT) is a fixed tap transformer (no adjustment of voltage ratios while in service) and is being replaced with a transformer which has the capability to respond to variations in grid voltage by changing voltage ratios (taps) while in service and carrying load. The replacement transformer is also larger in mega-volt-amperes (MVA) capacity. While the load tap changer (LTC) of the transformer can operate manually or automatically, this activity allows operation only in manual mode. The automatic control circuit is disconnected and voltage control will be performed by the Static VAR Compensator (SVC). New controls, power source for transformer cooling fans, and fire protection changes are also part of this activity. Also part of this activity is that the replaced transformer will be left in-place, full of oil, and the replacement transformer is being placed near the old one. Since the change is being performed in Operational Condition (Mode) 4 – Cold Shutdown, and the transformer (offsite power source) are declared INOPERABLE per Technical Specifications (TS), this activity will not increase the probability or consequences of an accident previously evaluated in the USAR. For the same reasons, this activity will not increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. Also, this activity will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than evaluated previously in the USAR. Lastly, for the same reasons (Mode 4 and INOPERABLE), this activity will not reduce the margin of safety as defined in the basis for any technical specification.

FEEDWATER LEAKAGE CONTROL MODIFICATION

Activity Evaluated: Modification FW-039 and USAR Change 8-187 Log Number: 98-110

Modification FW-039 installs a system to provide a water seal at the feedwater primary containment penetrations, thereby changing the testing requirement for the applicable containment isolation valves to a periodical functional water test. The new method of sealing the primary containment feedwater penetration isolation valves will be provided by the addition of a Feedwater Leakage Control (FWLC) mode of the Residual Heat Removal (RHR) system. The installation of the new FWLC will not affect operation or functional performance of any plant system since the new piping and electrical components have been designed and analyzed to the current design and licensing basis requirements. The impact of a failure in the new moderate energy FWLC piping on the RHR Emergency Core Cooling System (ECCS) functions is bounded by Updated Safety Analysis Report (USAR) Section 6.3.3.7.8.4 which evaluated a failure of the RHR Low Pressure Coolant Injection (LPCI) lines. The new motor-operated valves will be tagged out of service in the closed position and the manual block valves in the new FWLC piping will be tagged out in the closed position to prevent inadvertent operation of the FWLC. The tagouts ensure that there is no change to electrical load demand. The new components are seismically designed to ensure they will not fail in a manner which could prevent the proper function of safety related components located in close proximity. The installation phase of this activity does not impair system reliability by imposing transients not analyzed in the design basis for the system or equipment protective features, degrade support or attendant system performance, or reduce system redundancy or independence. Tests will be performed after installation of the new containment isolation valves and Closed Loop Outside Containment boundary valves to ensure the leak rates are within the acceptance limits of the existing Appendix J leak test program. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The only credible accident which can result from the new components of the FWLC system is a failure of the new high energy lines between the FW tap connections and the FWLC check valves. Failure of these lines is bounded by the spectrum of pipe break accidents analyzed in USAR Sections 6.3, 15.6.5, and 15.6.6. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The margin of safety for the offsite dose consequences of postulated accidents that is directly related to the primary containment leakage rate and the secondary containment bypass leakage rate will be maintained by meeting the acceptance criteria. All primary and secondary containment leakage testing continues to be performed in accordance with Technical Specification requirements. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FEEDWATER ZINC INJECTION TIE-IN

Activity Evaluated: ECN 30955 and USAR Change 8-188 Log Number: 98-111

Engineering Change Notice (ECN) 30955 installs the tie-ins for the zinc injection skid. The tie-in will be off the common Feedwater (FW) discharge line, 1FW01B30, and the tie-in to the Condensate system will be in the common condensate booster suction line, 1CB07B36. The installation of zinc injection tie-ins will not impact any of the initiating events for the design basis accidents associated with the FW system described in Chapter 15 of the Updated Safety Analysis Report (USAR). This activity does not affect any of the feedwater heater piping, valves, or controls, or any control devices or circuits associated with the FW controller. In addition, the capped lines with isolation valves could not lead to any operator errors since the tie-in lines are blocked and would not affect system performance if mispositioned. Also, this ECN only affect piping upstream of the outermost isolation valve, and therefore, does not affect any failure modes for this incident. Since, the flow through the system will not be affected, the system performance remains unchanged. The new piping is designed to the same codes and standards as the original piping. There are no interconnections with safety-related system and no specific regulatory requirements are imposed on the system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The only new credible failure would be a piping failure of the new piping. Pipe stress calculations have shown that the 1.5 inch piping has a negligible affect on the 30 and 36 inch run of feedwater and condensate booster piping. The new piping is installed to the same codes and standards as the original piping. Therefore, a piping failure in the new piping is bounded by the loss of feedwater accidents analyzed in Chapters 6 and 15. This ECN only affects the operation of the Condensate and FW systems which are not safety related systems and are not required for safe shutdown of the plant. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. These changes are made to non-safety related equipment and systems which are not associated with any margins of safety as specified in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

SHUTDOWN SERVICE WATER PUMP STRAINER BACKWASH VALVES

Activity Evaluated: USAR Change 8-189 Log Number: 98-115

Updated Safety Analysis Report (USAR) Change 8-189 adds the Shutdown Service Water (SX) system strainer backwash valves 1SX013D, 1SX013E, and 1SX013F to USAR Table 3.9-5. This will subject the valves to the functional testing requirements during the life of the plant. Adding these valves to USAR Table 3.9-5 is required to be consistent with the active safety function these valves perform. The SX system is designed to provide a reliable source of cooling water to mitigate the consequences of an accident; thus, the only accident that can be initiated through malfunction of the SX system is the flooding accident. This activity does not involve a change to personnel qualification/assignment or affect procedural steps. Valves 1SX013D, 1SX013E, and 1SX013F are located upstream of all station auxiliaries such that any change to these valves will not have an impact on the detection and isolation of a possible leak of radioactive material into the SX system. With cooling water flows unchanged, area temperatures within Secondary Containment will remain within design limits and the Standby Gas Treatment System will be capable of drawdown of Secondary Containment to -0.25 inch water gauge pressure as analyzed. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. No new flow paths are created, no electrical or mechanical separation requirements are affected, and no system will be operated outside of its design limits as a result of this activity. In addition, this activity does not does not alter or introduce any structures, systems or components or system interfaces. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. This activity does not alter any Technical Specification safety limits, acceptance values, limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

AR/PR MAIN CONTROL ROOM (MCR) LOCAL AREA NETWORK (LAN)

Activity Evaluated: Modification PR-040, Supplement 1; USAR Change   Log Number: 98-116 R/1

Modification PR-040, Supplement 1 upgrades the Area and Process Radiation Monitoring (AR/PR) system in the Main Control Room (MCR) and Technical Support Center (TSC). The Eberline Computer Control Terminal (CCT) (1H13-P864) is being replaced by a Central Server that performs the same function of communicating with and accumulating information from the various radiation monitoring field units. The CCT monitor in the TSC is being replaced with a state-of-the-art computer that performs the same functions. In addition, the TSC computer will act as the communication device for a new Radiation Protection (RP) Office computer. This new computer will provide the capability for RP to monitor status and parameters of the radiation monitoring field units. Another new computer (1H13-P870) will be connected to the central server allowing alarming capability and access to current and historical radiological information directly to the shift operating crew. The 1H13-P864, 1H13-P870 and TSC computers, a hub and printer located in the TSC, and interconnecting cables are collectively referred to as the AR/PR MCR Local Area Network (LAN). The AR/PR central monitoring system provides no information necessary for mitigating any design basis accidents. The accident scenarios in Updated Safety Analysis Report (USAR) Chapters 6 and 15 do not take credit for the AR/PR Central Monitoring System. The AR/PR MCR LAN has no affect on any accidents, cannot mitigate the effects of any accident, and will not alter the ability of any of the safety related radiation monitoring devices to withstand them. Operations is now the sole work authority for all of the centrally monitored portions of the AR/PR System; this change will support the direct monitoring of conditions without relying on information relayed by RP. Operations of the new interface will be significantly more efficient and operator friendly than the old system. These additional features improve the operator's ability to monitor and respond to plant conditions. All monitors are optically isolated such that a failure of the central monitoring system does not impact the safety-related function or devices. Each monitor has been designed to meet the environmental and seismic criteria for its own specific location. The new system will improve operator knowledge and better prepare them to identify radiological entry conditions. System reliability is enhanced by providing more direct operator involvement. All essential radiological information is still available where needed. Those monitors which are required to identify gross breeches of barriers, and those monitors which cause automatic actions are not dependent on the central monitoring system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The credible failure modes associated with this activity are hardware, software, and personnel error. A review of USAR Chapters 6 and 15 and the event with which the plant was designed to cope has indicated that the credible malfunctions are bounded by the existing USAR evaluations. This activity does not introduce any failures that can affect the ability of plant systems to perform their safety function. The system design and operating procedures will prevent or rapidly identify any inappropriate parameter change. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications address a number of radiation monitors and the requirement for that instrumentation to be operable. The central monitoring system is not addressed by the Technical Specifications. There are no Technical Specification acceptance values impacted by this modification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CARD READER INSTALLATION AT MAIN CONTROL ROOM ENTRANCE DOOR

Activity Evaluated: Modification SS-063 and USAR Change 8-194 Log Number: 98-121 R/1

Modification SS-063 converts the function of Main Control Room (MCR) Security Door SD-1-37 to a card reader controlled door, similar to that of MCR Doors SD-1-34, 36, and 38. These changes are consistent with descriptions and requirements set forth in the Updated Safety Analysis Report (USAR), Physical Security Plan (PSP), and other Licensing Basis Documents. However, the USAR and Plant Security Plan (PSP) both contain MCR floor plan figures which show MCR Door 37 as a security key controlled locked and alarmed door, labeled "Emergency Exit Only." This activity changes Door 37 from 'locked/alarmed emergency exit' to 'card reader controlled routine use' classification. USAR Section 6.4.2.5 describes Shielding Design. Radiation shielding for the control is based upon radiation sources released from the core following a postulated loss-of-coolant accident and protects the inhabitants of the MCR. The shielding design is unaffected due to the installation of penetration seals. The penetration seals installed by this modification meet the same design requirements as adjacent seals. This activity does not impair the ability of the control room ventilation system or degrade the performance by maintaining the integrity of the boundary. The availability of the ventilation system is not impacted because the changes are to structures and the changes maintain the pressure and seismic qualification of the structure. This activity does not degrade system reliability or performance. The operation of the emergency ventilation system has not been altered by this design change and the shielding provided by the penetration seals ensure that the analysis of the effects of an accident are not changed. Nor does this change alter fission product barriers. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The change in classification of Door 1-37 provides another means of access to the control room, during normal or accident operation and may actually increase the ability of the control room staff to respond to the plant by having additional means of ingress and egress. Door 1-37 was built and is maintained to the same standards as the other point of access to the MCR. The addition of the card reader ensures access to the MCR is maintained. The penetrations are designed to the same design requirements for the MCR for seismic, radiation, fire barriers and pressure boundaries. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification Bases 3.3.7.1 discusses Control Room Ventilation System; this activity does not degrade these requirements. The Technical Specification Bases references the PSP and Emergency Plan (EP) in the introductory operating license document. This document requires that the licensee fully implement and maintain in effect all provisions of the PSP and

EP. This activity is consistent with the existing PSP and EP. Therefore, this activity does not reduce a margin of safety as defined in the basis for any technical specification.

SECURING OF DIVISION 2 AND 3 SX PUMP ROOMS WATERTIGHT DOORS

Activity Evaluated: CPS 1893.04M800, M801, M803; USAR Change 8-199 | Log Number: 98-123

Condition Report 1-97-10-530-0 identified the watertight doors between Fire Zone M-2b and the screen house corridor and Fire Zone M-2a and the screenhouse corridor as having an indeterminate compliance with 10CFR50 Appendix R requirements for Fire Brigade access. These doors are secured in the closed position with a chain and clip. This procedure revision changes the note to the M-2 Pre-Fire Plan regarding the inside chain to eliminate reference to the Division 2 door and to indicate access from Division 2; it also adds a note that states: "Water tight door from Div. 2 SX pump room to screen house hallway is chained and padlocked closed from outside". These watertight doors are credited for flooding protection of safety-related equipment in the Circulating Water Screen House. The chaining arrangement for the access door to this zone has no adverse effect on the safe shutdown capability of this zone for a fire in another area. For a fire in Fire Zone M-2b, Updated Safety Analysis Report (USAR) Appendix F assumes a complete loss of the Method 2 Safe Shutdown capability and credits Method 1 for Safe Shutdown. The chaining arrangement has no impact on Method 1 Safe Shutdown. Also, this change does not effect the flood protection provided by the door. Nor does this change have any effect on the Physical Security Plan. In addition, this activity does not impact the primary or secondary containment boundaries or the Control Room boundary; systems which maintain the required pressures in the areas they serve and filter out the radioactive materials from the air; or any setpoints for the actuation of safety systems. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not adversely affect the availability, performance or reliability of any structure, system, or component. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Fire Protection Program is not included as part of the Technical Specifications. As previously discussed, this activity does not affect the ability of the watertight door to perform its intended function, nor does it adversely impact the ability to access the safety related equipment. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REDUCING IRM D AND H NOISE BY ADDING FERRITE BEADS AND GROUNDING PREAMPLIFIER

Activity Evaluated: ECN 31051 | Log Number: 98-124

Intermediate Range Monitor (IRM) channels D and H are experiencing increased noise levels on the input signal. Trouble shooting performed by the system engineer and an off-site noise reduction specialist determined that installation of ferrite beads and grounding the preamplifier chassis increases the signal to noise ratio significantly. No credible failure modes exist as a result of this plant change that can initiate a low power accident. The IRM system is an accident mitigator, not an initiator. Also, this plant change improves the current design in that it filters additional noise. Although the IRMs are not directly credited in the Updated Safety Analysis Report (USAR), they are indirectly credited in the accident analysis to provide a scram function due to high neutron flux. The fundamental function and objective of the system is not affected by this activity and the activity was analyzed not to have an adverse affect on any other system. No credible failure mechanism exists that would allow the ferrite beads to adversely affect the IRM circuit. The beads are passive components and are not intrusive. The installation of the ferrite beads and grounding of the preamplifier are seismically analyzed. In addition, this plant change meets the design, material and construction requirements of the IRM system. The design change does not degrade the performance below the design basis, by affecting the environmental, seismic, or separation criteria, of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure modes which could be introduced by this change are failure of a ferrite bead, seismic interaction, the burning of a bead jacket, and the possibility of a ground loop being created. Since the ferrite beads are not intrusive to the system, have no moving parts, do not require any external power, and are seismically supported, the failure of a bead is not credible. Since the bead supports are designed to be seismic, no failure of the supports or the connected beads due to a seismic event need be postulated. Since the energy level of the signal is so low, the beads can not be the initiator of a fire. A ground path will not be created unless a capacitor in the grounding strap shorts. The effect of a short would be considered a single failure and is bounded by the analysis in the USAR. Therefore, this change does not create the possibility of an accident or equipment malfunction of a different type. IRM channel operability is required by Technical Specification 3.3.1 for various modes of plant operation. Since the function and design basis of the IRM system is unaltered by the modification and the IRM system reliability and functionality is enhanced, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REDUCING SRM D NOISE BY ADDING FERRITE BEADS AND GROUNDING PREAMPLIFIER

Activity Evaluated: ECN 31023 Log Number: 98-125

Source Range Monitor (SRM) channel D is experiencing increased noise levels on the input signal. Trouble shooting performed by the system engineer and an off-site noise reduction specialist determined that installation of ferrite beads and grounding the preamplifier chassis increases the signal to noise ratio significantly. No credible failure modes exist as a result of this plant change that can initiate a Control Rod Drop Accident or a Fuel Handling Accident. The SRM system is an accident mitigation system, not an initiator. Also, this plant change improves the current design in that it filters additional noise. The fundamental function and objective of the system is not affected by this activity and the activity was analyzed not to have an adverse affect on any other system. No credible failure mechanism exists that would allow the ferrite beads to cause an initiating event for control rod accident or fuel handling accident. The beads are passive components and are not intrusive. The installation of the ferrite beads and grounding of the preamplifier are seismically analyzed. In addition, this plant change meets the design, material and construction requirements of the SRM system. The design change does not degrade the performance below the design basis, by affecting the environmental, seismic, or separation criteria, of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure modes which could be introduced by this change are failure of a ferrite bead, seismic interaction, the burning of a bead jacket, and the possibility of a direct current (DC) ground loop being created. Since the ferrite beads are not intrusive to the system, have no moving parts, do not require any external power, and are seismically supported, the failure of a bead is not credible. Since the bead supports are designed to be seismic, no failure of the supports or the connected beads due to a seismic event need be postulated. Since the energy level of the signal is so low, the beads can not be the initiator of a fire. A second ground path is created which is acceptable for high frequency applications such as the SRMs. Therefore, this change does not create the possibility of an accident or equipment malfunction of a different type. SRM channel operability is required by Technical Specification 3.3.1.2 for various modes of plant operation. Since the function and design basis of the SRM system is unaltered by the modification and the SRM system reliability and functionality is enhanced, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONTROL ROOM HVAC DATA ACQUISITION

Activity Evaluated: CPS 2800.98, Rev. 0 Log Number: 98-131

This activity creates Clinton Power Station Procedure 2800.98, "Control Room HVAC Data Acquisition Test". The purpose of these testing activities is to obtain data that will be sued to model the Control Room Ventilation System (VC) based on air flow and pressure drops. This data will be used as a basis for future changes to the VC system to minimize the noise level in key areas of the control room envelope in support of the Control Room Noise Reduction Project. It is noted that this procedure will be performed with the plant in Mode 4, with no Operations with a Potential to Drain the Reactor Vessel (OPDRV), no core alterations, or movements of irradiated fuel in primary or secondary containment. With the plant in these conditions, the VC system is not required to be operable and there will be no adverse impact to plant safety during the performance of this procedure. The VC system is not an initiator of the accidents evaluated in Chapters 6 and 15 of the Updated Safety Analysis Report (USAR). The VC system does provide support to the people and equipment in the Control Room. However, the changes to the VC system per this procedure will not adversely affect the ability of the VC system to perform its design basis functions. No new ignition sources are added as a result of this activity. Fire Protection has determined that none of the Fire Load Rating for the five fire zones involved will be changed from their current "Moderate" rating. The minor changes to the VC system per this procedure will not reduce the ability of the VC system to perform its design basis functions of controlling temperature, humidity, and radiological exposure for the Operator in the Control Room envelope. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. For impaired fire barriers, the use of compensatory measures, which are based on written procedures, meet the criteria established in the USAR, such that they are considered to be previously evaluated. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications 3.7.3 and 3.7.4 require the VC system to be operable during Modes 1, 2, 3 and during OPDRV, core alterations, and movement of irradiated fuel in the primary or secondary containments. This procedure will be performed during plant conditions when the VC system is not required to be operable. Also, the VC system will be able to perform its design basis functions for maintaining temperature, humidity, and radiological conditions within the required limits. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DIVISION 2 DEGRADED VOLTAGE MODIFICATION

Activity Evaluated: Modification AP-34, Supplement 2 Log Number: 98-132
USAR change 8-204

A detailed evaluation of electrical power distribution systems identified both under-voltage and over-voltage conditions could exist at various 120 volt circuit loads as a result of expected variations on the offsite electrical grid. Some loads were found to have under-voltage even at nominal grid conditions because of lengthy and/or heavily loaded circuits.
Modification AP-34 includes the following major activities:

1. Addition of five 480-208/120 volt 15 kVA, 3-phase regulating distribution transformers and modular distribution panels to Motor Control Centers (MCC) B, F2, H, 1B1, and 1B4.
2. Reconfiguring of MCC compartments as necessary by moving, adding, and deleting breakers, cables, and distribution panels.
3. Reassignment of loads to regulating or non-regulating transformers or movement of circuits from one MCC to another MCC within the same division.
4. Replacement of paralleling of cables to leads to reduce voltage drop.
5. Use of fast-acting fuses to mitigate the effects of saturation or regulating transformers on postulated faults in place of the individual circuit breakers.

The addition of new regulating transformers adds additional load on the Division 2 Diesel Generator (DG) (less than 20kW). The additional load is well within the bounds of the available excess capacity of the Division 2 DG, therefore, the additional load is acceptable.

The changes are being performed in Operational Condition (Mode) 4 – Cold Shutdown and the electrical circuits (distribution systems) are declared INOPERABLE per Technical Specifications (TS). Evaluations have shown that this activity will not increase the probability or consequences of an accident previously evaluated in the USAR or increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. Also, this activity will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than evaluated previously in the USAR. Lastly, for the same reasons, this activity will not reduce the margin of safety as defined in the basis for any technical specification.

DIVISION 2 DEGRADED VOLTAGE MODIFICATION

Activity Evaluated: Modification AP-34, Supplement 4 Log Number: 98-133
USAR change 8-206

A detailed evaluation of electrical power distribution systems identified both under-voltage and over-voltage conditions could exist at various 120 volt circuit loads as a result of expected variations on the offsite electrical grid. Some loads were found to have under-voltage even at nominal grid conditions because of lengthy and/or heavily loaded circuits.
Modification AP-34 includes the following major activities:

1. Addition of five 480-208/120 volt 15 kVA, 3-phase regulating distribution transformers and modular distribution panels to Motor Control Centers (MCC) B, F2, H, 1B1, and 1B4.

2. Reconfiguring of MCC compartments as necessary by moving, adding, and deleting breakers, cables, and distribution panels.
3. Reassignment of loads to regulating or non-regulating transformers or movement of circuits from one MCC to another MCC within the same division.
4. Replacement of paralleling of cables to leads to reduce voltage drop.
5. Use of fast-acting fuses to mitigate the effects of saturation or regulating transformers on postulated faults in place of the individual circuit breakers.

Supplement 4 consists of modifications to five control circuits which involve reduction of circuit load by adding new feeds, developing new feed sources for certain loads, and utilizing existing spare cables as additional feeds to reduce circuit loads.

The changes are being performed in Operational Condition (Mode) 4 – Cold Shutdown and the electrical circuits (distribution systems) are declared INOPERABLE per Technical Specifications (TS). Evaluations have shown that this activity will not increase the probability or consequences of an accident previously evaluated in the USAR or increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. Also, this activity will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than evaluated previously in the USAR. Lastly, for the same reasons, this activity will not reduce the margin of safety as defined in the basis for any technical specification.

LEAK-OFF LINE FLANGES AT VALVE 1G33-F101

Activity Evaluated: ECN 31184 Log Number: 98-147

Engineering Change Notice (ECN) 31184 installs a set of flanges at the leak-off line of valve 1G33-F101, located in the Reactor Water Cleanup (RT) system. Per Updated Safety Analysis Report (USAR) Section 5.4.8, the only portions of the RT system that could initiate or mitigate any accident are the portions of the system that make up the reactor coolant pressure boundary (RCPB); valve 1G33-F101 is part of the RCPB. The flanges are installed via welding, which is no different than the welding process associated with using a welded joint to restore pipe integrity. Testing will be performed during the weld installation process to assure the integrity of the welds. Analysis has been performed to seismically qualify the pipe and related piping supports for the flange installation. The new flanges are designed and installed to the original construction code and specification, and the materials used meet the original requirements for design, pressure, and temperature. The new flanges will not adversely affect RT or Leak Detection System performance. The proximity of the new flanges to the valve results in leakage at the gasket having the same zone of influence as valve packing leakage, and the same environmental conditions would be created. as a result, the flange leakage has the same affect on nearby equipment important to safety as a potential packing failure. The RT and Leak Detection System reliability and functionality are not adversely affected. Also, no fission product barriers are challenged due to this ECN. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The credible failure modes for a flanged joint is leakage around, or complete failure of, the gasket. The likelihood of leakage through the gasket is essentially the same as leakage through the packing. Gasketed joints already exist in the RCPB; which means that the flange leakage represents just another point of potential leakage from this line, not a new failure type. The leakage from the flanged joint decreases the threshold for corrective action assuming an actual pipe crack exists, and is therefore conservative. Should gasket leakage force a shutdown based upon the unidentified leakage Technical Specification, the shutdown is controlled, and is therefore not an accident or transient. The addition of the new flanged joint is no different than any other packed, sealed, or gasketed joints. As a result, gasket leakage will have the same affect upon drywell temperature and humidity as the failure of any of the before mentioned joints. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The safety margin associated with unidentified leakage is the difference between the capacity of the RCIC pump and the unidentified leakage rate limit of Technical Specification 3.4.5. Any unidentified leakage from any source counts in the quantification of the 5 gpm limit. Any leakage from an unidentified source other than the postulated pipe crack essentially lowers the threshold for corrective action for the postulated pipe crack since the 5 gpm limit represents an absolute maximum regardless of source. A leaking gasket then lowers the quantity of water that can be leaking from a postulated pipe crack; thus, lowering the threshold for action is conservative. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DEGRADED VOLTAGE CONTROL CIRCUIT UPGRADES

Activity Evaluated: Modification AP-033, Supplement 6; USAR Change 8-246 — Log Number: 98-160

Modification AP-033, Supplement 6 modifies nine control circuits to allow proper operation of the components in the circuits. They are being modified as follows: resupplying power to the circuits via a new power supply in the same division and reconfiguration of the loads, paralleling cable feeds to reduce resistance thus lowering voltage drop, rerouting/replacement of a feeder cable, changing the existing feeder cable with a larger cable size, or installing a new cable with a shorter cable route for an existing control cable. The Updated Safety Analysis Report (USAR) changes for these activities are limited to changes in the schematics listed in Table 1.7-1. These changes are only to accident mitigation or plant support systems. This modification does not change any systems that are design basis accident initiators in USAR Chapters 6 or 15. This activity does not change the function or operator control and interface of any of these circuits. This modification modifies the control circuits to improve operation of the components in the circuits during normal voltage conditions. Any changes to wiring and cable additions are designed to the same standards of separation and are bounded by existing seismic design, and safe shutdown requirements. The reliability of the modified control circuitry will improve the ability of the components to perform their intended design functions. The circuit changes modify the circuits from a parallel configuration to a master/slave configuration, however, the failure of any or all of the relays has the same result as it would in the original configuration. Periodic and post modification testing will verify the operation of the modified circuits. During the installation and testing portions of this modification, the plant will be in Cold Shutdown and in a Division 1 outage. Applicable voluntary Limiting Conditions for Operation (LCOs) will be entered to perform the design changes. In addition, Clinton Power Station procedural control will be in effect and all installation activities will be installed in accordance with Electrical Installation Specification K-2999, unless specifically justified and noted otherwise. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The final configuration associated with this modification does not change the function of the affected control circuits or components.

These changes improve the voltage to the devices, thereby improving their function to act as mitigators to a potential accident. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The final configuration does not change the function or method of operating the affected control circuits or components. Applicable LCOs, procedures, installation specifications and post-modification testing will be adhered to, to ensure the modification is installed in accordance with the design. This modification will provide a level of voltage at the devices above the minimum required as defined in the Technical Specification Bases. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

AIR COMPRESSOR LUBE OIL AND OIL COOLER COOLING WATER PIPING CHANGE

Activity Evaluated: ECNs 31029, 31139 and 31140 Log Number: 98-167

Engineering Change Notices (ECNs) 31029, 31139 and 31140 modifies the Component Cooling Water (CCW) lube oil piping and oil cooler for Service Air (SA) Compressors 0/1/2 SA01C. This activity consists of rerouting the CCW piping in a configuration such that the operation of a three-way valve allows cooling water to flow through one or both of the compressor oil coolers. Also, a thermostatic valve will be installed in the oil line downstream of the last oil cooler. Installation of this valve will significantly improve the lube oil system response time to CCW water temperature transients and will control the compressor lube oil temperatures. This modification will make it possible to use one or both of the oil coolers as required by the seasonal conditions. The most severe design basis accident that could occur is the Loss of the Service Air compressors on high oil temperature. This failure could result in the loss of all three SA compressors and consequently the SA system. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment is designed. Therefore, the fission product boundaries maintain their integrity and function as designed. Installation and operation of the new three-way valve and piping system will maintain operation of the system inside design limits. This activity could actually decrease the probability of an operator error in operating the system; use of a three-way valve in the CCW system and installation of a thermostatic valve in the lube oil system negates the requirement for operator to throttle a single cooling water valve to obtain the correct flow. This activity meets the applicable system design, material and construction standards. Fabrication, assembly, inspection and testing of the new installation meet the original system criteria as specified by the applicable piping codes and standards. This activity does not degrade system reliability, availability, independence, or redundancy. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The most severe malfunction of the SA compressors is that malfunction which causes the compressors to fail to perform their design basis function of supplying air to the plant. The consequences of this failure have been evaluated and documented in USAR Section 15.2.10. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. There are no Technical Specifications affected by this activity. Failure of the SA system to function would not affect any Technical Specification safety limit, limiting safety system setting, Limiting Condition for Operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONDENSATE POLISHING FILTER SYSTEM TOUCHSCREEN REPLACEMENT

Activity Evaluated: Modification CP-021; ECNs 31121 and 31242; USAR Change 8-250 Log Number: 98-169 R/1

Modification CP-021 replaces two touchscreen displays associated with the Condensate Polisher (CP) Filter System, located in panel 1PL03JA, with new TCI touchscreen displays. The third touchscreen display is being removed and the opening covered with a plate. The power switch and associated wiring for the third touchscreen is also being removed. Upgraded software packages are being provided as part of this change. The CP Filter System is part of the Power Conversion System by which condensate is returned to the reactor vessel and is not safety related, Seismic Category I, or Class 1E. The failure modes of the touchscreen and remote workstation are that the CP Filter Control System will not perform the required functions, the system will perform incorrect operations, or the touchscreens will provide erroneous annunciations of system status. The worst case failure would be that three demineralizers would be isolated which results in a reduction of feedwater flow. This activity does not alter the design, function, or method of performing the function of the CP Filter System. The hardware and software changes to the control system for the CP Filter System are designed, manufactured, verified, validated, installed, and tested in accordance with industry codes and standards, applicable Updated Safety Analysis Report (USAR) requirements, and plant specifications in order to ensure system reliability. The new touchscreens and their associated computers, and the new remote workstations, are system enhancements. The reduction of the number of touchscreens does not adversely affect the operation, control, or annunciation capability of the CP Filter System since the third touchscreen mainly provides local annunciation, and an alternate control station. These functions are provided by the remaining two touchscreens without a reduction in design or effectiveness. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure modes of the Condensate Polisher Filter Control System are that all valves fail into the filter bypass mode upon loss of instrument air, loss of control power, or any other Condensate Filter system failures. The changes are not adversely affecting the existing failure modes, and no new failure modes or types of failure modes are introduced. No new system interactions are introduced by this activity. As discussed above, the hardware and software changes are not adversely affecting the design, function, or method of performing the function of the Condensate Polishing Filter System. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The Technical Specifications and its associated bases do not address the Condensate Polishing Filter System. No Technical Specification parameters, limiting conditions of operation, surveillances or design features are impacted by the changes to the Condensate Polishing Filter System. Therefore, this activity does not reduce a margin of safety as defined in the bases for any Technical Specification.

DIESEL FIRE PUMP A RELIEF VALVE REPLACEMENT

Activity Evaluated: ECN 31218; USAR Change 8-358 Log Number: 98-188

Engineering Change Notice (ECN) 31218 replaces the 'A' diesel fire pump 6-inch relief valve with a 10-inch valve of the same type, replaces the sight glass cone, and reroutes the 1-1/2 inch diesel Heat Exchanger cooling water return line which necessitates the addition of a new support. The larger size valve of the same type does not affect the operation of the diesel fire pump in any way different from its operation with the existing relief valve. The pump maintains its capability to deliver water to the fire areas at the same flow rate and pressure as required by the existing fire analysis. Replacing the existing relief valve with a larger valve of the same make and model enhances the reliability of the system by eliminating the cavitation/erosion associated with the existing relief valve. This activity does not change the system redundancy or independence and has been analyzed to the same seismic criteria as the existing piping system. This design change does not affect the design basis function or any characteristics of the Fire Protection System. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The diesel fire pump relief valve is being replace with another larger valve of the same design, type, and make; the probability of its failure is the same as the existing valve. The increase in weight resulting from the new larger valve has been analyzed to the same code requirements used for the existing system and found to be acceptable. In addition, this activity has no adverse impact on the fire pump/system reliability, function, or method of performing its function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Clinton Power Station Fire Protection Program is not included as part of the Technical Specifications. The relief valve for the A diesel fire pump is a component of the Fire Protection System and is provided to protect the A diesel fire pump, the piping system from overpressurization when the pump operates at no flow or partial flow conditions. The only potential effect on any other equipment or systems by this modification would be an improvement of the relief valve performance. All acceptance values or design limitations involving the use of the relief valve are not altered by this change. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DIESEL FIRE PUMP B RELIEF VALVE REPLACEMENT

Activity Evaluated: ECN 31219; USAR Change 8-359 Log Number: 98-189

Engineering Change Notice (ECN) 31219 replaces the 'B' diesel fire pump 6-inch relief valve with a 10-inch valve of the same type, replaces the sight glass cone, and reroutes the 1-1/2 inch diesel Heat Exchanger cooling water return line which necessitates the addition of a new support. The larger size valve of the same type does not affect the operation of the diesel fire pump in any way different from its operation with the existing relief valve. The pump maintains its capability to deliver water to the fire areas at the same flow rate and pressure as required by the existing fire analysis. Replacing the existing relief valve with a larger valve of the same make and model enhances the reliability of the system by eliminating the cavitation/erosion associated with the existing relief valve. This activity does not change the system redundancy or independence and has been analyzed to the same seismic criteria as the existing piping system. This design change does not affect the design basis function or any characteristics of the Fire Protection System. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The diesel fire pump relief valve is being replace with another larger valve of the same design, type, and make; the probability of its failure is the same as the existing valve. The increase in weight resulting from the new larger valve has been analyzed to the same code requirements used for the existing system and found to be acceptable. In addition, this activity has no adverse impact on the fire pump/system reliability, function, or method of performing its function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Clinton Power Station Fire Protection Program is not included as part of the Technical Specifications. The relief valve for the B diesel fire pump is a component of the Fire Protection System and is provided to protect the B diesel fire pump, the piping system from overpressurization when the pump operates at no flow or partial flow conditions. The only potential effect on any other equipment or systems by this modification would be an improvement of the relief valve performance. All acceptance values or design limitations involving the use of the relief valve are not altered by this change. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION OF THE OFFSITE DOSE CALCULATION MANUAL

Activity Evaluated: ODCM Revision 18 Log Number: 99-022 R/2

Changes made to the Offsite Dose Calculation Manual (ODCM) in Revision 18 include following: change the requirements of the ODCM to reflect implementation of modification PR040, include modifications arising from an independent review of the ODCM by a consultant, add information governing the Accident Range Monitors, and make several editorial changes. The editorial changes include clarifying wording, correcting typographical errors, and modifying several unclear formulas; these changes are editorial in nature and do not alter or add any requirements in the ODCM. The ODCM provides the methodologies and parameters used by Clinton Power Station to assure compliance with the radioactive effluent dose limitations. The applicable accidents evaluated in the Updated Safety Analysis Report (USAR) deal with dose to unrestricted areas from radioactive effluents. The control methodology for the Area Radiation/Process Radiation (AR/PR) system being installed under Modification PR040 represents an equivalent method of performing the same function as currently specified. In addition, these changes do not alter or delete any commitment in the ODCM. The ODCM methodology is not an assumed initiator of any evaluated accident or malfunction of equipment important to safety as described in the USAR. Nor, does the ODCM provide a mitigatory role to limit radiological releases during any postulated accident. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not alter any monitoring requirements already in the ODCM or affect operation of any equipment important to safety. There are no credible accident or failure modes for equipment important to safety because the ODCM only provides requirements for monitoring effluents and methodology for calculating off site doses. The impact of Modification PR040 was discussed in Safety Evaluations 98-085 and 98-116 and no concerns were identified. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The changes incorporated in this revision of the ODCM are in full compliance with Technical Specifications for calculating dose to unrestricted areas as a result of liquid and gaseous effluents. The new AR/PR methodology introduced in Revision 18 is a different but equivalent way of fulfilling regulatory requirements for control of the AR/PR system. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

GAP SEALS IN LOW AND MODERATE FIRE LOAD ZONES

Activity Evaluated: USAR Change 8-283 Log Number: 99-058 R/1

This activity addresses the acceptability of structural gap seals between the top of fire-rated walls and the ceiling that were determined as being in non-compliance with American Society for Testing Materials (ASTM) E-119. An engineering evaluation determined that the gap seals are in compliance with Nuclear Regulatory Commission Generic Letter 86-10 and NUREG 1552, and are not required to comply with ASTM E-119. ASTM E-119 addresses the requirements for seals associated with cables. NUREG 1552 addresses all type of gap and penetration seals, including those that are required to meet ASTM E-119. The structural gap seals do not contain cabling, but do conform to the non-cable sealing criteria of NUREG 1552. This activity permits the use of test standards other than ASTM E-119 and allows the use of the structural gap seals in low/moderate fire areas. The engineering evaluation takes into consideration the type of combustible materials, the detection system, the suppression system, and the safety equipment located in the zone and determined that the installed gap seals are acceptable. Test results establish that ceramic fiber is the only material that provides any significant fire resistive properties in a seal. Walkdowns have shown that ceramic fiber is installed in all the required gap seals. There is no change to combustible loading, detection, or the suppression system. The only way improperly installed gap seals can increase the probability of malfunction of the equipment is by letting the fire spread to the other areas and damage the equipment. The structural gaps in question are located between the top of the non-load-bearing walls and under the concrete slab/metal deck at least 18 to 20 feet above the floors. The detection systems, combined with the fire brigade's use of available hose stations, will prevent the fire from becoming fully developed. Therefore, the integrity of the gap seals will not be challenged. Based on the fire endurance test, gap seals should provide an equivalent fire resisting rating to the concrete block wall. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). In the event of a fire, it is going to be contained in the area/zone where it started, and will not spread into other areas to degrade/damage the equipment important to safety. There is no change to combustible loading, detection systems, other fire hazards, or suppression systems. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. There are no explicit margins of safety delineated regarding fire rated gap seals. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

HFA RELAY 0UAY-CY505A REPLACEMENT

Activity Evaluated: Temporary Modification 99-022 Log Number: 99-060

Temporary Modification 99-022 replaces HFA Relay (0UAY-CY505A) on-line to satisfy the requirements of PEMCYA801. This relay de-energizes when the Condensate Storage Tank (CST) is at low-low level and provides a trip signal to Cycled Condensate (CY) Pumps A, B, and C. The relay replacement will be done while maintaining at least one Cycled Condensate pump running. Updated Safety Analysis Report (USAR) Chapters 6 and 15 do not identify any accidents that are initiated by a failure of any CY system, structure, or component (SSC). If the CY system pumps were to fail, numerous building Equipment and Floor drain pumps are exposed to pump seal damage. However, none of the Equipment and Floor drain pumps affected are required for safety or prevent safe shutdown of the reactor. To assure that Condensate Storage Tank level does not fall to a level where pump damage and system failure will occur, Operations will be coached to have a heightened awareness of CST level during the relay replacement activity, and a tag will be provided adjacent to the annunciator to reinforce the need for response. Existing procedures provide adequate pump protection during the time frame of relay replacement. The portions of the CY system associated with the containment isolation function are important to safety. The scope of this modification does not challenge the containment isolation boundary piping and valves portion of the CY system in any way. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The credible failure modes added by this activity are associated with the installation of the jumpers and lifted leads. The result of these failures is the same as previously evaluated in the USAR. The installation of the jumpers are done in accordance with Clinton Power Station maintenance procedures and the maintenance work order process provides adequate assurance that other systems with terminations in the vicinity of the jumpers are not adversely affected. Since failure of the CY pump does not compromise any accident initiator, the failure cannot create a credible malfunction that could directly or indirectly affect any plant system from performing the required safety functions. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The CY system is not directly or indirectly mentioned in the Technical Specifications. Thus, there are no acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with components affected by the activity, and no design limitations are adversely impacted. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE EXISTING WATER LEG PUMP FLOW RESTRICTION ORIFICES WITH MULTISTAGE ORIFICES

Activity Evaluate: ECNs 31252 and 31253; USAR Change 8-291 — Log Number: 99-063

Engineering Change Notices (ECNs) 31252 and 31253 replaces the Residual Heat Removal (RHR) and Low Pressure Core Spray (LPCS) water leg pump flow restriction orifices, 1E12D002 and 1E12D003 respectively, with multistage orifices. The orifices are designed to provide minimum flow for RHR and LPCS water leg pumps while at the same time keeping the water leg pump discharge pressure at the high point of the pump curve when system flow requirements are at a minimum. The original orifices were sized too conservatively, resulting in excessive minimum flow. These changes do not result in changes to the RHR or LPCS systems that could potentially cause Updated Safety Analysis Report (USAR) evaluated accidents. This modification maintains the RHR and LPCS design basis and improves the function of the water leg pump system. This modification meets the original system design, material and constructions standards. This activity does not affect the overall system performance of any other system, as the only function of the water leg pumps is to continuously maintain the RHR/LPCS discharge piping pressurized and filled with water in order to reduce injection time and minimize water hammer effects. The LPCS and RHR system are designed to mitigate the consequences of Loss of Coolant Accidents. This activity improves the capability of the water leg pumps to keep the LPCS and RHR piping full of water to prevent possible damage to core spray and injection piping when the systems initiate. The water leg pumps will operate at higher head due to the increased flow resistance provided by the new orifices. As the new orifices still pass the required minimum flow, the higher head is within the design capability and is desirable because it increase the margin above the required system head. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not degrade system performance, and does not reduce system redundancy or independence. The system function is unchanged by replacing the orifices. The piping and orifices are installed per ASME code requirements and have been analyzed to ensure pipe stresses and support loads remain acceptable. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Replacing the water leg pump orifices improves the ability of the water leg pumps to perform their design basis function. The ability of the RHR and LPCS systems to meet their Technical Specification and design requirements will not be impacted by this plant change. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF VACUUM BREAKER SPRAY DEFLECTOR

Activity Evaluated: ECN 31570; USAR Change 8-315 Log Number: 99-085

Engineering Change Notice (ECN) 31570 installs a vacuum breaker spray deflector at the open end of the butterfly valve of the vacuum breaker discharge line. When the Shutdown Service Water (SX) system is running, any water caught in the extension pipe will be pulled back into the SX system with subsequent operations of the vacuum breaker. The extension pipe assembly serves as a small reservoir to contain the slug of water which is periodically emitted when the vacuum breaker opens and to prevent the water from spilling on the floor. The vacuum breakers are not relied on to mitigate damaging transients in the SX system. They do help to reduce operational transients and minimize erosion/corrosion during normal operation. The addition of the SX spray deflector will not adversely affect the SX piping such that additional failures could occur. The SX system function is not affected, it will continue to provide a reliable source of cooling water. Failure of the vacuum breaker spray deflector would only cause the slug of water from the vacuum breaker discharge line to spill on the floor. The non-safety related spray deflector and the non-safety related butterfly valve are passive components; failure of one or both would not render SX inoperable, since failure would only increase the cavitation in the outlet line of the SX piping. Since the function of the vacuum breakers is not required for the SX system to perform its safety function, failure of this spray deflector is not an issue. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The SX system design basis function is not degraded as a result of this activity. Failure of the vacuum breaker spray deflector would only cause the slug of water from the vacuum breaker discharge line to spill on the floor. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications are not written at such a level of detail to include items such as vacuum breaker performance requirements. This change does not alter design pressure, flow, performance or functionality of the SX system. Nor does this change alter any Technical Specification operating limits or set points. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF VACUUM BREAKER SPRAY DEFLECTOR

Activity Evaluated: ECN 31571; USAR Change 8-314 Log Number: 99-086

Engineering Change Notice (ECN) 31571 installs a vacuum breaker spray deflector at the open end of the butterfly valve of the vacuum breaker discharge line. When the Shutdown Service Water (SX) system is running, any water caught in the extension pipe will be pulled back into the SX system with subsequent operations of the vacuum breaker. The extension pipe assembly serves as a small reservoir to contain the slug of water which is periodically emitted when the vacuum breaker opens and to prevent the water from spilling on the floor. The vacuum breakers are not relied on to mitigate damaging transients in the SX system. They do help to reduce operational transients and minimize erosion/corrosion during normal operation. The addition of the SX spray deflector will not adversely affect the SX piping such that additional failures could occur. The SX system function is not affected, it will continue to provide a reliable source of cooling water. Failure of the vacuum breaker spray deflector would only cause the slug of water from the vacuum breaker discharge line to spill on the floor. The non-safety related spray deflector and the non-safety related butterfly valve are passive components; failure of one or both would not render SX inoperable, since failure would only increase the cavitation in the outlet line of the SX piping. Since the function of the vacuum breakers is not required for the SX system to perform its safety function, failure of this spray deflector is not an issue. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The SX system design basis function is not degraded as a result of this activity. Failure of the vacuum breaker spray deflector would only cause the slug of water from the vacuum breaker discharge line to spill on the floor. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications are not written at such a level of detail to include items such as vacuum breaker performance requirements. This change does not alter design pressure, flow, performance or functionality of the SX system. Nor does this change alter any Technical Specification operating limits or set points. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

NON-SAFETY HVAC FREEZE PROTECTION

Activity Evaluated: USAR Change 8-308 Log Number: 99-096

Updated Safety Analysis Report (USAR) Change 8-308 adds the following procedural description, “During cold weather months, the cooling coils are manually isolated and drained to provide additional freeze protection”. This is a maintenance function, administratively controlled through maintenance procedures that alters the original, and historically ineffective, design method of providing freeze protection. This activity provides, by isolation and draining of cooling coils in cold weather, assured freeze protection to non-safety related cooling coils to ventilation systems in the Auxiliary Building, Fuel Building, Containment Building, and Radwaste Building. USAR Chapters 6 and 15 make no reference to the freeze protection function or the cooling coil function of these systems as part of the accident analysis. In regard to credible failure modes, none of the systems affected are either an initiator or mitigator of an accident. Application of freeze protection does not alter airflow, dampers, ducting, etc. and therefore does not alter or impact overall design considerations for that ventilation system. There are no safety systems important to safety affected by this activity. The method of freeze protection employed provides greater assurance that the system for cooling will be available in summer months when needed. This activity does not impair the availability or reliability of the ventilation systems, nor does it increase challenges to safety systems. Radiological conditions are not altered by this activity because the flow of air and the maintenance of building differential pressures are not impacted by the isolation and draining of chill water from the cooling coils of the affected units. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. A credible failure mode is to neglect to restore the system or neglect to install freeze protection when needed. The implementation of the scheduled maintenance procedures, for both the imposition of and restoration from freeze protection, are verified complete by Clinton Power Station procedure 1860.01, "Cold Weather Operation." None of the components involved with this change are either an initiator or mitigator of an accident and are not important to the safe shutdown or operation of the plant. The activity enhances the protection of the system. If freezing damage were to occur to the cooling units, repairs would be necessary before the cooling coil could be used in warm or hot summer month operation. This activity reduces the likelihood of freezing damage. The types of failures described have no impact or relevance to design basis accident or other transients that the plant is analyzed to withstand. In addition, the affected units do not include operation of the ventilation systems for the control room, the standby diesel-generator rooms, essential switchgear rooms, service water pump rooms, or the Emergency Core Cooling System equipment cubicles which are designed to operate and to maintain ambient conditions for equipment protection under postulated accident conditions as well as normal operation. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not address freeze protection for ventilation systems. The activity ensure protection from freezing for the affected system components and may be considered an enhancement to structure, system, and component reliability. This activity does not require alteration of the current Technical Specification as it does not involve any Technical Specification Limiting Condition for Operation value or requirement, Mode of applicability, Action or Surveillance requirement. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACEMENT OF DIVISION 1 AND DIVISION 2 AIR BOTTLE FILL VALVES FROM GLOBE TO BALL VALVES

Activity Evaluated: ECN 31647; USAR Change 8-351 Log Number: 99-097

Engineering Change Notice (ECN) 31647 replaces existing Division 1 and Division 2 Air Bottle Fill Valves, 1RA001A/1RA001B and 1RA002A/1RA002B, with new ball valves. The existing valves are one-inch globe valves and the replacement valves are one-inch full port ball valves. The new valves adequately meet the service pressure and temperature limitations. The replacement valves are heavier, which is acceptable due to decrease in weight on the seismically analyzed line. The operation or failure of the Emergency (Breathing) Air system is not considered with any of the accident scenarios described in Updated Safety Analysis Report (USAR) Chapters 6 and 15, or concurrently with other events such as a design basis fire. The ability of the new valves to seal enhances reliability by providing a leaktight boundary for the fill station. Valves 1RA002A and 1RA002B provide isolation between safety and nonsafety piping. This activity enhances the reliability of the system because it provides an isolation barrier to protect system integrity by limiting or eliminating system leakage. The safety related valves limiting condition is failing to provide a barrier of isolation for the breathing air system and the non-safety related fill system. The new positive sealing ball valves provide additional assurance against system leakage, thus maintaining system integrity. In addition, there is no change to the system redundancy or independence as a result of this design change. The valves are not interconnected to systems that transport or maintain fission products. The failure of the valves will not impede access to vital areas, will not increase the consequences to on-site or off-site dose, and will not impair systems that affect radiological consequences. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not affect the integrity, performance or reliability of any structure, system, or component (SSC) that is safety-related. The probability of the new valves failing to perform their function has decreased, because the ball valves are not susceptible to over tightening resulting in seat damage. Failure of the new valves is not anticipated because the valves are manufactured in accordance with the same standards as the valves being replaced and this activity does not affect the system capability to perform its design bases function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specification or Bases do not address the Emergency (Breathing) Air System. The emergency breathing air bottles are required for Control Room Habitability. The breathing air bottles are a passive subsystem to provide emergency air to the operators. The margins of safety of the Emergency (Breathing) Air System are unaffected by this activity on the basis that this will not affect the performance or the operability of the system. The activity does not impose any design limitations to the Emergency (Breathing) Air System or any other safety related SSC. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

IRM ROD BLOCK JUMPER INSTALLATION

Activity Evaluated: 8 Temp Mods (00-079 thru 00-086) Log Number: 99-101

This activity installs a jumper to defeat one Rod Block signal at a time from an Intermediate Range Monitor (IRM) channel that is tripped (tripped means to comply with the requirements of Technical Specification (TS) 3.1.1). The jumper is to be installed after the channel is placed in trip. The activity may be used to calibrate, perform maintenance, or take an IRM out-of-service to allow plant start-up. It is necessary to place the IRM in trip to satisfy LCO 3.3.1.1 requirements. Placing an IRM in trip provides a Rod Block signal to the Reactor Core & Information System (RC&IS). This Rod Block signal prevents further rod withdrawals and may place the plant and reactor start-up in an undesirable situation during a crucial phase of power ascension. Installing this temporary modification jumper bypasses the Rod Block input to RC&IS from the IRM channel in a trip condition. Thus, rod movement is allowed while not affecting the scram signal to the Reactor Protection System (RPS). The Rod Block functions and trips from other IRMs are not impacted by one jumper.

This activity will allow rod motion during plant startup with one IRM channel tripped to meet the Limiting Condition of Operation (LCO) requirements of TS 3.3.1.1. The facility review group determined that no unreviewed safety question is involved with this activity.

FREEZE SEAL IN LINE 1FP41A-4

Activity Evaluated: MWO D73156 Log Number: 99-102

Maintenance Work Order D73156 installs a freeze seal in the 4-inch fire protection line, 1FP41A-4, to allow the repair of the Standpipe Isolation Valve, 1FP057B. Valve 1FP057B could not be cycled with a reasonable amount of force. To isolate this valve, two valves would have to be closed, resulting in fifteen fire hose stations being removed from service in the Containment and Fuel Handling Buildings. By using a freeze seal, three hose stations are isolated in the Fuel Handling Building. Thus, performing this work with a freeze seal minimizes the impact upon the plant fire protection program and minimizes the plant's exposure to risk. As a result of the fire protection hose stations being taken out of service, this activity has potential impact upon Updated Safety Analysis Report (USAR) Chapter 3 Fire analysis and the Fire Protection Evaluation Report fire analysis. The installation of a freeze seal involves neither combustibles nor ignition sources. The failure of the freeze seal can lead to flooding, which is anticipated in the governing freeze sealing procedure. To make the risk of freeze seal failure acceptable, the installation of a freeze seal is part of a troubleshooting and repair plan, and is controlled via the maintenance process. The freeze seal procedure requires that a checklist be prepared. The checklist requires a plan that includes the identification of compensatory measures in the event of a freeze seal failure. The compensatory action identified for this activity that shall be taken if the freeze seal fails is the closure of certain isolation valves. This advance planning will allow Operations to respond quickly in the unlikely event that the freeze seal fails, and makes the risk of freeze seal failure acceptable. System redundancy is not affected beyond the loss of capability of the three hose stations. The freeze seal does not create a connection that affects the independence of any system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The credible failure mechanisms associated with this activity are freeze seal failure while the downstream piping is open and pipe failure due to the installation of the freeze seal. Either of these event could result in a flooding event or impede the functioning of the fire protection system. Also, installing a freeze seal will allow maintenance activities to be performed on the system without compromising the availability of Fire Protection System due to the required programmatic compensatory actions. If required, the USAR permits taking portions of the fire protection system out of service as long as compensatory measures are taken to extinguish fires in the immediate area where the system is out of service. The portion of the fire protection system affected by the activity can be isolated by valves if the freeze seal would fail. When these failure mechanisms are coupled with the procedural controls imposed, the failure mechanisms are no different than those previously analyzed, and the freeze seal failure does not represent a new failure type. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not discuss the fire protection program. Thus, the fire protection system does not have any safety limits or limiting conditions for operation stated in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ONE TIME USE OF ALTERNATE SAMPLING METHOD FOR CHARCOAL BED 0VC09SB FOR TECHNICAL SPECIFICATION TESTING

Activity Evaluated: PDRs 99-369 & 99-370 for CPS Procedures 9866.03 and 9866.03C001 — Log Number: 99-105

Clinton Power Station Procedure 9866.03, "VG/VC Charcoal Sample Analysis," and checklist 9866.03C001, "Charcoal Adsorber Sample Checklist," were revised in order to allow an alternate method of obtaining charcoal samples from the charcoal beds in the Control Room Heating, Ventilation, and Air-Conditioning (VC) and Standby Gas Treatment (VG) Systems. Neither the VC system or charcoal bed, 0VC09SB, are the initiators of any accident evaluated in the Updated Safety Analysis Report (USAR). Obtaining the charcoal sample directly from the bed does not physically alter the charcoal bed in any significant way. Charcoal bed 0VC09SB has a design basis function of removing the methyl iodine released from a design basis accident, to help ensure the radiological dose limits of General Design Criteria 19 are met for personnel in the Main Control Room. 0VC09SB does not impact the radiological dose received by the general public, since it only filters air entering the control room. The charcoal bed still meets its design basis. The procedure change only changed the method of obtaining a representative charcoal sample. This did not change any of the laboratory test parameters or the acceptance criteria given in the Technical Specifications for the testing of the sample. The alternate method is within the allowance of Regulatory Guide 1.52, Revision 2, position C.6.b, to which Clinton Power Station is Committed in Technical Specification 5.5.7.c. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Using this alternative sampling method does not change the design or function of the charcoal bed. The air flow through the bed and pressure drop across the bed will not be changed. There will be no reduction or alteration of the safety function of the charcoal beds. The removal of the sample per the alternate method will not cause any plant structure, system or component to become an initiator of any accident. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This alternative sampling method is equivalent to the regular sampling method. This activity does not change the testing parameters or the acceptance criteria of Technical Specifications 3.7.3.3 or 5.5.7.c. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

BYPASS THE 1FW002B 90-100% OPEN PERMISSIVE

Activity Evaluated: Temporary Modification 99-045 — Log Number: 99-107

Temporary Modification 99-045 installs a jumper around the 1FW002B contact in the "B" Turbine Driven Reactor Feed Pump (TDRFP) feedwater level controller logic. This contact is one of the required permissives to allow the "B" TDRFP to operate on the Start Up level controller in the automatic mode, allow closing the 1FW010B minimum flow valve to the condenser, and allow manual modulating control of the 1FW010B valve at panel 1PA05J. The feedwater control system is a power generation system for purposes of maintaining proper vessel water level. The transients or events pertinent to this activity are loss of feedwater flow and feedwater controller failure - maximum demand. The credible failure mode of this activity is the failure of the jumper to maintain circuit continuity. The purpose of the interlock, which is being defeated, is to ensure a flow path for the "B" TDRFP when it is running and supplying feedwater flow. This activity causes the "B" TDRFP minimum flow valve to shift to remote control. Strict procedural controls ensure this valve is positioned to provide an adequate flow path when the "B" TDRFP is running. In addition, this jumper is installed in a part of the feedwater control circuit that would not cause a maximum feedwater demand signal, therefore, this activity has the inability to result in system configurations that would alter the probability of a maximum feedwater demand signal failure. The result of an open circuit, when using the "windmilling" feedwater control mode or running the TDRFP, will open the "B" TDRFP minimum flow valve. This valve diverts feedwater flow from the reactor vessel to the main condenser. The result of an open circuit when running the TDRFP in the automatic mode with the Start UP level controller, results in shifting the control of the "B" TDRFP to manual control. The design of the feedwater control system ensures that the manual controller constantly tracks the output of the automatic control function to allow a "bumpless" transfer to manual control, thus, minimizing disturbances to the feedwater flow. This activity is implemented at lower, more conservative reactor power levels and at lower feed demands than assumed in the accident analyses of a loss of feedwater flow or feedwater controller failure. These events do not result in any pressure or temperature transients that would challenge the integrity of the fuel barrier, Containment or the pressure vessel that could increase off site radiation dose. Accident consequences that may result from this activity would be less severe and bounded by the analysis described in Updated Safety Analysis Report (USAR) Sections 15.1.2 and 15.2.7. The materials and installation requirements for this activity are consistent with the qualification of the existing panel and circuits. This activity installs a jumper to maintain continuity, which has less probability of failing then an active switch due to the inherent potential for high contact resistance. This activity meets the original design specifications and equipment qualifications. The jumper is installed in a non-safety, non-divisionalized panel. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not change the fundamental function or objective of the feedwater control system and does not result in unusual or untested system configurations or operating conditions. Also, this activity does not adversely impact an Operator during performance of routine duties or responses to plant transients or accidents. Since this activity maintains the design of the feedwater control system and creates no adverse conditions, this activity has no detrimental effects on the ability of plant systems to perform their intended safety function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not alter the feedwater system configuration or operation in a manner that degrades system capability. No design limitations, acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with the margins of safety impacted by the feedwater system are adversely affected by this activity.

INSTALLING JUMPER TO DEFEAT ROD BLOCK FROM APRM B

Activity Evaluated: Temporary Modification 99-047 Log Number: 99-110

Temporary Modification 99-047 installs a jumper to defeat the rod block signal from the "B" Average Power Range Monitor (APRM) channel that is tripped. The authorization to jumper the rod block signals is limited to one APRM channel at a time. This activity may be used to calibrate, perform maintenance upon, or take an APRM out of service to allow power ascension and/or continued plant operation. The rod block signal prevents further rod withdrawals, which imposes undesirable and unnecessary operational restraints. Installing a temporary modification to bypass only the rod block input to Rod Control and Information System will allow rod movement and does not affect the scram signal to the Reactor Protection System as it is already in trip. The rod block functions and trips from the other APRMs are not impacted. This activity is in compliance with Technical Specification, Operational Requirements Manual (ORM), and the current licensing basis. This plant change meets the design, material, and construction requirements of the APRM system. Safety related requirements are imposed on the system based on its ability to mitigate an accident. As such, the seismic and safety related qualification was considered in developing the design and was evaluated to be acceptable. Since this activity maintains the reliability and the original design function of the APRM system, there is no impact on the radiological considerations for any design basis accidents and transients. The offsite and onsite acceptance criteria are not changed and there is no affect on any of the fission product barriers as a result of this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). No safety function has been inhibited or degraded, no change has occurred to any sequence of events as anticipated in the USAR, and no new credible failure modes are introduced. In addition, this activity does not alter the way the APRMs or any other system currently fail and cannot increase the frequency or severity of the malfunction. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.3.1.1 determines the operability requirements of the APRMs for different operating conditions and ORM Section 2.2.1 addresses rod block requirements. No design limitations, acceptance values, safety limits limiting safety system settings, or limiting conditions for operation are adversely impacted by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION TO USAR TABLE 3.9-2(a) TO INCORPORATE UPDATED FATIGUE USAGE FACTORS

Activity Evaluated: USAR Change 8-321 Log Number: 99-112

Updated Safety Analysis Report (USAR) Table 3.9-2(a) provides stress limit criteria and calculated values for various parts of the Reactor Vessel and Internals. USAR Change Package 8-321 revises the table to change the vessel support skirt cumulative usage factor from 0.371 to0.8. The vessel support skirt is a structural support for the reactor vessel and does not form part of the reactor coolant pressure boundary. Failure of the vessel support skirt is not an analyzed accident, but could be a cause of pipe breaks such as the Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The large break LOCA and MSLB pipe breaks are considered limiting faults in the USAR and are evaluated without their causes being identified. The increase in the cumulative usage factor utilizes some of the available margin between the previous expected and the allowable usage factor of 1, permitted by ASME Code NB-3222.4. Since the cumulative usage factor is less than that permitted by the Code, there is no increase in the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Also, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type. Fatigue monitoring is required by Technical Specification 5.5.5. The revised calculated cumulative usage value reflects the results of calculations that are based on existing licensing requirements and operational practices that are already a part of the licensing basis. Vessel integrity is discussed in Technical Specification 2.1.2, primarily from an overpressurization standpoint. The Bases for this specification discuss the requirements of the ASME Code as related to overpressure protection. The margin relied upon in this bases is the margin provided by the ASME Code. This change is below the ASME Code allowable, and therefore, does not affect the margin described in this Technical Specification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

COVERSION OF PLANT CHILLED WATER (WO) UNITS TO R 134a REFRIGERANT

Activity Evaluated: Modification WO-016 and USAR Change 8-323 Log Number: 99-113

The Clean Air Act of 1990-Title 6, mandated a phase out of production of fluoro-chloro hydrocarbons (CFCs), in order to protect the earth's ozone layer. To ensure continued availability of the plant chilled water (WO) system it is necessary to convert WO chillers B and D to the new non-CFC refrigerant, R 134a. Without any equipment modifications, the full load capacity of a WO chiller will be reduced from 1100 to 977 tons, when operated with R 134a. As a result of Modification WO-016, Updated Safety Analysis Report (USAR) Change Package 8-323 provides the necessary revision on the reduction in cooling capacity for Plant Chilled Water refrigeration units 0WO02CB and CD. The WO system is only required to function in normal operating conditions. The WO system is not required to assure either the integrity of the reactor coolant pressure boundary or the capacity to shut down the reactor and maintain it in a safe shutdown condition. The WO system supplies chilled water to area coolers and fan-coil units in the drywell, and the containment, turbine, radwaste, fuel, and auxiliary buildings' ventilation systems. The system is non-safety related, except for components located between the containment isolation valves and drywell isolation valves. There is no failure analysis for the WO system evaluated in the USAR. This modification does not affect the seismically supported piping in the areas of seismic Category I buildings. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Design calculations and post modification testing prove that the original design of three WO units is still capable of meeting the summer cooling demand. The WO system will continue to provide an adequate quantity of chilled water to meet the cooling load requirements and maintain sufficient redundancy to ensure the power generation objective. With the exception of the change to a different refrigerant and a compatible lubricant, the operation of the B and D chiller units remains the same as before the modification. In addition, this modification did not affect the chillers built-in protection against freezing, high refrigerant pressure, low refrigerant pressure, high discharge temperature, motor overload, lubrication oil failure, and high motor temperature. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not govern the WO system, except for some WO valves that provide containment and drywell isolation and WO piping seismic supports. Modification WO-016 does not affect these components or any safety limits associated with them. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION OF DRAWINGS M01-1105 AND M01-1106 TO REFLECT AS BUILT STAIRWAY IN DRYWELL BASEMENT

Activity Evaluated: ECN 31686 and USAR Change 8-336 Log Number: 99-115

Engineering Change Notice (ECN) 31686 revises drawings M01-1105 and M01-1106 to show the stairway in the Drywell, going from elevation 737' to elevation 723'. This safety evaluation addresses impacts to the Fire Protection Program only, which was missed during the safety screening for ECN 28715. The stairway is part of the Fire Protection plan, since it would be used by fire-fighters in the event of a fire in this area, and is considered to be an enhancement to the Fire Protection program. This is an open stairway and does not change the fire area classifications. This activity does not affect the Design Basis Fire in Drywell 123' or 737'. The function of the stairway is to facilitate entrance/egress to the Drywell basement during outages, and performs no radiological function. The stairway and its supporting elements are non-Category I structures, but they have been designed as Category I structures for all the applicable loads to ensure it meets code allowables and that it will not fall on, damage, or impair any safety-related equipment. The stairway itself, has no safety function. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). Nor does this activity create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Adding a stairway in the Drywell does not affect any Technical Specification or its associated Bases. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE USAR FIGURE 9.2-11 SHEET 5

Activity Evaluated: USAR Change 8-324 Log Number: 99-116

Updated Safety Analysis Report (USAR) Change Package 8-324 removes 0CY075 from USAR Figure 9.2-11 Sheet 5 to reflect the actual plant configuration. There are no design basis accidents that are associated with the Cycled Condensate (CY) system. Failure of the system does not compromise any nuclear safety-related system or component and does not prevent safe shutdown of the reactor. The CY system performs no safety-related function, except for the piping and valves, which form the containment isolation boundary. Valve 0CY075 was never intended to perform a safety function and does not have any failure modes associated with it; nor could it have any impact on any other safety system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR; nor does it create an accident or malfunction of equipment important to safety of a different type than previously evaluated. The CY system does not have any Technical Specifications associated with it. No design limitations, acceptance values, safety limits limiting safety system settings, or limiting conditions for operation are adversely impacted by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF SECURITY BOOTH, OTHER PROTECTION IMPROVEMENTS INSIDE FENCE

Activity Evaluated: ECN 30993; USAR Change 8-327 Log Number: 99-118

Engineering Change Notice 30993 installs the following items outside the power block: a concrete pad north of the sealwell and inside the fence, in order to install a bullet-resistant booth on it, additional fences and cages inside the protected area, and two removable barriers in front of exterior doors 1-146 and 1-156, outside of the Turbine Building. These structures have no safety function and have been designed as non-category I structures, for all applicable loads. In addition, there is no safety related equipment nearby. They could conceivably become additional missiles in the event of a tornado. However, as a whole the security booth is considered too heavy to be a credible missile and if broken into smaller pieces it would be enveloped by the current missile hazard analysis. Doors 1-146 and 1-156 are part of the Fire Protection barrier, but they would only be blocked shut by the removable barriers, not blocked open, thus the fire protection feature is not affected. In addition, there are alternate means of entrance/egress into these areas in the event of a fire. The function of the removable barriers is to delay the entrance of terrorists into the power block; these doors perform no radiological function. The maximum distance from any point in the Turbine Building to an exit will exceed 150 feet, but not 300 feet. This increase has been evaluated and accepted by the Fire Protection community, as not causing a detriment to personnel safety in the event of a fire. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). This activity shows several new miscellaneous structures outside the power block; these structures are all passive components that serve no safety function and were designed for all applicable loads in these areas. No new failure modes have been introduced. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not affect any Technical Specification design limitations, acceptance values, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REMOVAL OF MAIN CONTROL ROOM HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEM SMOKE MODE FLOW RATE TESTING REQUIREMENT

Activity Evaluated: USAR Change 8-332 Log Number: 99-119

Updated Safety Analysis Report (USAR) Change 8-332 removes the requirement for verifying the Main Control Room Heating, Ventilation, and Air-Conditioning (VC) system smoke mode flow of 64,000 scfm +/- 10%. Procedures have been revised to reflect that there is no need for flow verification requirements for the VC system in smoke mode testing. Neither Chapter 6 or 15 take credit for the control room ventilation smoke mode for prevention of any type of accident. The components which perform the function of the control room ventilation smoke mode are safety related components. However, the smoke mode is not required for operability of the control room ventilation system. A failure of the control room ventilation smoke mode will not impair the main two safety related functions of the VC system, temperature control and radiation protection. These functions can be established regardless of the VC system mode of operation. Removal of the smoke mode has no effect on the VC system operation or function. This activity does not physically change the plant, and therefore will not change, degrade, or prevent actions described or assumed in the accident analysis for mitigating the effect of any accident or transient. The proposed activity will not affect any of the fission product barriers because the ventilation smoke mode is not required for operability of the VC system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Removing the requirement to test the VC system flow does not result in any change to the way the components of the control room ventilation smoke mode are operated. There is no flow testing requirement specified in the design documents for the VC system to handle its function of smoke and smoke odor removal. This activity does not change the way the components of the VC smoke mode are operated. Also, the VC smoke mode is not required for the operability of the VC system. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The control room ventilation smoke detectors and the control room ventilation smoke mode are not addressed in the Technical Specifications or Bases. This activity does not affect any Technical Specification design limitations, acceptance values, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REMOVAL OF TEMPORARY START-UP STRAINERS

Activity Evaluated: USAR Change 8-331 Log Number: 99-121

Condition Report 1-98-11-062 identified, during Refueling Cycle 6, that the temporary Component Cooling Water (CC) start-up strainers 1CC01MA, 1CC01MB, and 1CC01MC were still installed. Updated Safety Analysis Report (USAR) Figures 3.6-1, Sheet 34 and 9.2-3, Sheet 1 depict the strainers installed with no indication that they are temporary. According to Piping Specialty List for the CC system and drawing M06-1032-025, revision G the strainers are temporary and for initial start-up of the system only. Therefore, the strainers are no longer required. The CC system is a non-safety related system and is required for normal operations only. The only type of accident that could be impacted is the flooding accident. The failure mode that could lead to flooding is failure of the CC system pressure boundary. The pressure boundary has not been changed by this activity. Replacing the temporary strainers with a spacer, that meets the same design, material and construction standards as the CC system, reduces friction in the system. The affect this activity has on the function and operation of the CC system is negligible. The CC system is not required to assure safe shutdown of the plant; however, reliability of the CC system is important to the overall reliability of the plant due to equipment cooled by the CC system. The only credible failure mode related to removing the strainers is pump malfunction due to foreign material in the fluid stream. Introduction of foreign material in the CC system during normal operation is unlikely because it is a closed water system and cleanliness of the system is maintained in accordance with plant procedures. Removal of the strainers updates the configuration of the CC system to the original design intent. The design intent was for the strainers to be installed temporarily during initial system start-up. During initial system start-up, there is a possibility of construction debris being trapped in the piping. Following start-up, any construction debris that might cause pump damage would have been trapped by the strainers and removed. The CC system provides a barrier between cooling loads that are potentially radioactive and the service water which cools the CC heat exchangers and is discharged to the environment. A radiation monitor is located downstream of all potential sources to indicate leakage into the CC system. This design feature of the CC system is not affected by removal of the strainers. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Removal of the strainers does not have an effect on any radiological barriers, or more importantly, the barriers to fission products; and in addition, the removal of the temporary start-up strainers does not create any new credible failure modes. Since a failure of the CC system does not cause any type of accident previously evaluated in the USAR, and the system is expected to be more reliable without the temporary start-up strainers, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. There is a remote possibility that a change to the CC system could affect Technical Specifications 3.3.3.2, "Remote Shutdown System;" 3.4.7, "RCS Leakage Detection Instrumentation;" 3.6.1.3, "Primary Containment Isolation Valves;" and 3.6.5.3, "Drywell Isolation Valves." However, these Technical Specifications address operational limits for structures, systems, and components (SSCs). As removal of the temporary strainer does not affect the operation of any SSCs, there is no impact on any operational limits or technical bases contained in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE TECHNICAL SPECIFICATION BASES IDENTIFYING INSTRUMENT INDICATION CHANNEL UNCERTAINTIES CONSIDERATIONS

Activity Evaluated: TS Bases Change BL-98-004 Log Number: 99-127

Technical Specification (TS) Bases Change BL-98-004 adds a statement to identify what TS Surveillance Requirements (SRs) are to be considered "Nominal" or "Not Nominal". Additionally, this TS Bases change incorporates into each affected Bases section a reference to the applicable instrument channel uncertainty calculation number used in the "Nominal" or "Not Nominal" determination. This information is being incorporated to clearly identify which SR values require additional consideration of instrument uncertainties when indication instrumentation is used to verify the SR. This change only provides additional reference information to clarify the bases for TS SRs. There is no impact on any design basis accidents or radiological considerations described in the Updated Safety Analysis Report (USAR). This activity does not involve any changes to any structures, systems, or components (SSCs). This activity does not change any system's reliability or performance, nor reduce any system redundancy. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no credible failure modes associated with this change. This activity does not affect any plant systems to perform their intended safety function. This activity does not introduce any SSCs into the plant. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. No acceptance limits or allowable values found in the Technical Specifications are impacted by this change. This change is not a design change, so there are no new design limitations associated with this change. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REQUIRED COOLING WATER FLOW TO THE RCIC LUBE OIL COOLER

Activity Evaluated: ECN 31687, USAR Change 8-340 Log Number: 99-131

Engineering Change Notice 31687 and Updated Safety Analysis Report (USAR) 8-340 revises the cooling water flow rate to the Reactor Core Isolation Cooling (RCIC) lube oil cooler from 16 to 25 gallons per minute (gpm) to 8 to 25 gpm. This cooling water flow rate is bounded by analysis in Calculation IP-M-0559, Revision 00. The function of the RCIC system is to respond to transient events resulting in a loss of feedwater by providing sufficient makeup coolant to the reactor to keep from challenging Emergency Core Cooling Systems (ECCS). RCIC is an accident/transient mitigation system. Therefore, this change does not impact accident initiation. In addition, event limited to the RCIC system, such as high energy line break, are not affected by this change. This activity does not affect piping structural integrity since piping pressure and flow are not adversely affected. The only credible potential failure mode associated with this activity would be a failure of the RCIC turbine bearings due to a significant increase in lube oil temperature, which can subsequently affect the availability of the RCIC turbine. The design function for the RCIC lube oil cooler is to remove of heat from the turbine lube oil and transfer the heat to the cooling water. The heat in the lube oil comes from the RCIC turbine bearings, which are cooled by the lube oil. The GE design specification does not invoke a specific limit for normal operation but the bearing inlet high temperature alarm is set at 160°F. Analysis performed in calculation IP-M-0559, revision 00 showed a small increase to approximately 151.2°F in the bearing lube oil return temperature based on 140°F cooling water supply at 8 gpm. This is well below the operational lube oil return limit of 185°F and the 160°F alarm setpoint. As such there will be no adverse affect on the bearings as a result of the reduced cooling water flow. The reduced cooling water flow to the turbine lube oil cooler does not reduce the RCIC accident mitigation performance capability. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The reduced cooling water flow requirement to the turbine lube oil cooler does not change the ability of the RCIC system to perform its design basis function, nor does this activity introduce any new failure mechanisms. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The RCIC system will continue to function within operational limits and the accident mitigation functions and performance of the RCIC system under postulated accidents are not adversely affected; therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CPS MAINTENANCE ORGANIZATION CHANGE

Activity Evaluated: USAR Change 8-347 Log Number: 99-132

This Updated Safety Analysis Report (USAR) change eliminates the Maintenance Direct Support Group including the position of Director - Maintenance Direct Support. The reassignment of the Maintenance Direct Support Group's responsibilities, which are administrative in nature, to the other Maintenance groups does not result in a reduction of the commitments or effectiveness of the Maintenance organization and its responsibilities. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. The basis of the site organizational requirements established in Technical Specification 5.2.1 are not affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ELIMINATE ROTAMETERS FROM 0PL33JA

Activity Evaluated: ECN 30369 and USAR Change 8-346 Log Number: 99-133

Engineering Change Notice (ECN) 30369 replaces the existing rotameters on sample panel 0PL33JA with flexible tubing. The rotameters that are being replaced will be permanently removed from the panel and the panel cutouts will be covered with metal plate. This change also revises Updated Safety Analysis Report (USAR) Section 9.3.2.2 to clarify that administrative controls are used to control purge flow rate on sample lines that do not have rotameters. Chapters 6 and 15 do not discuss any accidents or transients that are initiated by the Process Sampling (PS) System. The flexible tubing meets the design, material, and construction standards applicable to the PS system. The flexible tubing is compatible with the existing panel tubing materials, exceeds the pressure rating of the rotameters, and is in compliance with USAR requirements. The tubing is made of a Teflon inner liner surrounded by stainless steel mesh. The Teflon product is a non-contaminating type and is qualified for the sample water temperatures. The purge flow water goes to the equipment drain sumps, where the water is treated prior to being returned to reactor service. Thus, no Teflon product will be introduced to the feedwater system or reactor vessel as a result of this modification. In addition, this modification does not affect the Fire Hazards Analysis. This activity does not adversely affect the design, functions, or method of performing the functions of the PS system or any other system. The function of the rotameters is to provide the chemistry technician with a convenient way to set and control sample purge flow. Controlling purge flow minimizes the potential for sample line plate out to occur due to low flow rates, and to minimize personnel exposure due to high flow rates. Clinton Power Station Procedure 3222.06, “Radwaste Building Sample Panel 0PL33JA,” contains guidance on sample purge rates and allows the use of alternate methods other than rotameters to measure sample purge rate. This modification does not change the sample purge flow collection process in any way other than creating more points where the alternate purge rate measurements methods will be used. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The potential failures of the flexible tubing are tube breaks resulting in leaks, leaking fittings, flow blockage, and incorrect sample purge flow rates. However, all of these failures are bounded by the failure of the rotameter. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Neither the Technical Specifications or the Bases discuss the PS system. Replacing the rotameters with flexible tubing does not adversely impact the ability to obtain grab samples at this panel. The PS system is a non-safety related system used to sample continuously or intermittently during plant operation and shutdown and is not required for safe shutdown of the plant. Removing the rotameters from the grab sample lines has no impact upon the acceptance values or design limitations for any structure, system, or component, and has no impact upon safety limits limiting safety system settings and limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ORM REVISION TO SHOW THERMAL OVERLOAD BYPASS

Activity Evaluated: ORM Change 26-7 Log Number: 99-135

Operational Requirements Manual (ORM) Change 26-7 revises ORM Attachments 3-3 and 3-4 to show thermal overload protection is continuously bypassed in the opening direction for Main Steam Isolation Valves (MSIV) in the Leakage Control System and to show thermal overload protection is continuously bypassed in the closing direction for valve 1HG001. The electrical schematic drawing for the valves, referenced in Updated Safety Analysis Report (USAR) Table 1.7-1, show the thermal overload protection is continuously bypassed in both directions. Revising ORM Attachments 3-3 and 3-4 will bring them into conformance with the Electrical Schematic drawings. Bypassing thermal overload protection for valve safety direction during a design basis accident is required by Regulatory Guide 1.106 because it provides the highest level of assurance that valves will perform their design basis accident functions. With the thermal overload protection bypassed, it will not prematurely stop valve movement during accident conditions before the valve has completed its design basis safety function. These valves will continue to meet Regulatory Guide 1.106 following this change. Additionally, bypassing of the thermal overload protection during accident conditions is in accordance with the USAR. Making this change will not have a deleterious affect on overall system response characteristics, cause operational transients within the system or cause adverse interaction with other systems. Valve failure due to bypassed thermal overload protection is not an initiating event for any of the accident previously evaluated in the USAR. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Updating ORM Attachments 3-3 and 3-4 to show the current licensing and design configuration for the thermal overload bypass will not create any new credible failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. System function and operation remain unchanged. This change does not reduce the reliability of these valves to function and meet Technical Specification requirements. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PLACEMENT OF A VALVE IDENTIFICATION TAG FOR VALVE 01A261

Activity Evaluated: ECN 31715 Log Number: 99-136

Engineering Change Notice (ECN) 31715 places a valve identification tag on inlet isolation valve 01A261 for Instrument Air (IA) regulator 1IA08MA and revises associated drawings, procedures, and Updated Safety Analysis Report (USAR) Figure 9.3-2. The effect of placing an identification tag on the valve is to add a very small amount of weight to the valve body. If the valve and/or IA tubing were to fail due to the addition of the valve identification tag the amount of air flow through the break would not be great enough to cause a reduced flow of instrument air to any component required to safely shut down the reactor. Due to the size of the 1/2" IA system tubing, even a catastrophic failure of the valve and/or tubing would not produce conditions that would result in loss of IA to plant components. The failure of a major IA line has been evaluated in Chapter 15 and the Frequency Classification is evaluated as an incident of moderate frequency. The addition of the valve and valve identification tag does not change the frequency classification or degrade the performance capability of any system important to safety. USAR Chapter 15 defines an accident/transient condition which is initiated by a loss of Service Air (SA) and the corresponding loss of IA. Since this design basis event does not result in any fuel failure or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event. This change actually enhances the availability of this portion of the IA system by providing a more comprehensive valve line up check list and the current design/configuration information required to make operational based decisions. Hence, functional or operational reliability is actually increased. System performance, equipment protective features, system redundancy and independence are not affected by this change since there are no physical changes to the equipment function. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The addition of a valve tag slightly increase the load on the valve/tubing in the vicinity of IA air regulator. This increase in load is so insignificant that it has a very small effect on the seismic load carrying capability of the system and has no effect on the systems operation. This would produce tubing and valve missles with a corresponding loss of instrument air to Continuous Air Monitors (CAMs) PR13 through PR17. These CAMs are not safety related, nor do the CAMs provide a function required to safely shut down the reactor. The most severe failure which is considered credible is the loss of IA initiated by the loss of SA. This failure has been evaluated in USAR Section 15.2.10. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specification Safety Limits and/or Bases for the functioning of the reactor are not affected by the IA system failure. Since there is no release of radioactive materials to the secondary containment or to the environment, there is no change to the radiological dose rate. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ELIMINATE ROTAMETERS FROM 0PL33JB

Activity Evaluated: ECN 30370 and USAR Change 8-348 Log Number: 99-137

Engineering Change Notice (ECN) 30370 replaces the existing rotameters on sample panel 0PL33JB with flexible tubing. The rotameters that are being replaced will be permanently removed from the panel and the panel cutouts will be covered with metal plates. This change also revises Updated Safety Analysis Report (USAR) Section 9.3.2.2 to clarify that administrative controls are used to control purge flow rate on sample lines that do not have rotameters. Chapters 6 and 15 do not discuss any accidents or transients that are initiated by the Process Sampling (PS) System. The flexible tubing meets the design, material, and construction standards applicable to the PS system. The flexible tubing is compatible with the existing panel tubing materials, exceeds the pressure rating of the rotameters, and is in compliance with USAR requirements. The tubing is made of a Teflon inner liner surrounded by stainless steel mesh. The Teflon product is a non-contaminating type and is qualified for the sample water temperatures. The purge flow water goes to the equipment drain sumps, where the water is treated prior to being returned to reactor service. Thus, no Teflon product will be introduced to the feedwater system or reactor vessel as a result of this modification. In addition, this modification does not affect the Fire Hazards Analysis. This activity does not adversely affect the design, functions, or method of performing the functions of the PS system or any other system. The function of the rotameters is to provide the chemistry technician with a convenient way to set and control sample purge flow. Controlling purge flow minimizes the potential for sample line plate out to occur due to low flow rates, and to minimize personnel exposure due to high flow rates. Clinton Power Station Procedure 3222.07, "Radwaste Building Sample Panel

0PL33JB," contains guidance on sample purge rates and allows the use of alternate methods other than rotameters to measure sample purge rate. This modification does not change the sample purge flow collection process in any way other than creating more points where the alternate purge rate measurements methods will be used. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The potential failures of the flexible tubing are tube breaks resulting in leaks, leaking fittings, flow blockage, and incorrect sample purge flow rates. However, all of these failures are bounded by the failure of the rotameter. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Neither the Technical Specifications or the Bases discuss the PS system. Replacing the rotameters with flexible tubing does not adversely impact the ability to obtain grab samples at this panel. The PS system is a non-safety related system used to sample continuously or intermittently during plant operation and shutdown and is not required for safe shutdown of the plant. Removing the rotameters from the grab sample lines has no impact upon the acceptance values or design limitations for any structure, system, or component, and has no impact upon safety limits limiting safety system settings and limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ELIMINATE ROTAMETERS FROM 0PL33JC

Activity Evaluated: ECN 30371 and USAR Change 8-349 Log Number: 99-138

Engineering Change Notice (ECN) 30371 replaces the existing rotameters on sample panel 0PL33JC with flexible tubing. The rotameters that are being replaced will be permanently removed from the panel and the panel cutouts will be covered with metal plates. This change also revises Updated Safety Analysis Report (USAR) Section 9.3.2.2 to clarify that administrative controls are used to control purge flow rate on sample lines that do not have rotameters. Chapters 6 and 15 do not discuss any accidents or transients that are initiated by the Process Sampling (PS) System. The flexible tubing meets the design, material, and construction standards applicable to the PS system. The flexible tubing is compatible with the existing panel tubing materials, exceeds the pressure rating of the rotameters, and is in compliance with USAR requirements. The tubing is made of a Teflon inner liner surrounded by stainless steel mesh. The Teflon product is a non-contaminating type and is qualified for the sample water temperatures. The purge flow water goes to the equipment drain sumps, where the water is treated prior to being returned to reactor service. Thus, no Teflon product will be introduced to the feedwater system or reactor vessel as a result of this modification. In addition, this modification does not affect the Fire Hazards Analysis. This activity does not adversely affect the design, functions, or method of performing the functions of the PS system or any other system. The function of the rotameters is to provide the chemistry technician with a convenient way to set and control sample purge flow. Controlling purge flow minimizes the potential for sample line plate out to occur due to low flow rates, and to minimize personnel exposure due to high flow rates. Clinton Power Station Procedure 3222.08, "Radwaste Building Sample Panel 0PL33JC," contains guidance on sample purge rates and allows the use of alternate methods other than rotameters to measure sample purge rate. This modification does not change the sample purge flow collection process in any way other than creating more points where the alternate purge rate measurements methods will be used. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The potential failures of the flexible tubing are tube breaks resulting in leaks, leaking fittings, flow blockage, and incorrect sample purge flow rates. However, all of these failures are bounded by the failure of the rotameter. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Neither the Technical Specifications or the Bases discuss the PS system. Replacing the rotameters with flexible tubing does not adversely impact the ability to obtain grab samples at this panel. The PS system is a non-safety related system used to sample continuously or intermittently during plant operation and shutdown and is not required for safe shutdown of the plant. Removing the rotameters from the grab sample lines has no impact upon the acceptance values or design limitations for any structure, system, or component, and has no impact upon safety limits limiting safety system settings and limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PARALLELS/TRANSFERS 1B21-N602D LOAD TO 1B21-N651D

Activity Evaluated: Temporary Modification 99-053 Log Number: 99-139

Temporary Modification 99-053 parallels the sensor for transmitter 1B21-N602D with transmitter 1B21-N651D loads. The additional load will not adversely affect transmitter 1B21-N651D or change the operating parameters or performance of the feedwater system or existing loads. These transmitters, sensors, and their loads are delineated on E02-1NB99-003, which is incorporated into the Updated Safety Analysis Report (USAR) by reference. The feedwater system is a power generation system for the purpose of maintaining proper vessel water level. The design basis accidents associated with feedwater are failure of the feedwater control system. However, the feedwater control system design or function as described in the USAR is not altered by this activity. Also, paralleling these loads does not affect system reliability or redundancy. The total burden on the transmitter being used is within the vendor specified limits. This activity does not affect the reliability or the accuracy of the indications provided to the operators. This activity does not change, degrade, or prevent actions described or assumed in the accident analysis for mitigating the effect of any accident. This activity will not impact the radiological consequences of any accident or affect any fission barriers. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not affect the reliability or the accuracy of the indications provided to the operators. The indication affected by this activity will be supplied by the same model transmitter which is calibrated the same. Neither the probability nor the magnitude of a miscalibration of thermal power is impacted by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification 3.3.1.1 describes the scram function on a failure of the feedwater control system. Since the feedwater control system design and function as described in the USAR is not altered by this activity, this activity does not challenge any Technical Specification. Also, paralleling these loads will not affect system reliability or introduce new failures that may degrade the reliability of the FW system. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

USAR VALIDATION FOR STEAM PACKING EXHAUSTER OPERATION

Activity Evaluated: USAR Change 8-352 Log Number: 99-140

The steam packing exhauster blower description in Updated Safety Analysis Report (USAR) Section 10.4.3.2 states that "The blower is designed to pass the maximum flow with a gauge discharge pressure of 6.8 inches of water. The blower also maintains a vacuum of 5 inches of water in the outlet of the shaft packing". Theses values can not be found in the vendor technical documents. USAR Change 8-352 revises this section to state "The blower is designed to adequately operate between 10 to 12 inches of water gage at the steam packing exhauster suction with normal packing clearance. With worn packing, the gage pressure may go up to 20 inches of water". This change brings the USAR description in line with the vendor technical requirements. Chapters 6 and 15 do not address the steam packing exhauster as being an initiator of any type of accident. The steam packing exhausters are not part of a safety system nor do they perform any safety function. Operation of steam packing exhausters do not support any other system or function important to safety. This activity does not impact Engineering Safety Feature systems nor does it impact any system to perform their required safety function during normal and postulated accident conditions. This activity does not impair system reliability, redundancy, or independence. In addition, this activity does not degrade, alter or prevent actions described or assumed in the USAR in terms of radiological consequence. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since no equipment or hardware is being introduced into the plant, the failure modes of existing equipment is not changed. This activity does not impact the ability of any system to perform its design basis function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The steam packing exhauster is a part of the turbine gland sealing system which is a non-safety related system. This system is not addressed in the Technical Specifications; nor, does this change affect any system that is addressed in the Technical Specifications. This activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

EXPERIENCE ASSESSMENT

Activity Evaluated: USAR Change 8-356 Log Number: 99-141

Updated Safety Analysis Report (USAR) Change 8-356 combines the Plant Staff Experience Assessment and Corrective Action Department functions into one organization, Experience Assessment. As a result, the Director - Corrective Action position will be eliminated. This activity does not result in a reduction of the commitments or effectiveness of the Corrective action or Operating Experience Assessment organizations and their responsibilities. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE USAR AND PROCEDURES TO DELETE A DEFINED NUMBER OF ISEG PERSONNEL AND DELETE ISEG IN THE ORM

Activity Evaluated: USAR Change 8-369 and ORM Change 27-1 Log Number: 99-142

Updated Safety Analysis Report (USAR) Change 8-369 deletes reference to a minimum number of Independent Safety Engineering Group (ISEG) personnel and deletes reference to an ISEG Supervisor. Operational Requirements Manual (ORM) Change 27-1 deletes reference to the ISEG in Section 6.2.3. The independent reviews/oversite (IR/O) function has no impact on equipment or how equipment is operated because any suggested changes from the ISEG are individually evaluated for acceptability and impact on plant safety. This is an administrative organizational change which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION OF FIRE PROTECTION SAFE SHUTDOWN EQUIPMENT LIST

Activity Evaluated: USAR Change 8-361 Log Number: 99-146

Updated Safety Analysis Report (USAR) Change 8-361 revises Table 1.8-2, "Safe Shutdown Equipment List (SSEL)," of Appendix F, the Safe Shutdown Analysis (SSA). This activity adds Note 8 to reflect that Motor Operated Valves (MOVs) 1SX173A and 1SX173B are disabled in the closed position, clarifies the elevations of Safety Relief Valves (SRVs), reflects the correct 1E Divisional power for various components, clarifies the Safe Shutdown Method for various components, adds the Charcoal Filter Drain Solenoid Valves and Makeup Check Valves for the Control Room Heating, Ventilation, and Air-Conditioning (VC) System, adds the Diesel Generator (DG) Starting Air Receiver Tanks for all Diesel Generators, adds the High Pressure Core Spray (HPCS) Water Leg Pump Check Valve, clarifies Notes 1 and 2, and corrects various typographical errors. These changes provide assurance that all systems and components required for safe shutdown are identified and will be evaluated to establish compliance with Appendix R. Appendix F demonstrates that for a fire in any single plant fire area at least one method exists that is free of fire damage to achieve and maintain a safe shutdown condition.

This activity does not add any fire hazards. The continuing review of compliance with 10CFR50 Appendix R has resulted in a revision to the Appendix F SSEL and thereby assures that there is no compromise in the capability to perform a Safe Shutdown. This activity does not involve change to the design, function, operation, or test of any equipment or systems affected. There is no affect on the environmental seismic, or separation criteria of the affected systems. While a postulated fire may have an effect on systems and components involved in the fire, there is no new impairment to system reliability, degradation of equipment protective features or system performance, or reduction of system redundancy or independence due to this activity. Therefore, this activity does not increase the probability or consequences of an activity or malfunction of equipment important to safety previously evaluated in the USAR. This activity assures that the appropriate safe shutdown systems and components are evaluated as being available to support reactor shutdown during a fire and does not reflect any change to or add any credible failure modes associated with any postulated fire involving the Fire Protection systems and other plant hardware, systems, or procedures. This activity does not involve any new equipment or a change to the design, function, operation, or test of any existing equipment or systems. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Clinton Power Station Fire Protection Program is not included as part of the Technical Specifications. All acceptance values and design limitation involving the Fire Protection System were previously documented in USAR Appendix E and F. There is no change to any control of systems, components, or functions as documented in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

REVISION OF TABLE 8.3-9 TO REFLECT THE NEW LOAD PROFILE FOR THE DIVISION 2 BATTERY

Activity Evaluated: USAR Change 8-364 Log Number: 99-147

Updated Safety Analysis Report (USAR) Change 8-364 revises table 8.3-9 which contains the load profile for the Division 2 battery. The time intervals shown in this table are based on the inrush current from loads being initiated as a result of Loss-Of-Coolant Accident (LOCA) signals where such loads are credited as occurring in the first minute and the emergency lights which come on at AC power loss and are credited as being turned off after one hour. Beyond these two events, all loads are shown for a four hour duration which is the analyzed time period for the battery to supply the essential connected loads. The change to the values listed in Table 8.3-9 of the USAR will not affect the operation of the battery or the DC system. Calculation 19-D-29 revision 11 concludes that the battery is capable of carrying the connected loads for the required time interval. Accordingly, the response to the LOCA can not be impacted by this activity since the loads placed on the DC system are included in the battery capability analysis. For the loss of AC power described in USAR Section 15.2.6, the only radiological consequence pertaining to this activity consisted of radioactivity being discharged to the suppression pool through the Safety Relief Valves (SRVs). The operation of the SRVs is not impacted by the source of the DC power for the solenoid valve. The DC bus can be fed by the battery or the battery charge and it will make no difference to the SRVs. The duration of the discharge of steam through the SRVs is determined by the operator's need to control/reduce the vessel pressure and allow operation of various systems for water injection and heat removal. This discharge is what determines the amount of radioactivity discharged to the suppression pool and is independent of the values listed in Table 8.3-9 of the USAR. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not introduce any new components into the plant. The revised values for Table 8.3-9 represent a more detailed analysis and not a new design or new equipment in the plant. Therefore, there are no failure modes that could be introduced. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The requirements for the DC system which includes the Division 2 battery are discussed in Sections 3.8.4 and 3.8.5 of the Technical Specifications. Surveillance Requirement 3.8.4.7 calls for a service test of the battery, but does not state numerical values for that test. Instead, it refers to the duty cycle specified in the USAR. The duty cycle for the Division 2 battery are listed in Table 8.3-9. A review of the results of the last service test performed for this battery confirm that the results were acceptable per the voltage criteria established and the test bounded the values of the new duty cycle. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

CONNECTION OF A TEMPORARY BATTERY CHARGER TO THE BOP (NON-SAFETY) DC BUSES

Activity Evaluated: Temporary Modification 99-056 Log Number: 99-148

Temporary Modification 99-056 installs a battery charger with a 300 amp rated output to maintain the Balance of Plant (BOP) DC 1F Bus and the BOP battery 1F on float charge. The charger will be located in the Unit 2 area of the 781' elevation of the Control building, in accordance with site procedures for control of transient materials. The temporary cables will be routed through several doors and the resultant impact on fire protection due to leaving the doors open while the temporary modification is in place will be handled in accordance with site procedures. There are no accident scenarios addressed in the Updated Safety Analysis Report (USAR) that are associated with the non-divisional DC buses. Loss of these buses will have an adverse impact on the plant, but the buses are not required for the response or mitigation of any accident. The temporary battery charger does not provide the same amperage; however, this temporary battery charger capacity exceeds the nominal 1F bus load. This temporary modification adds combustibles to the areas in which the temporary cable are installed; however these cables and the battery charger do not represent enough fire load to cause a change in the fire load ratings for the area. The temporary cables are not routed near any 1E electrical equipment; therefore, no electrical separation concerns are created by this activity. The only interface this temporary power supply has with equipment required for safe shutdown of the plant is the Reactor Protection System (RPS). Since the normal AC and back-up DC power supply to the RPS system are non-safety, the RPS system is designed to perform its safety function upon loss of power. The equipment used to maintain barrier integrity and prevent release of radiation to the public is not impacted by this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The review of the electrical and physical installation, including the walkdown performed to identify any seismic interaction or violation of electrical separation, indicates there is no potential for a new accident different from those evaluated in Chapters 6 and 15 from this temporary modification. Both the expected load and the maximum output of the charger were determined during this evaluation to be within acceptable operating limits for the DC buses. Accordingly, all equipment fed from the non-safety buses will remain fully functional. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The only items discussed in the Technical Specifications associated with this temporary modification are the RPS inverters. Technical Specification Bases 3.8.7, "Inverters-Operating," discusses the RPS inverters and their preferred feed from the DC source. That feed is not impacted by this temporary modification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE TS BASES SR 3.6.2.1.1 FOR SUPPRESSION POOL TEMPERATURE LIMIT OF 95 DEGREES TO "NOMINAL"

Activity Evaluated: TS Bases Change BL-99-024 Log Number: 99-150

Technical Specification (TS) Bases Change BL-99-024 revises Surveillance Requirement (SR) 3.6.2.1.1 to change the "Not Nominal" statement (added by Bases Change BL-98-004) to show that the 95°F LCO limit is a "Nominal" value, including the effects of instrument uncertainties. This activity also revises the acceptance criteria in the surveillance procedures used to verify the SR values for suppression pool average temperature. This change clarifies the TS Bases 3.6.2.1.1 to show that the design basis includes sufficient consideration of instrument uncertainties. These revised acceptance criteria are supported by calculation IP-0-0071 and conservatively support verification of the TS SR limits by imposing administrative limits where necessary. The Suppression pool temperature at the beginning of an event can affect the Design Basis Accident (DBA) Loss-of-Coolant Accident (LOCA) coping capability or the ability to cope with Safety Relief Valve (SRV) blowdown events. The calculation determined the various safety analyses do not assume a particular accuracy for the suppression pool bulk average temperature. However, the safety analyses incorporate sufficient mechanical margin to accommodate instrument inaccuracies of about 1.5°F. In addition, this activity does not have any design, material, or construction standards to be considered. This activity does not involve any changes to any systems, structures, or components (SSCs). This activity does not change any system reliability, performance, or reduce any system redundancy. Also, no fission product barriers are impacted by this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). Since this activity only adds a description to a TS Bases section to identify which SR values can be considered "Nominal" or "Not Nominal" and revises associated procedures to change the administrative limits, there are no effects, directly or indirectly, on any plant systems. Therefore, there is no possibility of creating an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This change does not impact the Technical Specifications, nor does it change any of the acceptance limits or allowable values found in the Technical Specifications. This change is not a design change, so there are no new design limitations associated with this change. This change does not impact any limits, limiting safety systems settings, or limiting conditions for operations. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

ON-LINE LEAK REPAIR BY INJECTING SEALANT

Activity Evaluated: Temporary Modification 99-058 Log Number: 99-152

Temporary Modification 99-058 authorizes the on-line leak repair to the piping downstream of the Feedwater Header 6A Emergency Drain Line Vent Valve, 1HD120A. This activity is limited to installing a leak sealant device on pipe line 1HD75AA which encapsulates valve 1HD120A, and to crimping pipe 1HD75AA downstream of the leak sealant device and sealing the line by filling it with leak sealant. This line is the vent path from Feedwater Header 6A Emergency Drain Line and represents a small steam line outside of containment. Updated Safety Analysis Report (USAR) Section 15.6.4 addresses pipe line breaks outside of containment, with a 24" main steam line break as the worst accident. A failure of this temporary modification would be bounded by this accident analysis. The Heater Drain (HD) system is non-safety related, not required for safe shutdown of the plant, and is not required during or after an accident. The pipe stress calculation was reviewed, and it was determined that the pipe can support the additional weight of the leak sealant device and the injection of the sealant. This activity will restore the HD system back to its original design function, with the exception to losing the ability to vent the system through this line. This activity will not affect the operation of the plant, since this valve and its discharge line are used to vent and fill the system prior to startup. This valve and line will be replaced when plant conditions can support their replacement. The enclosure is as reliable as the valve it is enclosing. Thus, neither off-site nor on-site doses will change as a result of this plant change. This activity does not affect any of the fission product barriers as it contains no change to pipe routing, radiological boundaries, radiological monitoring equipment, or structural configuration. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The credible failure mechanism associated with this activity is the failure of the sealant device or the sealed line to act as a pressure boundary. The potential for the piping system to fail to act as a pressure boundary has been previously addressed by USAR Section 15.6.4. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Although the operation and control of the HD system is not specifically addressed in the Technical Specifications, the HD system does have an impact on feedwater temperature, which is addressed in Technical Specification 3.2.2, "Minimum Critical Power Ration." This activity does not affect the operation of the HD system; therefore, there is no effect on the feedwater temperature and the requirements specified in Technical Specification 3.2.2. This line does not support the operation of and will not affect any safety related systems, structures, or components. Thus, there are no acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with components affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISED VALVE POSITION FOR VALVES 1IA851 AND 1IA852, AND ADDED VALVE 1IA1160

Activity Evaluated: ECN 31757 Log Number: 99-153 R/1

Engineering Change Notice (ECN) 31757 revises Instrument Air (IA) Piping and Instrumentation Drawing M05-1040 Sheet 22, Operational Schematic OS-1040 Sheet 4, and Updated Safety Analysis Report (USAR) Figure 9.3-2 Sheet 20 to show valves 1IA851 and 1IA852 in the open position supplying air to 1WO236 and 1WO204 respectively and to show a previously unidentified instrument air block valve upstream from 1IA852, the valve has been designated as 1IA1160. Loss of instrument air is a transient which is evaluated in USAR Section 15.2.10. Loss of the IA system will result in the shutdown of the reactor due to the opening of the control rod scram valves and/or the closing of the main steam line isolation valves, but that the failure of instrument air will not interfere with the safe shutdown of the reactor since all equipment using instrument air is designed to fail to a position that is consistent with the safe shutdown of the plant. Valve 1IA1160 is an integral part of the IA system; design change records do not identify valve 1IA1160 as being a design change. This indicates that the valve was part of the original installation; and thus, meets the same requirements as the remaining portion of the system. The portion of the IA system which is downstream from valves 1IA851, 1IA852, and 1IA1160 has the same design, material and construction standards as the upstream portion. This portion of the system will be subjected to the same pressure as the upstream portion of the system. The operating conditions of the IA system are within the design pressure and temperature limits in the applicable specifications; and is designed to not fail when operated in accordance with the specified conditions. The evaluation in USAR Chapter 15 concludes that the consequences of the postulated loss of the IA system would not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment is designed. As a result, these barriers would maintain their integrity and function as designed. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Even though the piping is designed to not fail when operated in accordance with the specified conditions, it could be postulated to fail. However, this type of transient has already been evaluated in Chapter 15 of the USAR. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Bases 3.5.1.3 discusses verification of adequate air pressure for reliable Automatic Depressurization System operation in the air supplied by the IA system. The Technical Specification Bases 3.6.5.3 discusses the fail-closed feature of the valves in the event of the loss of instrument air supply to the valves. This activity does not affect the acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with these Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

HIGH PRESSURE CORE SPRAY SYSTEM SUPPORT LOADS

Activity Evaluated: Operability Evaluation 1-99-07-165-OD-1 Log Number: 99-154

Condition Report (CR) 1-99-07-165 identified that the weight of valves 1E22-F010 and 1E22-F011 is 3754 pounds each versus 3260 pounds as shown on vendor drawings. The piping analysis used the originally supplied weight information of 3260 pounds. This newly identified weight causes additional stresses on supports for line 1HP02A-14", under postulated loading conditions, in excess of allowable stresses per the design rules of the ASME Code. However, Nuclear Regulatory Commission Generic Letter 91-18, Revision 1 provides alternate rules to demonstrate operability of supports. The supports have been analyzed to be within these alternate rules for stress allowables. Operability Determination/Operability Evaluation for CR 1-99-07-165 allows a disposition of "Use-As-Is" for the High Pressure Core Spray (HPCS) system piping. Updated Safety Analysis Report (USAR) Section 15.5.1 discusses the Inadvertent HPCS Pump Start-up accident. This activity does not affect the two possible modes of an inadvertent HPCS Pump Start. According to preliminary analysis, Level B (upset plant conditions) loads on the supports are greater than the Level B support capabilities. Level C (emergency conditions) and Level D (faulted condition) loads on the supports are within Level C and Level D allowable loads. The Level B loads are less than Level C allowable loads ensuring that the support members will not yield during any Level B event. The only concern of Level B loads exceeding Level B allowable loads would be metal fatigue of the support elements during multiple Level B events such as earthquakes. Therefore, a limited amount of time with the supports in this condition will not affect the operability or structural integrity of the system. As stated above, Nuclear Regulatory Commission Generic Letter 91-18, Revision 1 provides alternate rules to demonstrate operability of supports. These provisions exist in recognition of the conservatism in the piping analysis and the low probability of the various loads assumed in the analysis occurring simultaneously. Since this activity does not alter the system configuration, operation, response characteristics, or its interaction with other systems, the radiological consequences of a malfunction of the HPCS system are unchanged. This activity does not alter the physical characteristics or required functions of the HPCS system piping, and therefore, does not degrade or otherwise prevent the system from performing any actions assumed in the accident analysis. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not delete, add, modify, or relocate any equipment important to safety. This activity does not alter the operation, system configuration, or response characteristics of the HPCS system, or its interaction with other systems. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not affect the acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with any Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE TECHNICAL SPECIFICATION BASES DRYWELL TO CONTAINMENT PRESSURE VALUES FROM "NOT NOMINAL" TO "NOMINAL"

Activity Evaluated: Technical Specification Bases Change BE-99-027 Log Number: 99-156

Instrument Uncertainty Calculation IP-0-0092 Revision 1 evaluated the channel uncertainties and determined that Drywell-to-primary containment differential pressure uncertainty to be ±0.0987 psi. The channel uncertainty was then compared to the available margin for both the upper and lower limits in Technical Specification 3.6.5.4 and found that these limits include significant margin, adequately accounting for the calculated instrument uncertainties. Since there is sufficient margin to account for the instrument uncertainties, Technical Specification Bases Change BE-99-027 revises Section B3.6.5.4.1 from "Not Nominal" to "Nominal". The accidents associated with this change include those accidents which use Drywell or Primary Containment pressure as an initial condition to the event. Accidents, such as the Large break Loss-of-Coolant Accident (LOCA), rely on the Emergency Core Cooling System(s) which assumes a maximum Drywell pressure in the LOCA analysis. Additionally, the containment response analysis assumes that both the Drywell and containment initial pressure to be 0.0 psig. There is no impact on any of these assumptions or initial conditions due to this change. This change does not have any design, material, or construction standards to be considered. There are no changes to any systems important to safety associated with this activity. There are no automatic actions associated with these instrument channel indications, they provide pressure indication only. This activity does not change any system's reliability or performance, nor reduce any system redundancy. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not delete, add, modify, or relocate any equipment important to safety. This activity does not alter the operation, system configuration, response characteristics, or system interactions. The Surveillance Requirement limits have been demonstrated to be acceptable. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The SR limits have been evaluated in IP-0-0092 Revision 1 which determined that the instrument channel uncertainties are insignificant when compared to available mechanical margin. In addition, no new design limitations are associate with this change. This activity does not impact the acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with any Technical Specification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PROVISION FOR USAR/CLB PROJECT DISCREPANCY RESOLUTION PROCESS

Activity Evaluated: CPS 1005.06, Revision 13 Log Number: 99-157

Clinton Power Station (CPS) Procedure 1005.06, "Conduct of Safety Reviews," Revision 13 allows maintenance of the Updated Safety Analysis Report (USAR) and resolution of certain identified USAR discrepancies without 10 CFR 50.59 screenings or safety evaluations. This change is limited to those activities covered under Nuclear Safety and Performance Improvement (NSPI) Procedure L.19, "Conduct of CLB Discrepancy Resolution." Attachment 1 to NSPI Procedure L.19 identifies the specific criteria that a USAR change must meet in order to be eligible for this exclusion. Changes not meeting this exclusion criteria will continue to require safety evaluations consistent with current practices, already established in CPS 1005.06 step 8.2.3.3 and Attachment A. The impact of this action results in a reduction in the number of safety evaluations that the Facility Review Group (FRG) and Nuclear Review and Audit Group (NRAG) organizations will review. This reduction in review activities will have a negligible impact on the assurance that a change activity will not introduce an unreviewed safety question. This is because of the nature of the exception criteria established in NSPI Procedure L.19. The exception criteria were promulgated based on the limitation that the USAR change made do not invalidate or remove information establishing the 10 CFR 50.2 design bases, safety analyses, and associated USAR description. In addition, this activity does not alter the assumption regarding when FRG and NRAG reviews are required for initiated facility and procedure changes. Thus, the activity has no direct impact on either a facility change, procedure change, or test and experiment, that could impair the performance of a previously credited mitigatory action, sequence of events, method of propagation, fission product barrier, assumed release path, or source term for any previously evaluated accident. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Thus, the activity has no direct impact on the postulated frequency of failure of plant equipment. Nor will the activity result in new postulated failure modes or mechanisms for any plant structure, system or component. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. This activity does not directly alter the processes for performing change activities to either the facility or its procedures. Also, this activity does not alter the assumptions regarding when FRG and NRAG reviews are required for initiated facility and procedure changes. Therefore, this activity will not result in a reduction in any margin of safety associated with the Administrative Controls established in Technical Specification Section 5.0. Further, this activity has no direct impact on the plant structures, systems, or components. This activity does not directly alter the method of establishing and verifying operability of any plant component subject to a requirement in Section 3.0 of the Technical Specifications. This activity does not directly alter the requirements for systems or their design bases. Therefore, this activity does no directly alter the method of derivation of the requirements established in Sections 2.0, 3.0, or 4.0 of the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

RCIC STEAM LINE DRAIN POT HIGH LEVEL ALARM

Activity Evaluated: Temporary Modification 99-062 Log Number: 99-158

Temporary Modification 99-062 installs a time delay module for the Reactor Core Isolation Cooling (RCIC) Turbine Steam Exhaust drain trap high level alarm, a new computer point to monitor the frequency of the RCIC Turbine Steam Exhaust drain trap high level actuations. This Temporary Modification resolves the Main Control Room operator distraction related to frequent annunciator actuations due to excessive condensation build-up in the RCIC Turbine Steam Exhaust line drain trap. Updated Safety Analysis Report (USAR) Section 15.1.2 discusses Feedwater Controller Failure - Maximum Demand. In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high level scram and tripping of the main turbine and feedwater pumps, recirculation pump trip, and low water level initiation of the RCIC System and the High Pressure Core Spray (HPCS) System to maintain long term water level control following tripping of feedwater pumps. Temporary Modification 99-062 cannot possibly initiate any of these failure modes. The design basis of the RCIC Turbine Exhaust Drain Pot high water level alarm indicates that the purpose of the alarm is to detect the presence of condensation accumulation in the exhaust line which creates the potential for water hammer, damaging to the exhaust pipe upon injection of high pressure steam due to RCIC initiation. Excessive accumulation of condensation in the exhaust line is mainly caused by leakage from the steam admission and/or bypass valves. The bigger the leakage the higher the frequency of the alarm actuation. Therefore, an increasing frequency of alarm actuation is a clear evidence of a more advanced level of steam admission and/or bypass valve leakage degradation. After the installation of the temporary modification, an increased frequency of alarm actuation will not be noticed from the annunciator window. An alternate means for complying with the exceptions of USAR Section 7.7.1.23.1 must be established and controlled. This will be established by installing a digital computer point whose purpose is to provide a list of every high level actuation. This activity does not impact the safety function of the drain pot level actuation signal or of any component associated with the RCIC system. In addition, this activity does not change, degrade, or prevent any system or component from performing actions described or assumed in the accident analysis for mitigating the effect of the accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since this Temporary Modification installs a time delay module and computer input down stream of the electrically isolated safety related high level actuation, a failure of either time delay module or computer point will not impact the safety function of the level switch. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The operability of the RCIC system instrumentation is dependent on the operability of the individual instrumentation channel function specified in Technical Specification Table 3.3.5.2-1. The installation of a time delay in the drain high water level alarm circuit and the installation of a new computer point will not impact any of the instrument functions specified in this table. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE SWITCHES 1C41A-S004A AND 1C41A-S004B IN THE STANDBY LIQUID CONTROL SYSTEM INSTRUMENT PANEL

Activity Evaluated: ECN 31767 and USAR Change 8-389 Log Number: 99-159

Engineering Change Notice (ECN) 31767 replaces pushbutton switches 1C41A-S004A and 1C41A-S004B at the Standby Liquid Control system (SLC) Instrument Panel in the Containment Building. These switches are only used during performance of surveillance runs of the SLC pumps. For the remainder of the time, these switches are not needed and the only design consideration is that they should not operate spuriously or affect the operating circuit from the Main Control Room. The replacement switches will have maintained contacts to resolve a concern with spurious tripping of the pumps during surveillances. To prevent unintentional operation of the switch between surveillances, the design will require two actions rather than one for the switch to be placed in the test position. The SLC system provides the capability to respond to certain special events; for these events, the initiator that requires use of SLC is a failure of the control rods to insert or to receive a signal to insert. While the initiator is not defined in Chapter 15, it is expected that some aspect of the control rod drive system is involved. Chapter 7 states "SLCS is separated both physically and electrically from the control rod drive system." ECN 31767 is completely within the boundaries of the SLC system, thus it follows that this change is also physically and electrically separate from the control rod drive system. The new switches are installed in the same location using the same mounting configuration as the original switches. These switches are only used during testing of the pumps so they will not impact operation of the SLC system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). This activity uses a different model of switches in the local test control circuits. The functional difference between the original switch and the replacement is the maintained versus momentary nature of the contact operation. The change precludes the possibility of switch operation when the pump surveillance is not being performed. If such inadvertent switch operation were to occur, the resultant system conditions would stay within the system design parameters and would not result in any condition that would be considered an accident. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This switch replacement does not affect the system operation, which is discussed in Technical Specification 3.1.7 and its associated Bases. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FUSE REPLACEMENT IN DIESEL GENERATOR CONTROL PANELS

Activity Evaluated: ECN 31788 and 31789 Log Number: 99-160

Engineering Change Notices (ECNs) 31788 and 31789 replace the Gould Shawmut type A25X10 fuses in Diesel Generator (DG) control panels 1PL12JA/JB with Gould Shawmut A2K10R type fuses. These fuses isolate the Class 1E portion of DG 125 Vdc control circuits from the non-Class 1E Annunciator Power Supply in the event of a fault on the non-Class 1E side. The replacement fuses are class 1E qualified with the same 10 amp rating as the original fuses and are installed in the existing fuse blocks; there are no wiring changes associated with this design change. These replacement fuses meet the applicable design, material and construction standards applicable to the system and equipment being modified. The replacement fuses are Class 1E qualified. In addition, the analysis for this design change demonstrates that the replacement fuses will coordinate with the upstream breakers in the event of a fault. This activity does not affect overall system performance in a manner which could increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR)..Premature opening of the fuses would have no impact on the performance of safety systems since the annunciator system is non-safety related. The failure of these fuses to open, or delayed opening in the event of a short circuit on the power supply side, could interrupt direct current (dc) power to the DG and associated equipment which would result in loss of the DG. Based on analysis, the new fuses coordinate with the upstream breakers in the event of a fault thereby isolating the non-1E system from the Class 1E dc control circuit. Since the replacement of isolation fuses by this design change does not change the failure modes of affected equipment and does not cause any new credible malfunctions, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not specify any margin of safety for these fuses. Technical Specification 3.8 describes the requirement for the electrical distribution system. This design change does not alter this Technical Specification requirement or its associated Bases. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE BLANK FLANGE WITH WELDED PIPE CAP

Activity Evaluated: ECN 31768 and USAR Change 8-378 Log Number: 99-161

Engineering Change Notice (ECN) 31768 replaces the flanged clean-out connections upstream of valve 1B21SPDV1 with a welded pipe cap. This change includes revising Updated Safety Analysis Report (USAR) Figure 3.6-1 Sheet 7 and Figure 10.3-1 Sheet 3 to reflect a pipe cap instead of a blank flange. The location of the blank flange is between the turbine control valve and the high pressure turbine inlet. The failure mode associated with a welded pipe cap is limited to failure of the cap to function as a pressure boundary. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. The listed pipe cap material, pressure rating, and schedule meets or exceeds the relevant system piping design requirements. The cap will be welded and inspected per existing site procedures and appropriate post installation testing will be performed to assure that the welded cap will function as a pressure boundary. In addition, the change from a blank flange to a pipe cap will not adversely affect the pipe line support. The performance of welding introduces a fire source. In order to assure that the performance of the work does not initiate a fire, the controls Clinton Power Station procedure 1893.02, "Fire Protection - Control of Ignition Sources," will be established. This assures that the increased potential of a fire is compensated for through the use of a fire watch. The Main Steam (MS) line drain system is a non-safety related system and is not required to effect or support safe shutdown of the plant or to perform in the operation of reactor safety features. No equipment important to safety is located in the area of influence around the blank flanges. This activity does not result in system configurations that challenge the operation of any safety system or impair any systems reliability when relied upon for system protection. Therefore, this activity will not prevent actions described or assumed in the accident analysis for mitigating the effect of any transient, nor will any fission product barriers be challenged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure mode for the welded pipe cap is the same as the failure mode for the blank flange. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The MS drain system is not addressed in the Technical Specifications. Replacing the blank flange with a welded pipe cap does not adversely impact the ability to perform dirt and debris inspections of the steam line; nor does it have any impact upon the acceptance values or design limitations of any system, structure, or component. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE BLANK FLANGE WITH WELDED PIPE CAP

Activity Evaluated: ECN 31769 and USAR Change 8-379 Log Number: 99-162

Engineering Change Notice (ECN) 31769 replaces the flanged cleanout connections upstream of valve 1B21SPDV2 with a welded pipe cap. This change includes revising Updated Safety Analysis Report (USAR) Figure 3.6-1 Sheet 7 and Figure 10.3-1 Sheet 3 to reflect a pipe cap instead of a blank flange. The location of the blank flange is between the turbine control valve and the high pressure turbine inlet. The failure mode associated with a welded pipe cap is limited to failure of the cap to function as a pressure boundary. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. The listed pipe cap material, pressure rating, and schedule meets or exceeds the relevant system piping design requirements. The cap will be welded and inspected per existing site procedures and appropriate post installation testing will be performed to assure that the welded cap will function as a pressure boundary. In addition, the change from a blank flange to a pipe cap will not adversely affect the pipe line support. The performance of welding introduces a fire source. In order to assure that the performance of the work does not initiate a fire, the controls Clinton Power Station procedure 1893.02, "Fire Protection - Control of Ignition Sources," will be established. This assures that the increased potential of a fire is compensated for through the use of a fire watch. The Main Steam (MS) line drain system is a non-safety related system and is not required to effect or support safe shutdown of the plant or to perform in the operation of reactor safety features. No equipment important to safety is located in the area of influence around the blank flanges. This activity does not result in system configurations that challenge the operation of any safety system or impair any systems reliability when relied upon for system protection. Therefore, this activity will not prevent actions described or assumed in the accident analysis for mitigating the effect of any transient, nor will any fission product barriers be challenged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure mode for the welded pipe cap is the same as the failure mode for the blank flange. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The MS drain system is not addressed in the Technical Specifications. Replacing the blank flange with a welded pipe cap does not adversely impact the ability to perform dirt and debris inspections of the steam line; nor does it have any impact upon the acceptance values or design limitations of any system, structure, or component. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE BLANK FLANGE WITH WELDED PIPE CAP

Activity Evaluated: ECN 31770 and USAR Change 8-380 Log Number: 99-163

Engineering Change Notice (ECN) 31770 replaces the flanged cleanout connections upstream of valve 1B21SPDV3 with a welded pipe cap. This change includes revising Updated Safety Analysis Report (USAR) Figure 3.6-1 Sheet 7 and Figure 10.3-1 Sheet 3 to reflect a pipe cap instead of a blank flange. The location of the blank flange is between the turbine control valve and the high pressure turbine inlet. The failure mode associated with a welded pipe cap is limited to failure of the cap to function as a pressure boundary. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. The listed pipe cap material, pressure rating, and schedule meets or exceeds the relevant system piping design requirements. The cap will be welded and inspected per existing site procedures and appropriate post installation testing will be performed to assure that the welded cap will function as a pressure boundary. In addition, the change from a blank flange to a pipe cap will not adversely affect the pipe line support. The performance of welding introduces a fire source. In order to assure that the performance of the work does not initiate a fire, the controls Clinton Power Station procedure 1893.02, "Fire Protection - Control of Ignition Sources," will be established. This assures that the increased potential of a fire is compensated for through the use of a fire watch. The Main Steam (MS) line drain system is a non-safety related system and is not required to effect or support safe shutdown of the plant or to perform in the operation of reactor safety features. No equipment important to safety is located in the area of influence around the blank flanges. This activity does not result in system configurations that challenge the operation of any safety system or impair any systems reliability when relied upon for system protection. Therefore, this activity will not prevent actions described or assumed in the accident analysis for mitigating the effect of any transient, nor will any fission product barriers be challenged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure mode for the welded pipe cap is the same as the failure mode for the blank flange. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The MS drain system is not addressed in the Technical Specifications. Replacing the blank flange with a welded pipe cap does not adversely impact the ability to perform dirt and debris inspections of the steam line; nor does it have any impact upon the acceptance values or design limitations of any system, structure, or component. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE BLANK FLANGE WITH WELDED PIPE CAP

Activity Evaluated: ECN 31771 and USAR Change 8-381 Log Number: 99-164

Engineering Change Notice (ECN) 31771 replaces the flanged cleanout connections upstream of valve 1B21SPDV4 with a welded pipe cap. This change includes revising Updated Safety Analysis Report (USAR) Figure 3.6-1 Sheet 7 and Figure 10.3-1 Sheet 3 to reflect a pipe cap instead of a blank flange. The location of the blank flange is between the turbine control valve and the high pressure turbine inlet. The failure mode associated with a welded pipe cap is limited to failure of the cap to function as a pressure boundary. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange.

The listed pipe cap material, pressure rating, and schedule meets or exceeds the relevant system piping design requirements. The cap will be welded and inspected per existing site procedures and appropriate post installation testing will be performed to assure that the welded cap will function as a pressure boundary. In addition, the change from a blank flange to a pipe cap will not adversely affect the pipe line support. The performance of welding introduces a fire source. In order to assure that the performance of the work does not initiate a fire, the controls Clinton Power Station procedure 1893.02, "Fire Protection - Control of Ignition Sources," will be established. This assures that the increased potential of a fire is compensated for through the use of a fire watch. The Main Steam (MS) line drain system is a non-safety related system and is not required to effect or support safe shutdown of the plant or to perform in the operation of reactor safety features. No equipment important to safety is located in the area of influence around the blank flanges. This activity does not result in system configurations that challenge the operation of any safety system or impair any systems reliability when relied upon for system protection. Therefore, this activity will not prevent actions described or assumed in the accident analysis for mitigating the effect of any transient, nor will any fission product barriers be challenged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure mode for the welded pipe cap is the same as the failure mode for the blank flange. The welded joint, once installed, inspected, and tested, represents a joint much less likely to fail than the gasket blank flange. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The MS drain system is not addressed in the Technical Specifications. Replacing the blank flange with a welded pipe cap does not adversely impact the ability to perform dirt and debris inspections of the steam line; nor does it have any impact upon the acceptance values or design limitations of any system, structure, or component. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE TDRFP TRIP SETPOINTS ON LOW EXHAUST CONDENSER VACUUM

Activity Evaluated: Mod FW-034, ECN 29166, USAR Change 8-328 Log Number: 99-165

Modification FW-034 revises the Turbine-Driven Reactor Feed Pump (TDRFP) A and B exhaust condenser low vacuum trip setpoints from 19" Hg Vac to 18.5" Hg Vac with a tolerance of +/-0.5"Hg. This modification also revises the alarm setpoints for the same parameters from 22.5" Hg Vac to 21.4" Hg Vac with a tolerance of +/-1.0" Hg. In addition, this modification adds an Isolation Valve and Tee to the pressure sensing lines to allow in place calibration. This change does not affect the Main Turbine low condenser vacuum trip pressure settings or switches. Failure modes for the pressure switches and sense lines have not changed. Because the change lowers the vacuum setting at which the turbine trips, it can be viewed to maintain feed flow longer than before. Lowering the trip setpoint potentially causes the feedpumps to remain online for about one second after the turbine trip. Delaying the feed pump trip by one second is conservative due to the continued coolant injection and turbine steam flow resulting in reduced vessel pressure. This change in trip setpoint maintains the protection required for the TDRFP on high backpressure. The added loads to the piping section have been evaluated to be well within the allowable for 1/2" carbon steel piping and results in inconsequential changes in vibration. These changes have no affect on trip or alarm system performance other than trip and alarm at a lower value and increasing reliability of maintaining an acceptable setpoint because of the increased tolerance and in-place calibration. As indicated in Updated Safety Analysis Report (USAR) section 10.4.7.3, the feedwater system is not required to support the safe shutdown of the reactor. In addition, there are no safety systems dependent on the components affected by this activity. This activity does not degrade the standards of construction and maintenance. Any loss of feedpumps that would occur due to this activity is enveloped by event that do not result in fuel failure. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Loss of all feedwater pumps has been analyzed and envelopes any failures of the switch actuation points. This change in trip setpoints maintains the protection required for the TDRFP on high backpressure. Failure modes for the pressure switches have not changed. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications and its Bases do not address any controls over the TDRFP trip settings for high exhaust pressure. Technical Specification 2.1.2 and its associated Bases identifies the reactor coolant system (RCS) pressure safety limit as 1375 psig which is 110% of design pressure. As can be observed from USAR Section 15.2.5.4 the loss of condenser vacuum event does not challenge this limit. This activity, would actually increase the margin to the reactor pressure safety limit, for this transient, due to the additional subcooling and steam relief to the feedpump turbine for the additional one second to the TDRF trip. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

TRANSITION OF PROCUREMENT ENGINEERING AND PURCHASING INTO MATERIAL MANAGEMENT

Activity Evaluated: USAR Change 8-384 Log Number: 99-168

Updated Safety Analysis Report (USAR) Change 8-384 transfers the Procurement Engineering and Purchasing Groups into the Work Management Department, Material Management Section. Since this is considered a "reassignment of responsibility for an activity from one group to another" as identified in Quality Assurance Procedure 102.02, the change is not considered a reduction in commitment to a Quality Assurance Program Description. This is an administrative organizational change which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

RADIATION PROTECTION AND CHEMISTRY ORGANIZATIONAL CHANGES

Activity Evaluated: USAR Change 8-386 Log Number: 99-169

Updated Safety Analysis Report (USAR) Change 8-386 modifies position titles, position description, and the reporting structure. The title of Director Plant Radiation Protection and Chemistry changes to the Director - Plant Radiation Protection and the functions of the Director Plant Radiation Protection and Chemistry will be preserved by the Director - Plant Radiation Protection with the exception to the Chemistry functions. This activity changes the Supervisor - Chemistry title to the Director - Chemistry. Both these Director positions report directly to the Manager - Clinton Power Station. In addition, this activity eliminates the Supervisor - Radiological Operations position and changes the reporting structure of the Alara Coordinator and the Radiation Protection Shift Supervisors to the Radiation Protection Manager. The responsibilities of the Supervisor - Radiological Operations will be preserved by the Radiation Protection Manager. This is an administrative organizational change which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MOV 1E51-F095 CLOSING CIRCUIT MODIFICATION

Activity Evaluated: ECN 31793; USAR Change 8-388 Log Number: 99-171

Engineering Change Notice (ECN) 31793 involves a field wiring change for valve 1E51-F095 between the 125 VDC Motor Control Center (MCC) 1A compartment 5B and valve limit switch compartment. This activity places the torque switch in series with the limit switch contacts in the closing circuitry and allows the limit switch contacts to deenergize the motor. The torque switch remains in the control circuit to provide protection to the valve during testing. In addition, this activity revises Updated Safety Analysis Report (USAR) Section 7.4.1.1.3.6, "Actuated Devices," to provide an exception for double disk gate valves limit switch contacts to turn off the motor during closing. This is a wiring change only and no change to the Rod Control system will occur. Therefore, the failure modes for the initiating event for a Control Rod Drop Accident are not affected. No new components are added, so the design, material and construction standards are not affected. This activity only affects the limit switch seating of the valve and does not change the response characteristics of valve 1E51-F095. All systems still operate as designed and within the design limits. In addition, this activity does not adversely affect any fission product barriers, acceptance criteria, or actions described or assumed in the accident analysis for mitigating the effects of any accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This ECN does not introduce a new component, only a new function. There are two possible failures associated with the contact, fail to close and fail to open. If the contact failed to close, the valve would remain open under testing or surveillance conditions. The valve will still close in an accident condition and the motor would turn off by the set of limit switch contacts on separate rotor. If the contact would fail to open, the limit switch contact 9/9C or the torque switch would open the circuit and turn off the motor. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical specification 3.5.3 addresses the Reactor Core Isolation Cooling system. This activity does not change the safety function of valve 1E51-F095; thus, there is no change to the Technical Specifications, the safety limits, or the limiting conditions for operations. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

JUMPER TW INTERLOCK TO SUPPORT BUS 0AP28E OUTAGE

Activity Evaluated: Temporary Modification 99-063 Log Number: 99-172

Temporary Modification 99-063 bypasses the automatic trip of the Filtered Water (TW) Supply Pump B, 0TW01PB, on low Filtered Water Storage Tank level. The low level trip will be bypassed only during transfers of the power supply for control relay and will be restored as soon as the transfer is complete. The TW system is a support system for the Circulating Water (CW) system, in that it provides cooling water to the pump seals. Loss of Condenser Vacuum is a transient which would result as a loss of CW pumps. The failure modes associated with a jumper are shorts or opens, which (in this activity) results in premature pump stoppage or failure of the pump to stop in response to tank level signals. The wiring used for the jumper meet or exceed the current and voltage rating for the application, which minimizes the potential for the jumper to fail open. This type of jumper was selected since it minimizes the potential for inadvertent termination point disconnection. Using an insulated grabber minimizes the chance of shorts. The potential impact upon the Fire Hazards Analysis is limited by controlling the materials selected, location the materials are used, and by standard maintenance practice. The addition of standard electrical components and associated wiring to the internals of a control or instrumentation panel will not change the fire loading of the panel and therefore will not affect the fire rating of the area. The TW system is a non-safety related system and is not required to effect or support safe shutdown of the plant or to perform in the operation of reactor safety features. Since panel 0AP28E is non-safety, non-seismic, and not environmentally qualified, the installation of the jumper has no adverse impact upon electrical separation or equipment qualification. This activity does not increase challenges to safety systems by imposing more severe testing requirements, no new transients are imposed, and no system's redundancy or independence is affected. This activity does not prevent actions described or assumed in the accident analysis for mitigating the effect of any transient, no fission product barriers are challenged, and there are no radiological release limits associated with any accidents or transients. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The failure modes associated with a jumper are shorts or opens. These failures have been previously analyzed. No failures previously considered incredible are made credible, and no new failure mode types are created. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not address the TW pumps and relays addressed in this temporary modification. As stated in

USAR Section 9.2.3.3, the TW system is a non-safety related system and is not required to support safe shutdown of the plant. As a result, installing the jumper has no impact on the acceptance values or design limitations for any structure, system, or component, and has no impact on any safety limits limiting safety system settings or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

SPARING A PREVIOUSLY ABANDONED ANNUNCIATOR INPUT

Activity Evaluated: ECN 31811 Log Number: 99-173

Engineering Change Notice (ECN) 31811 spares the output of an unused optical isolator pair which feeds High Pressure Core Spray (HPCS) System Out of Service alarm. This ECN also revises several electrical drawings to reflect this change. HPCS is primarily an accident mitigation system, but can create an increase in reactor coolant inventory accident. The failure modes associated with this activity are associated with shorts and installation error. To minimize the potential for installation error, the work is performed using a maintenance work order, and double verification of wiring connections is performed per Clinton Power Station (CPS) procedure 8801.16, "Wire Removal/Jumper Installation." The potential for shorting to occur from the disconnected wire is eliminated by taping the wire per CPS procedure 8492.01, "Cable Terminations." The annunciation system is electrically independent of the plant safety systems to prevent the possibility of adverse effects. The isolation of this particular annunciator input is accomplished with an optical isolator, which eliminates the potential for a short or electrical transient from being conducted back to any safety class component. Thus, the likelihood of an inadvertent HPCS pump start is not affected. Since the work is limited to within a Main Control Room Panel, no potential impact upon the system mechanical components can occur that could cause water spray, pipe whip, or missiles. This activity does not affect electrical isolation, seismic, or separation criteria. In addition, this activity does not adversely affect the design, functions, or method of performing the functions of the HPCS or any other system important to safety. This activity does not adversely affect the radiological release limits associated with any accidents or transients and no fission product barriers are challenged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). No new failure paths are created, no electrical or mechanical separation requirements are affected, and no system will be operated outside of its design limits. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. HPCS is addressed in several Technical Specifications, however, this activity is limited to sparing the output of a spare optical isolator which has caused spurious annunciations. The Technical Specifications do not address the HPCS annunciator system. This activity restores reliable status indication of HPCS while maintaining system design requirements and not affecting HPCS operation. This activity has no impact upon acceptance values or design limitations for any structure, system, or component, and has no impact on safety limits limiting safety system setting or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

UPDATE OF USAR APPENDIX E, SECTION 3.3, 3.6, 3.7, 3.8, AND 3.9

Activity Evaluated: USAR Change 8-394 Log Number: 99-174

Updated Safety Analysis Report (USAR) Change 8-394 corrects deficiencies/inconsistencies in USAR Appendix E. This activity updates Appendix E Sections 3.3, 3.6, 3.7, 3.8, and 3.9 to provide missing information, and to reflect the physical configuration and fire protection features, including barrier fire rating, fire detection and protection features and locations. This activity reflects text descriptions in Appendix E consistent with the physical configuration of the plant as described on the Appendix E Fire Protection Figures, and therefore, no fire hazards are added by this activity. This activity does not involve a change to the design, function, operation, or test of any equipment or systems. There is no affect on the environmental, seismic, or separation criteria of the affected systems. No new impairment to safe shutdown system reliability, degradation of equipment protective features or system performance, or reduction of system redundancy or independence due to this activity. No fire hazards are added by this activity, and therefore, there is no adverse effect due to the severity of a fire, or no adverse effects due to the potential of a fire to spread. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity updates physical and fire protection features in sections of USAR Appendix E, and thereby, assures that the appropriate fire protection features are documented as being available to support reactor shutdown during a fire in accordance with the requirements of Appendix R. This activity does not add any fire hazards to affected areas, or compromise the capability to perform a safe shutdown in the event of a fire in the affected areas. In addition, this activity does not reflect any change to the design, function, operation, or test of any equipment or systems. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The Technical Specifications do not address the Fire Protection Program. All acceptance values and design limitations involving the Fire Protection System were previously documented in USAR Section 9.5.1, Appendix E, and Appendix F. There is no change to any control of systems, components, or functions as documented in the Technical Specifications. The changes associated with USAR Change 8-394 are all within regulatory design limitations. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

NORMAL POSITION OF VALVE 0WE030A, 0WE030B, 0WE044B, 0WE040, 1WE036, AND 1WE037

Activity Evaluated: ECN 31792, USAR Change 8-376 Log Number: 99-175

Engineering Change Notice (ECN) 31792 changes the position of valves 0WE030A, 0WE030B, 0WE044B, 0WE040, 1WE036, and 1WE037 from normally open to normally closed. The design basis accident most closely associated with this activity is the Liquid Radwaste Tank Failure described in Updated Safety Analysis Report (USAR) Section 15.7.3. This activity does not involve any changes to the Radwaste Tanks, and therefore, does not increase the probability of a tank failure. Depicting valves 0WE030A, 0WE030B, 0WE044B, 0WE040, and 1WE036 as normally closed does not change the operation of these valves or create any new interactions with other systems. Depicting these valves closed will result in the design documents being consistent with operating procedures. This will reduce conflicting information that could possibly result in operator error. The operation of valve 1WE037 changes from normally open to locked-closed. This will require operator action to unlock and manually open the valve prior to discharge of excess water inventory to the Service Water Discharge. This is conservative because it provides an additional barrier to an inadvertent discharge and is consistent with current operating philosophy. The accident analysis assumes failure of the concentrate waste tank because its average radioactivity inventory is the highest for the liquid radwaste system. The accident analysis in the USAR remains bounding because this activity does not increase radioactivity levels or tank inventory. The valves associated with this activity are not required to be manipulated to mitigate the consequences of an accident. As such, this activity does not change, degrade, or prevent actions for mitigating the affect of the Radwaste Tank Failure. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not create any new flow paths, introduce any new circuits, create or delete any automatic functions, or create any new interactions with any other systems. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. No Technical Specifications were identified that could affected by changing the normal state of valves 0WE030A, 0WE030B, 0WE044B, 0WE040, 1WE036, and 1WE037. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONDENSATE FILTRATION FOR POLISHERS D, E, AND F

Activity Evaluated: Modification CP-020 Supplement 2
and USAR Changes 8-399, 8-400, 8-401

Log Number: 99-176

Modification CP-020 Supplement 2 extends the application of Condensate Filtration (CF), reviewed and approved for use in Safety Evaluation 94-0075 Revision 1 (CP-020 Supplement 1), to Condensate Cleanup System (CCS) deep bed demineralizers D, E, and F. The modification adds a prefilter and three air accumulators, a backwash header, various piping, valving and hangers, and modifies Heating, Ventilation, and Air Conditioning (HVAC) drains, inlet lines, grating and structural steel to accommodate the new equipment, to each cubicle. The objective of this activity is to improve the ability to reduce exposure and volume by adding filters to demineralizers D, E, and F. Any failure of the CF subsystem is bounded by a feedwater line break outside containment, liquid radwaste tank failure, or loss of feedwater flow. The design, material and construction standards of the new equipment are identical to those used for installation of filters on condensate polishers A, B, and C. This modification does not affect any of the environmental, seismic, or separation criteria of any system important to safety. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). As stated above, any failure of the CF subsystem is bounded by one or more accident analyses in USAR Chapters 6 and 15. Since there is no equipment important to safety or needed to mitigate accidents located in the area of the Condensate Polisher rooms, there can be no direct or indirect effect of a failure in the condensate filtration system on equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Neither the Technical Specifications, nor their associated Bases discuss condensate cleaning or filtration. Operating design limits for the system have not been affected by this change. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification. Note: Also see Log Number 2000-113 in this summary.

INSTRUMENT AIR VALVES 1IA415 AND 1IA416

Activity Evaluated: ECN 31794, USAR Change 8-377 Log Number: 99-177

Engineering Change Notice (ECN) 31794 changes the position of Instrument Air (IA) Valves 1IA415 and 1IA416 from normally open to normally closed. The design basis accident most closely associated with this activity is the Loss of Instrument Air accident described in Updated Safety Analysis Report (USAR) Section 15.2.10. This event can occur as the result of a major line break in the system or as a result of mechanical or electrical failure of the normal IA supply and the backup Service Air (SA) supply air source. This change is a document change only; no physical changes are being made except tagging of valve 1IA415. The valves meet the design, material, and construction standards applicable to the IA system. Since valve 1IA415 and 1IA416 are located on capped connections of the IA system, maintaining the valves closed versus open does not change IA system response characteristics, cause IA system operation outside of its design limits, cause operational transients in the system or cause adverse system interaction with other systems. Maintaining the valves closed is expected to further reduce the possibility of leakage through the capped connection. A Loss of Instrument Air accident does not affect any of the fission product barriers because the consequences of this event do not result in any temperature or pressure transients in excess of the criteria for which the fuel, pressure vessel, or containment is designed. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The only credible failure mode related to this activity is failure of both a valve and the capped connection which would lead to a Loss of Instrument Air accident. This failure mode is bounded by the analysis in USAR Section 15.2.10. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specifications 3.5.1.3, 3.6.1.3, and 3.6.5.3 could all be affected by a change to the Instrument Air system. The above Technical Specifications address operational limits for systems, structures, and components (SSCs). Since the changes being made to design documents do not affect the operation of the SSCs discussed in those Technical Specification sections, there is no impact on any operational limits. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

RETUBING OF THE TURBINE BUILDING CLOSED COOLING WATER HEAT EXCHANGERS 1WT01AA AND 1WT01AB WITH TITANIUM

Activity Evaluated: ECN 31799 and 31800, USAR Change 9-008 Log Number: 99-179 R/1

Engineering Change Notices (ECNs) 31799 and 31800 modify the Turbine Building Closed Cooling Water System (WT) Heat Exchangers 1WT01AA and 1WT01AB. The first part of this activity is to install new titanium tubes in place of the Admiralty tubing. The second part of this activity is to install zinc anodes in the Heat Exchangers inlet/outlet and return waterbox covers. The new titanium tubes meet all of the necessary design requirements and are not susceptible to pitting, corrosion, nor erosion. The calculated heat transfer is well above the maximum required. The function of the WT heat exchangers is to transmit the Turbine Building heat loads to the Plant Service Water (WS) and to the Clinton Lake. There are no design basis accidents that could be impacted by the loss of one or both of the heat exchangers. The heat exchangers are only required for the power generation mode. All of the equipment modified per these two ECNs are non-safety related and have no impact on any safety related equipment or components and continue to function as before. Neither the WS, nor the WT systems are credited in mitigation of accidents, and as a result, do not impact any fission product barriers. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The loss of one or both WT heat exchangers has no affect on the plant's ability to safely shutdown following a design basis accident. The WT heat exchanger will continue to cool the turbine building loads. There are no direct or indirect malfunctions that are not already bounded. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. There are no Technical Specifications that control the WT heat exchangers, nor the components that are cooled by the heat exchangers. These ECNs do not impose or change any plant operation set point or operating limit. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE USAR SECTION 8.1.5.2.3.D

Activity Evaluated: USAR Change 8-406 Log Number: 99-180

Updated Safety Analysis Report (USAR) Change 8-406 revises the Nuclear System Protection System (NSPS) description in USAR Section 8.1.5.2.3.d. The USAR change consists of replacing component level details (specific voltage and frequency) with the overall system requirements, which is to maintain adequate voltage and frequency at the end device regardless of the input source. This USAR change maintains the NSPS system requirements as designed and does not affect overall system performance such that it changes system response characteristics. The divisional inverters design to provide the required capacity, capability, redundancy, and reliability has not been impacted by this USAR change. This activity does not impair any of the environmental, seismic, or separation criteria. The uninterruptible power will be maintained to the Emergency Core Cooling System components, which are assumed to operate during a design basis accident. This change does not alter, add, or remove any plant components, and neither the NSPS system nor the inverter operation is changed. In addition, the fuel, Reactor Coolant System, and containment design limits are maintained. This activity does not change, degrade, or prevent actions described or assumed in the accident analysis for mitigating the effect of the accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since there is no change to the NSPS system or its components, the system will continue to perform its safety function as designed. The failure modes of the NSPS power supply system or the supported systems remain unchanged by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The requirements for the preferred source of power for the uninterruptible NSPS buses are described in Technical Specification Section B3.8.7, B3.8.8, B3.8.9, and B3.8.10. This activity has not impacted any Technical Specification requirement or its associated Bases. This activity does not impose or change any plant operation set point or operating limit. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REROUTING OF POWER SUPPLY FOR DG ROOM 1A VENT DAMPERS INSTRUMENTATION

Activity Evaluated: ECN 31776; USAR Change 8-408 Log Number: 99-181

Engineering Change Notice (ECN) 31776 de-terminates and abandons a power supply cable and provides a new cable from a new power source to the Diesel Generator (DG) 1A ventilation instrumentation located in panel 1PL54JA. This cable provides power to various ventilation instruments, including temperature controllor, 1TICVD001, which controls the modulation of the DG Room 1A Ventilation Dampers. DG 1A ventilation instrumentation is required to support operation of the DG, which provides power to vital loads in the event of Loss of Auxiliary Power Transformer or Loss of All Grid Connections. In order to ensure the availability of power to these ventilation instruments in the event of a fire, the replacement cable and original cable are routed outside areas that credit Method R for safe shutdown. The replacement power supply cable is identical to the original power supply in voltage, source capacity, divisional separation and wiring. The components and material, such as cable, circuit breaker, and connectors are approved for use in safety related application and are identical to those originally installed. Installation and testing will be performed in accordance with approved procedures. The overall performance of the DG 1A ventilation system will not be affected in any way because the power supply to the instrumentation will satisfy the intent of the original design in all respects. This activity does not change, degrade, or prevent actions described or assumed in the accident analysis for mitigating the effects of an accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The new power source is identical to the original in all respects. In addition, the voltage regulation and reliability of the new power source is identical to the original power source. The effect of the failure of the new power supply is in no way different from the effect of the failure of the original power supply. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.8.2 discusses Electrical Power Systems. This activity does not affect, impact or change any parameters, limiting operating conditions, surveillances, or bases upon which the Technical Specification 3.8.2 is based. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF GEZIP SKID

Activity Evaluated: Modification FW-037 Supplement 1 Log Number: 99-182
USAR Change 8-414

Modification FW-037 Supplement 1 installs the General Electric Zinc Injection Passivation (GEZIP) system. The GEZIP system is a passive system designed to inject zinc ions into the reactor water via the Feedwater (FW) system. This will reduce the corrosion of internal surfaces of the stainless steel piping in the primary system, which in return will reduce the deposition of radioactive Cobalt-60 on these surfaces and lower radiation levels from primary system piping. This activity will not have an impact on any of the initiating events for the design basis accidents associated with the Feedwater system. This modification does not affect the ability of the Feedwater system to meet its design requirements. The addition of zinc will not result in degradation of the feedwater or reactor coolant systems or components. This modification only affects piping outside of containment upstream of the outermost isolation valve. A pipe break in the new 1-1/2 inch zinc injection line would have radiological consequences no worse than the large postulated FW line break outside containment. In addition, the affected portions of the condensate booster and feedwater systems are not safety related or seismic and are not required for safe shutdown of the reactor. There is no equipment important to safety located in the vicinity where the GEZIP skid will be installed. Also, the system is designed and installed in accordance with industry standards and plant specifications and procedures. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The injection of dissolved zinc into the feedwater results in an increase in zinc in the reactor and all interfacing systems. The increase in zinc gradually alters the oxide film formed on the interior of the piping systems, resulting in gradual film thinning and concurrent dose rate reductions as exposure to zinc continues. Other chemistry effects may include an effect on conductivity and pH. These effects are minor, and in some cases, beneficial and will not invalidate the USAR and Operational Requirements Manual limits. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications 3.4, Reactor Coolant System, and 3.7, Plant Systems, contain no requirements for feedwater or reactor coolant water chemistry. In addition, the operation of the FW and Condensate Booster systems is not addressed in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

APRM "B" MONITORING

Activity Evaluated: TMOD 99-066 Log Number: 99-184

This activity installs monitoring instrumentation in the Main Control Room (MCR) to monitor key parameters associated with Average Power Range Monitors (APRM) B since it intermittently indicates lower than other channels. The APRM signal is an average value of the Local Power Range Monitor (LPRM) signals. Each APRM has a reference signal based on Reactor Recirculation (RR) pump flow and the APRM logic compares the two signals. If the difference between the signal circuit and the reference signal exceeds a preset value, an annunciator relay actuates. Faulty electrical connections or damaged wiring could cause voltage fluctuations to an APRM or LPRM causing a deviation in the APRM signal circuit. The monitoring equipment has a high impedance to minimize adverse effects on the logic circuitry and is powered from a power supply different than the APRM circuit power supply. No APRM trip signals are affected by this activity, there is one division (B) affected out of four.

The changes are being performed on one of four divisions of APRM and all APRM trips and functions remain unaffected. Evaluations have shown that this activity will not increase the probability or consequences of an accident previously evaluated in the USAR or increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. Also, this activity will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than evaluated previously in the USAR. Lastly, for the same reasons, this activity will not reduce the margin of safety as defined in the basis for any technical specification.

ELMINATING NSPI DEPARTMENT AND TRANSFERRING NSPI RESPONSIBILITIES TO OTHER DEPARTMENTS

Activity Evaluated: USAR Change 8-412 Log Number: 99-185

Updated Safety Analysis Report (USAR) Change 8-412 eliminates the Nuclear Safety and Performance Improvement Department. As a result, the Licensing Group will report to the Manager – Clinton Power Station and the Independent Safety Engineering Group (ISEG) and the Employee Concerns Program will report within the chain of command of the Quality Assurance Department. These changes are organizational only and will not change the duties or responsibilities of Licensing, ISEG and Employee Concerns Program. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. The basis of the site organizational requirements established in Technical Specification 5.2.1 are not affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FIRE PROTECTION PIPE REPLACEMENT

Activity Evaluated: Modification FP-106, ECN 31801, and USAR Change 8-413 Log Number: 99-186

Modification FP-106 involves replacing approximately 120 feet of 8" carbon steel piping with 10" polyethylene pipe and 6" carbon steel pipe. This modification replaces a leaking section of piping with a modified design that minimizes construction impact and galvanic corrosion concerns. The polyethylene pipe material is used for the outdoor portion of the piping being replaced. Inside the Turbine Building carbon steel piping of the same specification as the piping being replaced is used. Calculations have been performed to ensure that the modified system will meet its hydraulic design requirements and to ensure the structural adequacy of the change. This activity does not adversely impact the ability of the Fire Protection system to meet its design requirements. This activity does not change the Fire Protection system function, capability, operation, or operator actions. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The only credible failure mode for this modification is a leak or break in the Fire Protection system piping. This would be no different than a break in the existing piping and does not introduce a new or unanalyzed condition. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not discuss the fire protection program. Thus, the fire protection system does not have any safety limits or limiting conditions for operation stated in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CLB/USAR DISCREPANCY RESOLUTION FOR PROCESS SAMPLING SYSTEM

Activity Evaluated: USAR Change 8-410 Log Number: 99-187

Updated Safety Analysis Report (USAR) Change 8-410 revises the description of the Process Sampling (PS) system within subsection 9.3.2 and Table 9.3-3. This activity includes changing the temperature control of the reactor water sample panel from 77°F ±1°F to 25°C ±1°C; deleting the on-line instrumentation ranges, alarms and computer points; changing the listed instrumentation to agree with that shown on USAR Figure 9.3-4; indicating several on-line instrumentation are no longer actively in use; and changing the potential activity levels of several samples. The PS system is not mentioned in any assumption made for the initiation of a design basis accident. In addition, the PS system is not required to ensure safe shutdown of the plant. This activity does not degrade the ability of the system to sample and analyze the process systems or to identify out-of-specification chemistry. Thus, the changes to the PS system will not impact the systems sampled or monitored by the PS system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. These changes do not affect the capability of the PS system to meet its objective of monitoring any safety-related systems. There are no systems, structures, or components being introduced into the plant, or being changed by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications discuss requirements for the Post Accident Sampling System, but a review did not locate any mention of the Process Monitoring System. This change does not make any changes to the operation of the Post Accident Sample System. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REASSIGNMENT OF MANAGER-CPS RESPONSIBILITY FOR APPROVAL OF ADMINISTRATIVE PROCEDURES

Activity Evaluated: USAR Change 9-001 Log Number: 99-189

Updated Safety Analysis Report (USAR) Change 9-001 reassigns the Manager – Clinton Power Station (CPS) responsibility for approval of administrative procedures to the applicable Department Head/Designee. This activity does not alter the requirement for Facility Review Group (FRG) review of administrative procedures or the requirement that the review and approval be obtained prior to implementation. The assignment of this administrative function is not by itself a barrier preventing the occurrence of any evaluated accident. The act of approving administrative procedures and execution of this function does not contain a credible failure mode that could initiate an evaluated accident. This change does not alter the assumed performance capability or standards for performance of equipment or standards for activities establishing station operation and maintenance of structures, systems, or components (SSCs) considered to be equipment important to safety. This activity does not alter the assumed release mechanisms, pathways, rates, duration, source term, or protective functions, nor does it degrade the effectiveness of SSCs used to mitigate assumed and evaluated accidents. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no SSCs being introduced into the plant, or being changed by this activity. This activity does not introduce the potential for a new operator induced error. The Manager – CPS and Department

Head/Designee responsibilities are unaffected by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 5.0 discusses the Manager – CPS's position, responsibilities and qualifications; these do not result in the establishment of a margin of safety that functions to preserve the integrity of a fission product barrier. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

TURBINE MISSILES

Activity Evaluated: USAR Change 9-010 Log Number: 99-191

Updated Safety Analysis Report (USAR) Change 9-010 revises USAR Section 3.5.1.3, "Turbine Missiles," to differentiate the design stage turbine missile damage probability from the current probability values due to variations in material condition of the equipment and testing intervals that impact how this probability is calculated. Chapter 15 discusses events related to turbine and generator trips. The probability of missile damage due to turbine wheel missile has been evaluated to be sufficiently below the threshold to be considered a credible accident and consequently has not been included in the design basis accident analysis. As long as the calculated missile damage probability remains under the threshold value of 10–7 per year, any variation of the calculated turbine missile generation during plant operation is not to be considered a credible accident. The turbine and overspeed protection equipment are not safety related, nor do they support functions of safety related components. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The threshold for consideration for a new type of accident would be a turbine damage probability greater than 10-7 per year. This activity allows operation of the turbine with a different material condition than the original turbine and also allows a change in the test frequency within the bounding missile damage probability of 10-7 per year. This activity does not physically or functionally alter a function of any structure, system, or component important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity recognizes that the turbine missile generation probability may change over the years, due to turbine material condition and changes in the inspection/testing program of the turbine; the overall probability of the missile hitting a safety related SSC must remain below the threshold value of 10-7 per year. The threshold value of the missile damage is considered small enough to be excluded from the design basis accident analysis and the technical specification. There is no Technical Specification action or requirement for protection against missile damage due to the low threshold value of the acceptance criteria. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE TECHNICAL SPECIFICATION BASES CONTAINMENT/DRYWELL HYDROGEN MIXING SYSTEM FROM "NOMINAL" TO "NOT NOMINAL"

Activity Evaluated: Bases Change BE-99-034 Log Number: 99-192 R/1

TS Bases Change BE-99-034 prepared based on the results of instrument uncertainty calculation IP-O-0076 Rev. 1. This TS Bases change will change the "Nominal" statement in

B.3.6.3.3.2 to "Not Nominal". LCO 3.6.3.3 reads in part "Operation with at least one OPERABLE Containment/Drywell Hydrogen Mixing System provides the capability of controlling the hydrogen concentration in the drywell without exceeding the flammability limit." Tech Spec Bases Section 3.6.3.3 describes the function of the Hydrogen Mixing system compressors for removal of hydrogen from the drywell under post LOCA conditions. Calculation IP-O-0076 Rev. 1 is completed and determined that the instrument indication used to satisfy SR 3.6.3.3.2 are to be considered "Not Nominal" using the methodology in CPS CI-01.00, Rev. 1. As a result of the "Not Nominal" conclusion (the uncertainty associated with this channel is +/-0.22 psid. - TS limit is 800 scfm not including any margin for calculated instrument uncertainties), a Tech Spec Basis Change is required for section 3.6.3.3.2. CR 1-97-07-105 identified a condition which concerned the horsepower rating of the Hydrogen Mixing System compressors. The condition was associated with the degraded voltage issue (under the conditions of a lower voltage the horsepower rating for the hydrogen mixing compressors would exceed the rating used in the associated equipment qualification package). Calculation 01HG08 Rev. 0 was created which included performance curves for the compressor. The performance curves were not used in developing IP-O-0076 Rev. 0. The effects of the instrument channel uncertainties for the Hydrogen Mixing System compressor was evaluated. Revision 1 of IP-O-0076 identified the associated instrument indication limit, in SR 3.6.3.3.2 was "Not Nominal". This activity does not degrade any system's reliability or performance, nor reduce any system redundancy. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). There are no new performance requirements being proposed due to this activity. There are no SSCs being introduced into the plant, or being changed by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. No new design limitations are associated with this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CLB/USAR DISCREPANCY RESOLUTION FOR POST ACCIDENT SAMPLING SYSTEM

Activity Evaluated: USAR Change 9-012 Log Number: 99-193

Updated Safety Analysis Report (USAR) Change 9-012 item number 5 clarifies several statements in subsection 9.3.7.2 by changing the "will be" and "is" statements to "may be". Thus, the ability of the Post Accident Sampling System (PASS) is demonstrated, but the actual post accident activities performed are not constrained. The PASS system is not considered an initiator for any accident described in Chapter 6 and 15 of the USAR. This activity affects only the potential sampling and analyses performed after a postulated accident. Since only the post-accident operation is affected, there can be no effect upon the actual initiation of an accident. The actual sampling and analysis methods are not changed. The only safety-related portion of PASS is the containment penetration isolation ability, which is not affected by this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no structures, systems or components being introduced into the plant, or being changed by this activity. There is no change in the actual methods used to obtain and analyze specific samples. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications 5.5.2 and 5.5.3 require certain administrative program controls be established related to PASS. This activity does not compromise any of these administrative program controls. This change does not reduce the capability of PASS to perform any sampling, and does not limit the method of PASS operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PARALLEL THE SWITCHYARD 125 VDC DISTRIBUTION BUSSES BY INSTALLING JUMPERS

Activity Evaluated: Temporary Modifications 99-067 and 99-068 Log Number: 99-194

Temporary Modifications (TMs) 99-067 and 99-068 parallel the switchyard 125 VDC distribution busses to allow removing a battery charger from service for inspection and maintenance. This activity includes cross-tying the buss to provide control power while the charger is removed from service. One battery charger and its associated battery provide control power. Updated Safety Analysis Report (USAR) subsection 15.2.6 discusses Loss of AC power, including loss of all grid connections, which would be the design basis accident that could be impacted by this activity. The jumpers used are consistent with the design requirement. Failure of the available charger would result in an accelerated discharge of the available battery, since it is supplying both busses. However, the off-line switchyard battery would be at full capacity and this activity provides steps to reconnect it to supply control power if needed. Loss of control power could result in loss of an automatic trip of a breaker if required, but the relay system continues to function as designed. Since paralleling of the batteries will only occur during positive control and the period of time the batteries are paralleled will be minimized. The worst case scenario for this activity would be failure of a line-relaying breaker to trip due to a circuit fault. The component which could be adversely affected by failure of the breaker to trip would be the Reserve Auxiliary Transformer (RAT) or safety systems connected electrically to the RAT; however, each safety-related structure, system, or component is protected from electrical faults by two qualified isolation devices. To ensure control power and 345kV switchyard system reliability is maintained, the disconnected battery will be restored to prevent an adverse affect on the ability of the battery to provide back-up control power. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does this activity create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. There are no specific requirements related to the 125 VDC switchyard distribution buss addressed in the Technical Specifications. Technical Specification Bases 3.8.1, "AC Sources-Operating", discusses operating requirements for offsite power circuits including the breakers and circuit switches which feed the RAT and the Emergency Reserve Auxiliary Transformer. This activity will be performed while the unit is operating, so Technical Specifications for shutdown conditions will not be addressed. This activity maximizes DC control power reliability and will not adversely affect the switchyard control system, breakers, and circuit switchers from performing their design function. This activity will not adversely affect compliance with the licensing conditions as addressed in any Technical Specification or Operational Requirements Manual requirement because the original design and reliability is maintained. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACEMENT OF MOTOR OPERATED VALVE TYPE TEC 36007 BREAKER WITH TEC 36015 BREAKER

Activity Evaluated: ECN 31897 and USAR Change 9-014 Log Number: 99-195

Engineering Change Notice (ECN) 31897 replaces a TEC 36007 breaker for motor operated valve (MOV) 1E22-F012, Suppression Pool Minimum Flow Bypass Valve, with a TEC 36015 breaker. The seven amp frame size listed on the KEY diagram for breaker cubicle 4B will be changed to reflect the fifteen amp size of the new breaker. Updated Safety Analysis Report (USAR) Section 15.5.1, "Inadvertent HPCS/RCIC Pump Startup", is the principal accident that can be caused by the High Pressure Core Spray (HP) system. There are no credible failure modes associated with replacing the HP minimum flow valve breaker that can initiate an inadvertent HP system start. The control logic for the HP minimum flow valve is separate from the HP start circuitry, and the breaker control logic is unchanged by this activity. This activity does not change the motor or circuit loading, the breaker control circuitry, or adversely impact the electrical loading of the associated motor control center. This activity will be controlled via the maintenance process, which will minimize the potential for installation errors resulting in circuit damage or misoperation. This activity meets the design, material, and construction standards applicable to the HP system. The replacement breaker is a qualified replacement for 480 VAC applications and has been procured to the correct quality standards. This activity does not degrade the performance of the valve or decrease the HP system availability. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The replacement breaker is a qualified and electrically suitable replacement for the existing breaker. The breaker replacement will not affect overall system performance such that system response characteristics change, cause HP system operation outside of its design limits, cause operational transients in the HP system, or cause adverse system interaction with other systems. Per calculation 19-AN-20 revision 2, the new breaker will provide an electrical trip when needed to protect the circuitry, motor, and motor control center, and will not create spurious trips. Additionally, divisional separation is preserved because the replacement breaker wiring details are the same as the existing breaker and does not require additional wiring. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The operability of the HP minimum flow valve is required to support Hp system operability per Technical Specification section 3.5.1, "ECCS Operating". Replacing the breaker will not adversely impact HP minimum flow valve operability. Replacement of the seven amp breaker with a 15 amp breaker has no adverse affect upon the operation of the HP system, including functional capabilities, response times, and capacities. Consequently, no acceptance values, safety limits limiting safety system settings, or limiting conditions for operations associated with these components are adversely affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE USAR DESCRIPTION FOR DIESEL FUEL TESTING, FUEL POOL WATER PH, AND RADWASTE RESIN

Activity Evaluated: USAR Change 9-015 Log Number: 99-196 R/1

Updated Safety Analysis Report (USAR) Change 9-015 clarifies the diesel fuel oil analysis requirements, corrects conflicting pH limits for fuel pool chemistry (keeping the most conservative), and deletes reference to a vendor product name for radwaste resin type. Failure of the emergency diesel generators to start and carry the vital loads is one of the initiating events for station blackout. Condition Report (CR) 1-99-02-122 documents that there are three diesel fuel oil analyses required by the diesel generator vendor, that are not require by American Society for Testing Materials (ASTM) D975 (final boiling point, distillation recovery, and pour point). There are three analyses that are performed by different methods (corrosive sulfur, conrad carbon residue, and filtration cleanliness test). Regulatory Guide 1.137 requires the manufacturer tests only if they are more restrictive than the ASTM-D975 requirements. The diesel-generator manufacturer's testing requirements are not being performed, because the analysis requirements are similar to or redundant to analyses required by ASTM-D975. This activity does not degrade the reliability of the emergency diesel engines or equipment powered by vital AC power. The pH of the fuel pool water and the ion exchange resin in the liquid radwaste demineralizer do not have any effects on an initiating event for any accident previously evaluated in the USAR. The changes in fuel pool pH control and liquid radwaste demineralizer resin changeout procedures are in compliance with design specifications and vendor recommendations. The replacement resin has been shown to be equivalent to the vendor product referenced in the USAR, so the radionuclide concentration of the radwaste demineralizer effluent is unchanged. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The bounding fuel handling accident is the drop of a channeled spent fuel bundle onto unchanneled spent fuel in the spent fuel pool racks. The water quality control in the spent fuel pool maintains the strength and integrity of the fuel bundles and cladding. The liquid radwaste demineralizer resin changeout procedure change meets the vendor manual and EPRI Guidelines requirements, so the function of the liquid radwaste demineralizer is not degraded. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Diesel fuel oil sampling methods, testing requirements, and associated acceptance criteria are specified in Technical Specification Surveillance Requirement 3.8.3.3 and its bases, and in Technical Specification Section 5.5.9. There are no acceptance values, safety limits limiting safety system settings, or limiting conditions for operation associated with components affected by the activity, and no design limitations are adversely impacted. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

UPDATE TO THE NUCLEAR PROGRAM QUALITY ASSURANCE MANUAL

Activity Evaluated: Quality Assurance Manual Revision 27 Log Number: 99-197

Quality Assurance Manual (QAM) Revision 27 includes editorial changes, amends the QA audit/assessment functions, and incorporates recent Updated Safety Analysis Report (USAR) changes which correct position titles and responsibilities. The title of the document is also being changed from "Illinois Power Nuclear Program Quality Assurance Manual", to "Clinton Power Station Quality Assurance Manual". A majority of the changes are excluded from this evaluation, due to the fact that they were previously evaluated. The changes in question add amplifying information, expand existing responsibilities, and amend the QA audit/assessment functions. The changes to the QAM are primarily administrative changes and do not alter any system, structure, or component (SSC) either physically or functionally, nor do these changes result in a change in QA requirements or reclassification of any SSC. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This activity does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any SSC. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. This activity does not reduce the QA program requirements for SSCs used to preserve and demonstrate the integrity of the principle fission product barriers. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

USAR CHANGES INVOLVING THE TRANSITION OF CPS FROM IP TO AMERGEN AND ORGANIZATION CHANGES

Activity Evaluated: USAR Change 9-020 Log Number: 99-201

Updated Safety Analysis Report (USAR) Change 9-020 adjusts the total acreage to be owned by AmerGen, to allow IP to maintain three parcels of property; revises the alternate supplier of emergency diesel fuel oil from "another IP station" to "another supplier"; removes all references, descriptions, and responsibilities of the Manager – Recovery; and removes all references, descriptions, and responsibilities of the Assistant Plant Manager – Operations. The change in land ownership, change in supplier for diesel fuel oil and the organizational changes are unrelated to the initiation of the accidents evaluated in USAR Chapters 6 and 15. In accordance with the NRC License Transfer Application, the impact of the transfer of ownership and license of CPS from IP to AmerGen will not reduce the level of any NRC commitments. In addition, it also specifies that AmerGen will sufficiently control the area surrounding Clinton Power Station to preclude activities that would constitute a hazard to the station and its associated equipment important to safety. The requirements of Technical Specification 5.5.9, "Diesel Fuel Oil Testing Program", will continue to apply and function to preserve the integrity of the diesel fuel oil supply. The organizational change to delete the Manager – Recovery was deleted, due to CPS returning to Mode 2. The organizational change to delete the Assistant Plant Manager – Operations is a result of the Manager – CPS assuming these roles and responsibilities. In addition, these changes do not compromise the design, material, or construction standards to which the plant was originally built. This activity does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any SSC. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no structures, systems or components being introduced into the plant, or being changed by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This change will not negatively impact any margin of safety provided by the diversity, redundancy, and capacity of the electrical distribution system as described in the Bases for Technical Specification (TS) Limiting Condition for Operability (LCO) 3.8.1, "AC Sources – Operating". The reduction in the acreage owned by AmerGen does not impact the facility's control over the Exclusion Area and does not negatively impact the method of dose consequence determination; therefore, the assumptions used in deriving the Technical Specifications remain valid. The changes involving the deletion of the positions of Manager – Recovery and Assistant Plant Manager – Operations are not associated with any margin of safety as defined in the basis for any Technical Specification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MONITORING TURBINE STOP VALVE #4 CLOSURE CONTACT PERFORMANCE

Activity Evaluated: Temporary Modification 99-070 Log Number: 99-202

Temporary Modification (TM) 99-070 installs a recorder to monitor the performance of the limit switch contacts that indicate when the Main Turbine Stop Valve (TSV) #4 is open. Spring loaded and insulated grabber type test leads will be used to make the connection and will be secured with tie-wraps, tape, or other suitable material. While the turbine trip and some of the signals that initiate a turbine trip are included in the accident analysis, the TSV closure is not an initiating event for any of the accidents or transients. TSV closure is an accident mitigating function initiated by turbine trip to protect the turbine, initiate a reactor scram, and trip the recirculation pumps. The limit switch being monitored by this TM is not in the TSV trip circuit and cannot initiate a turbine trip. The function of the TSVs to provide a scram signal or Recirculation Pump Trip (RPT) is not impacted by this activity. The installation only affects one division, which assures electrical separation and redundancy is maintained. In addition, there are no credible seismic interactions associated with installing the recorder and test leads per this temporary modification. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The failure modes of the recorder and associated test leads are associated with shorts and opens. An open would cause the recorder to no longer indicate limit switch position, but has no consequences upon the operation of the TSV limit switch or Reactor Protection system. A short could artificially create a scram and RPT trip signal. Since this activity complies with the single failure, redundancy, and independence criteria, no potential for a failure in Division 4 affecting the other divisions exists. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation", delineate when the TSVs are required to be operable and Operational Requirements Manual (ORM) Tables 1, 3, 13, and 14 provide trip setpoint and response time limits. This activity only monitors the 125 VDC input from the actuation device and will not affect the valve position setpoint or response time of this signal during installation, removal, or while this TM is installed. As a result, using a recorder to monitor the performance of the TSV #4 open limit switch has not impact upon the acceptance values or design limitations for any system, structure, or component. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

HFA RELAY 1UAY-CC516D REPLACEMENT

Activity Evaluated: Temporary Modification 99-073 Log Number: 99-203

Temporary Modification (TM) 99-073 replaces the HFA Relay, 1UAY-CC516D, on-line to satisfy the requirements of PEMCCA808. This relay energizes when the Component Cooling Water (CC) Storage Tank, 0CC01TA, is at low-low-low level and provides a trip signal to CC Pumps A, B, and C. This relay also provides a permissive to start each of these pumps when level is above the reset point for low-low-low level. Loss of CC would ultimately result in a trip of a recirculation pump or loss of other auxiliary systems, which would result in a scram. The jumpers installed by this activity bypass a contact that is intended to be a start permissive for the pumps, so the pump cannot be started without CC Tank level above the low-low-low level setpoint. The lifted leads defeat a trip signal to the pump breaker strip coil. Defeating the low-low-low CC Storage Tank level CC pump trip does not impact the ability to maintain the CC Storage Tank level in the normal band. Level is manually maintained between 105" and 117" by monitoring a computer point. With the TM installed, the CC pumps are lost at low-low-low level due to a loss of pump suction or due to the operator de-energizing them in anticipation of loss of suction. The methods used by the Operator to control CC Storage Tank level are unaffected by this activity, and this activity does not increase the Operator's burden in controlling CC Storage Tank level or the probability of low tank level. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The credible failure modes associated with this activity involve the installation of the jumpers and lifted leads. The termination points are clearly labeled, minimizing the potential for incorrect connections. Ring lugs will be used which minimizes the potential for shorts due to a jumper coming loose or rubbing against another termination point. The scope of work represents a short term maintenance activity and will be conducted in accordance with approved plant procedures. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Bases Section B3.3.3.2, Remote Shutdown System, considers the CC system as a safety support system and states the instrumentation and controls required are listed in the Operational Requirements Manual. However, as described in USAR Section 9.2.2.3, failure of the CC system does not compromise any safety related system or component and does not prevent safe shutdown of the reactor. The scope of this TM does not challenge or impact the containment isolation boundary piping and valves portion of the CC system. As a result, defeating the contacts associated with the low-low-low level CC pump trips has no safety significance. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ABANDONMENT OF WATER MAKEUP SYSTEM CONDUCTIVITY RECORDER

Activity Evaluated: ECN 31764 Log Number: 99-204

Engineering Change Notice (ECN) 31764 abandons the Demineralized Water Makeup (WM) System conductivity recorder, 0CR-WM067. This involves removing power to the recorder by removing the input power fuse, without affecting the power supply to other instruments in the panel. The conductivity recorder is not part of the piping boundary for the WM system or any other water systems in the plant; thus, this activity could not initiate any pipe breaks. Adequate breaker/fuse coordination is provided to preclude an electrical fault in the WM system from propagating through the power distribution system and initiating a loss of AC power event. The removal of the input power fuse associated with the conductivity recorder removes power from the recorder, and isolates the recorder from the power distribution system. Therefore, a postulated electrical fault in the recorder is unlikely and will not adversely affect the power distribution system. The WM system is not required for safe shutdown of the plant or for mitigation of any accidents or transients. The WM system is not safety related, Seismic Category I, or Class IE. There is adequate separation and isolation provided for the recorder to preclude any malfunctions from adversely affecting any equipment important to safety. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). No new components or system interactions are being introduced by the abandonment of the WM conductivity recorder. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Demineralized Water Makeup System is not addressed in the

Technical Specifications. As such, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGES REGARDING ATWS-RPT TRIP LOGIC

Activity Evaluated: TS Bases Change BL-99-036; LDI 99-04 Log Number: 99-206

Technical Specification (TS) Bases Change BL-99-036 revises the description in TS Bases Section 3.3.4.2, Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT) Instrumentation, to agree with a Licensing Document Interpretation (LDI) 99-04 on the entry and applicable Actions of TS 3.3.4.2. LDI 99-04 specifies not using Action A and using an allowable out-of-service time (AOT) for Action B of 48 hours versus 72 hours. Further, the Bases change restores the system description in the licensing basis documents to reflect what was in effect prior to Amendment 64 and 95. The changes to use a more conservative allowable out-of-service time do not affect the performance capability of the system. The Nuclear Regulatory Commission's (NRC) ATWS rule states that the Alternate Rod Insertion system, Standby Liquid Control system, and the automatic ATWS RPT are required to mitigate the consequences of an ATWS event. The changes do not affect any Updated Safety Analysis Report (USAR) evaluated initiators and the changes have been determined to continue to meet the requirements for the ATWS rule. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since the ATWS event has been previously analyzed in the USAR and the evaluation presents a bounding analysis, the change does not affect the performance capabilities of the ATWS-RPT system, and the change imposes a more conservative AOT; then, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The margins of safety as defined in the basis for TS Limiting Condition for Operation (LCO) 3.3.4.2, Anticipated Transient Without Scram-Recirculation Pump Trip (ATWS=RPT) Instrumentation, were premised on the functional characteristics of the system as presented and described in the USAR. These margins of safety are not affected by the changes made to the TS Bases in bringing them into conformance with the USAR, nor are they impacted by reducing the AOT to the value originally established within the TS. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PRIMARY CONTAINMENT ISOLATION CLB/USAR DISCREPANCY RESOLUTION

Activity Evaluated: USAR Change 9-022 Log Number: 99-208

Updated Safety Analysis Report (USAR) Change 9-022 eliminates the requirement to perform stroke (closure) time testing for the following valves: Containment Heating, Ventilation, and Air-Conditioning (HVAC) Outboard Isolation Valve, 1VR002A; Containment HVAC Inboard Isolation, 1VR002B; Drywell Purge Outboard Isolation, 1VQ006A; and Drywell Purge Inboard Isolation, 1VQ006B. These valves are not discussed in any assumptions made for the initiation of any design basis accidents or other plant events. However, the four containment isolation valves are expected to close, to maintain containment integrity, and limit the release of radioactivity from containment following a loss of coolant accident (LOCA) inside containment. USAR Appendix D, subsection II. E.4.2, Operational Requirements Manual (ORM) Attachments 4-10 and 4-35, and Clinton Power Station Procedure 3408.01 require these valves be sealed closed to assure they cannot be inadvertently opened in Modes 1, 2, and 3. These valves satisfy 10CFR50, Appendix A, General Design Criteria 56 by providing one locked isolation valve inside and one locked isolation valve outside containment. Thus, the 1VR002A/B and 1VQ006A/B valves, with respect to containment isolation have been moved from "active" to "passive", and as such are considered manual valves. Since they are entirely passive in establishing and maintaining the primary containment boundary, there is no credible failure mode that can affect any equipment important to safety. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR; nor does it create an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification Sections 3.6.1.3 and Bases Section 3.6.1.3 contain operability requirements for the primary containment isolation valves and discuss the need to ensure that the primary containment boundary is maintained during and after an accident. Since the 1VR002A/B and 1VQ006A/B valves are considered to be manual valves and are "sealed closed", deleting the requirement to monitor their stroke (closure) time has no effect upon establishing or maintaining the primary containment boundary. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE THE SHIFT TECHNICAL ADVISOR (STA) POSITION TO ALLOW LICENSED SENIOR REACTOR OPERATORS TO FULFILL FUNCTION OF STA

Activity Evaluated: USAR Change 9-042 Log Number: 99-210

Updated Safety Analysis Report (USAR) Change 9-042 changes the Shift Technical Advisor (STA) position to allow licensed Senior Reactor Operators to fulfill the function of the STA. This activity does not change or delete any of the required functions that a STA previously performed, it only alters who would perform those functions. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. Technical Specification section 5.2.2g discusses providing advisory technical support to the SS in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant. This function is not being changed or deleted; it will still be performed by a qualified individual. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DELETING THREE OCA BUILDINGS, CLOSING TWO FIRE PROTECTION VALVES AND REVISING PROCEDURES

Activity Evaluated: USAR Change 9-034 Log Number: 99-212

Updated Safety Analysis Report (USAR) Change 9-034 deletes three temporary construction units, the Ice House building, the Time Keeper building, and the Air Compressor building from USAR Figure 9.5-1 Sheet 3. These three buildings are badly deteriorated, were defined as temporary construction units, are not included as Principle Station Structures discussed in USAR Section 1.2.2.2, and are planned to be demolished. In addition, USAR Change 9-034 revises Figure 9.5-1 Sheet 3 to show the fire protection isolation valves, 0FP361 for the Compressor building and 0FP230 for the Time Keeper building, as normally closed and the valves downstream piping as having no building service. This activity affects three procedures, which will be revised to change valves 0FP230 and 0FP361 from "Locked Open" to "Locked Close". The buildings are located outside the protected area and the only design basis impact is the potential of producing missiles as a result of the buildings being destroyed during extremely high winds or a tornado. The removal of the buildings will eliminate the potential source of debris from the buildings. This activity changes the fire protection system by deleting a fire suppression load; however, the elimination of the fire suppression load has a beneficial impact on the operation by improving the reliability of the fire suppression system upon closing the valves, which serve the specific buildings. This eliminates a potential fire protection water inventory demand and the fire pump operation as a result of the specific building suppression system activation. This change does not alter the operation or performance characteristics of any structure, system, or component that could affect any equipment important to safety. Also, this change does not directly or indirectly impact the ability of the operating crews to mitigate an accident or other off-normal plant event. Therefore, this activity does not increase the probability or consequences of an activity or malfunction of equipment important to safety previously evaluated in the USAR. The only credible failure mode associated with his change is the failure of the fire protection valve isolation capability, which would result in fire protection water inventory loss. This would result in the fire protection pump initiation until the leak could be isolated at the header. The failure would necessitate an operator being dispatched to manually isolate the leak, but the point of isolation would not impact fire protection to any Principle Station Structures. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. There are no Technical Specifications which control the structures being removed by this change or that control the fire protection system, components, or functions; thus there are no margins of safety associated with the fire protection system. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALL DRAIN LINES ON THE FIRE PROTECTION OUTBOARD CONTAINMENT ISOLATION VALVES

Activity Evaluated: ECN 29371, 31900, and 31901; USAR Change 9-030 Log Number: 99-213

Engineering Change Notices (ECNs) 29371, 31900, and 31901 add drain lines to the Fire Protection Outboard Containment Isolation Valves, 1FP051, 1FP054, and 1FP092. The purpose of these new drain lines is to provide a means to drain the between-the-seat area of the valves prior to Local Leak Rate Testing (LLRT). The only possible accidents associated with this activity are moderate energy line break and internal flooding. However, the probability of failure is the same as the existing penetration piping because the new drain lines will be designed and installed in accordance with the original code of construction of the Fire Protection system. Also, per Updated Safety Analysis Report (USAR) Section 3.6.2.1.4, moderate energy piping and piping of nominal size one inch or less is exempted from pipe break evaluation and protection requirements. The purpose of these valves, which are Primary Containment Isolation Valves, is to minimize leakage from containment during certain accident conditions. The new drain lines will be outside the System Boundary and, therefore, will not be a factor in containment leakage during postulated accidents. The drain connections are passive components, which cannot interfere with valve operation. This activity does not alter the safety function of these valves and will continue to be normally closed. The additional weight of the drain connections relative to the weight of the valves is insignificant and not part of the extended structure of the valve such that the seismic qualification of the valve is unaffected. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The new drain line isolation valves will be normally closed and the lines capped such that there are no new flow paths. No new circuits will be introduced and no changes in automatic functions will be made. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs), defines the basis for the PCIVs as for the valves to be closed or function to close within the required isolation time following event initiation to minimize potential leakage. Valves 1FP051, 1FP054 and 1FP092 will continue to meet these requirements. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLING A FREEZE SEAL TO PERFORM MAINTENANCE ON 1FP113

Activity Evaluated: MWO D78191 Log Number: 99-214

Maintenance Work Order D78191 installs a freeze seal, in order to rebuild or replace valve 1FP113, while the Fire Protection (FP) system is in-service. Flow to the drywell purge filter train C is isolated during the performance of this activity; no other parts of the FP system will be affected. Updated Safety Analysis Report (USAR) Chapters 6 and 15 do not address an accident analysis due to the loss of the fire protection system in the Diesel Generator Building Drywell Purge Filter Train. This activity has potential impact upon USAR Chapter 3 flood analysis and Fire Protection Evaluation Report (FPER) fire analysis. The installation of a freeze seal involves neither combustibles nor ignition sources. The failure of the freeze seal can lead to flooding, which is anticipated in the governing freeze sealing procedure. To make the risk of freeze seal failure acceptable, the installation of a freeze seal is part of a troubleshooting and repair plan and is controlled via the maintenance process. The freeze seal procedure requires that a checklist be prepared. The checklist requires a plan that includes the identification of compensatory measures in the event of a freeze seal failure. This activity does not alter the valve's function or performance or change system response characteristics. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no credible failure mechanisms being introduced during this activity. Installing a freeze seal will not prevent Control Room personnel from monitoring annunciation, indication, and instrumentation that is affected by the system. Freeze seal activities will be installed and monitored by the appropriate site procedures. A design change is not required because the original design parameters will not change, and the system will be restored to its original design condition. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not discuss the fire protection program. Thus, the fire protection system does not have any safety limits or limiting conditions for operation stated in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION OF USAR SECTION 1.7.1, TABLE 1.7-1, AND REGULATORY GUIDE 1.70 COMPLIANCE STATEMENT

Activity Evaluated: USAR Change 9-032 Log Number: 99-215

Updated Safety Analysis Report (USAR) Change 9-032 removes the statement in Section 1.7.1 that the figures listed in USAR Table 1.7-1 are incorporated by reference and explains that this is a historical compilation of the figures provided to the Nuclear Regulatory Commission (NRC) during initial licensing. It also revises USAR Table 1.7-1 to remove the columns presenting Revision Number and Date of Revision. In addition, this activity revises the statement of compliance with Regulatory Guide 1.70, Revision 3 as presented in USAR Section 1.8. Inherently, these activities do not have the ability to influence the probability of an evaluated accident because these activities have no impact on the operational behavior, performance characteristics, or reliability of site structures, systems, or components. Further, the absence of this material does not result in the loss of a barrier that prevents the occurrence of an accident. Although the material is removed from the USAR, elements of operational activities that possess accident or malfunction initiation characteristics or consequences will continue to be identified through the 50.59 review process because of their contradiction with system design and performance bases. Compliance with the Regulatory Guide only provided a mechanism for ensuring the content and format of the USAR was acceptable to the NRC during initial licensing. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. None of these activities possess operational characteristics capable of inducing credible failure modes that could result in new events with radiological consequences comparable to those previously evaluated. The removal of the information incorporated by reference does not alter the defined bounds of the accident analysis, nor does it alter the threshold at which an event could be considered credible. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The material removed from the USAR did not define performance or operability criteria pertinent to the Technical Specification requirements. Further, the material removed did not define the acceptance value(s) for determining acceptable performance of the fission product barriers. Thus, this activity would not affect information used to define a margin of safety.

USAR DESCRIPTION OF SERVICE AIR/INSTRUMENT AIR INSTRUMENTATION AND ALARM FEATURES

Activity Evaluated: USAR Change 9-035 Log Number: 99-216

Updated Safety Analysis Report (USAR) Change 9-035 revises USAR Section 9.3.1.5 to correctly describe the design of the instrumentation features of the Service Air (SA) and

Instrumentation Air (IA) systems which provide control room alarms and automatically isolate the building ring headers in the event of the low pressure in the header. USAR Section 15.2.10 evaluates the complete loss of the IA system that could occur as the result of a major line break in the system or as a result of mechanical or electrical failure of the normal IA supply and the backup SA source. The turbine building and auxiliary building IA headers are not equipped with automatic isolation valves or low pressure alarms. In the event of a pipe rupture or excessive air usage in the ring headers in these IA headers, there would not be a control room alarm or automatic header isolation. However, the process for identification and isolation of IA/SA headers for those headers that do not have control room alarms and automatic isolation features is similar to those that do have them, since they are fed from ring headers that do have those features. Those features allow for identification and isolation in the event of a pipe rupture or excessive leakage prior to loss of the entire SA/IA system. The absence of the IA/SA low pressure alarms and automatic ring header isolation features does not change the safety function of those air-operated components which perform safety-related functions. The portions of the IA system, which are safety related and are relied upon to respond to other plant events, are not affected by this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The absence of control room alarms and automatic isolation features in some of the building ring headers could result in low pressure in those headers until detected by the headers from which they are fed. This is a less severe event and is bound by the complete loss of IA. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Surveillance Requirement 3.5.1.3 discusses the requirement to verify every 31 days that Automatic Depressurization System (ADS) accumulator supply pressure is greater than or equal to 140 PSIG to assure adequate air pressure for reliable ADS operation. Air is supplied to the ADS through the IA ring headers in the turbine and auxiliary buildings. The absence of automatic isolation features on low pressure o on these ring headers does not compromise the reliable source of air to these items. Technical Specification Bases 3.6.5.3 discusses the drywell vent and purge isolation valve which fail closed on loss of IA or power. The absence of control room alarms and automatic isolation features in headers that supply IA to these valves does not adversely affect the fail closed features of these valves. Therefore, this activity does not reduce a margin of safety as defined in the basis for any technical specification.

INSTALLING A FREEZE SEAL TO PERFORM MAINTENANCE ON 1SX041B

Activity Evaluated: MWO D86732 Log Number: 99-218

Maintenance Work Order D86732 installs a freeze seal, in order to rebuild valve 1SX041B, while the Shutdown Service Water (SX) system is in-service. Flow to the High Pressure Core Spray (HPCS) room coil cabinet will be isolated during the performance of this activity. During normal plant operation, the Division 3 Diesel Generator cooling loads and the HPCS pump room is cooled by the SX system; the Division 1 and 2 SX system will be operable and in standby during the entire freeze seal and subsequent maintenance activities. The design basis accident that could be impacted is a Moderate Energy Line Break (MELB); however, the failure of a freeze seal is bounded by a larger 14" Plant Service Water pipe that was previously analyzed for safe shutdown following a postulated internal flooding. The failure of the freeze seal can lead to flooding, which is anticipated in the governing freeze sealing procedure. To make the risk of freeze seal failure acceptable, the installation of a freeze seal is part of a troubleshooting and repair plan and is controlled via the maintenance process. The freeze seal procedure requires that a checklist be prepared. These controls mitigate the possibility of a freeze seal failure. This activity does not alter the valve's function or performance or change system response characteristics. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no credible failure mechanisms being introduced during this activity. Installing a freeze seal will not prevent Control Room personnel from monitoring annunciation, indication, and instrumentation that is affected by the system. Freeze seal activities will be installed and monitored by the appropriate site procedures. A design change is not required because the original design parameters will not change, and the system will be restored to its original design condition. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity is being completed under applicable Limiting Conditions for Operation as required by the Technical Specifications. The freeze seal will affect operability of the Division 3 Diesel Generator, Division 3 Shutdown Service Water System, and the High Pressure Core Spray System. This activity does not adversely affect the ability to achieve and maintain safe shutdown. The SX system is designed so that maintenance can be performed on Division 3 systems without adversely impacting Divisions 1 and 2. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

SX PUMP MOTOR REPLACEMENT

Activity Evaluated: ECN 31957 and USAR Change 9-040 Log Number: 99-219

Engineering Change Notice (ECN) 31957 replaces a safety related 1500 HP, 4kV motor in the Division 1 Shutdown Service Water (SX) system with an equivalent, but not like for like, motor. The SX system provides cooling water for safety related systems to maintain equipment within its operating temperature and to remove reactor heat from the containment building. In the event of a loss of off-site power or service water (WS) pump failure, the SX pumps respond automatically to provide the required water supply for the SX system. Since the motor is in standby, the only accident, which this motor could initiate, would be the failure of Residual Heat Removal shutdown cooling discussed in Section 15.2.9 of the Updated Safety Analysis Report (USAR); however, the new motor has the same potential failure modes as the original motor. Since the basic design and construction of the two motors are equivalent, there are no new failure modes introduced. The replacement motor has been refurbished using material of equivalent or better grade than the original motor. The USAR Change reflects the load change due to motor replacement. Calculations confirm the Auxiliary Power (AP) system's capability to supply the connected loads, whether fed from the offsite or onsite source. This activity does not impair the availability or degrade the performance of any system. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not change system operating logic or introduce any new failure modes into the system. The replacement relay will have the same setting as the original relay and as such will provide the same protection. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification 3.7.1 and its associated Bases section discuss the SX system. This discussion centers around the ability of the system provide the required cooling water flow to the connected equipment. The refurbished motor has been tested and confirmed as being capable of powering the pump. After the replacement of this motor, the SX system will continue to perform in accordance with site requirements.

Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CORRECT SOLENOID AND VALVE BODY SYMBOLOGY TO AGREE WITH DESIGN

Activity Evaluated: USAR Change 9-025 Log Number: 99-220

Updated Safety Analysis Report (USAR) Change 9-025 corrects non-safety related valve symbology on USAR Figure 9.2-11. Review of the USAR showed that neither the valves nor the pumps to which they provide/supply seal water are initiators for any of the events or transients discussed in the USAR. Changing the solenoid valves' functional failure mode has no impact on the operation, reliability or design, material or construction standards of these valves. Failure of these valves to operate, to either open or close, will not impact any accident previously evaluated in the USAR. The Cycled Condensate (CY) system has no nuclear safety-related function, except for the piping and valves, which form part of the containment isolation boundary. Failure of the system, with this exception, does not compromise any nuclear safety-related system component and does not prevent safe shutdown of the reactor or impact any fission product barriers. The subject valves are located upstream from their respective pumps and are not part of the containment or isolation boundary. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not create any new flow paths, introduce any new circuits, create or delete any automatic function, or create any new interactions with any other systems. In addition, this change will not reduce overall valve reliability. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The solenoid valves or the pumps to which they provide seal water are not an input into any of the Technical Specifications or their Bases. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DIESEL GENERATOR TANK VENT SEPARATION

Activity Evaluated: USAR Change 9-044; OD/OE 1-99-12-073 Log Number: 99-221

Condition Report 1-99-12-173-0 identified that there is a discrepancy between the Updated Safety Analysis Report (USAR) description and the as-built design with respect to separation of the vent lines of the diesel generators fuel oil storage and day tank vent lines. The Operability Evaluation determined that the condition is in accordance with design basis requirements, but USAR Change 9-044 revises the USAR description to correctly describe the 14-foot separation between the storage tank vents. The diesel fuel oil system is not an initiator for any design basis accidents evaluated in the USAR. Its purpose is to supply fuel to the emergency diesel generators, which provide 4160 VAC power for mitigation of postulated accidents described in USAR Chapters 6 and 15. The configuration of the tank vents does not degrade the capability of the diesel generators to perform the required accident mitigation functions because there is not a credible common cause failure. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The applicable portions of the tank vent lines are located outside the building housing the diesel generators, are not in the vicinity of equipment important to safety, and are not positioned to create the possibility of any accident or malfunction of equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Bases Section 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," discusses details of the supply of fuel oil to the diesel generators and the compliance with regulatory requirements. The day tank vents are not required to function for the diesel generators to perform the required accident mitigation functions. There are no aspects of the separation of the diesel generator tank vent lines that affect the margin of safety specified in Technical Specification 3.8.3 or any other Technical Specification. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

STEAM BYPASS VALVE TESTING AT GREATER THAN 85% REACTOR POWER

Activity Evaluated: CPS Procedure 2800.13 Revision 0 Log Number: 99-222

Clinton Power Station (CPS) Procedure 2800.13 demonstrates that the Main Steam Bypass Valves can be surveillance tested at up to and including 98% reactor power. One or more valves will be stroked open and closed (one at a time) near 90% power and all six valves will be stroked open and closed (one at a time) near 98% power while test data is collected. With the exception of the initial power level, this bypass valve test is the same test as the normal bypass valve surveillance test and the bypass valve test during initial start-up testing. The opening of a bypass valve at any power level does not initiate any of the events described in the Updated Safety Analysis Report (USAR). Bypass valve testing causes a mild reactor system pressurization event under normal plant operation that increases nuclear power by less than 2%. This event is bounded by the pressurization transients, i.e., anticipated operational occurrences (AOOs), that are analyzed in the USAR which may cause Average Power Range Monitor (APRM) power to briefly exceed 300% core-wide. The fuel operating thermal limits consider the AOOs and ensure that the safety limits on the fuel will not be exceeded. Testing of the bypass valves will not affect equipment reliability because it will use installed equipment within normal, licensed limits. Since this test uses installed plant equipment that has been analyzed, it does not affect environmental, seismic, or separation criteria. The overall system response will be evaluated at each successive power level in order to ensure that the reactor and balance of plant response is within acceptable limits for valve stroke performance, heat flux, system stability, and APRM power. This will ensure that no unanticipated conditions occur that could increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not change any systems, structures, or components (SSCs), nor does it introduce any SSC into the plant. Also, the small pressure change during the bypass valve testing will be within the USAR analysis so there will be no new failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.2, "Power Distribution Limits," and Operational Requirements Manual, Attachment 2-2, Table 1, "Reactor protection System (RPS) Instrumentation Trip Setpoints," controls the operating ranges of the nuclear fuel and system pressure. These operating limits protect the safety limits for the fuel and primary system pressure. The bypass valve test will cause a very mild transient that will not exceed the operating limits on the fuel or dome pressure. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FUEL BUILDING HVAC SYSTEM (VF) CLB/USAR DISCREPANCY RESOLUTION

Activity Evaluated: USAR Change 9-050 Log Number: 2000-001

Updated Safety Analysis Report (USAR) Change 9-050 makes several changes to the description of the Fuel Building Heating, Ventilation, and Air-Conditioning (HVAC) (VF) System. Included in this USAR Change are changing the supply air filter efficiency listed on Table 9.4-3 from 85% to 55% and changing the exhaust fan static pressure listed on Table 9.4-3 from 8.6" H_2O to 7.7" H_2O. The VF system is not mentioned as the initiator for any of the accidents analyzed in USAR Chapters 6 and 15, or any other plant events discussed in the USAR. Following an accident, the VF system is shutdown, and supply and exhaust duct dampers close to isolated the secondary containment. Thus, the supply filter and exhaust fans are not operating in a post-accident condition. The supply filter efficiency and the exhaust fan static pressure are being changed to that originally specified by vendor specifications. Calculations determined the VF system performance is adequate and this activity does not affect the capability of the VF system to supply ventilation to the area containing safety-related equipment. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no components or systems being introduced into the plant, or being changed by these changes to the USAR. Thus, no credible failure modes or mechanisms are being introduced. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.6.4.1, "Secondary Containment," and 3.6.4.2, "Secondary Containment Isolation Dampers," include requirements, which may potentially be affected by changes to the VF system. These sections require maintaining the secondary containment at a negative pressure and maintaining the ability to isolate secondary containment. This activity does not affect the operation of the VF system during normal operation, prior to any postulated accident. In addition, these changes have no affect upon the ability of the secondary containment isolation dampers to close following a postulated accident. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF A GAG ON RELIEF VALVE 1B21F408

Activity Evaluated: Temporary Modification 00-001 Log Number: 2000-002

Temporary Modification (TM) 00-001 installs a gag, which prevents relief valve 1B21F408 from lifting/opening, while in Modes 4 or 5. This relief valve is installed between the Main Steam (MS) and Auxiliary Steam (AS) systems. This portion of the MS system is non-safety related, has no safety function, is not required to operate during or after a design basis event, and is not required for safe shutdown of the plant. Updated Safety Analysis Report (USAR) Chapters 6 and 15 describe different accidents related to the safety-related portion of the MS system. Since this activity does not affect safety-related piping and components and is being completed in operational modes 4 or 5, this activity does not introduce any new failure mechanisms. The integrity of the relief valve will be maintained. In addition, this activity does not change the capability of the MS system to perform its safety or non-safety related function. The material of the gag is carbon steel, which is compatible with the chemical properties of the valve cap. The gag is a mechanical device, which will not affect overall system performance of the MS or AS systems such that it changes system response characteristics, causes system operation outside of its design limits, causes operational transients in the system, or causes adverse system interaction with other systems. This activity does not affect any of the fission product barriers as it contains no change to pipe routing, radiological boundaries, radiological monitoring equipment, or structural configuration. Therefore, this activity does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The gag will not affect the reliability or operation of the valve. This activity does not change the failure modes of the relief valve and does not affect the operation or reliability of the MS or AS system. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The MS system is addressed in the Technical Specifications, but is not applicable in operating modes 4 or 5. This relief valve is non-safety related, has no safety function, is not required to operate during or after a design basis event, and is not required for safe shutdown of the plant. This temporary modification has no impact upon any Technical Specification, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

TIE-INS FOR FUTURE HYDROGEN WATER CHEMISTRY (HWC)

Activity Evaluated: Modification CD-006; ECN 31919 Log Number: 2000-004
USAR Change 9-054

Modification CD-006 installs the process header tie-ins required for the future Hydrogen Water Chemistry (HWC) Injection system. There are no design basis accidents in the Updated Safety Analysis Report (USAR) directly associated with the Condensate (CD) system. The relevant design basis accident associated with this activity would be the pipe breaks in the Reactor Water Clean-Up (RT) and CD systems. The only potential failure mode for this modification, which could act as an initiating event for the above accidents is a pipe break or leak. The CD system is not safety related or seismic Category I and is not required for safe shutdown of the reactor. Installation of the HWC tap connections will not have any impact on this initiating event because the modified piping is designed to the same codes and engineering standards as the existing piping. Calculations have been performed to ensure the pipe stresses and support loads remain within allowables. The tie-ins are dead-end capped pipes and therefore the flow though the RT and CD systems will not affected and system performance remains unchanged. A pipe break in the new ¾ inch tie-in lines would have radiological consequences no worse than the large postulated Feedwater Line Break or the postulated Main Steam Line Break. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The piping tie-ins installed by this modification are dead ended capped pipes and will not affect the performance of the RT and CD systems. The only credible failure mode for this modification is a leak or break in the modified system piping. The break in the modified piping would be no different than a break in the existing piping and does not introduce a new or unanalyzed condition. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. There are no Technical Specification requirements relative to the RT or CD systems. This modification does not adversely affect the function or operation of these systems. The new piping tie-ins will be designed and installed in accordance with industry standards and existing plant specifications and procedures. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MAINTENANCE ORGANIZATION CHANGE

Activity Evaluated: USAR Change 9-060 Log Number: 2000-005

Updated Safety Analysis Report (USAR) Change 9-060 combines the positions of Director - Electrical Maintenance and Director – Mechanical Maintenance into one position, Director – Maintenance. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.0 addresses the management responsibilities/requirements of the Plant Manager and Operations personnel and Technical Specifications 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

UPDATE HYDROGEN GENERATION CORROSION RATES IN USAR

Activity Evaluated: USAR Change 9-053 Log Number: 2000-007

Corrective Action to CR 1-99-12-060. This CR was written to document the failure to update the Updated Safety Analysis Report (USAR) following the revisions of Calculation 3C10-0383-005 and IP-F-0075. USAR Tables 6.2-50a & b and 6.2-51a & b were not updated to show the correct corrosion rates for aluminum alloys and continuously immersed zinc. These tables will be updated along with additional text to Section 6.2.5.1.3.2.2 in order to clarify the contents of aluminum alloys. Section 1.8 of the USAR dealing with RG 1.7 will be revised to remove reference to Figures 6.2.131a & b (figures were deleted in Rev 6 of the USAR). Correct the legend of Figure 6.2-125 to reflect RG 1.7 not RG 1.70. Other minor editorial and typographical errors will be made to Table 6.2-49, 6.2-50a & b, and 6.2-51a & b. This activity does not affect the inventories of aluminum alloys or zinc in the containment or drywell, nor does it affect the allowances of future modifications for aluminum allows or zinc in containment or drywell. No impacts on design or operation of hydrogen control/mitigation systems, this activity does not affect the ability of these systems to perform their design basis functions nor does it impose any new constraints on the operation of these systems.

EVALUATION TO SUPPORT COMPENSATORY ACTION FOR OPERABILITY DETERMINATION 2-00-01-048

Activity Evaluated: CCF 2000-0047 for CPS 3402.01 Log Number: 2000-008

Compensatory Action for Operability Determination 2-00-01-048 requires an operator to manually position the Control Room Heating, Ventilation, and Air-Conditioning (VC) system "A" train control room backdraft dampers to a closed position during "B" train normal operating conditions. This activity requires the addition of an operator action to Clinton Power Station (CPS) Procedure 3402.01, "Control Room HVAC (VC)", to ensure that the "A" train backdraft dampers are in closed position while "B" train is in normal mode of operation. The VC system has not been postulated to initiate any accident previously evaluated in the Updated Safety Analysis Report (USAR). Ensuring the dampers close when placing a VC train in service during procedural evolutions is easily accomplished and will not increase the probability of operator error. This activity places the backdraft damper of the idle train in its design position and will not have an adverse affect on any normal flow path. The degraded condition, increased friction, prevents gravity closure of the damper; however, the increased frictional force which prevents gravity closure of these dampers is insignificant compared to the opening force from the supply fan and failure to open when required is not considered credible. The VC system is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The subject backdraft dampers and isolation dampers assist in maintaining the Main Control Room (MCR) design basis pressure by mitigating reverse flow through the non-operating train of the VC system. Closing or ensuring a damper in the VC system is in its design position cannot increase radiation exposure to MCR personnel following any accident. This activity does not take credit for a manual action in place of an automatic action for protection of Safety Limits. Ensuring the damper is closed occurs on 825' Control Building, which is not a harsh environment and ingress and egress to this area is not a concern. Procedural guidance for this activity is being incorporated into CPS Procedure 3402.01. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure modes associated with this activity are failure of the operator to correctly position the backdraft dampers and for the damper to stick shut after being manually closed. Because the increased frictional force which prevents gravity closure of these dampers is insignificant compared to the opening force from the supply fan, sticking closed is not considered credible. Failure to correctly position a damper is expected to have a minimal impact. Failure of the damper to close is neither expected to result in an increased temperature and have an adverse affect to MCR equipment/instrumentation, nor to result in an increase in radiation exposure to the plant operator. Based on the design of the system and flow data collected, there is reasonable assurance that positive pressure can be maintained during high radiation mode with the backdraft dampers open. Upon detection of high radiation in control room air intake, the VC system operating train will shift automatically to high radiation mode or requires manual start of the other train within 20 minutes in the event of loss of operating train. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications 3.3.7.1, 3.7.3, and 3.7.4 are the Technical Specifications associated with the VC system. The purpose of these Technical Specifications is to ensure radiation exposure of the MCR personnel, through the duration of any one of the postulated accidents, does not exceed the limits of General Design Criteria 19 and to maintain the control room temperature within a specified band for operator comfort and that equipment in the control room is not adversely affected. Since there is neither an increase in the dose to the operator nor an adverse affect on MCR temperature, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ALTERNATE PART FOR OBSOLETE IJCV RELAY USED FOR DIVISION II DIESEL GENERATOR OVERLOAD PROTECTION

Activity Evaluated: ECN 31948 Log Number: 2000-009

Engineering Change Notice (ECN) 31948 replaces the 12IJCV51A13A relay for the 51V-1, 51V-2 and 51V-3 time overcurrent relays with voltage restraint with a safety related GE Model No. 12IFCV51AD1A. These relays are installed in Diesel Generator (DG) 1B Control Panel 1PL12JB for diesel generator overload protection. The panel is designed for the Type IJCV relay. The Type IFCV relay has a smaller case, so a mounting plate is required to install the IFCV relay. Failure of this relay is not an event initiator for any of the design basis accidents defined in Updated Safety Analysis Report (USAR) Chapters 6 and 15. The alternate IFCV relay meets the same design, material, and construction standards that were applicable to the IJCV relay. This activity will not affect the overall electrical system and the Division II DG system performance. Calculations concluded that the replacement relay will provide adequate overload protection with proper coordination that will meet the intended design and protection function. The design change does not impair the availability of the Division II DG system and does not degrade the performance by affecting the environmental, seismic or separation criteria. Mounting holes for the plate are required to be field drilled prior to installation. The replacement relay installation, testing and operations is identical to the original relay. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The diesel generator is used as backup AC electrical power source to mitigate consequences of an accident. Since the new relay is identical in performance and function, the design change does not alter the function or operation of the Division II DG systems and does not impact other safety related systems. Failure modes of the IFCV relay such as inadvertent tripping, premature tripping, late tripping or failure to trip are the same as for the IJCV relay. The effects of these failure modes on the Division II DG system have not been changed. Therefore, this activity does not create an activity or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.8.1, AC Sources – Operating, and its associated Bases specify the requirement s for the Diesel Generator systems. The margin of safety is defined as "sufficient capacity, capability, redundancy, and reliability". Calculations indicate sufficient capability is provided for the IFCV relay to perform its safety related function. Since the IFCV relay is built and installed to the same standards as the IJCV relay, the capacity, redundancy, and reliability remains unchanged. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

OFF-SITE REPORTING FOR QUALITY ASSURANCE DEPARTMENT

Activity Evaluated: USAR Change 9-073 Log Number: 2000-010

Updated Safety Analysis Report (USAR) Change 9-073 changes the reporting relationship for the Manager – Quality Assurance (QA) from the site Vice President to PECO/AmerGen Director – Nuclear Quality Assurance. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALL NEW BUFFER SYSTEM

Activity Evaluated: Modification BS-002; USAR Change 9-214 Log Number: 2000-011

Modification BS-002 installs the new Buffer System (BS) computer hardware and software in the Nuclear Training Building and adds a description of the BS to the Updated Safety Analysis Report (USAR). This modification does not impact any design basis accident analyzed in the USAR. The affected hardware is in the Nuclear Training Building and is of the appropriate material and design to operate in a normal environment. The new Buffer System is capable of higher performance and is year 2000 compliant. Overall system response characteristics will no measurably change, the system will continue to operate within design limits, and there will be no operational transients as a result of the proposed activity. Transmissions from the Buffer System are sent to the Illinois Department of Nuclear Safety and to the Nuclear Regulatory Commission, however, no credit is taken for the receipt of these transmissions in any accident evaluation. This activity is limited to the removal of the old hardware and software and replacing it with the new Buffer System; this activity does not affect any safety-related structures, systems, or components. The Buffer System is not utilized to control or operate any systems that are used to maintain radiological barriers for protection of the public or plant personnel, or to mitigate the consequences of any accidents. There are no new system interactions introduced by these changes. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no Technical Specifications associated with the Buffer System. In addition, these changes have no affect upon any safety limits limiting safety system settings or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ADD NEW FILTERS UPSTREAM FROM PRESSURE REGULATORS 1IA044A/B IN INSTRUMENT AIR SUPPLY TO THE AUTOMATIC DEPRESSURIZATION SYSTEM

Activity Evaluated: ECNs 31830 and 31831; USAR Change 9-081 Log Number: 2000-016

Engineering Change Notices (ECNs) 31830 and 31831 install new high pressure filters upstream of Instrument Air (IA) system pressure regulators 1IA044A/B along with required changes to piping and piping supports. It also removes the control panels and rupture discs and various isolation valves located in the panels. Isolation valves will be added upstream and downstream of the filters and pressure regulators to facilitate maintenance activities. The new filters replace low pressure filters which were installed downstream of the pressure regulators. The pressure regulators are in the IA system, which supplies air from the air bottle tank farm to the Automatic Depressurization System (ADS). The ADS is designed to mitigate postulated accidents that are described in Updated Safety Analysis Report (USAR) Chapters 6 and 15. However, the ADS is not postulated to initiate any accident or transients. The use of a finer filter cartridge could create the possibility of clogging of a filter if it is placed in a system with a large quantity of particulate material. However, since this is a clean system, which is recharged with dry filtered air and the air bottles are routinely cleaned, it is unlikely that a large quantity of particulate matter could accumulate. Also, the filters are oversized for this type of application making it unlikely that the filters would clog sufficiently to prevent air flow. The modified portion will meet the same design, material and construction standards as the applicable portions of the original installation. The operating conditions will be within the design temperature and pressure limits in the applicable specifications and industry codes. The system is designed to not fail when operated in accordance with the specified conditions. Consequently, the modified portion of the system will not be degraded from the original installation, will not affect the capability of the ADS to perform its design basis functions, and will not affect the results of the accident analyses in USAR Chapters 6 and 15. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. As stated above, the modified portion will meet the same design, material and construction standards as the original installation. The ADS is designed to perform its design basis function even if the air supply to the ADS fails because each of the safety relief valves used for automatic depressurization is equipped with an air accumulator and check valve arrangement that are designed to operate under the postulated accident conditions. In addition, this activity does not introduce any new failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Bases 3.5.1, ECCS – Operating, discusses the design basis for the ADS. It does not identify specific requirements for the operability of the individual components in the ADS; however, the filters are an integral part of the air supply to the ADS. This activity is being implemented to improve the operability of the air supply to the ADS. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

UPDATE TO 1995 IEEE-450 REVISION

Activity Evaluated: USAR Change 9-092; TS Bases Change BL-00-001 — Log Number: 2000-017

This activity revises the Clinton Power Station (CPS) commitment in the Updated Safety Analysis Report (USA) and Technical Specification (TS) Bases to reference the latest revision of the Institute of Electrical and Electronics Engineers (IEEE) Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations". The latest revision to the standard does not affect the design, load capacity, voltage, or any of the hardware of the station batteries. The changes are to be implemented by CPS procedures, which impact periodicity of certain inspections and maintenance practices and add options for certain conditions. This activity does not involve any hardware changes or changes to any battery parameters or system components. Since there is no change to the scope or intent of the standard, there is no change to the function or performance of the batteries, and therefore, no change to the accident analyses involving the batteries. Regardless, failure of any of the plant batteries is not an initiator of an accident analyzed in the USAR. In addition, this activity does not affect overall battery system performance and does not impact the design, material, function, or hardware of any plant systems. The intent of IEEE Standard 450 is to minimize human error as an initiator of battery failure. None of the existing failures for which batteries have been evaluated are changed as a result of this activity. Also, there are no changes to the types of tests performed on batteries, so no new failure mechanisms would be introduced. The batteries will still be maintained in accordance with the standard, which disseminates information intended to minimize failure modes or mechanisms of the batteries. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There is no direct reference to IEEE Standard 450 in the Technical Specifications. This activity does not impact the operation of any of the station batteries or any plant structure, system, or component. Referencing the latest revision of IEEE Standard 450 does not impact any Technical Specification safety limits, limiting conditions for operation, surveillance requirements, acceptance requirements, or allowed out-of-service time. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION AND REMOVAL OF TEST EQUIPMENT DURING THE PERFORMANCE OF CPS 2800.41D009

Activity Evaluated: CPS 2800.41D009 Revision 0 Log Number: 2000-020

This activity installs and removes test equipment in order to perform a pre-static, dynamic, and post-static VOTES testing of Residual Heat Removal (RHR) "B" heat exchanger inlet and outlet valves, 1E12-F014B and 1E12-F068B. The pre-static test will be conducted to confirm that the setup is consistent with operability. The installation of the test equipment meets or exceeds the design pressure rating of the piping line rating. The moderate energy line break as described in Updated Safety Analysis Report (USAR) Chapter 3 is not an initiator for any of the events described in Chapter 15. The pressure transducers used have a range of 0-200psi, which is compatible for this application. There will be no mechanical or electrical connections to any equipment important to safety that would cause a malfunction of that equipment. The pressure boundary integrity of the piping will be maintained. The test procedure affects only one train of the RHR heat exchanger and associated supporting systems. Since the "A" train is not affected and is completely separated from the "B" train, a failure of the "B" train will not increase the consequences of an accident. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The pressure boundary line break that has been evaluated in the moderate energy line break in Chapter 3 is a limiting line break that encompasses any failure mechanisms possible. There are no new failure modes introduced by this activity. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Since no functional changes to the RHR and service water systems will occur, their Technical Specification function will be accomplished. This activity does not impact any Technical Specification safety limits, limiting conditions for operation, surveillance requirements, acceptance requirements, or allowed out-of-service time. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONVERSION OF WO CHILLER 0WO02CE FROM R-500 TO R-134A REFRIGERANTS

Activity Evaluated: ECN 31969; USAR Change 9-091 Log Number: 2000-021

The Clean Air Act of 1990-Title 6, mandated a phase out of production of fluoro-chloro hydrocarbons (CFCs), in order to protect the earth's ozone layer. To ensure continued availability of the plant chilled water (WO) system it is necessary to convert the WO chillers to the new non-CFC refrigerant, R 134a. Without any equipment modifications, the full load capacity of a WO chiller will be reduced from 1100 to 1047 tons, when operated with R 134a. As a result of Engineering Change Notice (ECN) 31969, Updated Safety Analysis Report (USAR) Change Package 9-091 provides the necessary revision on the reduction in cooling capacity for Plant Chilled Water refrigeration unit 0WO02CE. The WO system is only required to function in normal operating conditions. The WO system is not required to assure either the integrity of the reactor coolant pressure boundary or the capacity to shut down the reactor and maintain it in a safe shutdown condition. The WO system supplies chilled water to area coolers and fan-coil units in the drywell, and the containment, turbine, radwaste, fuel, and auxiliary buildings' ventilation systems. The system is non-safety related, except for components located between the containment isolation valves and drywell isolation valves. There is no failure analysis for the WO system evaluated in the USAR. This modification does not affect the seismically supported piping in the areas of seismic Category I buildings. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Design calculations and post modification testing prove that the original design of three WO units is still capable of meeting the summer cooling demand. The WO system will continue to provide an adequate quantity of chilled water to meet the cooling load requirements and maintain sufficient redundancy to ensure the power generation objective. With the exception of the change to a different refrigerant and a compatible lubricant, the operation of the 0WO02CE chiller unit remains the same as before the modification. In addition, this modification did not affect the chiller's built-in protection against freezing, high refrigerant pressure, low refrigerant pressure, high discharge temperature, motor overload, lubrication oil failure, and high motor temperature. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not govern the WO system, except for some WO valves that provide containment and drywell isolation and WO piping seismic supports. ECN 31969 does not affect these components or any safety limits associated with them. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE TO THE HIGH PRESSURE CORE SPRAY DIESEL GENERATOR COMBUSTION AIR INTAKE AND EXHAUST SYSTEM INTAKE AIR FILTER/SILENCER CAPCITY

Activity Evaluated: USAR Change 9-093 Log Number: 2000-022

Updated Safety Analysis Report (USAR) Change 9-093 revises the High Pressure Core Spray (HPCS) Diesel Generator (DG) combustion air intake and exhaust system intake air filter/silencer capacity from 10,700 cfm to 9,040 cfm. In addition, this USAR change revises the exhaust silencer capacity from 23,000 cfm each at 735°F to 19,650 cfm at 745°F. The HPCS diesel generator combustion air intake and exhaust system is not a contributor or initiator for any accident scenario evaluated in the USAR. The change in rated capacities is consistent with the design of the installed equipment and will not adversely affect equipment performance. This activity does not affect environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no components or systems being introduced into the plant, or being changed by these changes to the USAR. Thus, no credible failure modes or mechanisms are being introduced. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The component data included in USAR Table 9.5-6 was not used in the calculation or determination of any margin of safety defined in any Technical Specification. This activity does not negatively impact any Technical Specification safety limits, limiting conditions for operation, surveillance requirements, acceptance requirements, or allowed out-of-service time. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MONITORING POSITION CONTROL SIGNALS FROM REACTOR RECIRCULATION "A" FLOW CONTROL VALVE

Activity Evaluated: Temporary Modification 00-003 Log Number: 2000-024

Temporary Modification 00-003 installs test leads to existing terminal points and utilizes spare GETAR points to monitor Reactor Recirculation (RR) "A" Flow Control Valve (FCV) feedback circuitry signals. As a result, any accident that relies upon operation of the RR FCV or can be initiated by uncontrolled FCV motion could be impacted. Regardless of the exact cause of the failure, the result is the same – the FCV will open, close, or fail to move. The connection points are clearly labeled which minimizes the potential for incorrect connections to be made. The connections will be made with terminal lugs, which essentially eliminates the chance of shorts due to a temporary lead coming loose or contacting any other electrical termination point. These components are not safety-related and there are no single failure criteria or electrical separation requirements, except to ensure that the temporary installation does not impact which have these requirements. The materials selected for this activity meet the design, material, and construction standards applicable to the RR and GETAR systems. Spare GETAR cables will be used to transmit the test signal to GETAR. The jumper wiring will be constructed with materials typically used for panel wiring and exceeds the voltage and ampacity requirements. Also, no ignition sources are being added; the addition of electrical wiring to the internals of a control panel does not change the fire loading of the panel or adversely affect the fire analysis. The high impedance of the detection device ensures that there is no adverse affect to the signal being monitored. This activity has no adverse impact upon system response characteristics, will not cause system operation outside of design limits, will not cause operational transients in the system, and will not cause adverse system interaction with other systems. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The failures that could occur would not be specific to a FCV controller failure caused by the temporary modification; the failures would be no different than for any FCV controller failure. As a result, no new failure modes are indirectly caused by installation of the temporary modification. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification 3.4.1 and 3.4.2 address the RR system. Technical Specification 3.4.1 delineates requirements for matched loop flow; this activity does not adversely impact the ability to control RR FCV position and loop flows are manually controlled by the operator who selects the FCV position by monitoring flow indications. Technical Specification 3.4.2 specifies maximum FCV speeds of 30%/second maximum in the opening direction and 60%/second in the closing directions. These limits are not physically achievable by the valve hydraulics. In addition, Technical Specification 3.4.2 specifies that a FCV shall be operable in each operating loop. Since FCV reliability could be lost during installation and removal, the FCV will be considered inoperable. The FCV will be locked out during this evolution, which is consistent with the required action. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FIRE BARRIER DEVIATION FOR CB-2/A-3F PENETRATION SEAL

Activity Evaluated: ECN 30028; USAR Change 9-097 Log Number: 2000-026

Engineering Change Notice (ECN) 30028 documents a fire barrier deviation for a 2-hour fire rated penetration seal installed in the 3-hour fire rated wall between Fire Area CB-2 and Fire Zone A-3f. Updated Safety Analysis Report (USAR) Appendix F, Safe Shutdown Analysis (SSA), demonstrates that for a fire in any single plant fire area, at least one method exists that is free of fire damage to achieve and maintain a safe shutdown condition. For a fire originating in Fire Area CB-2, the estimated fire severity exceeds the estimated fire resistance rating of the penetration seal in the barrier enclosing the area. However, automatic detection and sprinkler systems are provided and, in the event of a fire in the area, the sprinkler system would control the fire and limit its severity until fire brigade action is initiated. Even if the available fire protection features became inoperable, and a fire occurred in either Fire Zone A-3f or Fire Area CB-2 breaching the barrier, only Safe Shutdown Method 2 would be lost and an alternate method, Safe Shutdown Method 3, is available for safe shutdown. While a postulated fire may have an effect on systems and components involved in the fire, safe shutdown system reliability is maintained by Safe Shutdown Method 3 continuing to be available to safely shutdown the plant. Similarly, the equipment protective features will provide adequate protection to preclude the fire from breaching the barrier until the fire is extinguished. As the same divisional systems exist on both sides of the barrier, there is no reduction of system redundancy or independence. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Regarding the introduction of a different fire hazard in either Fire Area CB-2 or Fire Zone A-3-f, whether a penetration seal is fire rated at two hours or at three hours, there is not contribution by the seal to a fire hazard. This activity does not involve any new equipment or a change to the function, operation, or test of any existing equipment or systems affected. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Clinton Power Station Fire Protection Program is not included as part of the Technical Specifications. All acceptance values and design limitations involving the Fire Protection System were previously documented in USAR Appendix E and F. There is no change to any control of systems, components, or functions as documented in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

CHANGE TO OPERATING PRESSURES/TEMPERATURES OF DG ROOM PIPING

Activity Evaluated: USAR Change 9-098 Log Number: 2000-028

Updated Safety Analysis Report (USAR) Change 9-098 changes the description of the Diesel Generator (DG) components. This package revises the operating temperature/pressure for the DG exhaust lines from 745°F/20 psig for all three diesels to 823°F/5 psig for Division I and II and 745°F/5 psig for Division III. It also revises the operating pressure for the Division III DG starting air from 240 psig to 250 psig. This change is associated with classification of piping within the DG rooms as high or moderate energy lines. According to ANSI/ANI-58.2-1980, the changes to the piping maximum operating temperatures and pressures do not result in re-classification of any lines. Since the pressure of the Division III starting air line operating pressure is still well below the piping design pressure, there is not a significant increase in the probability of this moderate energy line rupturing. The piping associated with this USAR change is not an initiator for any other accident or transient. The changes are within current design limits for the piping and will not increase the likelihood of piping failures. These changes have no effect on environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The new values are within design limits for the piping systems and do not affect the classification of this piping. This activity does not involve any new equipment or a change to the function, operation, or test of any existing equipment or systems affected. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications 3.8.1, 3.8.2, and 3.8.3 and their associated Bases address Diesel Generators; however, the values being revised were not used in the calculation or determination of any margin of safety. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MINOR DOCUMENTATION DISCREPANCIES

Activity Evaluated: ECN 32109; USAR Change 9-104 Log Number: 2000-030

Engineering Change Notice (ECN) 32109changes two Service Building basement sprinkler control valves from "open" to "locked open" and revises plant drawings, Fire Protection (FP) system description, valve and equipment lists, instrument data sheets, and the Updated Safety Analysis Report (USAR) to make these documents consistent with the plant configuration. These USAR revisions are documentation corrections associated with systems and equipment in non-safety related areas, with the exception of the Fuel Building Railroad Bay Automatic Preaction System. The Fuel Building revision only changes the suppression system designator from Deluge Preaction System (DPS) to Automatic Preaction System (APS). The ECN also makes a change to the Service Building basement sprinkler control valves at the request of Nuclear Electric Insurance Limited (NEIL) to be consistent with other plant sprinkler control valves. These changes will not increase the fire hazards in the areas affected, or compromise the capability to perform a safe shutdown. This activity does not affect the design, material, or construction standards of the FP system. In addition, the overall FP system performance and effectiveness will not be affected by these changes. This activity does not involve a change to the function, operation, or testing of any equipment or systems affected. There is no new impairment to safe shutdown system reliability, degradation of equipment protective features or system performance, or reduction of system redundancy or independence due to this activity.

Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. The Clinton Power Station Fire Protection Program is not included as part of the Technical Specifications. All acceptance values and design limitations involving the Fire Protection System were previously documented in USAR Appendix E and F. There is no change to any control of systems, components, or functions as documented in the Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

SAFETY RELATED AIR REGULATORS QUALITY CLASSIFICATION CHANGE

Activity Evaluated: USAR Change 9-061 Log Number: 2000-031

Updated Safety Analysis Report (USAR) Change 9-061 revises the quality group classification for certain Instrument Air (IA) Regulators from "C" to "N/A" in Table 3.2-1. The IA regulators affected by this USAR Change are on non-safety related lines. There is a specific event discussed for the Loss of Instrument Air, but it states that the cause is a break in a major line. Since these instrument air regulators are not on major lines and these IA regulators are the same as many installed in non-safety applications throughout the plant, this activity does not increase the probability of Loss of IA. The equipment served by these IA regulators is designed to fail in the safe position and this design is unchanged by this activity. The functions of the equipment and methods of performing these functions are unchanged by this activity. The failure modes associated with the regulators are the same whether they are classified as "C" or "N/A". Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not involve any new equipment or a change to the function, operation, or test of any existing equipment or systems affected. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. There are no specified acceptance values established for the Control Room Air Conditioning system, Standby Gas Treatment system, and the Common Station Heating, Ventilation, and Air Conditioning Exhaust Hi-Range Radiation Monitors that would be affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

AOV INSTRUMENT AIR SUPPLY LINE MODIFICATION

Activity Evaluated: ECNs 30391, 30392, 30393, 30394, and 30395 Log Number: 2000-033

This activity installs an isolation valve, tee and quick disconnect between the solenoid valve and air operated valve (AOV) diaphragm operator in the Instrument Air (IA) system tubing. All changes are to mechanical components; this activity does not modify or delete instrumentation, electrical systems, or power sources. This activity will not cause the affected systems or components to operate outside of the design or testing limits. A new test interface is created, but the interface is only functional during the AOV Flow Scan testing and at that the AOV is classified as inoperable for the length of the test. Therefore, the new connection made to the AOVs does not affect the valves operating function or ability to perform that function except during AOV testing. This activity does not affect environmental, seismic, or separation criteria.

The original design criteria of the AOV air supply system is applied to the new components installed by this activity. Hence, the design standards are maintained and have not changed as a result of this activity. The material associated with this change will be of equal or better quality than originally installed. This activity does not require the addition of any operator actions, nor does the activity modify or delete any existing operator actions assumed in the accident analysis for mitigating the effect of an accident or transient. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The possible malfunctions that could be caused by implementation of this activity and several improbable malfunctions have been investigated and have been found to be bounded by the accident/transient analysis contained in the USAR. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not negatively impact any Technical Specification safety limits, limiting conditions for operation, surveillance requirements, acceptance requirements, or allowed out-of-service time. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CORRECTIONS TO DRYWELL PENETRATION LISTS

Activity Evaluated: ECN 32132; USAR Change 9-108 Log Number: 2000-034

Updated Safety Analysis Report (USAR) Change 9-108 corrects the descriptions and line sizes in USAR Table 3.8-5 for some Drywell Penetrations. The function of the penetration sealing mechanism is to maintain drywell integrity during accident/events which require it. The penetrations are passive components that cannot act as accident initiators and are not described as such in the accident analyses in USAR Chapters 6 and 15. Since the functions of these lines are being changed such that there will not be fluids present, this change does not increase the probability of flooding, nor does it affect accident initiating systems. The spare penetrations were designed in accordance with the appropriate codes and standards to maintain structural integrity. The spare piping stubs were designed in accordance with the appropriate division ASME Section III. The spare penetrations are designed, tested and fabricated to the same criteria as those previously evaluated. Since these penetrations are designed, tested and fabricated to the same criteria as those previously evaluated structural and pressure integrity of the drywell will be maintained. The drywell will continue to perform its function of mitigating accidents. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since the functions of these lines are being changed such that there will not be fluids present, this change does not introduce the possibility of new flooding events that were not previously considered. The penetrations that are changed to spare do not add any components to the plant. The penetrations, spare piping stubs and electrical conduit are designed, tested, and fabricated to the same standards as previously evaluated. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Maintaining drywell integrity, pressure and temperature is addressed by Technical Specifications 3.6.5.1 through 3.6.5.6. The configuration of the spare penetrations meets the requirement in 3.6.5.1 that states, "The drywell penetrations required to be closed during accident conditions are either: (1) capable of being closed by an OPERABLE automatic drywell isolation valve, or (2) closed by a manual valve, blind flange, or de-activated automatic valve secured in the closed position except as provided in LCO 3.6.5.3, 'Drywell Isolation Valves'". Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALL JUMPER TO BYPASS CYCLED CONDENSATE TRANSFER PUMP TRIP

Activity Evaluated: Temporary Modification 00-009 Log Number: 2000-035

Temporary Modification 00-009 installs a jumper to prevent trip of the Cycled Condensate (CY) Transfer Pumps (A, B, C) due to low level in the CY Tank. The event Loss of Feedwater (FW) could be related to the loss of the CY transfer pump. The initiator for this event is a loss of all feedwater pumps and bounds any impact on condensate as a result of CY transfer pumps. Disabling the CY transfer pump trip has no impact on FW pump failure. If CY is not available, Operator action is taken to overcome the restraint or the plant is shutdown. The result of loss of the CY transfer pump, whether due to trip on low tank level or due to pump failure from cavitation, is the same. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. The CY system is not mentioned in the Technical Specifications, because it does not perform any safety function. Failure of the CY transfer pumps does not affect Technical Specifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

PLANT CHILLED WATER SYSTEM WATER CHEMISTRY UPGRADE AND RECORD OF VALVE POSITIONING

Activity Evaluated: CPS 2800.14 & C001 Revision 0 Log Number: 2000-036

Clinton Power Station (CPS) procedure 2800.14, "WO System Water Chemistry Upgrade," and associated Checklist 2800.14C001, "Record of Valve Positioning," provide instruction to reduce the suspended corrosion products and impurities from the Plant Chilled Water (WO) system while the system remains in service. The WO system is non-safety related, except for components located between the containment isolation valves and drywell isolation valves. A review of Updated Safety Analysis Report (USAR) Chapters 6 and 15 indicates that the various ventilation units supplied by the WO system are not assumed initiators for any evaluated accident. The WO piping components are seismically supported in Seismic Category I buildings to preclude damage to safety related equipment. Since the drains are continually monitored while open, there is no increase in the probability of flooding due to overflow of the drain system. Compliance with this new procedure and constantly monitoring the compression tank and open drains, reasonable assurance is provided that the level and pressure in the compression tank will be maintained high enough to prevent loss of WO water flow due to loss of pump suction head. The location and size of the drains, which may be opened simultaneously, were selected to ensure that the drain flow rate would not adversely impact the operation of the chilled water pumps or the chillers. Portions of the system located within the Primary Containment and Drywell automatically isolate in the event of a Loss-of-Coolant Accident; therefore, no new release path is expected to occur and radioactive contamination is not expected to increase. Failure of the WO system does not compromise any safety-related system or component and does not prevent a safe reactor shutdown. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure of the WO system does not compromise any equipment important to safety and this activity does not create any new interaction mechanisms. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The isolation capability described in the Operational Requirements Manual are unaffected by the feed and bleed operations in this procedure. The functionality of the Containment Isolation System is unaffected by this activity. This activity will not affect the functional capability of the containment isolation valves or the leak detection system to perform their intended safety function and will not negatively impact any margin of safety. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FUNCTIONAL VERIFICATION OF THE HIGH PRESSURE CORE SPRAY DIESEL GENERATOR SYNCHRONIZING CIRCUIT

Activity Evaluated: CPS 2800.08 Revision 0 Log Number: 2000-037

Clinton Power Station (CPS) Procedure 2800.08, Diesel Generator 1C Synchronizing Circuit Functional Check, is a special test for backfeeding the Division 3 High Pressure Core Spray (HPCS) Diesel Generator (DG) synchronizing circuit to verify its functionality. This is a one-time test that will be performed on the Division 3 Diesel Generator with the affected DG output breaker closed while the diesel is out of service and its power cables disconnected. This special test procedure will be cancelled after the test has been completed. HPCS is required to perform safety-related functions to mitigate certain accident scenarios. This test will be conducted while the HPCS is in a 14-day Limiting Condition for Operation (LCO) in accordance with Technical Specifications, and therefore, HPCS is not required to be available during the test period. The special test configuration will assure that interfacing systems and interlocks that have the potential for creating new or different failure modes or to degrade the performance of protected systems and their support systems below their design basis are tagged out of service, jumpered, or removed from service. The DG will be fed from an offsite power source different than that used for Division 1 and 2 systems, thereby assuring that there is no interaction between the test and other safety systems. All design, material, and construction standards applicable to the HPCS system will be used for the special test. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). During this test there is a potential for two failure modes to occur: the potential for a fault due to the backfeed and the potential for the closed DG output breaker interlocks to impact other systems. The potential for a fault to the Division 3 DG synchronizing circuit while being backfed by offsite power is negated by maintaining an existing differential protective relay that will trip the DG output breaker and protect other safety-related systems should a fault occur. In regard to the output breaker interlocks, the test procedure provides for tagging out the Division 3 DG engine, generator control, and 5kV regulating circuit; jumpering the Division 3 DG speed and voltage relay; removing protective relays from service; and lifting leads on Division 3 DG support systems that would, under normal operating conditions, open the breaker. With the Division 3 DG output breaker closed for the special test, a Loss of Offsite Power (LOOP) during the special test would serve to close an already closed breaker, and since this test is conducted with the HPCS already inoperable, there would be no affect on the LOOP scenario. With these features in place, this activity does not introduce any failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The safety system involved with this activity is HPCS, with its supporting Division 3 Diesel Generator, which is already inoperable in a 14-day LCO in accordance with Technical Specification requirements. These systems are controlled by the Technical Specifications and are not required to be available during the special test. No other Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation are affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MODIFY ULTRASONIC RESIN CLEANER

Activity Evaluated: ECN 32103 Log Number: 2000-038

Engineering Change Notice (ECN) 32103 removes the flange cover bolts and installs a hinge, handle, and latches on the Ultrasonic Resin Cleaner (URC) access lid, installs a basket strainer, and installs a drain line that connects to the bottom of the URC. This activity does not impact any accidents analyzed in Updated Safety Analysis Report (USAR) Chapters 6 and 15. It will not affect any structure, system or component (SSC) that is required to mitigate an accident or required to safely shutdown the reactor. The integrity and reliability of the URC will be maintained. The drain line and valve are being installed to the original piping specifications. This activity will not affect overall system performance of the Condensate Polishing (CP) system, such that it changes system response characteristics, causes system operation outside of its design limits, or causes operational transients in the system or adverse system interaction. The drain line does not introduce new release paths or increase release rates due to its failure. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The new drain line will not create any new failure modes and will not affect system response characteristics or performance. The failure of the new drain line or the action of an operator leaving the drain valve in the open position is bounded by previous analysis. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The CPS system is not addressed in the Technical Specifications. This is a non-safety related maintenance activity that installs a drain line to a non-safety related system. The new drain line does not adversely impact the margin of safety or safety limits to any SSC that is required for the safe shutdown of the reactor. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

POST MAINTENANCE TESTING FOR THE REPLACEMENT DIVISION 3 HIGH PRESSURE CORE SPRAY DIESEL GENERATOR

Activity Evaluated: CPS 2808.02 Revision 0 Log Number: 2000-039 R/1

This activity addresses a special test procedure, Clinton Power Station (CPS) Procedure 2808.02, Diesel Generator 1C 24 Hour Run and Hot Restart, which verifies that the Diesel Generator will start on a manual start signal and that it will reach the required voltage and frequency. High Pressure Core Spray (HPCS) is required to perform safety-related functions to mitigate certain accident scenarios. This test will be conducted while the HPCS is in a 14-day Limiting Condition for Operation (LCO) in accordance with Technical Specifications, and therefore, HPCS is not required to be available during the test period. The special test configuration will assure that existing interfacing systems and interlocks are in place, and there will be no potential for creating new or different failure modes or to degrade the performance of protected systems and their support systems below their design basis. The Diesel Generator (DG) 1C will be fed from an offsite power source different than that used for Division 1 and 2 systems, thereby assuring that there is no interaction between the test and other safety systems. All design, material, and construction standards applicable to the HPCS system will be used for the special test. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). During this test there is a potential for two failure modes to occur: the potential for a fault in the DG 1C system and the potential for the closed DG 1C output breaker interlocks to impact other systems. The potential for a fault in the DG 1C system is negated by maintaining an existing differential protective relay that will trip the DG 1C output breaker and protect other safety-related systems should a fault occur. In regard to the output breaker interlocks, the test procedure provides for having all normal interlocks in place and functioning. With the DG 1C output breaker closed for the special test, a Loss of Offsite Power (LOOP) during the special test would serve to close an already closed breaker, and since this test is conducted with the HPCS already inoperable, there would be no affect on the LOOP scenario. With these features in place, this activity does not introduce any failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The safety system involved with this activity is HPCS, with its supporting Division 3 Diesel Generator, which is already inoperable in a 14-day LCO in accordance with Technical Specification requirements. These systems are controlled by the Technical Specifications and are not required to be available during the special test. No other Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation are affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE IN REPORTING RESPONSIBILITIES

Activity Evaluated: USAR Change 9-125 Log Number: 2000-041

Updated Safety Analysis Report (USAR) Change 9-125 changes the reporting relationship for the Director – Projects/Contracts from the Site Vice President to the Manager – Outage Management and for the protected area and owner controlled area facilities groups from the Director – Plant Support Services to the Director – Work Support. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.0 addresses the management responsibilities/requirements of the Plant Manager and Operations personnel and Technical Specifications 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

RESIDUAL HEAT REMOVAL/LOW PRESSURE CORE SPRAY KEEP FILL SYSTEM UPGRADE

Activity Evaluated: Modification M-084 Supplements 0 and 1; USAR Changes 9-131 and 9-132 — Log Number: 2000-042

Modification M-084 Supplements 0 and 1 involve various upgrades to the Residual Heat Removal/Low Pressure Core Spray (RHR/LPCS) Keep Fill Systems, including piping and pump changes. Pipe leak/break and pump failure are the two credible failure modes associated with this activity. These failure modes are not initiating events for any of the accidents described in Chapters 6 and 15 of the Updated Safety Analysis Report (USAR). The higher head water leg pumps increase the maximum operating and design pressure of a portion of the pumps' discharge piping. However, the piping, valves, and components affected by the higher pressures are capable of operating as intended at the higher pressure. The modified piping has been analyzed to ensure pipe stresses and support loads are acceptable. The increased design pressure does not change the "moderate energy" classification of these lines. This modification meets the original design, material and construction standards. The RHR and LPCS are designed to mitigate the consequences of loss of coolant accidents. This modification improves the capability of the keep fill systems to keep the LPCS and RHR piping full of water and helps ensure that LPCS and RHR can perform their design basis functions. In addition, this modification does not degrade system performance and does not reduce system redundancy or independence. The revised diesel generator loading is still within the 2000-hour rating of the diesel generators and meets the guidelines provided in Regulatory Guide 1.9. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This modification does not introduce any new failure modes or mechanisms, and a break in the modified piping would be no different than a break in the existing piping. The increased power demand on the electrical distribution system and the fuel oil system has been evaluated and found to be within the capacity of these systems. Also, this modification does not introduce any new active components to the plant. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The ability of the RHR and LPCS systems to meet their Technical Specification and design requirements will not be impacted by this activity. The Technical Specification Bases specify that the diesel generators satisfy the requirements of Regulatory Guide 1.9 and the continuous service rating of each of the diesel generators is given. The Technical Specifications state that the diesel generator must be capable of accepting required loads within the assumed loading sequence intervals and must continue to operate until offsite power can be restored to the ESF buses. These Technical Specifications are still valid are not affected by this activity. Technical Specification 3.8.3 and its associated Bases address the minimum fuel oil storage requirements for the diesel generators. The fuel oil storage capacity required by the Technical Specifications has sufficient margin to accommodate the small change in fuel oil consumption due to the pump motor replacement. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ADD RELIEF VALVE TO CONTAINMENT ISOLATION PIPING

Activity Evaluated: ECN 30072 Log Number: 2000-044

Generic Letter 96-06 identified that piping systems that penetrate the containment may be susceptible to overpressure due to the thermal expansion of fluid that is heated. To prevent this overpressurization, Engineering Change Notice (ECN) 30072 adds a relief valve to the containment penetration piping between the containment isolation valves of penetration 1MC-065. Containment isolation and the Solid Radwaste Reprocessing and Disposal (WX) system are not initiators of any design basis accidents. The changes to the piping meet the same design, material, and construction standards as the original piping. Containment integrity is required to limit dose to the control room and the general public. The addition of the relief valve provides overpressure protection for the penetration piping, thereby decreasing the likelihood of a containment penetration failure. The Updated Safety Analysis Report (USAR) is revised to allow testing of the relief valve in the reverse direction, which tends to lift the relief valve, which is a conservative leakage test. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The only design function for containment penetration is containment isolation during accident conditions. Failure of a containment penetration would therefore not result in any accident. The addition of the relief valve and piping is performed to the same quality standards as other containment piping and valves. The valve is also included in the IST testing program and will be periodically tested to demonstrate that the valve will function as intended. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification, Limiting Condition for Operation 3.6.1.1 states: "Primary containment shall be OPERABLE". This activity does not adversely affect the containment isolation function. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ADD RELIEF VALVE TO CONTAINMENT ISOLATION PIPING

Activity Evaluated: ECN 30076 Log Number: 2000-045

Generic Letter 96-06 identified that piping systems that penetrate the containment may be susceptible to overpressure due to the thermal expansion of fluid that is heated. To prevent this overpressurization, Engineering Change Notice (ECN) 30076 adds a relief valve to the containment penetration piping between the containment isolation valves of penetration 1MC-050. Containment isolation and the Make-up Condensate Storage (MC) system are not initiators of any design basis accidents. The changes to the piping meet the same design, material, and construction standards as the original piping. Containment integrity is required to limit dose to the control room and the general public. The addition of the relief valve provides overpressure protection for the penetration piping, thereby decreasing the likelihood of a containment penetration failure. The Updated Safety Analysis Report (USAR) is revised to allow testing of the relief valve in the reverse direction, which tends to lift the relief valve, which is a conservative leakage test. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The only design function for containment penetration is containment isolation during accident conditions. Failure of a containment penetration would therefore not result in any accident. The addition of the relief valve and piping is performed to the same quality standards as other containment piping and valves. The valve is also included in the IST testing program and will be periodically tested to demonstrate that the valve will function as intended. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. Technical Specification, Limiting Condition for Operation 3.6.1.1 states: "Primary containment shall be OPERABLE". This activity does not adversely affect the containment isolation function. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE UPDATED SAFETY ANALYSIS REPORT TO DELETE QUALITY ASSURANCE SUPERVISORY POSITIONS

Activity Evaluated: USAR Change 9-073 Log Number: 2000-047

Updated Safety Analysis Report (USAR) Change 9-073 removes the four Quality Assurance (QA) supervisory positions from USAR Figure 13.4-1 and replaces them with one block labeled Quality Assurance staff. This change is to indicate that all QA personnel report to the Clinton Power Station Quality Assurance Manager. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.0 addresses the management responsibilities/requirements of the Plant Manager and Operations personnel and Technical Specifications 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

SVC MANUAL FREEZE SWITCH TEST

Activity Evaluated: CPS 2808.03 Revision 0 Log Number: 2000-048

This activity addresses a new special test procedure Clinton Power Station (CPS) Procedure 2808.03, "LOCA Signal Response on Div 1 ERAT SVC Freeze Test." This procedure tests the capability of a manual freeze switch installed in the freeze logic for the Static Var Compensator (SVC) and the capability of a Loss of Coolant Accident (LOCA) signal to automatically remove a manually initiated freeze. This is a one-time test and will be cancelled upon completion of the test. This test will be conducted while the Emergency Reserve Auxiliary Transformer (ERAT) and Division 1 Diesel Generator are in a Limiting Condition for Operation (LCO) and are not required to be available during the test period. The ERAT will be fed from an offsite power source different than that used for Division 2 and 3 systems, thereby assuring that there is no interaction between the test and other safety systems. The test configuration will allow Division 1 to operate normally with normal protective circuits in operation to isolate on a fault. This will assure that a fault in the Division 1 bus will not affect the offsite power source feeding other safety related systems. During the test, the ERAT SVC will be in the freeze mode and grid voltage is expected be relatively stable for the short duration of the test. If grid voltages are considered unstable, the test will be stopped and the SVC function will be restored. Therefore, the test will not affect overall system response characteristics, cause system operation outside of its design limits, cause operational transients in the system, or cause adverse system interaction with other systems. All design, material, and construction standards applicable to the Division 1 system will be used for the special test. Also, the special test configuration will assure that existing interfacing systems and interlocks are in place, and there will be no potential to degrade the performance of protected systems and their support systems below their design basis or to affect their protective features or their environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). During this test there are two potential failure modes that are introduced: the potential for a fault in the Division 1 system and the potential for the closed Division 1 interlocks to impact other systems. The potential for a fault in the Division 1 system is negated by maintaining an existing differential protective relay that will trip the Division 1 breaker and protect other safety-related systems should a fault occur. In regard to the output breaker interlocks, the test procedure provides for having all normal interlocks in place and functioning. With Division 1 configured for the special test, a LOCA or Loss of Offsite Power (LOOP) occurring during the special test is provided for with the LOCA signal circuitry for the manual freeze switch, and the breakers in place for a LOOP. With these features in place, this activity does not introduce any failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The ERAT and Division 1 DG are already inoperable in a LCO in accordance with Technical Specification requirements. These systems are controlled by the Technical Specifications and are not required to be available during the special test. No other Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation are affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DISABLING OF "D" CHANNEL INPUT TO MAIN STEAM LINE HIGH RADIATION ALARM 5067-3F

Activity Evaluated: Temporary Modification 00-013 Log Number: 2000-049

Temporary Modification 00-013 disables channel "D" divisional input to the Main Steam Line (MSL) Radiation Monitor high radiation annunciator in the Main Control Room (MCR). The purpose of this monitor is to provide early indication of possibly damaged fuel or reactor water chemistry problems allowing higher than normal levels of radioactive material to be carried by the plant's main steam supply. The MSL radiation monitoring system does not directly communicate with any equipment or process variable that could initiate an accident. The circuit affected by this change is isolated from the trip part of the circuit by an isolator assembly. The open socket relay that it is removed from would have no exposed terminals or wires to short to anything. Seismic, environmental and separation criteria are not affected by this activity. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no Technical Specifications that address the Main Steam Line Radiation

Monitoring Function. The Reactor Protection System is covered by Technical Specification section 3.3.1.1, but it does not mention the Main Steam Line Radiation Monitoring Function Trips. However, this function is found in Operational Requirements Manual section 2.2.17, which specifies function, setpoint, and allowable values. It also specifies Limiting Conditions for Operation requirements for inoperable channels. Revising the alarm circuits as specified in this temporary modification will in no way impact any values or operability requirements for the trip portion. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

TURBINE FIRST STAGE PRESSURE SENSING PROTECTIVE TRIP FUNCTIONS DURING BYPASS VALVE TESTING

Activity Evaluated: TS Bases Change BL-99-031 Log Number: 2000-050

Technical Specifications (TS) Bases Change BL-99-031 clarifies wording that verification of protective trip functions are not bypassed during the time that main turbine bypass valves are not closed. Specifically, the Reactor Protection System (RPS) and End of Cycle – Recirculation Pump Trip (EOC – RPT) functions of Turbine Stop Valve Closure and Turbine Control Valve Fast Closure are not bypassed during the time a main turbine bypass valve is open. Chapter 15 of the Updated Safety Analysis Report (USAR) contains analyses of transients involving bypass valves. This activity does not affect these analyses since the functions of the bypass valves or protective trips are not being changed. There are no changes to be made to plant operating procedures as a result of this activity, and as such, no new transient initiators or changes to initiating events to design basis accidents are created. There is no change to the system hardware, parameters monitored, or the range that is being monitored. There is no change to the response characteristics or function of the bypass valves or protective trip functions or how they are used. There is no change to the operation and control of the affected systems; therefore, there is no change to the availability of the systems or degradation of the performance of the affected systems. This activity does not alter the design, material, or construction standards applicable to the affected systems. Malfunctions or failures of the turbine bypass valves or the protective trip functions have been evaluated to not cause a failure that would result in a release of radioactive material. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not challenge or exceed design or operational limits of the affected systems. All equipment qualifications of the affected systems will remain unaltered. This activity will not, directly or indirectly, impact any plant structure, system, or component from performing its safety function. There is no equipment that will be operated differently or change to the testing methods or requirements as a result of this activity. There is no new equipment or changes to existing equipment that would cause any new failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not change acceptance criteria for the applicable Technical Specifications. The Technical Specification Surveillance Requirements, Actions, and Limiting Conditions for Operations of these sections are not affected by this activity. There are no changes to design limitations or requirements. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

SVC MANUAL FREEZE SWITCH TEST

Activity Evaluated: CPS 2808.04 Revision 0 Log Number: 2000-051

This activity addresses a new special test procedure Clinton Power Station (CPS) Procedure 2808.04, "LOCA Signal Response on Div II ERAT SVC Freeze Test." This procedure tests the capability of a manual freeze switch installed in the freeze logic for the Static Var Compensator (SVC) and the capability of a Loss of Coolant Accident (LOCA) signal to automatically remove a manually initiated freeze. This is a one-time test and will be cancelled upon completion of the test. This test will be conducted while the Emergency Reserve Auxiliary Transformer (ERAT) and Division 2 Diesel Generator are in a Limiting Condition for Operation (LCO) and are not required to be available during the test period. The ERAT will be fed from an offsite power source different than that used for Division 1 and 3 systems, thereby assuring that there is no interaction between the test and other safety systems. The test configuration will allow Division 2 to operate normally with normal protective circuits in operation to isolate on a fault. This will assure that a fault in the Division 2 bus will not affect the offsite power source feeding other safety related systems. During the test, the ERAT SVC will be in the freeze mode and grid voltage is expected be relatively stable for the short duration of the test. If grid voltages are considered unstable, the test will be stopped and the SVC function will be restored. Therefore, the test will not affect overall system response characteristics, cause system operation outside of its design limits, cause operational transients in the system, or cause adverse system interaction with other systems. All design, material, and construction standards applicable to the Division 2 system will be used for the special test. Also, the special test configuration will assure that existing interfacing systems and interlocks are in place, and there will be no potential to degrade the performance of protected systems and their support systems below their design basis or to affect their protective features or their environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). During this test there are two potential failure modes that are introduced: the potential for a fault in the Division 2 system and the potential for the closed Division 2 interlocks to impact other systems. The potential for a fault in the Division 2 system is negated by maintaining an existing differential protective relay that will trip the Division 2 breaker and protect other safety-related systems should a fault occur. In regard to the output breaker interlocks, the test procedure provides for having all normal interlocks in place and functioning. With Division 2 configured for the special test, a LOCA or Loss of Offsite Power (LOOP) occurring during the special test is provided for with the LOCA signal circuitry for the manual freeze switch, and the breakers in place for a LOOP. With these features in place, this activity does not introduce any failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The ERAT and Division 2 DG are already inoperable in a LCO in accordance with Technical Specification requirements. These systems are controlled by the Technical Specifications and are not required to be available during the special test. No other Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation are affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF 0WO670 INTO THE PLANT CHILLED WATER SYSTEM AIR SEPARATOR DRAIN LINE

Activity Evaluated: ECN 32031 Log Number: 2000-052

Engineering Change Notice (ECN) 32031 installs isolation valve 0WO670 into the Plant Chilled Water (WO) system Air Separator drain line. The WO system is a non-safety related system except for components located between the containment isolation valves and drywell isolation valves. The credible failure modes for the drain line and isolation valve are structural failure of the valve or drain piping to a seismic event, functional failure of the valve internal components to operate as designed, weld defects, or loss of freeze seal. Neither the valve installation, nor the freeze plug installation will affect the frequency of occurrence of design basis accidents or transients because the failure of both installations combined, or one installation alone, does not produce the results required for accident initiation. Failure of the drain line or any of its components to perform its design basis function will not compromise any safety-related system or prevent the safe shutdown of the reactor. The new valve meets the same design, construction, and material standards as the rest of the system. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The possible failure modes associated with this activity have been evaluated in the Flood Analysis and seismic event analysis and were found to have no effect on the accident analysis. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications or its Bases do not address the WO system. There are no Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE UPDATED SAFETY ANALYSIS REPORT 8.2.1.1 WORDING FOR 12KV SUBSTATION MODES OF OPERATION

Activity Evaluated: CPS 3505.02 Revision 12; USAR Change 9-146 Log Number: 2000-053

This activity updates the alignment of the 12kV substations to allow for one electrode boiler to be supplied from either the SCAB transformer or the construction transformer and recognizes the cross-connection capability of the 12kV substations. Updated Safety Analysis Report (USAR) Subsection 15.2.6 discusses loss of Alternating Current (AC) power, including loss of grid. Nuclear Station Engineering Department Instruction EE-6 identifies those items, which have the potential to disturb the off-site AC electrical sources. The 12kV system has been evaluated as an input parameter for maintaining grid stability compliance with General Design Criteria 17 and acceptable range of operation and the proper setpoints for the Diesel Generator second level undervoltage relays. The 12kV system is modeled as two loads on a common 138 kV bus, thus only the total of the two loads is of concern, regardless of its substation configuration. Therefore, the loading and configuration of the 12kV distribution system will not change the previous response analysis. The specific limitation for only running one electrode boiler for compliance with USAR accident analysis has been retained and reinforced. The 12kV substation loading capability and response characteristics have not changed. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

STATIC VAR COMPENSATOR MANUAL FREEZE SWITCH TEST

Activity Evaluated: CPS 2808.05 Revision 0 Log Number: 2000-054

This activity addresses a new special test procedure Clinton Power Station (CPS) Procedure 2808.05, "LOCA Signal Response on Div III ERAT SVC Freeze Test." This procedure tests the capability of a manual freeze switch installed in the freeze logic for the Static Var Compensator (SVC) and the capability of a Loss of Coolant Accident (LOCA) signal to automatically remove a manually initiated freeze. This is a one-time test and will be cancelled upon completion of the test. This test will be conducted while the Emergency Reserve Auxiliary Transformer (ERAT) and Division 3 Diesel Generator are in a Limiting Condition for Operation (LCO) and are not required to be available during the test period. The ERAT will be fed from an offsite power source different than that used for Division 1 and 2 systems, thereby assuring that there is no interaction between the test and other safety systems. The test configuration will allow Division 3 to operate normally with normal protective circuits in operation to isolate on a fault. This will assure that a fault in the Division 3 bus will not affect the offsite power source feeding other safety related systems. During the test, the ERAT SVC will be in the freeze mode and grid voltage is expected be relatively stable for the short duration of the test. If grid voltages are considered unstable, the test will be stopped and the SVC function will be restored. Therefore, the test will not affect overall system response characteristics, cause system operation outside of its design limits, cause operational transients in the system, or cause adverse system interaction with other systems. All design, material, and construction standards applicable to the Division 3 system will be used for the special test. Also, the special test configuration will assure that existing interfacing systems and interlocks are in place, and there will be no potential to degrade the performance of protected systems and their support systems below their design basis or to affect their protective features or their environmental, seismic, or separation criteria. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). During this test there are two potential failure modes that are introduced: the potential for a fault in the Division 3 system and the potential for the closed Division 3 interlocks to impact other systems. The potential for a fault in the Division 3 system is negated by maintaining an existing differential protective relay that will trip the Division 3 breaker and protect other safety-related systems should a fault occur. In regard to the output breaker interlocks, the test procedure provides for having all normal interlocks in place and functioning. With Division 3 configured for the special test, a LOCA or Loss of Offsite Power (LOOP) occurring during the special test is provided for with the LOCA signal circuitry for the manual freeze switch, and the breakers in place for a LOOP. With these features in place, this activity does not introduce any failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated. The ERAT and Division 3 DG are already inoperable in a LCO in accordance with Technical Specification requirements. These systems are controlled by the Technical Specifications and are not required to be available during the special test. No other Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation are affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLING A FREEZE SEAL AND TEMPORARY PIPE SUPPORTS FOR THE REPLACEMENT OF VALVE 1FC026A

Activity Evaluated: MWO D31038 Log Number: 2000-055

Maintenance Work Order D31038 installs a freeze seal in order to replace valve 1FC026A, while the Fuel Pool Cooling and Clean-up (FC) system is in-service. Flow to the FC "A" Heat Exchanger is isolated during the performance of this activity by closing inlet "A" Heat Exchanger isolation valve. Updated Safety Analysis Report (USAR) Chapters 6 and 15 do not address an accident analysis due to the loss of the Spent Fuel Pool Cooling System. The USAR takes credit for other interconnected systems to be available and the pools have been designed so that water level will be maintained to cover the fuel at all times. The FC system will be available and inservice during the implementation of this activity. The spent fuel pool is designed so that no single failure of structures or equipment will cause inability to maintain irradiated fuel submerged in water; or to establish normal fuel pool water level; or to remove fuel safely from the station. The installation of a freeze seal involves neither combustibles nor ignition sources. The failure of the freeze seal can lead to flooding, which is anticipated in the governing freeze sealing procedure. To make the risk of freeze seal failure acceptable, the installation of a freeze seal is part of a troubleshooting and repair plan and is controlled via the maintenance process. The freeze seal procedure requires that a checklist be prepared. The checklist requires a plan that includes the identification of compensatory measures in the event of a freeze seal failure. This activity does not alter the valve's function or performance or change system response characteristics. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. There are no credible failure mechanisms being introduced during this activity. Installing a freeze seal will not prevent Control Room personnel from monitoring annunciation, indication, and instrumentation that is affected by the system. Freeze seal activities will be installed and monitored by the appropriate site procedures. A design change is not required because the original design parameters will not change, and the system will be restored to its original design condition. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The margin of safety as defined in Technical Specification Bases section 3.6.2.4 is the temperatures between 120°F and fuel cladding failure temperature of 2200°F, and the minimum required volume of water needed for suppression pool makeup, which is greater than or equal to 14, 652 cubic feet. Equipment being replaced during this activity is an exact replacement; therefore, flow performance to the upper pools will not change. The FC system will be available and inservice to spent fuel pool during the entire implementation and surveillance requirements to verify temperature will not be affected. In addition, a contingency plan is in force if the freeze seal would fail. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REMOVAL OF RESERVE AUXILIARY TRANSFORMER MINIMUM LOAD RESTRICTION/ALARM

Activity Evaluated: ECN 32222; USAR Change 9-152 Log Number: 2000-057

Engineering Change Notice (ECN) 32222 removes the Reserve Auxiliary Transformer (RAT) 4 kV low load alarm from the Main Control Room annunciator and restores the Unit Auxiliary Transformers (UATs) as the normal power source for the non-safety related loads when the main generator is on-line. Updated Safety Analysis Report (USAR) Section 15.2.6, Loss of AC Power, describes the design basis accident related to the Auxiliary Power (AP) system power sources. Removal of minimum loading restriction from the RAT and designating the UATs as primary source of power for the Balance of Plant (BOP) buses do no act as initiators for Loss of AC Power. Calculations demonstrate that a transient overvoltage condition, caused by a RAT Static VAR Compensator (SVC) trip condition with a high 345 kV grid voltage, will not cause failure of the Class 1E components. Alarm and system operating procedures will be revised to reflect required operator actions needed to maintain, restore and recover from an unlikely safety bus overvoltage event as required to maintain system availability and operability. This activity does not affect overall AP system performance and does not impact design, material, and construction standards of any plant system. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. The margin of safety associated with degraded voltage is ensuring adequate voltage to plant safety loads. The minimum voltage required to avoid a transfer from offsite power to the diesel generators is defined by the degraded voltage allowable values in Technical Specification Table 3.3.8.1-1. The degraded voltage relay setpoint is not affected by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

NONESSENTIAL PIPING AND NON-CATEGORY I CONDUIT

Activity Evaluated: USAR Change 9-141 Log Number: 2000-060

Updated Safety Analysis Report (USAR) Change 9-141 revises USAR Section 3.2.1 concerning the seismic category of conduit and associated supports in the Lab Area of the Control Building. The revision identifies that there are some Category I supports and Class 1E conduits in this area. The reclassification of conduit and associated supports is not involved in the initiation of any accident previously evaluated in the USAR since the change provides for a more conservative seismic design of conduit supports which thereby precludes the initiation of an accident. This activity does not affect overall system performance and does not cause adverse system interactions with other systems. Also, this activity does not affect the environmental or separation criteria of any safety system important to safety, nor does it degrade system performances or reduce system redundancy or independence. The activity does affect the seismic criteria, however, the performance is not degraded below the design basis since the seismic criteria is more conservative than originally stated in the USAR. In addition, calculation DC-ME-17-CP indicates that those cable tray supports, conduit supports, bus duct supports and lighting system support attachments that are not designed seismically are analyzed to verify that the ability of safety-related systems and components to perform their safety function will not be impaired by the failure of these supports. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Since the seismic supports and conduit are more conservatively designed than before, the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The revision of the seismic category of conduits and associated supports in the Lab Area of the Control Building does not alter any acceptance limits or design failure points. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REPLACE THE HEATING, VENTILATION, AND AIR-CONDITIONING/STANDBY GAS TREATMENT SYSTEM/ AND OFF-GAS POST TREATMENT PROCESS RADIATION MONITOR EPROMS AND REPLACE MEM I BOARDS WITH MEM II BOARDS

Activity Evaluated: ECNs 29583, 32032, 32033, 32034, 32035, 32036; USAR Changes 9-159 through 9-164

Log Number: 2000-061

This activity upgrades the Heating, Ventilation, and Air-Conditioning (HVAC)/Standby Gas Treatment System (SGTS)/Off-Gas Post Treatment Process Radiation Monitor (PRM) EPROMs with a newer firmware version. It also upgrades the MEM I boards to MEM II boards. The HVAC/SGTS/ Off-Gas Post Treatment PRMs are not accident initiators and only perform a monitoring function of effluents released to the environments. Accidents previously evaluated in the Updated Safety Analysis Report (USAR), which result in a release of gaseous activity to the environment, would not be affected by this activity. There are no credible failures associated with this activity because the functional changes to PRMs will be tested by the manufacturer and as part of the post installation testing to demonstrate that the monitors function as intended. This activity will affect the response of the Low Fail alarm in that the time to identify a detector failure will be changed to 30 minutes versus 10 minutes. This increase in the detector response time is small compared to the compensatory sample requirements employed when a detector failure occurs. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Each PRM channel will undergo a full functional testing to demonstrate that the Low Fail alarm functions as intended. There are no effects, direct or indirect, that cause a failure of a plant safety system to perform its safety function, because these instruments are non-safety related and do not support any safety related components. Reducing the unnecessary auto initiation signals to the HVAC High Range accident monitors improves their reliability. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The requirements for operating the HVAC/SGTS/ Off-Gas Post Treatment systems are specified in Off-site Dose Calculation Manual Chapter 3.0. There are no Technical Specifications that directly or indirectly establish any operating requirements for these instruments. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DRYWELL CHILLED WATER SYSTEM WATER CHEMISTRY UPGRADE AND RECORD OF VALVE POSITIONING FOR DRYWELL CHILLED WATER

Activity Evaluated: CPS 2800.15 Revision 0 and CPS 2800.15C001 Revision 0 — Log Number: 2000-063

Clinton Power Station (CPS) Procedure 2800.15, VP System Water Chemistry Upgrade, provides instruction to reduce the suspended/soluble corrosion products and impurities from the Drywell Chilled Water (VP) system while the system remains in-service. This is accomplished by simultaneous and repetitive feed and bleed of chilled water system in a controlled manner to prevent loss of VP chilled water inventory. The VP system is non-safety related, except for the components located between and inclusive of the containment isolation valves and drywell isolation valves in order to ensure primary containment isolation. Loss of the VP system could result in the requirement to administratively shut down the plant, however, a review of the Updated Safety Analysis Report (USAR) indicates that the VP chilled water system and the various drywell cooling units supplied by the chilled water are not assumed initiators for any evaluated accident. This activity is consistent with current design basis and continues to meet system design, material, and construction standards. Failure of VP components during this activity would not compromise or impair availability of any system/component required for mitigating the effect of an accident or transient. System reliability is not impaired since stationed operators will continuously monitor and control the VP Compression Tank levels and system pressures. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no Technical Specification limits, safety limits, limiting safety system settings, or limiting conditions for operation negatively impacted by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

DRY SOLID WASTE STORAGE INSIDE THE PROTECTED AREA

Activity Evaluated: USAR Change 9-179 — Log Number: 2000-066

Updated Safety Analysis Report (USAR) Change 9-179 supports an additional temporary storage location for filled ISO boxes or SeaLand containers located outside within the Protected Area. The total number of full containers stored outside in the Protected Area shall not exceed 13. This limit does not apply to empty containers, containers of scaffold or shielding storage, full containers staged inside buildings within the Protected Area, and containers removed from the Power block and either returned inside or shipped off-site the same business day. The storage of filled ISO boxes and SeaLand containers awaiting processing and disposition as radioactive waste is not responsible for the initiation of any accident evaluated in Chapters 6 and 15 of the USAR. Of the evaluated events, only tornadic and flooding considerations are pertinent. Flooding is not considered credible due to storage location elevation and drainage capability. The closes safety related structure, the Diesel Generator Building, is approximately 500 feet away and the temporary storage location is outside of the Tornado Missile Exclusion Area. As a result, the impact as a potential missile during a Design Basis Tornado is not a credible missile threat to plant structures. Calculations prove that the low activities and low associated dose rates from the storage of the containers and the dose to the public is below the limits of 10CFR20.1301. The handling and storage of radioactive waste materials is located such that it can not affect overall system performance, system response characteristics, system design limits, system operational transients, or system interactions. No hardware changes are made that can compromise any assumptions made in the design and construction of Clinton Power Station. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not impact the design or operation of any structure, system or component. No hardware additions or changes were made to the plant. Also, no component or system level failure modes are introduced. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The dry active waste management system is not directly represented in the Technical Specifications. The radiological release limits of Technical Specification 5.5.4.g, 5.5.4.i, and 5.5.4.j are not violated by the outdoor storage of the dry active waste containers. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISE AVERAGE POWER RANGE MONITOR UPSCALE ALARM/ROD BLOCK TO USE SIMULATED THERMAL SIGNAL VICE NEUTRON FLUX

Activity Evaluated: ECNs 32158, 32172, 32173, 32174; USAR Change 9-204; ORM Change 29-5 — Log Number: 2000-067

This activity modifies the Average Power Range Monitor (APRM) upscale rod block/alarm function to use the Simulated thermal Power (STP) signal instead of the neutron flux signal. Under normal operating conditions, the neutron flux and the STP signal are the same except that the neutron flux signal will show more of the short duration noise-type transients, including those that can result in nuisance alarms and temporary rod block signals. Several accidents described in the Updated Safety Analysis Report (USAR) could be related to this activity; they are: Main Steam Isolation Valve (MSIV) Closure – High Flux Scram, Pressure Regulator Failure – Downscale, Loss of Feedwater Heating, Closure of one MSIV, Recirculation Flow Control Failure – Fast Opening of One Loop at 30% per second, and Rod Drop Accident. The initiating failure modes for these events do not reside with the APRM monitoring functions or control room panels that this activity affects. The activity only affects APRM circuit cards and wiring. The change utilizes the same standards for design, construction, and installation as the original configuration. The modifications to the card do not alter the boards physical characteristics or strength. The small mass added to the card has no significant impact on the seismic capability of the card. This change increases the loading on the trip reference card but the increase is well within the circuit's output capability. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The failure modes of the quad trip card and STP signal output of the thermal trip card have not changed. The minor changes to the board and substitution of the STP signal for the neutron flux signal have not caused the APRM system to become an accident initiator or to cause a malfunction of a different type. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. None of the sections addressing APRM functions concern the Upscale Alarm/Rod Block. The APRM Upscale Alarm/Rod Block in itself is not credited with a safety function. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

EMERGENCY CORE COOLING SYSTEM CHECK VALVE MODIFICATION

Activity Evaluated: Modification M-079 Revision 5; USAR Change 9-182 — Log Number: 2000-068

Modification M-079 modifies the testing features of Emergency Core Cooling System (ECCS) Check Valves and associated equalizing valves. Each valve can be modified, tested, and design released independently of each other. Post maintenance testing of the check and equalizing valves will be performed to verify operability. The primary functions of the ECCS check valves are to maintain the Reactor Coolant Pressure Boundary (RCPB), provide pressure isolation capability, and allowing ECCS injection flow when required. The primary functions of the equalizing valves are to maintain the Reactor Coolant Pressure Boundary and to equalize differential pressure across the check valve disk during stroke testing. This modification installs an electrical jumper plug connection to the equalizing valve solenoid to allow for stroking the valve during local outage testing. There is no credible failure mode of the plug that can initiate the design basis accidents described in the Updated Safety Analysis Report (USAR). A breach of the RCPB is an accident initiator. The check valve actuator and limit switch components being removed are not part of the ASME pressure boundary and thus does not impact the RCPB. This modification maintains the ASME valve classification, seismic qualification, and flow characteristics of the check valves and equalizing valves. The instrument air line to the check valve air actuator is being disconnected. Loss of Instrument Air is an accident initiator. This modification isolates the air line at the closest isolation valve and the line is capped. Removing the air connection to the check valve actuator will reduce the potential for an air leak and an overall Loss of Instrument Air. The check valve's ability to prevent flow in the reverse direction and allow flow in the forward direction have not been changed or degraded. No new electrical or mechanical failure modes have introduced by this activity. No radiological release paths are modified and there is no impact to fission product barriers. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. The pressure isolation capability, valve opening capability, and flow characteristics have not been degraded by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

FUEL STORAGE & HANDLING CLB/USAR DISCREPANCY RESOLUTION

Activity Evaluated: USAR Change Package 9-183 — Log Number: 2000-070

USAR subsection 9.1.4.2.3.7 describes the jib crane and states the crane has two full capacity brakes and two sets of independent limit switches. Contrary to these statements, vendor manual K2801-157 indicated the jib crane has only a single brake assembly and one set of independent limit switches. The following portions of the description of the jib crane within subsection 9.1.4.2.3.7 are changed. The brake description changed from "two full capacity brakes" to "a full capacity brake". The description of the limit switch(es) which automatically stop the hoist cable terminal approximately 8 feet below the jib crane base is changed from "...two... switches" to "...a...switch". The description of the limit switch(es) which automatically cuts the hoist power at the maximum safe up-travel limit is changed from "Two additional independent switches.." to "The additional independent switch..." This activity does not involve any field modifications. This activity does not increase the probability of an accident, does not increase the consequences of an accident, does not increase the consequences of a malfunction of equipment important to safety, and does not reduce the margin of safety as defined in the basis of the TS.

WASTE WATER TREATMENT FACILITY

Activity Evaluated: Modification ST-009; USAR Change 9-197 Log Number: 2000-072

Modification ST-009 replaces the Sewage Treatment (ST) Facility/System with a Waste Water Treatment Facility. The new treatment consists of primary and secondary aerated lagoons, tertiary sand filters for solids removal, and a lift station, which has lift out pumps that do not require entry into a confined space for rework. The Sanitary System is not required to perform or in any way assist to perform any nuclear safety function, such as safe shutdowns of the power station, or to mitigate the consequences of a nuclear accident. However, the system is designed to meet the effluent quality limits set by the Illinois Environmental Protection Agency. It is also designed so that no connections are made to systems that have a potential for containing radioactive materials. The use of aerated lagoons and tertiary sand filters to treat waste water is a well-established industry practice, and proven process. The design, material, and construction standards applicable to the Waste Water Treatment Facility are consistent with industry practice. The primary function affected by the design will be an increase in treatment capacity of the Waste Water Treatment Facility. The capacity of ST will be increased by fifty-five percent, after the NPDES Permit is modified to allow this discharge rate. The volumes of the lagoons are designed to retain the sludge for about thirty years of plant operation. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. The systems and components involved in waste water treatment and structure of the Sewage Treatment Facility are not addressed in the Technical Specifications. Modification ST-009 does not affect the safety limits, limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REVISION TO INSTRUMENT LINE EXCESS FLOW CHECK VALVE SETPOINTS AND DOCUMENTATION OF DESIGN BASIS

Activity Evaluated: ECN 31745; TS Bases Change BL-00-006; ORM Change 29-3; and USAR Change 9-193 Log Number: 2000-075

This activity revises the Technical Specification Bases by removing the reference to closure testing the Excess Flow Check Valves (EFCVs) by differential pressure and clarifies that the EFCVs are used to minimize consequences of an accident, but are not credited in the approved accident analysis. Also, Operational Requirements Manual (ORM) Change 29-3 revises the "Maximum Isolation Time" for each EFCV by deleting the differential pressure testing criteria for all EFCVs and adding a minimum flow limit of 0.58 scfm for those valves tested with air. The minimum flow limits are being added to the EFCV valve data sheets via Engineering Change Notice (ECN) 31745. In addition, the elimination of the minimum and maximum closure limit for the water-tested valves is also being evaluated, since this was not properly evaluated in ORM Change 13-2. Instrument line break accidents are discussed in USAR sections 62.1.1.3.5 and

15.6.2. None of the automatic signals are accident initiators, although an inadvertent upper pool dump is discussed in USAR Section 6.2.7.3.3. Calculation IP-M-0506 determines the air flow that would be generated due to a 1 psi pressure spike during normal operation and establishes the minimum closure setpoint above this value. The use of the 1 psi pressure transient as a basis for preventing spurious isolation of the EFCVs is consistent with the existing licensing basis. In addition, the EFCVs have an internal equalizing orifice, which will allow the EFCV to reopen if closed when the downstream instrument line is not broken. This change reinstates a minimum closure limit for EFCVs connected to air, but does not reinstate a minimum limit for valves in water filled lines. Calculation IP-M-0506 determines that there is no mechanism to establish flow in the water filled lines other than a line break and therefore, no minimum limit is required for these valves. This activity does not involve a physical plant change, a change in quality or type of materials used for fabrication of the instrument lines or components, or an increase in the temperature, pressure, stress or fatigue to which the instrument lines are exposed. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The EFCVs have two active failure modes: they may inadvertently close, preventing the downstream instruments from performing their intended functions or they may fail to close upon an instrument line break and not limit the release of radioactive material. The change in setpoints and method of testing has no affect on any of these failure modes. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This modification does not adversely affect the function or operation of any structure, system, or component. Nor does this modification adversely impact any Technical Specification, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

ELMINATION OF FACILITY REVIEW GROUP REVIEW OF IMPLEMENTING PROCEDURES

Activity Evaluated: USAR Change 9-178 Log Number: 2000-076

Updated Safety Analysis Report (USAR) Change 9-178 deletes the requirement for the Facility Review Group (FRG) to review changes to the implementing procedures for Security, the Emergency Plan, and the Fire Protection Program. For the Emergency and Security Plan implementing procedures, the change is consistent with the philosophy established by the Nuclear Regulatory Commission (NRC) in Generic Letter 93-07. For the Fire Protection Program implementing procedures, the change is consistent with the philosophy established by the NRC in Regulatory Issue Summary 99-02. This activity is not associated with the assumed initiator of any evaluated accident nor any design basis event. Procedural controls are in place to provide reasonable assurance of the continued effectiveness of these plans and programs. This activity does not alter the assumed performance capability or standards of performance of any equipment important to safety. Nor does this activity not alter the assumed release mechanisms, pathways, rates, duration, source term, or protective functions, nor does it degrade the effectiveness of structures, systems, or components used to mitigate assumed and evaluated accidents. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not introduce a credible failure or degradation mechanism for any structure, system, or component, nor does it introduce the potential for a new operator induced error that could result in an accident or malfunction of equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. No pertinent margins of safety are established that are potentially affected by the elimination of the FRG review of implementing procedures for the Security Plan, the Emergency Plan, or the Fire Protection Program. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

BREATHING AIR CONTAINMENT/DRYWELL ISOLATION VALVES

Activity Evaluated: ECN 32187; USAR Change 9-199; ORM Change 29-4 — Log Number: 2000-077

Engineering Change Notice (ECN) 32187 changes Containment Isolation valves, 0RA026 and 0RA027, and Drywell isolation valves, 0RA028 and 0RA029, from normally open to normally closed. The affected portions of the Breathing Air (RA) and Instrument Air (IA) systems do not initiate any accidents described in Updated Safety Analysis Report (USAR) Chapter 6 or 15. In addition, there are no accidents evaluated in the USAR that requires the valves to be open. The purpose of the Containment Isolation valves and the Drywell Isolation valves is to minimize potential leakage of fission products from Containment/Drywell during accident conditions. Maintaining these valves in the closed position puts them in their post-accident position, which assures isolation of the Containment/Drywell Breathing Air and assures they do not become a flowpath out of the Containment/Drywell for fission products. As stated in USAR Section 15.2.10, loss of IA has no impact on safe shutdown because the affected components are designed to fail to their intended post-accident positions. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. The RA system in the Containment and Drywell has no interaction with any other system. Thus, isolating the RA system to the Containment and Drywell by maintaining the valves normally closed cannot create the possibility of an accident or malfunction of equipment important to safety. No changes are made to the Containment or Drywell isolation logic. Since, there are no hardware changes being performed due to this activity, the design, material, and construction standards are maintained. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specifications do not address the Breathing Air valves specifically. Technical Specification Bases 3.6.1.3 discusses primary Containment isolation system and Technical Specification 3.6.5.3 discusses Drywell isolation. Valves 0RA026, 0RA027, 0RA028, and 0RA029 are air operated valves with an active function to close and isolate the Containment/Drywell in the event of an accident. By isolating their Instrument Air supply, the Breathing Air valves are maintained in their safety function position. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

REORGANIZATION OF CHEMISTRY & RADIATION PROTECTION

Activity Evaluated: USAR Change 9-201; — Log Number: 2000-079 R/2

Two sections of USAR Chapter 13 changed to reflect recent reorganization of Chemistry and Radiation Protection functions. Supervisor Radiological Programs is eliminated and the responsibilities under that position will be divided between the Director-Chemistry and Director-Radiation Protection. The Radiological Effluents and Radiological Environmental Monitoring

Program will remain or be consolidated in Radiation Protection Department. Radioactive waste and materials shipping and other non-radiological environmental responsibilities will be retained in the Chemistry department. These changes will bring CPS in line with the PECO & Commonwealth Edison organizational models. It also streamlines the organization, places similar programs and processes under one owner and thereby provides greater accountability for those processes. The probability of a malfunction of equipment and the probability of an accident are not increased. The margin of safety as described in the bases of the TS is not decreased. The Facilities Review Group determined that this was not an unreviewed safety question.

UPDATE TO REGULATORY GUIDE 8.15, REVISION 1 OCTOBER 1999

Activity Evaluated: USAR Change 9-208 Log Number: 2000-080

Updated Safety Analysis Report (USAR) Change 9-208 updates Clinton Power Station's (CPS's) position in regard to Regulatory Guide 8.15, Revision 1. This regulatory guide describes a respiratory protection program that is acceptable to the Nuclear Regulatory Commission and provides guidance on performing evaluations to determine whether the use of respirators to optimize the sum of internal and external dose and other risks. This activity does not impede, degrade, or prevent actions taken by the control room operators against an accidental toxic gas release or by the emergency response personnel who wear respirators as part of their emergency team assignment. Nor does this activity impede, degrade, or prevent actions taken by fire brigade members who are trained in fighting fires protecting all areas of the plant including structures, systems, and components important to safety. This activity does not compromise the design, material, or construction standards to which CPS was originally built. In addition, this activity does not compromise the performance of any system or involve the operation, maintenance, or design of equipment important to safety. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. No hardware additions or changes are made as a result of this activity. This activity does not affect the design or operation of any equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Respiratory Protection Program is not directly addressed in the Technical Specifications. This activity does not impede, degrade, or prevent actions taken by the control room operators, emergency response personnel, or the fire brigade members who wear respirators. The CPS Respiratory Protection Program complies with all the requirements specified by Regulatory Guide 8.15, Revision 1. This activity does not compromise those responsibilities or required qualifications. There are no operational considerations associated with the changes being made. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGES TO PLANT STAFF ORGANIZATION

Activity Evaluated: USAR Change 9-212 Log Number: 2000-084

Updated Safety Analysis Report (USAR) Change 9-212 combines the Licensing and Experience Assessment Department functions into one Licensing organization. As a result of this change, the Director – Experience Assessment position is eliminated. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INCORPORATE ENVIRONMENTAL PROFILES FROM CALCULATIONS AND CORRECTION OF MISCELLANEOUS ERRORS

Activity Evaluated: ECN 32226; USAR Change 9-216 Log Number: 2000-085

Engineering Change Notice (ECN) 32226 incorporates the results of calculation 3C10-0699-001, Revision 0 that affect the temperature profiles for various areas of the Auxiliary and Fuel Buildings (post High Energy Line Break (HELB) operation only) into the Updated Safety Analysis Report (USAR). Also, this activity changes the maximum temperature in Battery Room 1A1 and 1B1 under all conditions to 95°F. These areas house multiple systems that are important to safety, including potential accident initiating systems such as Emergency Core Cooling Systems and Main Steam Isolation Valves. The equipment in the areas affected have been reviewed and determined to be qualified for the modified environments without reduction in qualified life. Since the peak temperatures are not increased, calculations for pressure locking, thermal binding, setpoint, and manual operated valve capability are not affected. This activity does not reduce the single failure protection of systems or alter the ability to accomplish required operator actions. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity does not introduce any new modes of plant operation or introduce any new equipment into the plant. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. No Technical Specifications directly address environmental qualification or the temperature profiles in the various plant areas. However, implicit in the definition of "operable" is the ability of equipment to function in the environment in which it is intended to operate. This activity does not change the environmental qualification of the equipment required to be operable. Since the Instrument Setpoint calculations use the peak Loss of Coolant Accident/HELB temperature for the instrument uncertainty determinations, the margin of safety for any specification is unaffected. Technical Specifications 3.8.4, 3.8.5, and 3.8.6 provide the requirements for the batteries. Minimum electrolyte temperature is included, but not ambient temperature. The reduction in maximum battery room temperature does not hinder meeting any of the Technical Specification requirements or affect the margin of safety for the battery in any way. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE OF QA DEPARTMENT NAME AND REPORTING STRUCTURE

Activity Evaluated: USAR Change 9-217, QAM Rev. 27a, ORM Change 31-3 Log Number: 2000-087

Changed Illinois Power and Illinova in Quality Assurance Manual (QAM) to AmerGen (previously evaluated by USAR Change 9-020). Transferred Independent Safety Engineering Group (ISEG) to the Nuclear Oversite (NO) department (previously evaluated by USAR Change 8-412. Deleted section 6.4 of the Operational Requirements Manual (ORM) comprising the QA program description and added USAR Section 13.4. (previously evaluated under USAR Change 9-088 and ORM Change 28-6). Changed the QA Department to Nuclear Oversight Department. Changed reporting relationship for CPS Manager-N.O. to the report to the Regional Operating Group (ROG) and revised the QA Manual and the USAR Chapter 13 to indicate this title name change and reporting change. The responsibility for supplier evaluations and maintenance of the Qualified Supplier List from CPS QA to the ROG Supply Management Supplier and maintenance of the Qualified Suppliers list will be accomplished under the Exelon Quality Assurance Program (reviewed and approved by the NRC). Exelon Supply Management implements the Nuclear Utilities Procurement Issues Committee (NUPIC) process for performance of supplier evaluations as did Clinton Power Station (CPS) Quality Assurance. Nuclear Review Board will now be known as Nuclear Safety Review Board and reports to the Chief Nuclear Officer (CNO), as well as advises the CNO. QA Manual to reflect Manager-NO reporting to the ROG Director-NO reporting to Vice President Nuclear Oversight (change covered by safety screening/evaluation for USAR 9-278). The reporting relationship change and transfer of responsibility for external audits and supplier evaluations for the Nuclear Oversight Department to the ROG will eliminate duplication of efforts to maintain qualified suppliers common to AmerGen/Exelon Midwest Operating Group. The department name change from QA to NO will be consistent within the AmerGen/Exelon corporation and reporting to the ROG rather than to PECO will bring the organization to a closer geographical location. The name change from the Nuclear Review Board to the Nuclear Safety Review Board is to better describe its function. Changes to the appointment and reporting for the NSRB reflect the current corporate structure subsequent to the merger of PECO and UNICOM.

OPERATION OF CONTINUOUS CONTAINMENT PURGE SYSTEM IN PLANT MODES 4 AND 5

Activity Evaluated: USAR Change 9-219; ECCN 32228; CPS 3408.01 Log Number: 2000-088

This activity allows the option of operating the Continuous Containment Purge (CCP) system in modes 4 and 5. The CCP system is not required to function in any but normal station operating conditions to limit airborne radioactivity and maintain proper pressure boundaries in the containment. Therefore, this system has no safety design bases except for the containment building penetration isolation valves. The CCP system isolation valves at the containment penetration and the intermediate pipe between the valves are required during and after all abnormal station operating conditions to maintain the containment boundary integrity. The main isolation valves are spring loaded, air operated, and fail closed on loss of electric power or station air. The isolation valves close upon receiving a Loss-of-Coolant accident signal, or a high radiation signal from any of the associated radiation monitoring design system, thus preventing an accidental release of radioactivity. The CCP system will still meet all applicable design, material, and construction standards. Overall system performance will not change as a result of this activity. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The malfunction of the containment isolation valves has already been evaluated and the containment isolation valves have redundancy on both the supply and the exhaust side to withstand single failure criteria. The closure signals provided to the containment isolation valves are still active in all plant modes of operation and are not disengaged in plant modes 4 and 5. The failure of the isolation valves is the only safety related accident of concern and changing the modes of operation allowed to CCP would not create a different type of accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification sections 3.3.6.1, 3.3.6.2, and 3.6.5.3.2 do not stipulate modes of operation and there are not discrepancies with the changes to the USAR. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM DIFFERENTIAL TEMPERATURE INSTRUMENTATION

Activity Evaluated: ORM Change 29-6; USAR Change 9-223 Log Number: 2000-089

Operational Requirements Manual (ORM) Change 29-6 deletes the requirement for the Differential Temperature – High Trip Functions from Table 3.2.16-1, Containment and Reactor Vessel Isolation Control System (CRVICS) Instrumentation. The trip functions affected include Main Steam Line Isolation function, Reactor Water Cleanup System Isolation function, Reactor Core Isolation Cooling System Isolation function, and Residual Heat Removal System Isolation function. As a result of this ORM Change, the specific parameter value for the Trip Setpoint and Allowable Value and explicit testing interval requirements for the CRVICS instrumentation have been deleted. CRVICS requirements for differential temperature instrumentation are provided to mitigate the consequences of leakage from systems containing reactor coolant. Postulated accident scenarios pertinent to this activity involve those events that result in a loss of Reactor Coolant System (RCS) pressure boundary integrity. None of the revised or deleted actions affect the probability of occurrence of any of the evaluated accidents because the actions and result of the actions are not associated with the initiation mechanism that would contribute to the RCS pressure boundary failure. These instruments will continue to function to detect and initiate automatic isolation of the affected systems. The requirements of Updated Safety Analysis Report (USAR) Sections 7.3.2.2.2.1.6 and 7.6.1.4.5 for periodic testing of the instrumentation preserves the reliability of the differential temperature initiated isolation function. This activity has not altered the USAR credited mitigatory functions that would initiate isolation of one of the associated flow paths. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This activity has not introduced a new causal mechanism and has not altered the bounding nature of the USAR Chapter 6 and 15 accident analyses. The deletion of the ORM action requirements based on component inoperability will not introduce a failure mechanism for equipment important to safety different than those already present because the equipment is unchanged. This activity does not involve a change that impacts the operational behavior of any structure, system, or component classified as equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Although the differential temperature isolation function interfaces with a number of systems, this isolation function does not constitute an operability requirement for those systems or the associated isolation valves. Technical Specification (TS) Limiting Conditions for Operation (LCOs) pertaining to these functional areas include TS 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," TS 3.4.5, "RCS Operational Leakage," TS 3.4.7, "RCS Leakage Detection Instrumentation," TS 3.6.1.3, "Primary Containment Isolation Valves," and TS 3.6.5.3, "Drywell Isolation Valves." For each of these LCOs, the assumed functional performance capabilities of equipment subject to the LCO are unchanged following this activity. The functions provided by the Technical Specifications in ensuring the preservation of the integrity of the RCS, drywell and containment pressure boundaries are unchanged and the margins of safety afforded by these requirements will not be affected by this activity. The location at which the setpoint resides is not associated with a condition that would create a margin of safety associated with preservation of the integrity of any Technical Specification or the integrity of any fission product barrier. The explicit testing interval does not provide a margin of safety as defined in the basis for any Technical Specification. The isolation valves that actually perform the mitigatory action to isolate a ruptured pipe remain subject to Technical Specification control. The ORM requirements for testing of the differential temperature instrumentation are unnecessary in preserving any margin of safety that may exist. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

NSPS STS ACTIONS AND BASES CLARIFICATION

Activity Evaluated: ORM Change 29-7; Log Number: 2000-092

Operational Requirements Manual (ORM) is revised for Nuclear System Protection System (NSPS) Self Test System (STS) to clarify certain requirements for when manual or partially automatic operations are acceptable in lieu of automatic mode requirements. Specifically, the asterisked section (footnote) is being revised to allow partial automatic or manual operations as an equivalent to fully automatic operations as long as all required tests are performed at least once per seven days; or during Operational Modes 4 & 5, at least once per 90 days. Secondly, ORM Action 3.2.14.1.a is being revised to state that when the STS is not operating in the required mode (i.e., the fully automatic mode or being operated in either the manual or partially automatic mode such that all required test are performed at least once per seven days during Modes 1, 2, or 3, or once per 90 days during Modes 4 or 5), the STS must be restored to the required mode within 30 days, or be in at least Mode 3 within the next 12 hours, and in Mode 4 within the following 24 hours, when the plant is in Modes 1, 2, or 3. Thirdly, for Modes 4 or 5, Action 3.2.14.1.b is being revised to require restoring the STS to the required mode within 90 days or suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel, verify all insertable control rods to be fully inserted and lock the reactor mode switch in the SHUTDOWN position within one hour. A change to the associated Bases, Section 5.2.14, is being made to reflect the changes made to the Operation Requirement and the Actions and to clarify that methods other than the process computer and/or diagnostic terminal may be utilized to observe that the STS is operating in the automatic mode or operation pursuant to the requirements of Testing Requirement (TR) 4.2.14.1. Additional information regarding the licensing basis for the STS, particularly as established in two reports prepared for Clinton Power Station (CPS) by General Electric (GE), is also being incorporated into the Bases discussion.

It was determined that this activity will not increase the probability of an accident previously evaluated in the USAR. Since STS is neither an accident initiator or accident mitigator, this activity will not increase the consequences of an accident previously evaluated in the USAR.

Any STS failure will not degrade the NSPS function since STS is isolated from NSPS, hence eliminating failure propagation. Also, all interdivisional links are optically isolated. A complete failure of STS will not prevent the NSPS from performing its function, therefore, the activity will not cause any increased probability of equipment failure. Based on the above, the change will not increase the probability or consequences of a malfunction of equipment important to safety evaluated previously in the USAR. The existing failure analysis is bounding for this change and this change will not create the possibility of an accident or malfunction of a different type than previously evaluated in the USAR. Since required tests are performed every 7 days in Modes 1, 2, or 3 and every 90 days in Modes 4 & 5, the change will not reduce the margin of safety as defined in the basis for technical specification.

INSTALL DRAIN LINE ON VALVE 1E12-F009

Activity Evaluated: ECN 32224; USAR Change 9-236; ORM Change 29-8 Log Number: 2000-093

Engineering Change Notice (ECN) 32224 installs a ¾" drain line from the drain connection on Primary Containment Isolation Valve (PCIV) 1E12-F009 to the reactor side of the valve. The purpose of the PCIVs is to minimize leakage from containment during certain accident conditions. As such, PCIVs are accident mitigating components and will not cause an accident with radiological consequences. As part of the installation, the bypass line around valve 1E12-F009 containing check valve 1E12-F475 will be removed and the connection upstream of valve 1E12-F009 capped. The new drain line thus becomes a thermal relief path for both the bonnet of valve 1E12-F009 and the volume of piping between Containment Isolation and Reactor Coolant System Pressure Isolation valves 1E12-F008 and 1E12-F009. Calculations demonstrate there is sufficient flexure in the outboard disc of valve 1E12-F009 to produce a gap sufficient to relieve pressure through the new line. For the Failure of Residual Heat Removal Shutdown Cooling event, the Shutdown Cooling suction valves 1E12-F008 and 1E12-F009 are assumed to fail to open. Since this activity is intended to improve the reliability of valve 1E12-F009 to open, it does not increase the probability of a failure of Shutdown Cooling suction valves. This change does not compromise the design, material, or construction standards to which the plant was originally built. The additional weight of the drain connections relative to the weight of the valves is insignificant and not part of the extended structure of the valve such that the seismic qualification of the valve is unaffected. The Containment Isolation function of valve 1E12-F009 is not affected by this activity because it will continue to perform the sealing function, be normally closed, and actuate automatically to the closed position upon a Containment Isolation signal. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The new flow path from the valve bonnet to the reactor side of the valve does not affect Primary Containment and Reactor Coolant System Pressure Isolation functions. This activity does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. This activity does not introduce a credible failure or degradation mechanism for any structure, system, or component, nor does it introduce the potential for a new operator induced error that could result in an accident or malfunction of equipment important to safety. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Technical Specification Section 3.6.1.3, Primary Containment Isolation Valves (PCIVs), requires the PCIVs to either be closed or function to close within the required isolation time following event initiation to minimize potential leakage from containment. This activity does not affect the closure time or automatic actuation of valve 1E12-F009 or its ability to perform its isolation function. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CONTROL ROOM INSTRUMENTATION LOCATIONS

Activity Evaluated: USAR Change 9-007; Log Number: 2000-095

Correction of USAR Description of Reactor Operator Information Displays. Correction of typographical errors and internal USAR cross references, clarification of the method used to determine Division II RHR pump flow when operating from the remote shutdown panel, and revision to the instrumentation description to correct the stated location of control room indications that are used by the operator to verify operation of the Emergency Core Cooling and RCIC systems following an accident. The following sections of the USAR are revised: 7.1.2.3, 7.2.2.1.2.3.9, 7.4.1.4.4.3, 7.5.1.4.2.3.1 and 7.6.2.5.4.2.3. The USAR package was revised in accordance with CPS 1038.03 and per CPS 1005.06 Section 2.2 the changes were evaluated for exemption from 10CFR50.59 evaluation. There is no physical work to be performed and no revisions to plant staff procedure are required to support these changes. For the purposes of this safety evaluation discussion, the changes are treated as a modification to the facility. The need for the change was found during the Current Licensing Basis (CLB) USAR validation. Several discrepancies between specific USAR statements and supporting plant documentation (controlled drawings, procedures, specifications, etc.) were documented. This USAR change resolves the discrepancies between USAR Subsection 7.5.1.4.2.3.1 and design drawings MO5-1002, M10-9002 (Sheets 1 & 2) and E04-1P870-61B.

SEISMIC MONITORING INSTRUMENT UPGRADE

Activity Evaluated: Mod EM-018, ECN 31413; Log Number: 2000-096

The existing equipment is obsolete, no longer supported by the original vendor and is not functioning properly. This design change replaces several obsolete components of the non-1E seismic monitoring instrumentation located in Main Control Room Panel 1H13-P865. The components removed are as follows: DCA-300P-12 Digital Cassette Recorder 1VRC-EM007, SMR-102 Cassette Playback unit 1VY-EM008, RSA-50 Response Spectrum Analyzer 1VXEM009. The replacement equipment is as follows: 15 Channel, 12 Bit, Central Recorder (1VRC-EM007), Dot Matrix Printer (1VY-EM008), and Data Analysis System - laptop PC (1VX-EM009). Electro-Magnetic Interference (EMI) / Radio Frequency Interference (RFI) shield doors are being installed over the PC and Printer openings in the panel to ensure EMI/RFI to and from the panel is within acceptable industry standards. New equipment is supplied by the original vendor and meets RG 1.12 requirements. The panel and equipment are non-1E but are classified as seismic Category I. They are seismically qualified to the requirements of IEEE 344. The overall weight and power consumption of equipment in the panel IS being decreased by the modification and electrical load will continue to be fed from the same non-1E distribution circuit. Existing sensors are compatible with the new equipment and are being reused. The new equipment provides the same alarms to the operator and provides the required time-history data to determine the severity of an earthquake. This allows operator to determine whether the measured response spectra has exceeded the predicted response spectra so that appropriate action can be taken.

This activity does not introduce any new failure mechanism in addition to those previously evaluated in the USAR. Therefore, the activity will no increase the probability of an accident as previously evaluated in the USAR. There are no accidents that can be affected by this modification to upgrade the Seismic Monitoring Instrumentation, therefore this mod does not increase radiological consequences of an accident previously evaluated in the USAR. There is no equipment important to safety affected and the EMI/RFI values are within acceptable industry standards, this activity will not increase the probability of a malfunction of equipment important to safety as previously evaluated in the USAR and will not create the possibility of an accident of a different type than any previously evaluated in the USAR. The seismic monitoring equipment is non-1E and has no impact on any equipment described in the TS. There are no Margins of Safety associated with the use and operation of the Seismic Monitoring Instrumentation. Therefore, this activity does not reduce the Margin of Safety as defined in the basis for any Technical Specifications.

CYCLE 8 RELOAD AND CORE DESIGN MODIFICATION

Activity Evaluated: Modification NB-034, Supplement 1; ECN 32298 — Log Number: 2000-097

Modification NB-034, Supplement 1 updates the design and licensing basis for the receipt of the GE14 fuel bundle and subsequent storage in the new fuel storage vault. In this phase of implementation of the modification, the GE14 bundle only interacts with the new fuel inspection stand and new fuel storage vault in the manner described in the Updated Safety Analysis Report (USAR) for new fuel preparation. There are no design basis accidents that involve these structures. These structures have seismic qualifications and the rack maintains the stored fuel subcritical. These are important safety features, but are not initiators for other events. The GE14 bundle compliance with the seismic qualification of these structures and the applicable design requirements in GESTAR II ensure that the design change does not affect the performance of these structures. In the process of performing the fuel receipt activity, the GE14 fuel is not stored with or moved over any irradiated fuel. The GE14 fuel bundle has a mass and dimensions comparable to previous bundles used at Clinton Power Station. The comparable mass ensures that damage to the NFSV is bounded by analysis and the comparable dimensions ensure that the height the bundle is dropped from is unchanged. The new fuel storage vault is designed to maintain a fully loaded rack of fuel in a sub-critical configuration under dry, flooded, and optimum moderation conditions. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). In this process there are no credible failure modes for an accident different than evaluated in the USAR. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The storage vault has a design requirement that the bundles, when analyzed in an infinite array, will have a k-effective less than 0.95. The GE14 fuel bundles have been analyzed in accordance with the procedure described in GESTAR II. The procedure confirms that the GE14 fuel, when loaded in the rack in an infinite configuration, have a k-effective less than 0.95. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

CHANGE REFUEL HOIST POWER CUTOFF FROM 10 mR/hr to 50 mR/hr

Activity Evaluated: ORM Change 29-9 Log Number: 2000-099

Since plant startup, area background readings for the Containment Refueling Platform bridge have increased due to normal plant operation. Operational Requirements Manual (ORM) 29-9 changes the high alarm setpoint for detector instrument 1RE-AR037 from 10 mR/hr to 50 mR/hr, while retaining step-change warning capability. There are no design basis accidents or equipment important to safety affected by this activity. This monitor is not connected to any other component of the Process Radiation Monitoring (PR) system, so the proposed activity does not affect any other PR component, or PR system response, in any way. The monitor is only connected to the refueling hoist for the purpose of halting upward fuel movement under certain conditions. The monitor itself will respond differently in that it will no longer generate nuisance alarms. It has been concluded that 50 mR/hr is the desired setpoint since it prevents nuisance alarms yet will still halt fuel bundle movement before the are is not habitable for necessary work. The purpose of this monitor is to sense when a fuel bundle is being raised too high; activation of the high alarm stops upward fuel movement. The monitor is not used as a fission product barrier. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). This activity does not introduce or alter any components in the plant. There is no direct or indirect effect of either failure mode on the ability of plant systems to perform their safety function. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not negligibly impact any Technical Specification, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

MOVEMENT OF NEW (NON-IRRADIATED) FUEL TO THE UPPER POOLS DURING PLANT OPERATION

Activity Evaluated: ECN 32315; USAR Change 9-249; TS Bases Change BL-00-015 Log Number: 2000-101

This activity allows the storage of new (non-irradiated) fuel in the upper pools during plant operation and addresses the transport of new fuel from the fuel building fuel pools to the containment building upper pools using the Inclined Fuel Transfer System (IFTS). Fuel handling accidents are discussed in Updated Safety Analysis Report (USAR) Section 15.7.4. These accidents involve dropping a spent fuel bundle either onto the reactor core or onto the spent fuel storage racks in the fuel building. Both cases involve handling of spent fuel; this activity does not change the handling of spent fuel in either the fuel building or containment. The activity does not affect any other accident initiating systems or components discussed in USAR Chapter 6 or 15. This activity allows movement and storage of new fuel within the containment building while the plant is operating, whereas previously it was only allowed with the plant in cold shutdown or refueling modes.

ONE-TIME EXTENSION OF 18-MONTH TESTING REQUIREMENT

Activity Evaluated: ORM Change 31-2 Log Number: 2000-102

Operational Requirements Manual (ORM) section 2.5.2 requires that thermal overload protection for each safety-related motor-operated valve (MOV) with a bypass device integral with the motor starter shall be bypassed continuously for those directions for which the valve performs an active safety function. This requirement is verified by executing surveillance procedure 9381.01, MOV Thermal Overload Bypass Verification, every 18 months in accordance with ORM testing requirement 4.5.2.1a. ORM Change 31-2 allows a one-time extension of the 18-month testing requirement. The valves affected by this activity supply stand-by water to the control room Heating, Ventilation, and Air-Conditioning (HVAC) makeup and supply filter manual deluge systems and spent fuel pool make up system. Portions of the Shutdown Service Water (SX) system, including the valves affected by this change, are used to mitigate the consequences of a design basis fire. The one-time change to the surveillance interval for testing of the thermal overload bypass circuits has no impact on the initiators, failure modes or mechanisms that may cause a design basis accident. This activity does not physically involve any changes to the plant, nor does it impact any design or functional requirements of the associated systems. The operability requirements for systems, structures, and components required by the ORM and the plant design basis remain unchanged. This activity is not expected to have an impact on the operability of these valves to perform their function for the time extension proposed for this testing requirement surveillance. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). This activity does not introduce any failure mechanisms or accident initiators of a different type. Also, the surveillance test requirement itself and the way the surveillance test is performed will remain unchanged. This activity does not introduce any new structure, system, or component or system interactions, nor does it alter existing components, systems, or methods of operation. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. This activity does not negligibly impact any Technical Specification, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

INSTALLATION OF SPARE BATTERY CHARGER 1DC27E

Activity Evaluated: ECN 32306; USAR Change 9-252; Log Number: 2000-103

Clinton Power Station (CPS) has no established method to cross connect 125 volts – direct current (VDC) Motor Control Center (MCC) 1E and 1F to support maintenance activity on the Balance of Plant (BOP) chargers. Permanent installation of the spare BOP charger will improve overall BOP DC system availability. Temporary connections is a proven method of connecting replacement (spare) charger to either of 125 VDC BOP bus (1E or 1F) during planned or emergency maintenance activities on regular charger. Activities evaluated include ECN 32306 and Updated Safety Analysis Report (USAR) change package 9-252. ECN 32306 permanently installs spare BOP Battery Charger 1DC27E to use it as a temporary replacement for either of presently installed BOP battery chargers 1DC25E or 1DC26E. The scope of the ECN is provide 125 VDC power to termination box on elevation 781' in vicinity of 125 VDC BOP MCCs. Permanent installation of spare BOP battery charger 1DC27E per ECN 32306 requires the following changes to facility as described in the USAR. The spare BOP charger will be shown on M01-1107 sheets 1 and 6 and on USAR Figure 1.2-6, Mezzanine Floor Plan El. 762', Figure 10, Cable Tray Figure FP-12a, Fire zone boundaries and Figure FP-12b, Fire Protection Features. The spare BOP battery charger will be added to drawing E02-1DC06, which is included in USAR as Figure 8.3-7. USAR subsections 1.2.2.6.1.8 and 8.1.3.3 will be revised to add new spare BOP battery charger to Unit Auxiliary DC Power Supply description and to make both sections consistent with each other. This activity does not increase the probability of malfunction of equipment or increase the consequences of an accident. There in no reduction in the margin of safety as described in the bases of the TS.

REACTOR RECIRCULATION PUMP DVP BREAKER REPLACEMENT

Activity Evaluated: ECNs 32188, 32189, 32190, and 32191; USAR Change 9-256; TS Bases Change BL-00-014 — Log Number: 2000-104

Engineering Change Notices (ECNs) 32188, 32189, 32190, and 32191 replaces the reactor recirculation pump motor 6.9 kV Westinghouse circuit breakers with new Cutler-Hammer model retrofit breakers. The new breakers were purchased as nuclear qualified components for safety related applications and seismically qualified to Clinton Power Station specific seismic response criteria. The new breakers are form, fit, and functional replacements designed to fit into the existing cubicles such that no physical or wiring modifications are required in order to install the replacement breakers. Replacing these breakers does not impact the control circuit logic (manual, automatic or protective relaying) for the breakers. The increase in load current for the trip coils has been analyzed and demonstrated to have no adverse impact on the ability of the breakers or their supporting systems to perform their design functions. Existing electrical divisional separation has been maintained and not degraded, thus maintaining the independence between the two reactor recirculation pumps. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The replacement breakers are of a newer design than the existing breakers. The new design incorporates different sub-components and mechanisms than the original breakers. These sub-components and mechanisms naturally exhibit different failure mechanisms and effects than those on the original breakers. These differences are at the sub-component level. Therefore, the credible failure modes associated with this activity are control circuit failure and power circuit failure. These events are included in the accident analyses in the USAR. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The breaker capability to open in response to an End of Cycle Recirculation Pump Trip signal has not been degraded by this change, and thus the margin of safety that depend on the Recirulation Pump Trip to aid Reactor Protection System in protecting fuel integrity have not been reduced.

REACTOR RECIRCULATION FLOW CONTROL VALVE MINIMUM POSITION FOR PUMP STARTS AND SPEED TRANSFERS

Activity Evaluated: CPS 3302.01 Revision 24a, CPS 3302.01C001 — Log Number: 2000-105

This activity changes the minimum position of the Reactor Recirculation (RR) Flow Control Valve (FCV) to less than or equal to 10% open and bypasses the minimum valve position interlock. It is applicable to idle pump starts, shifts from slow to fast speed, shifts from fast to slow, and pump trips. This will be controlled administratively rather than by the physical position switch on the valve actuator. Because the RR FCV will be further open, there will be a greater initial increase in the amount of core flow, causing more moderation, producing a greater positive reactivity insertion, resulting in a more rapid power increase. GE has evaluated the equipment affects from this change and they are as follows: the jet pumps and core internals have acceptable loads; jet pump stalling will not occur; the RR pump cavitation is protected by the interlock for differential temperature; the thermal stresses and fatigue for the jet pumps, vessel nozzles and piping are unchanged because the idle loop temperature is the same; and the RR motor has no significant impact on its integrity or duty, due to the longer starting time to achieve a higher flow. The margin to the Minimum Critical Power Ratio (MCPR) safety limit is maintained protecting the fuel. The jumper addition is the only new component added; its failure would place the interlock in service. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The new component introduced by this activity is the jumper for the minimum position interlock. The jumper will be installed by double verification into the panel, thus preventing mispositioning. If the jumper would fail open or disengage in any way, the interlock would no longer be bypassed; this does not create any new failure or initiating event. Bypassing the interlock during pump starts and up-shifts does not add any failure modes or initiating events. The interlock prevents pump starts and up-shifts unless the FCV is at its minimum position. Disabling it prevents the hardware from preventing an operator error in starting the pump in a different FCV position, but this is precluded by the verification that the valve is in its correct position. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The reactor recirculation system is addressed in Technical Specification section 3.4. This change to the start, stop or shift conditions does not impact any of the Limiting Conditions for Operation or surveillance requirements. The power distribution limits in section 3.2 are not affected by this change. The GE analysis demonstrated that the operating limits for MCPR are bounding for the abnormal start event with the higher initial flows. None of the principal barriers are impacted by this activity. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

TEMPORARY POWER SUPPLY FOR TURBINE BUILDING CRANE AND TURBINE BUILDING ELEVATORS T1 AND T2

Activity Evaluated: Temporary Modifications 00-034, 00-035, and 00-036 — Log Number: 2000-109

During Refueling Outage 7, maintenance being performed on the 6.9 kV Bus 1A will result in the loss of power to Turbine Building 480V Unit Sub 1J and Radwaste Building Unit Sub E. These Unit Subs normally provide power to the Turbine Building passenger elevator T1, the Turbine

Building freight elevator T2, and the Turbine Building crane. In order to keep the elevators and crane operational, Temporary Modifications 00-034, 00-035, and 00-036 will provide temporary power. Elevators T1 and T2 and the Turbine Building crane are part of the Hoist and Crane (HC) system, which is classified as non-seismic and non-safety related and are not required to operate during any postulated accident. These temporary modifications will not change the normal operation of the elevators or crane outside their analyzed design parameters. In addition, this change affects only the non-safety portion of the Auxiliary Power (AP) system. The circuit breakers used to provide equipment protection are the same circuit breakers, which are used during normal plant operations. The alternate power source being used for the temporary power supply provides the same level of circuit protection as the normal power supply. All temporary cables used during the installation of the temporary modifications are of the same conductor size, insulation rating as the permanent cables. Since all loadings/short-circuit contributions are within equipment and cable ratings, the equipment will operate normally with either the temporary power supply or the normal power supply and no equipment will be operated outside its analyzed design parameters. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The equipment will be operated within its design parameters and the circuit breakers are the same circuit breakers which are used during normal plant operations and are coordinated properly to provide overload protection as well ass coordination with upstream breakers. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The operation of the elevators, crane or the affected buses are not governed by any Technical Specification. This activity does not negligibly impact any Technical Specification, safety limits limiting safety system settings, or limiting conditions for operation. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

STARTING AND OPERATING CONTROL ROOM HEATING, VENTILATING AND AIR-CONDITIONING IN NORMAL MODE WITH DAMPER CLOSED

Activity Evaluated: ARR 00-0352 to CPS 3402.01 Revision 18b Log Number: 2000-110

This activity deletes steps to allow Control Room Heating, Ventilation, and Air-Conditioning (VC) Trains to start and/or stop with the locker room Exhaust Fan 11C Isolation Damper, 0VC69Y, failed closed. This if for a one time use and is not to be incorporated permanently without further investigation. The Main Control Room (MCR) Habitability Envelope is required to provide habitability and recourses to Operators in both normal and emergency conditions. Isolation dampers 0VC69Y and 0VC70Y allow flow to exit the MCR envelope through the 11C exhaust fan. During normal operating conditions, this allows for 1000 cfm of the 4000 cfm air exchange as described in the Updated Safety Analysis Report (USAR). All other modes of the MCR ventilation system, smoke, chlorine, purge, and high radiation modes, require the 0VC69Y and 0VC70Y dampers to be closed. The affected area being addressed is outside the "At the Controls" area of the MCR and function as personnel support areas both during normal operation and in the event of an emergency, thus no having any impact upon initiation of accidents or transients. This activity does not affect the design or construction of any structure, system, or component (SSC); thus, it will not affect any applicable standards related to the SSC. This activity will cause a higher differential pressure than normal, but the higher differential pressure is in the conservative direction and aids in maintaining required pressure.

LOOSE PARTS MONITORING SYSTEM CHANNEL #3 SETPOINT CHANGE

Activity Evaluated: Temporary Modification 00-051 Log Number: 2000-111

This activity installs temporary modification (TMOD) that changes the current setpoint for Loose Parts Monitoring System (LPMS) Channel #3. LPMS Channel #3 monitors the vibration and the presence of loose parts of the Reactor Recirculation (RR) Loop "B" suction. This activity will raise the existing Channel #3 setpoint to its maximum setting, under TMOD 00-051, to prevent unnecessary alarms when the plant is operating with a high background noise condition such as operation near 100% full RR flow. The temporary setpoint is based on an evaluation of the "B" Loop RR vibration spectrum and the known "A" loop vibration spectrum recorded when the primary system is being operated with no loose part detected and high background noise conditions present. To establish the temporary setpoint, channel #3 will be changed to maximum to account for the high background. Upon implementation of this temporary modification, channel #3 will have reduced capability to alarm for loose parts. The normal audio and manual recording functions will not be impacted by this activity. This change only deals with the Main Control Room (MCR) alarm and auto starting of the 4-track recorder. Since this activity will raise the setpoint, without validating the channel's ability to alarm at the required impact energy, the channel will be operating with reduced capability.

This activity does not increase the probability of any accidents, does not increase the consequences of any accidents, does not increase the probability of a malfunction of equipment important to safety, does not create the possibility of an accident of a different type than previously evaluated in the USAR, and does not reduce any margin of safety as defined in the Technical Specification or Bases.

CONDENSATE FILTRATION STABILIZATION

Activity Evaluated: Modification CP-20 and Log Number: 2000-113
USAR changes 8-400 R/1 and 8-401 R/1
ECNs 32304, 32305

Modification (Mod) CP20 Supplement 2 Revision 2 documents the design changes necessary to reflect "as-built" configuration of the condensate filtration in cells "E" and "F". Actual routing of 2" and under piping, specific core drill locations for plate mounting and actual hanger locations/configurations are examples of the kinds of changes being incorporated by Engineering Change Notices (ECN) 32117 and 32178. All changes of this type are encompassed by the original safety evaluation for CP-20 Supplement 2 (Log Number 99-176, also in this summary report) and are not covered by this evaluation. CP20 Supplement 2 Revision 2 corrects coordinates on Piping & Instrument Drawings (P&ID) in ECNs 32177 and 32178 (this item does not require an evaluation). CP20 Supplement 2 Revision 2 installs 1/4" flex tubing in sample panel 1PF32J to bypass all 3/8" panel internals for condensate filtration effluent sample lines in "D", "E", and "F" cells, the bypassed components are retired in place by ECNs 32304 and 32305. This change is necessary because the rough preliminary drawing coordinates provided by Black & Veatch do not correspond to the coordinate numbers finally assigned by Clinton when the drawings were incorporated. The existing 3/8" tubing, components and volume chamber of 1PL32J reduces sample velocity which, in turn, could result in particle settling in the lines. The internals are bypassed by 1/4" flex tubing. The probability of an accident is not increased by this activity, nor is there an increase in the malfunction of equipment. There is no reduction in the margin of safety as defined in the Technical Specifications or Bases.

EXELON MERGER

Activity Evaluated: USAR Change 9-278 Log Number: 2000-114

Updated Safety Analysis Report (USAR) and other Licensing Basis Documents (LBD) changes to reflect merger of companies PECO and UNICOM and the formation of Exelon. Also added commitment to RG 1.181. The merger of PECO and UNICOM and resultant formation of Exelon necessitated Facility Operating License Amendments to accurately reflect the transfer of PECO's ownership interest in AmerGen. This activity incorporates the resultant changes from this licensing activity into the Clinton Power Station (CPS) USAR (and other License Basis Documents) as required by 10 CFR 50.71e (and other governing LBD change processes). The deletion of Assistant Vice President position was made at the request of CPS management. Inclusion of the Regulatory Guide 1.181 commitment and the editorial changes were made as a matter of convenience. This change does not increase the probability of an accident or the possibility of malfunction of equipment. There is no reduction in the margin of safety as defined in Technical Specifications or Bases.

OPERATION OF REACTOR RECIRCULATION AUXILIARY SEAL INJECTION PUMP

Activity Evaluated: USAR Change 9-283 Log Number: 2000-115

Updated Safety Analysis Report (USAR) Change 9-283 adds a paragraph describing the use of the third Reactor Recirculation (RR) seal injection pump without the loss of offsite power. The purpose of the Auxiliary seal injection pump is to provide cooling water to the reactor recirculation pump seals. The system as installed is connected to turbine building Motor Control Center (MCC) 1M and the configuration is unchanged. The MCC is powered from 480V Bus 1B, which can be supplied from the Division II Diesel Generator if normal station power is lost. The turbine building MCC 1M is tripped in the event of a Loss of Coolant Accident (LOCA) and power to the Auxiliary Seal Injection Pump will not be available. This will not be altered by allowing the use of the system when normal station power is available. Cooling of the pump seals can be accomplished by an alternate backup cooling pump, which is installed to prevent seal damage from a loss of offsite power event and thus improves plant availability. In the event of loss of offsite power (LOOP), the pump will be actuated manually. The change allowing the operation of the pump during normal station operation would not require any additional operator action. The effects of single failure and operator error are not impacted, because operation of the Auxiliary seal injection pump is not intended to mitigate the consequences of an accident, but to limit damage to the seals. Operation of the Auxiliary seal injection pump will not prevent or degrade any activities for mitigating the effects of an accident. Operation of the pump when a LOOP does not exist and the normal Control Rod Drive (CRD) is not available is conservative and should help preserve the integrity of the seal. Also, the pump will continue to be available for use when normal station power has been lost. Therefore, this activity does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). Operation of the pump during a LOOP or LOCA has not been altered. This activity will not affect overall system performance, such that it changes system response characteristics, causes system operation outside of its design limits, or causes operational transients in the system or adverse system interaction. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the USAR. The Technical Specifications do not address the operation of the Auxiliary Seal Injection Pump and operation of the pump in conjunction with the operations of the Division II Diesel Generator has not been altered. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specifications.

NUCLEAR SUPPORT DEPARTMENT ORGANIZATIONAL CHANGES

Activity Evaluated: USAR Change 9-288 Log Number: 2000-116

Updated Safety Analysis Report (USAR) Change 9-288 makes the following revisions: the position of Director – Plant Support Services has been deleted (the responsibilities of the Director will be redistributed to the Manager – Nuclear Station Engineering Department); Human Resources will report directly to the Site Vice President; Security, Emergency Preparedness, and Personnel Processing will report to the Manager – Clinton Power Station (CPS); the "Nuclear Support" Department will be renamed "Business Operations"; the title of the "Manager – Nuclear Support" will be renamed "Manager – Business Operations"; the title of "Director – Financial Services" will be renamed "Controller"; and "Information Technology" will be renamed "Information Services". All of these changes are being implemented to support the transition to the Exelon Organizational Structure/Model. This is an administrative organizational change, which does not modify plant design or operation. No change is being made to the operation of the facility or to the availability of any equipment. In addition, this change does not compromise the design, material, or construction standards to which the plant was originally built. This organizational change does not compromise or impact compliance with seismic, fire loading, separation, or environmental design considerations of any structure, system, or component. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, nor does it create the possibility of an accident or malfunction of equipment important to safety of a different type. There are no operational considerations associated with the organizational changes made. Technical Specification 5.2 and 5.3 address manning and qualifications. This organizational change does not compromise those responsibilities or required qualifications. Therefore, this activity does not reduce a margin of safety as defined in the basis for any Technical Specification.

JET PUMP PLUG INSTALLATION

Activity Evaluated: CPS 8117.04 Rev. 9 & 8225.11, Rev. 8 Log Number: 2000-117

Jet pump plugs are being installed during RF-7 in all Reactor Recirculation (RR) system jet pumps to allow draining of the RR loop piping between the jet pump nozzles and the RR suction isolation gate valves. Draining of the RR loop discharge piping is required to support maintenance activities on the RR loop discharge valves 1B33F067A/B. Valve internals replacement is to address potential internal damage concerns from turbulent flow induced vibration as documented in GE SIL 528. A valve bonnet cover plate is utilized during the time period when the valve internals are removed and the bonnet is not required to be open for maintenance activities. Only one RR loop is to be worked on at a time. All work on installing and removing jet pump plugs is performed with the vessel head removed (Mode 5 - Refuel). Maintenance work on the RR discharge valves is treated as an operation with the potential to drain the reactor vessel (OPDRV). The jet pump plugs are designed to ASME section III code requirements and a seismic analysis has been performed to demonstrate that the plugs will maintain structural and leakage boundary integrity both during and after operational basis earthquake (OBE) and safe shutdown earthquake (SSE). A fuel movement evaluation concluded that fuel moves should be restricted during the time the "A" bonnet is open for maintenance activities. There were no unreviewed safety questions as a result of this 50.59 evaluation and was approved per FRG Meeting Minutes 2000-053.

ONE TIME EXTENSION OF TR 4.5.2.1a FOR 1E12F042C

Activity Evaluated: ORM Change to TR 4.5.2.1a to support RF7 Log Number: 2000-118

Change Operational Requirements Manual (ORM) section 4.5.2.1a to allow a one-time extension of the performance interval of the testing requirement (TR) from 10/20/00 to 11/30/00 for the RHR system valve 1E12F042C (RHR C LPCI injection). ORM 1.3.2, General Testing Requirement expiration date, incorporating 1.25 times the specified interval for valve 1E12F042C, is 10/22/00. The TR requires thermal overload protection for each safety-related motor operated valve (MOV) with a bypass device integral with the motor starter shall be bypassed continuously for those directions for which the valve performs an active safety function. This requirement is fulfilled by procedure 9381.01 every 18 months in accordance with TR 4.5.2.1a. The surveillance history & results evaluation shows that 40 days is the maximum time of extension, no history of failure of the thermal overload bypass feature of this valve was found for the surveillance history during the last two operating cycles and no history of failure was found in the equipment history since plant startup. This valve is interlocked with vessel pressure (it is the high to low pressure interface) and cannot be operated when the plant is on-line. The facility review group determined that no unreviewed safety question is involved with this activity.

INSTALLATION OF NEW GE NUCLEAR MAIN STEAM LINE PLUGS

Activity Evaluated: ECN 32271; USAR Change 9-292 Log Number: 2000-119

General Electric Nuclear Energy (GENE) has supplied newly designed Main Steam Line Plugs with associated tooling system. The new Main Steam Line Plugs may be remotely installed/removed with the reactor well filled (Wet Lift) using a remote steam line plug installation tool. The purpose of the ECN is to incorporate the information provided by GENE, for the set up, installation, removal and operation of newly designed Main Steam Line Plugs with an Installation Tool Assembly, into the Vendor Technical Document. The USAR will be revised to clarify quality related issues on the Main Steam Line Plugs and eliminate potential interpretation issues of refueling procedures that would prohibit the "Wet Lift" capability of the newly designed Main Steam Line Plugs and to eliminate a conflict regarding removal of the Steam Separator. The new design plugs provide the flexibility to install and remove the plugs with the reactor well flooded. Two major advantages are the time saved by eliminating the need to lower reactor water level to below the Main Steam Line Nozzles and lower radiation exposure because there will be water for shielding between personnel manipulating the plugs and the Main Steam Line

Nozzles. Changes will be made to USAR Sections 9.1.4.2.5.2, 9.1.4.2.10.2, 9.1.4.2.10.2.5, USAR Tables 3.2-1 and 9.1-4, USAR Figure 9.1-15. The probability of an accident or malfunction of equipment will not increase as a result of this activity. Nor is there a reduction in the margin of safety as defined in the basis of the TS.

CYCLE 8 FUEL RELOAD AND CORE DESIGN

Activity Evaluated: Mod NB-034, Supp 2, ECN 32329 Log Number: 2000-120

ECN 32329 (Supplement 2 of Mod NB-034) allows the core to be reconfigured to the cycle 8 configuration which includes the first-time use (at CPS) of the GE14 fuel bundle. The scope of this design change includes the transfer of the GE14 bundles from the new fuel storage vault to the spent fuel pool and movement into containment through IFTS, where they are staged in the upper containment pool racks and ultimately loaded in the core. The cycle 8 core configuration uses 188 new GE14 fuel bundles and shuffles the older fuel. The cycle 8 (18 months) core design has been reviewed by GE and confirmed to be consistent with GESTAR II licensing basis. The scope of this supplement to the modification addresses the concerns of the core for shutdown conditions.

PERMANENT REMOVAL OF DRYWELL BULDHEAD GRATING

Activity Evaluated: ECN 32338; USAR Change 9-294 Log Number: 2000-121

A stainless steel removable grating floor and the grating securing fasteners and supporting floor is installed within the Drywell Head Cavity and is supported by built-up structural steel tees. The tees are welded to the Bulkhead Plate and the Bulkhead Plate is supported by 24 Bulkhead Brackets. Foreign material accumulates around the built-up structural steel Tees and the Bulkhead areas during plant operations, then during refueling outages these areas are cleaned. Removal of this grating will significantly reduce the cleaning time, resulting in less radiation exposure and contamination. The grating provided a secure walkway for maintenance and observation of the RPV flange and bellows. The grating allowed foreign material to pass through to the steel plate collection area to protect the bellows and flange from unwanted foreign material contamination during work activities. A walkway way will now be provided via a recessed steel plate which will also continue to serve as a catch basin for potential foreign material. The 1 1/2" stainless steel Bulkhead grating, the grating securing fasteners will be eliminated. The removable grating performs no safety functions during any transients or design bases accidents per referenced USAR Sections. The facility review group determined that no unreviewed safety question is involved.

ECCS LOCA ANALYSIS CHANGE TO SAFER GESTR METHODOLOGY

Activity Evaluated: USAR 9-233 and TS Bases Change BL-00-012 Log Number: 2000-122

These changes revise the LOCA ECCS performance licensing basis from the SAFE REFLOOD methodology to the SAFER/GESTR methodology. The SAFER and GESTR are 2 computer codes used for the ECCS and LOCA analysis, developed in the mid 1980s after Clinton's licensing calculation were complete. SAFER does the vessel water level/pressure response calculations while GESTR provides fuel stored energy and pellet gap conductance calculations. Together they form a methodology that is used for large breaks, small breaks, steam breaks and feedwater loss or breaks. SAFER/GESTR calculate more realistic (yet conservative) peak clad temperatures (PCT). We also add the results for local clad oxidation fraction and core wide metal water reaction fraction. Applicable portions of USAR section 6.3.3 are being changed to describe the new methodology and results. The NRC issued an SER approving the use of the method in March 1984. The USAR and ITS Bases changes are made because the ECCS LOCA analysis has been upgraded to a new methodology and results. The new methodology was necessary for GE14 fuel.

INCORPORATING DC TEMPORARY MODIFICATIONS INTO CHECKLISTS

Activity Evaluated: Procedure Checklists 3503.01C001, 002 and 006 Log Number: 2000-123

Checklists CPS 3503.01C001, 002, and 006 are prepared as proceduralized Tracking Temporary Modifications (TTM) to support maintenance activities on Division 1 and 2 Direct Current (DC) systems during refueling outage RF-7. Only one checklist will be implemented at any given time in plant modes 4 or 5. Each checklist is independent, and can be performed independently provided initial conditions specified are maintained. No two battery chargers or batteries are connected on any class 1E DC system at the same time. The batteries connected during the temp mod will be fully charged in all cases except when a spare class 1E charger is used. The parallel core alteration activities will be in progress within limitations of Technical Specifications (TS) 3.8.5, 3.8.6 and 3.8.10. DC system design allows for a single failure or loss of any redundant DC subsystem during simultaneous accident and loss of offsite power conditions without adversely affecting safe shutdown of the plant. Only Div 1, 2, and 3 125-VDC subsystems are required to be considered for safe shutdown analysis of the plant. Batteries are kept fully-charged during normal operation. The 125-VDC motor control centers and each 125-VDC distribution panel are normally fed from their primary (charger) and secondary (battery) sources operating in parallel in a "float-charge" configuration. Loss of either source does not interrupt power flow to the bus. If Alternating Current (AC) power is lost to the battery charger, it will be restored through an alternate or standby AC power source within 15 seconds. For CPS 3503.01C001, the TTM involves connecting Motor Control Center (MCC) 1F to the Division 1 MCC 1A through two class 1E protective devices; MCC 1A spare breaker - compartment 13A (225A) and safety related fuse (200A). The cross-tie connection from a 1E to a non-1E system is acceptable and is in conformance Regulatory Guide 1.75 as amended by CPS Updated Safety Analysis Report (USAR) Section 8.1.6.1.14. This configuration allows for isolation of the non-1E power source from the 1E system during fault condition. For CPS 3503.01C002, this TTM involves connection MCC 1F to the Division 2 MCC 1B through two class 1E protective devices; MCC 1B test breaker compartment 6A to the permanently mounted test switch 1DC24E and a safety related fuse (200 A). For Loss of Coolant Accident (LOCA) conditions, the test breaker at 1DC24E contains a shunt trip that will separate the non-1E system from the 1E bus.

CPS 3503.01C006 provides the installation and removal details for the TTM. The TTM will install electrical power cables to spare charger 1DC11E. Temporary 1E class cables will be re-routed in accordance with applicable separation criteria. Annunciation circuits from the existing charger are disabled. The TTM disables annunciator 5061-4E Trouble Battery Charger 1B for 1DC07E. Operations will use other means to compensate for the defeated annunciation. The required 125 VDC system will be operating to support the ECCS mitigation within TS 3.8.5, 3.8.6 and 3.8.10 limitations, therefore radiological consequence of an accident is not increased, probability of a malfunction of equipment important to safety is not increased, and the margin of safety as defined in the basis for any TS is not reduced.

EOC-RPT BREAKER MAINTENANCE AND TESTING INTERVAL CHANGE

Activity Evaluated: Tech Spec Bases Change BL-00-017 Log Number: 2000-125

This evaluation address Technical Specification (TS) Bases Change BL-00-017 associated with Engineering Change Notices (ECN) 32188, 32189, 32190, and 32191. Plant changes replace the existing Reactor Recirculation Pump Motor EOC-RPT circuit breakers with new Cutler-Hammer (model 75DHP-VR500) retrofit breakers. The ECNs were evaluated under Safety Evaluation Log No. 2000-104 and found to not involve an unreviewed safety question. This evaluation is the End of Cycle – Recirculation Pump Trip (EOC-RPT) breaker testing interval change in the Bases for the TS Surveillance Requirement SR 3.3.4.1.5 (BL-00-017). Manufacturer requires new breaker maintenance interval to be 18 months instead of 36 months. No unreviewed safety question was determined to be involved with this activity.

INSTALLATION OF FLOORING ON CONTAINMENT REFUEL FLOOR

Activity Evaluated: CPS 1019.05 Change Log Number: 2000-126

Evaluation of the grating on the refuel floor (elevation 828'3") in Modes 3, 4, and 5 is a normal refueling activity that has been performed in previous outages, but will now be started in Mode 3 instead of Mode 4. Calculation IP-M-0635 analysis used to evaluate the installation of sheet vinyl on the Refuel Floor during Modes 3, 4, and 5. The GOTHIC Computer Program Models were used to determine that containment pressure remains below required values during an event after the grating on the refueling floor is covered. Analysis included qualitative assessment of the effects of covering the grating on containment temperature, hydrogen mixing, and other technical or safety issues. Evaluation included staging and storing items on the refuel floor and the installation of the material on the refuel floor. When installing the material ensure no "low-spot" occur where water could pool, keep one equipment hatch and two stairways open, use the specified type and quantity of material described in the analysis and install in accordance with CPS 1019.05. Pre-accident environmental conditions remain within the required limits after the grating is covered. Structural and mechanical load changes on the refuel floor and on the suppression pool strainer are insignificant, currently established standards remain unchanged. This change does not increase the probability of a malfunction of equipment important to safety, does not reduce the margin of safety as defined in the basis to a Technical Specification.

ONE TIME EXTENSION OF SURVEILLANCE INTERVAL FOR VALVE 1B21-F065B

Activity Evaluated: ORM Change 31-5 Log Number: 2000-127

Proposal that the implementation of Surveillance CPS 9381.01 – Motor Operated Valve (MOV) THERMAL OVERLOAD BYPASS VERIFICATION be deferred past its overdue date of 10/8/00 until 11/30/00 for valve 1B21-F065B. This would enable the surveillance to be completed during RF-7. This proposal requires approval of one-time extension of Operational Requirements Manual (ORM) Section 4.5.2.1a. which states, "The thermal overload protection of each valve in safety systems with a bypass device(s) integral with the motor starter shall be by-passed continuously for those directions for which the valve performs an active safety function." ORM Section 4.5.2.1a stipulates that the thermal overload protection for the required valves shall be verified to be bypasses continuously in the valve's safety direction(s) at least once per 18 months. This verification performed under Surveillance 9381.01 - MOV THERMAL OVERLOAD BYPASS VERIFICATION and requires plant shutdown. NOTE: Other valves in the Shutdown Service Water (SX) and Residual Heat Removal (RHR) systems have had similar one-time extensions for this surveillance approved earlier under SE 2000-102 & 2000-118. No design basis accidents are impacted by this activity, the activity does not increase the probability of accident previously evaluated in the SAR. Radiological consequences of a LOCA event as described in the SAR bound this activity and for that reason this activity will not increase the consequences of an accident previously evaluated in the SAR. The activity will not increase the probability of a malfunction of equipment important to safety evaluated. The function and operation of 1B21-F065B will not be changed or impacted by this deferral, the leakage rate from this valve is not expected to degrade by this deferral. The margin of safety discussed in the Technical Specifications in not reduced.

ADDITION OF A STAIRWELL DOOR IN THE RADWASTE BUILDING

Activity Evaluated: ECN 32351; USAR Change 9-302 Log Number: 2000-128

The addition of a door, 1DR1-657, in the column 131 wall between column lines D/E and 737 of the Radwaste Building is to provide additional access to a stairwell. A door is required because the stairwell is a fire zone boundary. The new door is a fire door with a self-closing feature for smoke control as required by BTP 9.5-1, Appendix A as documented in USAR, Appendix E, Section 4.0.D.4.f.; structure evaluated in Calculation SDQ17-23DG04, Rev. 4, Vol. A. Revised the following USAR Figures to reflect new door: 12.3-9; 12.3-49; Appendix D, Section II.B.2.d, Fig. D-4, Sheet 2; Appendix E, Fig. 16, FP-18a, and FP-18b. Door added by Engineering Change Notice (ECN) 32351. Generic Letter (GL) Letter 86-10, allows licensees to apply 10CFR50.59 in evaluating Fire Protection (FP) program changes with the determination of an unreviewed safety question based on the "accident...previously evaluated" being the postulated fire in the Fire Hazards Analysis (FHA) for the Fire Area affected by the change. No fire hazards (i.e., combustibles, ignition sources, etc.) are added by this activity (USAR Appendix E, F). There are no safe shutdown components or systems in the Radwaste Building. Note that smoke control is not addressed by the USAR, but is addressed by the Pre-Fire Plans of CPS 1893.04. No unreviewed safety question is involved with this activity.

FINAL FEEDWATER TEMPERATURE REDUCTION

Activity Evaluated: COLR Reload 6 cycle 7 R/2; CPS 3102.01 R/13a; CPS 3102.01C002 R/0; USAR Change 9-298 — Log Number: 2000-129

Final feedwater temperature reduction (FFWTR) change is to provide for end of cycle 7 operation with reduced feedwater temperature. This mode of operation will extend full power operation. The change considers a feedwater temperature reduction of up to and including 50 degrees F, corresponding to a maximum decrease from 420 degrees F to 370 degrees F at rated power conditions. Feedwater temperature is lowered by removing both #6 high pressure heaters from service. This change is based on "Final Feedwater Temperature Reduction Evaluation for Clinton Power Station," GENE-L12-00877-00-01P, Rev. 0, Sept 2000 and GENE Technical Services Letter ECE-2000-10, NSA: 00-354, September 22, 2000, References 1 & 2. The FFWTR Evaluation allows a FFWTR of 90 degrees F, however CPS is electing to implement a FFWTR of only 50 degrees F for Cycle 7. This change will be implemented in the Core Operating Limits Report (COLR), CPS 3102.01, CPS 3102.01C002, USAR Section 10.4.7.2.3. The COLR provides thermal limits for operating CPS. Plant operation within these operating limits is addressed in the applicable TS. GENE-L12-00877-00-01P R/0 (Ref. 1) is being added to the COLR as Reference 11. CPS 3102.01 and CPS 3102.01C002 are being updated to implement FFWTR. USAR Section10.4.7.2.3 added operational flexibility to allow the removal of feedwater heaters from service for the purpose of maintenance or cycle extension. This change is being processed to extend the length of the fuel cycle to maximize power generation.

INCORPORATING TEMPORARY MODIFICATIONS INTO CHECKLISTS

Activity Evaluated: CPS 3509.01C001 and 3509.01C002 — Log Number: 2000-130

The evaluated activities consist of revisions to CPS 3509.01C001 and CPS 3509.01C002, which incorporate proceduralized temporary modifications (temp mod) in these procedure checklists. The changes for the procedures are within the limits of prescreening criteria for the procedure changes and therefore the further evaluation of the change to the above procedure checklists are not required. CPS 3509.01C001/C002 are used for Nuclear System Protection System (NSPS) Div 1 or 2 power outage. The temp mod changes maintain operation of Reactor Water Clean-Up (RWCU) Recirculation Pumps during NSPS 1 or 2 power outage, which de-energizes pump interlock circuits. The RWCU pump controls are interlocked with RWCU inlet flow instrumentation and RWCU Pump Suction Inboard and Outboard Isolation valves (1G33-F001, 1G33-F004) control such that the low flow or closing of 1G33-F001 or 1G33-F004 will trip all pumps. The interlock circuits are de-energized during NSPS buses outage initiating pump trip. Temp mod disables RWCU inlet flow interlock by removing the fuse feeding interposing relay and steps which bypass valves 1G33-F001 & 4 position interlock, by installing jumper across the contact of position interposing relays. The RWCU inlet low flow instrumentation is fed from NSPS Division 1 and therefore fuse removal is needed for CPS 3509.01C001 only. In CPS 3509.01C001 indicating open position of 1G33-F001 is de-energized. The bypass jumper in CPS 3509.01C002 is installed across contacts of interposing relay K7, which is normally actuated on open position of 1G33-F001. The description of RWCS interlocks in USAR 7.3.1.1.2.4.1.9.6 bypasses and interlocks states that "The RWCU inlet flow signal interlocks in the RWCU pumps to stop the pumps when flow is below a predetermined value". Disabling the

RWCU inlet flow interlock impacts this section. However, section will not be revised because the changes are temporary. No unreviewed safety question is involved with this activity.

DROPPED FUEL BUNDLE WARNING SYSTEM RADIATION MONITOR CHANGE

Activity Evaluated: USAR Change 9-304 Log Number: 2000-131

The original wording "Fuel Transfer Drywell access. Incorporated into the PRM/ARM system the intended purpose of personnel warning." is vague. This change specifically addresses the type of monitor to use and the reason for the Dropped Fuel Bundle Warning System. This will give the station flexibility to use ARMs that are equivalent to the Eberline AR/PR ARM for use with the Dropped Fuel Bundle Warning System (DFBWS). There is no design basis accident (DBA) that could be impacted by this activity. This activity does not interface with systems or subsystems that involve fuel handling and the mitigation of a fuel handling accident. The DFBWS function providing local warning to personnel in drywell in the event of a dropped fuel bundle in RPV or RPV refueling pool is not changed. USAR section 12.4.1.4.1(c) rewording to address the DFBWS and its intended purpose to allow use of monitors that are equivalent to the Eberline AR/PR ARM monitors. No changes to ARM setpoints, or to the location of the monitors. This change does not affect refueling operations or the probability of a refueling accident nor will this activity pose any personnel hazards and will not increase the potential to exceed 10CFR20 limits. No new credible failure modes or accidents would be introduced by the use of an ARM that is equivalent to the Eberline AR/PR ARM as long as the ARM fulfills the radiological design objectives and ARM equipment design requirements list in USAR chapters 12.3.4.1 and 12.4.3.4.1.1. The DFBWS is neither mentioned in any sections in the Tech Spec nor does it interface with any system(s) that is/are related to any Tech Spec, safety limits, limiting safety settings or limiting conditions for operation.

ADDITION OF 900 MHz CELL PHONE SERVICE

Activity Evaluated: Mod CQ-023 Supp 1, ECN 32343, USAR Chg 9-299 Log Number: 2000-132

Numerous problems experienced where poor plant communications contributed to delays in equipment restorations and emergency plan response drills. Letter U-603231 committed per the CPS 1999 Business Plan to provide communication systems upgrades. Mod CQ-023 Supplement 1 (ECN 32343) adds additional wireless 900MHz personal communication service (PCS) system that is usable in all areas of the power block, including radio exclusion zones. The PCS system is non-safety related, non-class 1E and non-seismic. The SpectraLink PCS 900MHz cellular phone system will supplement the existing plant communication systems (CQ - dial phones, public address, sound powered jacks, microwave, fiber optic, emergency and intraplant radio). The system does not replace or supplement any of the USAR described functions of the existing communication systems and will not be required for any plant shutdown, emergency, or design bases events.

PROVIDE SERVICE AIR TO VR/VQ DAMPERS & ISOLATION VALVES DURING RF-7

Activity Evaluated: TMOD 00-060 Rev. 1 Log Number: 2000-134 R/1

Due to the containment building Instrument Air (IA) outage and a Division 1 electrical bus outage scheduled concurrently towards the end of outage RF-7 (after refueling activities have completed), the loss of either instrument air or electrical power will initiate the isolation function of the Containment Building HVAC (VR) and Drywell Purge (VQ) systems. This temporary modification (TMOD) provides a temporary source of air and necessary piping change to allow continued operation of the VR and VQ ventilation systems during these outages to support maintenance activities inside containment during the latter portion of RF-7. This TMOD will provide a temporary air supply from the Service Air (SA) system to operate particular Drywell Purge (VQ) and Containment Building HVAC (VR) dampers (1VY017Y, 1VR18Y, 1VR055Y, 1VQ003, 1VR001B, 1VQ004B) during the containment building Instrument Air (IA) outage. Outboard containment isolation valve 1VQ004A will continue to have IA available, but no power will be available to the actuator solenoid valves 1FSV-VQ033B & C during a concurrent bus outage. This TMOD will hold 1VQ004A open by providing IA to both the actuator solenoid valve's supply and the vent ports simultaneously. Outboard containment isolation valve 1VR001A will not have power or IA available, and will be held open by providing SA to both the supply and vent ports for solenoid valves 1FSV-VR008B & C. Section 3.6.1.3 of the Clinton Power Station (CPS) Technical Specifications (TS) requires operability of these primary containment isolation valves during modes 1, 2, and 3. Since 1VR001A/B and 1VQ004A/B are also secondary containment bypass valves, they are required to be operable during core alterations, irradiated fuel handling, and operations with the potential to drain the reactor vessel. Installation of the TMOD is limited to during cold shutdown (mode 4) only. After installation of the TMOD the VR/VQ systems will be limited to the containment ventilation mode of operation only. As stated in the USAR, this mode of operation is the preferred means of providing ventilation to containment during cold shutdown and refuel modes, although this TMOD will not be used in any plant mode other than cold shutdown.

INSTALLATION OF MANUAL INTERLOCK IN IFTS CONTROL CIRCUIT

Activity Evaluated: TMOD 00-070 Log Number: 2000-135

Inclined Fuel Transfer System (IFTS) tube full sensors prevent the operation of the flap valve or movement of the fuel in the upward direction if the IFTS tube is not full. These sensors have degraded and no longer function properly. The cable tubes communicate with the top of the IFTS tube, and fill with water after the IFTS tube is completely full which allows IFTS operators to rely on observation of water exiting the top of the cable tubes. The IFTS operator has a manual switch to simulate the lever signal to indicating lamps and to the section of the logic affected by the IFTS tube full sensors. The transfer tube will be verified to be empty with the IFTS "tube empty" sensors, which are unaffected by this activity.

These "tube full" sensors determine when the IFTS tube is full to allow the carriage to be raised and the operation of the IFTS Upper Pool Flap valve. Use of these sensors prevents the opening of the upper valve if the IFTS tube is not completely full. The sensor output has degraded to the point that the IFTS system can no longer be operated without bypassing these sensors and relying on some other form of indication that the IFTS tube is full. Therefore, a manual interlock is being installed on temporary modification (TMOD) until permanent repairs can be made.

INSTALLATION OF BLIND COUPLINGS IN RR DISCHARGE VALVE BONNET VENT LINES

Activity Evaluated: ECN 32360 (USAR Change 9-313) and ECN 32361 (USAR Change 9-315)   Log Number: 2000-136

Install blind couplings in the bonnet vent lines off of the Reactor Recirculation Pump Discharge Valves, 1B33F067A/B which in turn renders the vent line unusable and removes process pressure from vent line isolation valves 1B33F068A/B and 1B33F069A/B. The blind couplings installation is on 3/4" vent lines (1RR10AA 3/4 and 1RR10AB 3/4). Associated documentation to update the Piping & Instrumentation Diagrams (P&ID)s and piping drawings is addressed in the evaluation. USAR change packages 9-313 and 9-315 are used to update Figures 3.6-1, sheets 57 and 58 and 5.4-2, sheets 1 and 2. The bonnet vent line is not used. Installing the blind coupling eliminates the possibility of leak-by through the vent line isolation valves or packing leaks from these valves, thus eliminating possible paths for drywell leakage. Maintenance requirements for the 3/4" vent line isolation valves are also eliminated.

MANUAL SWITCH INSTALLED FOR LOWER UPENDER VERTICAL SENSOR

Activity Evaluated: TMOD 00-077   Log Number: 2000-137

Temporary Modification (TMOD) 00-077 installs a manual interlock for the Fuel Building fuel transfer system upender, 1F42-N021A, in the IFTS control circuit. The TMOD is developed as a contingency for upender inclined/not-inclined position sensor failure, preventing IFTS operation. It allows the IFTS operator to select the manual switch using the control system selector switch, provides a temporary manual switch in the Inclined Fuel Transfer System (IFTS) control circuitry for the inclined/not-inclined position. The switch is manually operated when the operator observes the Fuel Building upender in the correct position. Visual observation is considered an equivalent method of determining the upender position. No other interlocks associated with the IFTS control circuitry is affected by the proposed activity. TMOD 00-077 is a contingency temporary modification where both Fuel Building upender position sensors for the inclined/not-inclined position have failed, thus preventing IFTS operation. If these sensors fail, they are not readily accessible for replacement during fuel transfer operations. Therefore, the TMOD allows IFTS operation to continue using a manual switch activated by an operator until such time as permanent repairs can be made.

The proposed activity does not increase the probability of an accident, does not increase the consequences of an accident, does not increase the probability of a malfunction of equipment important to safety, and does not reduce the Margin of Safety as defined in the basis for any Technical Specification.

DEFERRAL OF ELECTRICAL PM TASKS

Activity Evaluated: PEMAP1301, PEMAP1303 Log Number: 2000-138

One time deferral of these Preventive Maintenance (PM) activities is desired in order to minimize their impact on other scheduled outage activities. As stated earlier, these PMs are normally scheduled during a Division 1 AC bus outage and there is no Division 1 AC bus outage in the schedule for RF-7. The next scheduled Division 1 bus outage is planned during RF-8 in Spring of 2002. These PMs are past their identified due date and will expire in Spring of 2001.

PEMAP1301 & PEMAP1303 are associated with functional testing of certain 120 volt circuit breakers. Both PM tasks are required to be performed to satisfy the requirements of ORM Section 2.5, "Electrical Power Systems" Subsection 2.5.1, "Containment Penetration Conductor Overcurrent Protective Devices." The ORM requirements were created as a result of CPS Technical Specification Amendment 95. Implementation of the new ORM requirements resulted in re-zeroing each PM clock and establishing new PM due dates which to correspond to the appropriate schedule windows applicable at that time. These two PM activities were scheduled with the Division 1 bus outage, scheduled for RF-8. At that time RF-8 was scheduled for 10/15/1999. With 1.25 allowance provided for by ORM 1.3.2, and the re-zeroing of the PM clock in June 1995, the actual maximum late date allowable for these PMs under the ORM requirements is past the current RF8 (4/15/2002).

The intent of the frequency of this test program is to insure that adequate reliability of equipment is verified while not unduly removing equipment from service. The function for which these breakers are considered "risk significant" by the Boiling Water Reactor Owners Group (BWROG) is to provide power to loads important to safety. The function for which the breakers are being tested (fault interruption) is considered low safety significant, according to the site maintenance rule program, GE NEDO-31466, and the NRC SER for TS Amendment 95 (page 79). Therefore ORM 4.5.1 requirement of six years is conservative. One time deferral to RF8 will still meet the BWROG recommendations of 10 years maximum form the previous actual performance of testing of these breakers.

This activity will not increase the probability of an accident, will have no effect on the consequences of any design basis accidents, will not increase the probability of a malfunction of equipment important to safety, will create the possibility of an accident of a different type than any evaluated previously in the SAR, will not reduce the Margin of Safety as defined in the basis for any Technical Specification.

CYCLE 8 FUEL RELOAD AND CORE DESIGN

Activity Evaluated: Modification NB-034, Supp. 3, ECN 32299 Log Number: 2000-139

This activity is Supplement 3 to modification NB-034, Supplement 2 was approved under 50.59 evaluation number 2000-120. The supplement to the modification is necessary to issue the reload licensing information for cycle-8 core loading to allow continued operation. The updates to the Core Operating Limits Report (COLR) and USAR are necessary to reflect the cycle-8 core configuration and reload licensing in these documents. The introduction of the GE14 fuel bundle supports the strategic and economic goals for operating the facility.

This evaluation performs the assessment of the Modification NB-034 Supplement 3, which issues the necessary licensing documentation including: the reload licensing analysis and core operation reports (ECN 32299); the cycle-8 COLR, which contains the cycle specific core operating limits; USAR change 9-311, which implements the USAR changes to support the modification; and the use of the new GE14 fuel bundle.

This change will not increase the probability of an accident, will not change the consequence of an accident previously evaluated in the USAR, and will not change the probability of malfunction of equipment important to safety. The evaluation on the radiological consequences showed that the original design basis consequences still bound the consequences due to the reload. This change does not increase the possibility for an accident of a different type from any previously evaluated in the USAR. The margin of safety as defined in the basis for any Technical Specification is not reduced.

EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE

Activity Evaluated: USAR Change 9-327 Log Number: 2000-140

CR 1-99-02-307 identified that USAR Appendix D, Item II.K.3.44, needed to be revised such that the potential effects of the ADS bypass timer(s) are considered in connection with mitigation of the loss-of-feedwater transient (assuming a failure of HPCS with RCIC unavailable) using ADS with low pressure makeup, thus ensuring that either the core remains covered (water level above the top of active fuel) or, with partial core uncovery, that no fuel damage occurs, using appropriate evaluation or analysis. The USAR change reflects the analysis performed by GE to support that no fuel damage occurs during the core uncovery.

This activity involves revising USAR Appendix D (II.K.3.44), "Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure." The current USAR discuss the evaluation performed by GE that even with degraded conditions involving one stuck open relief valve in addition to the worst case transient with the worst single failure, the core remains covered. The generic evaluation was performed prior to a generic BWR/6 design change that incorporated a bypass timer into the ADS initiation logic. The USAR is now being revised to reflect a June 2000 CPS-specific analysis which will correct the USAR to state that the core will experience uncovery briefly, however the fuel heatup due to core uncovery is minor such that no fuel damage occurs.

This change will not increase the probability of an accident previously evaluated in the SAR, will not increase the consequences of an accident previously evaluated in the SAR, will not increase the probability of an equipment malfunction important to safety previously evaluated in the SAR, will not create an accident of a different type than previously evaluated in the SAR, and does not reduce the margin of safety as defined in the basis for any Technical Specifications.

LICENSING DEPARTMENT REVIEW OF PROCEDURES IMPLEMENTING TECHNICAL SPECIFICATION REQUIREMENTS

References: LER 89-025, LER 89-026

This commitment was written in 1989 to ensure compliance with Technical Specifications (TS) by having the Licensing Department review all procedure changes which implemented TS requirements as a result of failing to meet TS surveillance requirements (SR) because of inadequate procedures. The inadequate surveillance procedures were determined to be caused by inconsistent interpretation of confusing TS notes.

The corrective action for these events was "to provide additional assurance that consistent interpretations of the TS requirements are incorporated into Clinton Power Station (CPS) operating procedures. Administrative CPS procedures 1005.01 and 1005.07 will be revised to require that Licensing review any change to sections of station procedures if those sections implement TS requirements."

Since 1989, there have been a number of improvements in TS implementation. Improved TS (ITS), which resulted in a complete change in the format of the CPS TS, were implemented in January of 1995. As part of this conversion effort, each of the procedures that implement TS requirements were reviewed to determine required changes to implement the changes to ITS format. In addition, Licensing has had the opportunity to review each of the procedures several times through the biennial review process. Finally, there have been a number of initiatives to improve the licensed operator's knowledge and application of TS at CPS. These initiatives provide added assurance that procedure change preparers and reviewers in Operations will continue to maintain compliance with the CPS TS.

Based on the above, there is no longer a need for Licensing personnel to review plant procedures for proper implementation of TS requirements and these two commitments (one for each site procedure) are closed.

The following revisions to the Clinton Power Station (CPS) Operational Requirements Manual (ORM) have been made in accordance with approved station procedures. Changes to the ORM are reviewed per 10 CFR 50.59. These changes have been implemented since the submittal of the CPS USAR Revision 8 to the NRC, through November 12, 2000.

Revision Number	Scope of Revision
Revision 26	• Revised section 6.5.3.1.a, 6.8.2 & 6.8.3.c to reflect NRC approval of a 10CFR50.54 (a) submittal which allows certain safety-related procedures to be given final approval by the department head rather than the plant manager. ANSI N18.7-1976/ANS 3.2 requirements are still met by the Clinton Power Station program. • Revised Attachment 3-4, MOV Thermal Overload Protection and CIV Attachments 4-6, 4-9 & 4-14 to reflect deenergized valves 1FP051, 1FP054, 1FP078 and 1FP079. • Revised CIV Attachments 4-4 and 4-11 add "#" under applicable modes and to show certain valves as being secondary containment bypass paths. • Relocated Section 6.5.1.6 discussion of the on-site review group to USAR Subsection 13.4.1. • Revised Attachments 3-3 and 3-4, MOV Thermal Overload Protection, to reflect change thermal overload directions for certain MSIV leakage control & hydrogen recombiner valves. • Revised Section 2.6.3, Refueling Platform to include Inclined Fuel Transfer System (IFTS).
Revision 27	• Relocated discussion of Independent Safety Engineering Group (ISEG) to USAR Subsection 13.4.3.
Revision 28	• Revised Attachments 3 (MOV Thermal Overload Protection) and 4 (Containment Isolation Valves) to add components as a result of feedwater keepfill modification, FW-39.

Revision Number	Scope of Revision
Revision 29	• Revise Sections 6.8.2 (Review and Approval) and 6.8.3.c (Temporary Changes) to replace reference to Section 6.5.1.6 (on-site review group) with USAR 13.4.1. • Retitled the Illinois Power QA Manual to Clinton Power Station QA Manual. • Revise titles in Sections 6.5.2, 6.6 & 6.7 to reflect change in ownership from Illinois Power to AmerGen. • Relocated Section 6.5.2 (Nuclear Review Board) to USAR Subsection 13.4.2. • Clarified Attachment 3 (MOV Thermal Overload Protection) with a note explaining the ORM requirement applies to the valve's safety function direction and may not apply to both directions. • Revise Attachment 4 (Containment Isolation Valves) to show Breathing Air valves closed, since Instrument Air to them was isolated per a design change.
Revision 30	• Delete requirements for differential temperature instrumentation from Section 2.2.16. • Revise Section 2.2.14 (NSPS Self Test System) to clarify Operational Requirements and Action Times to prevent unnecessary entry into Actions.
Revision 31	• Removed excess flow check valve (EFCV) differential pressure test methodology and allows only flow testing. Restored minimum closure setpoint for some EFCVs. • Revised radiation setpoint in Section 2.6.3 (Refueling Platform) from 10 mR/hr to 50 mR/hr. • One time extension of surveillance interval for certain SX valves under TR 4.5.2.1 a). • One time extension of surveillance interval for valves 1E12-F042C and 1B21-F065B under TR 4.5.2.1 a).

The following deletions from the Clinton Power Station (CPS) USAR were made during the since the submittal of the CPS USAR Revision 8 to the NRC, through November 12, 2000.

Subsection Number	Subject of Deletion
1.8	• Deleted obsolete references to voided E02 drawings in the CPS position on Reg Guide 1.6. • Deleted obsolete reference to a one-time exemption to the surveillance requirements of Division 4 battery in Reg Guide 1.129.
2.3.3	• Deleted details of meteorological tower (height, base elevation, & frequency of chart paper change).
2.5.2.6	• Deleted non-existing reference to CPS PSAR.
3.6.1	• The time the turbine building siding blows off following a postulated simultaneous rupture of a main steam line and a feedwater line in the steam tunnel outside containment. Considered excessive detail.
D3.6.2.10.2	• Delete actual number of restraints in discussion about containment isolation valve restraints.
3.8.3.5.1.1.c 3.8.3.5.1.2.c	• Deleted obsolete references to CPS PSAR.
C3.8	• The calculations for three representative masonry walls are deleted and updated design criteria for masonry walls provided.
Table 3.9-1a	• Removed thermal transient cycles parameters & columns involving initial, intermediate, final temperatures, time intervals, rate of temperature change parameters and the short term ramp function. Reg. Guide 1.70 requires listing the transients & number of events/cycles but not require listing all of the event temperatures, times, etc.
4.3.1.2	• Delete obsolete reference to NEDE 24011-P-A.
5.2.3.3.4	• Deleted specific temperature information and statement concerning "unused electrodes" from moisture control of arc-weld electrodes.
5.4.1.3	• Removed statement referring to the use of high-speed recirculation (RR) pumps for the purpose of plant heat-up for hydrostatic tests and statement about the type & heat transfer rate of thermal insulation for RR components & piping. • Deleted specific thrust value of the hydraulic actuator of the RR flow control valves.
5.4.6.2.2.2 (3)	• Deleted statements referring to the specific accuracy of flow & pressure instrumentation.
5.4.7.1.2	• Deleted specific flow value for RHR minimum flow valve actuation. • Deleted statement that "setpoint tolerance" of RHR relief valves is provided on USAR Figure 5.4-13.
5.4.8.2	• Deleted specific conductivity setpoints for the condensate polishing demineralizers.

Subsection Number	Subject of Deletion
6.1.1.1.1	• Deleted values of corrosion allowances for pressure retaining components of the Emergency Safety Features (ESF).
6.1.5.1.2.1	• Deleted standby gas HVAC air filter package discussion on adsorber frames are in accordance with ERDA 76-21 since it does not apply.
6.2.2.2	• Deleted vendor/brand name for anti-sweat insulation used on cold piping and equipment in containment.
Table 6.2-47	• Deleted "test type" & "type C test direction" columns from containment isolation valve list. • Deleted Note 22 (cross-reference to USAR P&ID numbers). • Deleted Note 23 (specific description of each valve).
6.4.6	• Deleted discussion of locked local control panels located inside equipment rooms under the administrative control of operators from instrumentation requirements relating to control room habitability.
6.5.1.1.3	• Removed SGTS calculated post-LOCA temperature for the carbon filters.
6.5.1.2.1	• Deleted the calculated heating requirement for the SGTS electric heaters and SGTS HEPA filter size, frame and gasket material.
6.5.1.2.2 6.5.1.2.3	• Deleted MCR HVAC air filter package discussion on adsorber frames are in accordance with ERDA 76-21 since it does not apply.
Table 6.5-1	• Deleted SGTS maximum demister water removal rate
Table 6.5-3	• Deleted actual clearance provided for filter replacement.
6.7.2.3	• Removed MSIV-LCS valves specific stem speeds. Added reference to correct controlling document.
Table 7.1-13	• Note 16 revised to remove vendor & model number of calibration unit. • Note 17 revised to remove reference to detector model numbers.
7.3.1.1.6.10 (1)	• Removed wording that identifies divisional power to motor-operated deluge valves in MCR HVAC.
7.3.1.1.7.3	• Removed CGCS time delay setpoints.
7.3.2.2.2.3.1.1 Table 7.3-12	• Deleted table with ranges for the types of sensing instruments used for isolation and deleted reference to the table.
7.6.1.2.6.3.1 7.6.1.2.6.3.2 7.6.1.2.7.3.1	• Removed specific radiation detector model numbers.
7.3.1.1.6.3 7.7.1.13.3.1 7.7.1.14.3.2.2 7.7.1.14.3.3 7.7.1.17.3.1.2 7.7.1.17.3.2 7.7.1.18.3.2.2 7.7.1.18.3.3	• Deleted specific value for chiller minimum temperature shutdown for cooling coil freeze protection.

Subsection Number	Subject of Deletion
8.3.2.1.2.1 8.3.2.2.2.4	• Deleted obsolete reference to a one-time exemption to the surveillance requirements of Division 4 battery during battery replacement.
9.1.2.3.1.1	• Deleted actual percentage of fuel compaction from dropped fuel assembly in design analyses of the spent fuel pool.
9.1.3.1.2	• Deleted values of SX & FPC&C system flow rates.
9.1.4.1	• Deleted values of allowable stresses for fuel handling platform.
9.1.4.2.1	• Deleted method of the backup cooling for the spent fuel cask.
9.1.4.2.3.10	• Deleted actual value of inclined fuel transfer system movement during installation of blank flange.
9.1.4.2.5.1	• Deleted design safety factor of individual reactor vessel service tools.
9.1.4.2.5.6	• Deleted discussion of inspections following the dryer & separator strongback load testing.
9.1.4.2.6	• Deleted discussion of replacement method of incore guide tube seal.
9.1.4.2.10.2.1.1	• Deleted dimensions & weights of new fuel shipping crates.
9.1.4.2.10.2.2	• Deleted reactor vessel water level during cooldown prior to refueling.
9.2.8.1.2	• Deleted partial list of control building water chillers trips/alarms.
9.2.8.2.2	• Deleted partial list of drywell water chillers trips/alarms.
9.2.8.3.2	• Deleted number of WO chillers running & in standby. (seasonal)
9.2.8.4.2	• Deleted partial list of service building water chillers trips/alarms.
Table 9.2-8	• Deleted CCW heat exchanger tubes & shell inlet & outlet temperatures.
9.3.1.3	• Deleted statement regarding sharing of safety-related portions of the compressed air systems.
9.3.1.5	• Deleted partial list of service air compressor trips/alarms.
9.3.6.2	• Removed the particular type of resin used in the FPC&C demineralizers. • Removed obsolete discussion on a two unit plant site.
Table 9.3-3	• Deleted on-line sampling instrumentation ranges, alarm setpoints, and computer points.
9.4.3.5	• Deleted type (keylock) of control switch for the auxiliary building exhaust fan control switches.
9.4.4.4	• Deleted statement about ability to change turbine bldg. filters on-line.
9.4.5.3.2	• Deleted statements about having shutoff valves near each ECCS cooling coil for ease of maintenance and about auto-start signal overrides manual switch for ECCS cooling system.
9.4.5.4.2	• Deleted statement about having shutoff valves near each SX pump room cooling coil for ease of maintenance
9.4.7.1.2	• Deleted discussion of maximum allowable drywell temperatures during hot shutdown from drywell cooling system subsection.
9.4.7.2.2	• Deleted the locally-mounted location of drywell purge instrumentation.
9.4.9.2	• Deleted discussion of ducting for machine shop HVAC and deleted listing of indications provided on local panels for machine shop HVAC.

Subsection Number	Subject of Deletion
9.4.11.2	• Deleted specific laboratory HVAC parameters on local control panels. • Deleted discussion of presence of a laboratory HVAC audible trouble alarm located in the radiation chemistry office.
9.4.12.1.2	• Deleted temperature & humidity ranges that are maintained in the Record Storage Facility by service building HVAC.
9.4.12.2	• Deleted specific service building HVAC parameters that are displayed on the local control panel.
9.4.13.1.2	• Deleted discussion of radwaste area ventilation system isolation damper interlocks.
Table 9.4-3	• Deleted pressure drop across the fuel building HVAC supply filter.
Table 9.4-5	• Deleted pressure drop across the auxiliary building HVAC supply filter.
Table 9.4-7	• Deleted pressure drop across the turbine building HVAC supply filter.
Table 9.4-9	• Deleted pressure drop across the diesel room HVAC supply filter.
Table 9.4-11	• Deleted pressure drop across the main & standby switchgear heat removal coil cabinet filters.
Table 9.4-19	• Deleted pressure drop across the containment HVAC supply filter.
Table 9.4-19a	• Deleted pressure drop across the continuous containment purge supply filter.
Table 9.4-23	• Deleted pressure drop across individual filters in drywell purge trains. • Deleted weight of activated charcoal in drywell purge filter trains.
Table 9.4-25	• Deleted chilled water flow requirements through individual components of the off-gas vault refrigeration skid. • Deleted condenser capacity from off-gas refrigeration skid.
Table 9.4-27	• Deleted efficiency classification (medium or high) of the machine shop exhaust train pre-filters. • Deleted pressure drop across machine shop supply and exhaust trains pre-filters and HEPA filters. • Deleted design details and equipment numbers of machine shop exhaust dust collector and moisture separator.
Table 9.4-31	• Deleted flow rates for laboratory HVAC electrical heaters. • Deleted pressure drops across laboratory HVAC filters.
Table 9.4-32	• Deleted pressure drops across the service building HVAC filters. • Deleted design details of service building HVAC humidifiers.
Table 9.4-33	• Deleted pressure drops across the radwaste building HVAC filters.
9.5.5.2	• Deleted estimate of engine heat rejection by the diesel generator. • Deleted design operating pressure & inlet service water temperature range for the diesel generator heat exchanger.
10.2.2.1	• Deleted number of extraction points with the high & low pressure turbines for feedwater heating.
10.2.2.2.1	• Deleted distance between hydrogen storage and safety-related equipment.

Subsection Number	Subject of Deletion
10.2.3.6	• Deleted description of "cast" in the discussion of material used for the extraction steam check valve discs and disc arms. • Deleted description of extraction steam disc arm free swing cut-out.
10.4.1.1.2.4	• Deleted listing of total non-condensable loads from main condenser.
10.4.1.5.2	• Deleted setpoints of condenser protective instrumentation.
10.4.2.1.1	• Deleted expected hydrogen removal rate from main condenser.
10.4.2.2	• Deleted specific value of the main condenser vacuum when the steam jet air ejector takes over from the mechanical vacuum pump.
10.4.5.2	• Deleted design temperature range of water supply for main condenser.
Table 10.4-1	• Deleted circulating water pump discharge valve stroke times.
11.2.2.2	• Deleted evaluation of viable alternatives for the disposal of contaminated oil collected by oil separators.
Fig 11.3-2	• Deleted descriptive details concerning off-gas main process routing.
Appendix A	• Deleted definitions of "Major Load Changes", "Manual Component", "Place in Isolated Condition", "Site Features", "Source Material", "Special Nuclear Material", and "Technical Specifications".
Appendix D	• Sections I.C.4 & II.B.1. Deleted last sentence of the CPS response concerning the availability of procedures for NRC Region III review.

Attachment E
Revised USAR Pages