DUKE POWER COMPANY

OCONEE NUCLEAR STATION

Summary of Nuclear Station Modifications and Exempt Change Variation Notices Completed under 10CFR 50.59.

ON - 733

DESCRIPTION: This modification adds red head switches in series with the auxiliary switches on air circuit breakers 1,2,3, and 4. This will allow control circuits to be opened during maintenance to prevent a unit from tripping accidentally and will reduce the time required for checking continuity of the auxiliary switch contacts.

SAFETY EVALUATION: This modification involves installation of a different type of terminal block. There will be no functional change to any system affecting nuclear safety. No unreviewed safety questions are created by this modification.

STATUS:	Keowee	Keowee
	Unit 1	Unit 2
	Complete	Complete

ON - 908

DESCRIPTION: This modification installs a radiation monitoring unit designed for particulate and iodine sampling in the Interim Radwaste Building ventilation system.

SAFETY EVALUATION: This modification will allow for the monitoring of particulates and iodine in the Interim Radwaste Building ventilation system. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 1578

DESCRIPTION: This modification replaced the Teledyne seismic switch with a Kinemetrics Triaxial Seismic Switch.

SAFETY EVALUATION: The seismic trigger feeds control room annunciator alarms when design basis seismic motion is detected. This system has no effect on controls. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

ON - 1826

DESCRIPTION: This modification involved the addition of circuitry to allow testing of the control oil pressure switches on both the motor driven and turbine driven emergency feedwater pumps.

SAFETY EVALUATION: This change was incorporated to satisfy a regulatory commitment contained in NUREG 0578. The additional circuitry allows operators to check the operability of the automatic actuation feature for the motor driven and turbine driven emergency feedwater pumps.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 1862

DESCRIPTION: This modification installs three storage tanks, two pumps, and connected piping for collection, processing, and discharge of contaminated turbine building sump water.

SAFETY EVALUATION: This modification will prevent an uncontrolled release of radioactivity in the event Turbine Building sumps accumulate water faster than can be processed. No safety system will be degraded, and no functional change to any system will be made. No unreviewed safety questions will be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 1890

DESCRIPTION: This modification installed a cross-connect with an isolation valve between the fuel transfer canal drain line and the fuel transfer canal fill/drain header.

SAFETY EVALUATION: No safety systems will be degraded and no functional changes will be made to any system as a result of this modification. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 2004

DESCRIPTION: This change to the HPI system allows pressurizer auxiliary spray lineup without reactor building entry. Also, flow control to the auxiliary pressurizer spray nozzle is remotely controlled in the control room instead of manually from the penetration room. Properly sized orifices assure adequate nozzle warming flow with indication in the control room.

SAFETY EVALUATION: Adequate warming flow to the affected HPI nozzles will be maintained. Spurious operation of HP-355 is prevented, as such no unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Incomplete	N/A

ON - 2174

<u>DESCRIPTION:</u> The purpose of this modification is to change the emergency power switching logic to prevent the repeated cycling of startup transformer feeder breakers if the startup transformer power source cannot carry Oconee loads at acceptable steady state voltage levels. Under the new logic, if the startup transformer feeder breakers cycle a single time (close-open-close-open) within one minute, the emergency power source will be switched to the standby busses. Failure of the new circuit to operate results in a return to the current configuration.

<u>SAFETY EVALUATION:</u> This modification provides protection of the startup transformer feeder breakers in event of breaker chattering due to fluctuating voltage. Spurious operation of the modified circuit will cause premature switch to the standby busses but this will not degrade the system performance or its safety function. The new circuit would not degrade any safety system, therefore no unreviewed safety question is posed by this modification.

<u>STATUS:</u>	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 2180

DESCRIPTION: This modification provides the interim upgrade (Phase I) of the operator aid computer and peripherals to accommodate SPDS and PT displays. SPDS is provided per NUREG 0737 and Generic Letter 82-33.

SAFETY EVALUATION: No safety system will be degraded by this modification and no functional change to any system will be made. The operator aid computer capacity will improve the detail and amount of information available to the operators. No unreviewed safety question will be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Incomplete	Incomplete	Complete	Complete

ON - 2338

DESCRIPTION: This change replaces the core flood tank fill header isolation valve 3HP-154 with a gate valve with chain operator.

SAFETY EVALUATION: The portion of the HPI system which is affected by this NSM is not included in the initiation or mitigation of any accident previously evaluated in the FSAR. Valve 3HP-154 serves no safety function other than its role as a passive boundary of the HPI system. Therefore, neither the probability nor consequences of such an accident will be increased as a result of this NSM. Since the new valve performs essentially the same function as the existing valve, no possibility of an accident different from any already evaluated in the FSAR will be created. Due to the previously mentioned similarity in the valve function, there is no increase in probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR and no possibility of malfunctions of equipment important to safety different from any already evaluated in the FSAR is created. No margin of safety as defined in the bases to any Tech Spec. is reduced. There are no unreviewed safety questions associated with this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

ON - 2371

DESCRIPTION: This modification makes necessary changes to the wall separating the Turbine and Auxiliary Buildings to provide a flood barrier.

SAFETY EVALUATION: No safety system will be degraded and no functional change will be made to any system as a result of this modification. This modification eliminates the possibility of a turbine building flooding event causing damage to auxiliary building equipment. No unreviewed safety question is judged to be created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 12428
ON - 22428
ON - 32428

DESCRIPTION: This modification upgrades the post accident liquid sampling system to meet the criteria of NUREG 0737. This modification affects FSAR Section 9.3.6.

SAFETY EVALUATION: This modification was required for compliance with NUREG 0737 Item II.B.3. This modification improves system accuracy and enhances system operation. No adverse impact to any safety or non-safety system will result. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12439

DESCRIPTION: This modification removes the stop assembly on fuel transfer carriages in order to provide clearance for combinations of MK B-4 and MK B-5 fuel assemblies and components.

SAFETY EVALUATION: This modification prevents fuel assemblies from hanging up in the fuel carriage. The probability or consequences of a fuel handling accident are not increased by removal of the stop assembly. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

ON - 32441

DESCRIPTION: This modification replaced the existing obsolete fuel assembly hoist load sensing and readout device with an improved digital readout type.

SAFETY EVALUATION: Replacement of the load sensing device will improve the safety of fuel handling operations. This modification will not create any unreviewed safety questions.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Incomplete	N/A	Complete	N/A

ON - 12443
ON - 32443

DESCRIPTION: This modification replaced the existing air powered motor driven chain drive spent fuel transfer carriage system with an electric motor driven winch system.

SAFETY EVALUATION: This modification does not increase the probability or consequences of fuel handling accidents since interlocks and travel stops are not eliminated. There are no unreviewed safety questions created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	Complete	N/A

ON - 12460
ON - 22460
ON - 32460

DESCRIPTION: This modification increases the capacity of the SSF reactor coolant makeup pump, improves pump flow measurement capability and revises the pump test loop to accommodate pump flow rate changes.

SAFETY EVALUATION: No safety system will be degraded nor will any functional changes take place. This modification assures sufficient pump capacity is available. No unreviewed safety questions are judged to be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12487
ON - 22487
ON - 32487

DESCRIPTION: This modification replaces component cooling system gate valve CC-8 with a butterfly valve. This change affects FSAR Table 6.2-3.

SAFETY EVALUATION: The replacement valve meets or exceeds the original design specification. There are no unreviewed safety questions associated with this change.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Incomplete	Complete	Complete	N/A

ON - 12626
ON - 22626
ON - 32626

DESCRIPTION: The incore instrumentation trench is a high radiation area during refueling when the incore detectors are withdrawn from the core. The area was shielded with temporary shielding requiring excessive personnel time and increased radiation exposure to remove and install this shielding. Permanent shielding has been installed on existing structural steel and was seismically designed to QA 4 requirements to remedy this problem.

SAFETY EVALUATION: The incore instrumentation trench contains the guide tubes which are used for insertion and withdrawal of the incore detectors. These detectors are used to provide a history of power distribution and fuel burnup data to be used in fuel management decisions. During refueling outages, these detectors are withdrawn from the core via the incore instrument tank to allow free transfer of the fuel assemblies. While withdrawn, the radiation levels are quite high, requiring shielding if personnel are to be allowed in the area. Temporary shielding was used, but this involved increased personnel time and exposure. Permanent shielding reduced manhours and radiological exposure associated with handling temporary shielding during outages.

The incore detectors are not safety related. The excore nuclear instrumentation (NI's) are used for inputs to the Integrated Control System (ICS) and Reactor Protective System (RPS). Therefore, with respect to plant operation and safe shutdown, the incore detectors are not important. However, it is imperative that the guide tubes, which are part of the RCS pressure boundary, maintain their structural integrity. This requires the permanent lead shielding and structural supports to be seismically designed.

The small voltages and currents associated with the incore detectors will not be disturbed by the presence of permanent lead shielding during plant operation. Also, there is no known mechanism of corrosion that would be unacceptable caused by the presence of lead and borated water near the stainless steel guide tubes, following a loss of inventory accident and subsequent flooding of lower containment.

24 of the 52 incore detectors have thermocouples which provide temperature indication for Post-Accident-Monitoring concerns. This is not of greater importance than preserving the RCS pressure boundary, and seismic design of the permanent lead shielding supports should preserve this function.

The QA Condition 4 designation adequately covers any safety concerns associated with this modification.

Accordingly, this modification will have no effect on the probability, consequences or possibility of new accidents evaluated in the FSAR. Nor will it affect the probability, consequences or possibility of malfunctions of equipment important to safety evaluated in the FSAR. The margin of safety defined in the bases of the Tech. Specs. is unaffected. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12630
ON - 22630
ON - 32630

DESCRIPTION: Single input failures to the ICS/NNI can lead to transients induced by the ICS/NNI acting on erroneous input information. This change modifies the ICS to provide to automatic selection of input signals from valid sensors over failed ones. This was accomplished by supplementing each of the present manual selecting circuits for 15 inputs and one spare with an automatic selecting circuit known as Smart Automatic Signal Selector (SASS).

SAFETY EVALUATION: The purpose of the subject NSMs is to modify the Integrated Control System (ICS) to provide the automatic selection of input signals from valid sensors over failed ones. This is accomplished by the installation of a self-contained, microprocessor-based system designed to evaluate the validity of redundant analog process control signals. In the event of a gross failure of one of the redundant signals, the system automatically selects the non-failed signal. Affected input signals are as follows:

FDW Flow (A&B)
OTSG Start Up Level (A&B)
OTSG Operating Level (A&B)
Turbine Header Pressure
RCS Pressure*
RCS Flow (A&B)*
T Hot (A&B)
T Cold (A&B)
Power Range Flux*
*RPS inputs

The ICS is a non-safety control system designed to provide coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions in order to match megawatt generation to unit load demand. Major system parameters, such as those affected by this modification, are available to the ICS from redundant sensors. The ICS is normally powered from a dedicated static inverter system, which receives input from the Vital I&C batteries, and is backed by an AC input from one of the plant's regulated non-load shed buses. Essential plant parameters necessary for shutdown are arranged with their power supplies independent of the ICS source. Power for the new selector system is fed from the ICS source, so from a loss of power standpoint, the new system is as reliable as the ICS itself.

As strictly a monitoring system, the selector does not interact with the process signal, and thus cannot add error to the control signal. Also, the selector does not introduce any new failure modes into the ICS and so will not increase the probability of any resulting plant transients. The selector will, in fact, significantly enhance the ability of the ICS to moderate plant response to accidents or transients by automatically selecting valid input signals and alleviating the need for operator action. A failure of the selector will stop the selection process, but will not adversely affect the plant.

ON-12630
ON-22630
ON-32630
(cont'd)

Though most FSAR accident analyses have been performed without credit for ICS availability, the addition of signal selectors will not degrade the ability of the ICS to provide its intended design functions. Also, the manual mode will still be ultimately available to the operators. Therefore, the consequences of any previously analyzed accidents or equipment malfunctions will not be increased by this modification. No new accidents or equipment malfunctions will be created.

The addition of the signal selector system to the ICS does not involve unreviewed safety questions.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12637
ON - 22637
ON - 32637

DESCRIPTION: This modification replaces Power Circuit Breakers (PCB) in the 230kV switchyard. Studies have indicated that fault current could exceed the PCB interrupting capability.

SAFETY EVALUATION: The relaying changes in this modification will not affect any safety related components or the ability of the external grid trouble protection system to function as designed and no new failure modes will be created. Replacing the PCBs with ones which are capable of interrupting the possible fault current will decrease the possibility of a unit separation from the transmission system. This modification will increase the reliability of the 230kV switchyard. As such, no unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Incomplete	Incomplete	Complete	N/A

ON - 12642
ON - 22642
ON - 32642

DESCRIPTION: This modification replaces component cooling system gate valve CC-7 with a soft seated butterfly valve. This change effects FSAR Table 6.2-3.

SAFETY EVALUATION: The replacement valves meet or exceed the original design specification. There are no unreviewed safety questions associated with this change.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12643
ON - 22643
ON - 32643

DESCRIPTION: This modification replaced valve LPSW-15 with a soft seated butterfly valve.

SAFETY EVALUATION: The only safety function performed by these valves is containment isolation on high reactor building pressure (ES5&6-3 psig). The intent of the modification is to enhance this function by providing a valve with better seating characteristics. No new or different failure modes will exist as a result of its installation.

Valve LPSW-15 is located outside the reactor building. It is electric motor operated and qualified seismically. Piping stress analysis has been reviewed and will not be impacted by the modification.

The modification will not impact any FSAR accident analyses and will not create the potential for any new type of accident. Also, no equipment important to safety will be adversely affected and no margins of safety as defined in Tech Spec bases will be reduced. Therefore, no unreviewed safety questions are involved with the modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12667
ON - 22667
ON - 32667

DESCRIPTION: This modification reroutes cables for valve HP-4 through the west penetration room to the SSF. This modification is the result of an NRC Appendix R audit.

SAFETY EVALUATION: This modification assumes that Appendix R shutdown train separation requirements are met. The power sources for the cables and the valve locations will not be changed. There are no unreviewed safety questions associated with this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12678
ON - 22678
ON - 32678

DESCRIPTION: Design analysis showed that auxiliary steam header safety valves are too small to prevent overpressurization as required by applicable codes -- in the event valve MS-126 and MS-129 fail open or bypass valve MS-131 is full open. This modification provides additional relief capacity by providing more safety valves with larger capacity.

SAFETY EVALUATION: The modification is non-QA Condition and non-safety related. The assured source of steam to the EFW pump turbine is from the Main Steam System, not the Auxiliary Steam System. The expansion joint on the relief valve discharge will absorb the relief valve force. Drain lines will be added to the relief valve discharge to drain rainwater or condensate.

The new relief valves will have a setpoint higher than the system design pressure of 350 psig, but the ASME Code requires only one of the valves have a setpoint equal to (or less than) the design pressure of the system if two or more safety valves are used. The rest of the valves can have a slightly higher set pressure. Existing safety valve AS-23 will be retained as the first line of overpressure protection and can handle minor overpressure protection incidents by itself. The two new valves will probably not have to open. During a design basis event, all three safety valves will open and maintain proper overpressure protection.

Sections 10.3 and 10.4.7 of the FSAR address the Main Steam System (including the Auxilary Steam System) and the EFW System. This modification will increase the reliability of the EFW pump turbine since the addition of the relief valves will reduce the probability of overpressurizing the Auxiliary Steam System and the supply piping to the EFW pump turbine. If the new relief valve inadvertently opens, the QA Condition 1 main steam supply of steam to the EFW pump turbine will not be released out the relief valve since a check valve in the Auxiliary Steam System prevents the main steam flow from reaching the relief valve. Therefore, the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The FSAR accidents that could be associated with this NSM and the mitigative ability of the turbine driven EFW pump will not be degraded. Therefore, the probability or consequence of any accidents previously evaluated in the FSAR will not be increased. The FSAR accidents that could be associated with this NSM are Loss of Electric Power accidents (Section 15.8) and the Main Steam Line Break (Section 15.13). These accidents will not be caused by this NSM and the mitigative ability of the Turbine driven EFW pump will not be degraded. Therefore, the probability or consequences of any accident previously evaluated in the FSAR will not be increased. There are no additional failure modes postulated so the possibility of any new accidents or malfunctions of equipment important to safety not evaluated in the FSAR are not created. Since this modification increases the reliability of turbine driven EFW pumps, no margin of safety is reduced. There are no unreviewed safety questions associated with this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Incomplete	Incomplete	Complete	N/A

ON - 12683
ON - 22683
ON - 32683

DESCRIPTION: This modification provides circuitry to automatically trip the main feedwater pumps on the loss of Integrated Control System (ICS) Hand or Auto power. Procedures required a manual trip of the main feedwater pumps.

SAFETY EVALUATION: The effect of this modification is to replace a manual operation with an automatic operation. This modification eliminates the possibility of an overcooling or undercooling transient due to the loss of ICS Hand or Auto power. Thus, no unreviewed safety questions are judged to be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 12685
ON - 22685
ON - 32685

<u>DESCRIPTION:</u> A reanalysis of the pressurizer spray line indicated that snubber supports on valve RC-1 may fail during a postulated operational earthquake. This modification removes both snubbers attached to RC-1 and adjusts one constant spring.

<u>SAFETY EVALUATION:</u> This modification will result in this portion of the Reactor Coolant System being more capable of withstanding design basis events. No unreviewed safety questions are judged to be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

ON - 32735

DESCRIPTION: This modification secured manual butterfly valve 2LPSW-133 in the open position by inserting a retaining rod thru a wet tap on each side of the valve. These rods will hold the butterfly valve disc in the open position. The wet taps' installation are also part of the modification. This valve, if closed, could have isolated all LPSW flow from the Unit 3 Auxiliary and Reactor Buildings. A pin, which prevented the valve disc from rotating on the shaft, had the potential of rusting out and allowing the disc to rotate freely. Failure of the pin could possibly allow the valve to inadvertently go shut with no ability to reopen.

SAFETY EVALUATION: The LPSW System provides cooling water for normal and emergency services throughout the station. Safety related functions served by this system are:

(1) Reactor Building cooling units
(2) Decay heat removal coolers
(3) High Pressure Injection pump motor bearing coolers
(4) Emergency Feedwater pump motor air coolers
(5) Turbine Driven Emergency Feedwater pump turbine bearing oil coolers.

The modification is QA Condition 1 and safety related. The pipe stresses were reviewed by stress analysis for the effect of the wet tap installation. This valve was originally installed so isolation could be possible during the installation and connnection fo the Unit 3 LPSW piping to the Unit 1 and 2 CCW System discharge piping. Securing this valve will eliminate a potential failure mode. No unreviewed safety questions are judged to be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

ON - 22736
ON - 32736

DESCRIPTION: Cold springing guidelines have been issued since the last maintenance was performed on the Unit 2 and 3 RCPs. These guidelines will make realignment of RCP seal related piping difficult if not impossible following maintenance. This modification makes realignment easier by:

a) Removing interferances such as piping and related dummy pressure breakdown coils which are no longer needed and blind flanging the nozzels.

b) Providing removable (bolted) pipe supports where needed.

Additional graylock type flanges will be installed to speed the maintenance process to keep exposure ALARA. Existing pressure taps in the piping to be removed will be relocated.

SAFETY EVALUATION: Piping and related dummy pressure breakdown coils were installed when new type shaft seals made the original breakdown coils functionally unnecessary. The dummy units were intended to restore the pump seismic characteristics. This has been judged to be unnecessary due to the relatively small mass of the coils. Therefore their removal will have no effect on equipment performance and there will be no increase in the probability of malfunctions of equipment important to safety.

Removable pipe support/restraints, when properly designed and installed, have no inherent weaknesses which make them less reliable then welded supports and the added graylock flanges will be QA-1. Therefore the RC pressure boundary is not degraded and there is no increase in the probability of accidents previously evaluated in the FSAR.

Since affected piping is not relied upon to mitigate accidents and no RCS/Core parameters are affected, there is no increase in consequences of accidents previously evaluated in the FSAR.

These NSMs will not cause any new failure modes since the design has been properly addressed. Therefore, the possibility of any new malfunctions of equipment important to safety will not be created. Likewise, the possibility of new accidents different from any already evaluated in the FSAR is not created.

Since no set points or other plant parameters are altered by these NSMs, there is no reduction in the margin of safety as defined in the bases to any Tech. Specs.

There is no unreviewed safety questions associated with this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	Complete	N/A

ON - 12754
ON - 22754
ON - 32754

DESCRIPTION: This modification repairs the 2B Reactor Building Cooling Unit (RBCU) isolation dampers. The old damper linkage is replaced with a flat steel bar with pressure fit brass bushings.

SAFETY EVALUATION: The damper serves no safety-related function. It is designed to direct air flow from the center RBCU fan (2B) to either the A or B steam generator cavity when the 2B fan is running.

The modification will have no impact on the ductwork and will not degrade the operation of the dampers. Therefore, the probability or consequences of a malfunction of equipment important to safety will not be increased and no different malfunction will be created. Also, since the ductwork will operate as designed in a LOCA and no increases in hydrogen generation from additional amounts of aluminum or zinc will be created, the consequences of a LOCA will not be increased. This modification does not initiate any accidents analyzed in the FSAR so the probability of accidents analyzed in the FSAR will not be increased. No new accidents are postulated. No design/safety limits are adversely affected so no margins of safety as defined in the bases to any technical specifications are reduced.

There are no unreviewed safety questions associated with this NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

ON - 52601

<u>DESCRIPTION:</u> This change relocates meterological equipment to a new tower. The previous tower location became unacceptable with construction of the new administration building due to the affects of building generated turbulence.

<u>SAFETY EVALUATION:</u> This change does not place the plant in a cenfiguration different than that described in the FSAR, nor does it change the bases of any Technical Specification. As such, no unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52646

DESCRIPTION: This modification involves the addition of interlocks and time delays to prevent transfer valves VR-125, 126, 130, 395, 396 and 446 from being closed when hopper valves VR-141 and 150 are open.

SAFETY EVALUATION: The subject modification is to a QA Condition 2 sub-system of the Oconee Radwaste Facility (RWF). The facility and all of its associated systems are non-nuclear safety and non-seismic (except for a portion of the building's foundation which is designed to prevent the runoff of its liquid inventory following a seismic event). The facility does contain radioactive materials which are available for release to the environment. However, analyses using conservative, worst-case assumptions have determined that the maximum hypothetical release would result in doses below 10CFR20 limits. The subject modification is an operational enhancement which does not increase the probability nor the consequences of such a release, does not create the possibility for any new type of accident, and does not adversely affect the design bases of the plant or the RWF.

The RWF is a remote facility which does not contain any equipment important to safety. Also, the potential for interactions, either direct or secondary, with equipment which is important to safety does not exist. The subject NSM, therefore, does not increase the probability nor the consequences of malfunctions of such equipment and does not create the possibility for any new type of malfunction of such equipment.

The Technical Specifications applicable to radioactive effluents and waste handling are not affected by the NSM, and no margins of safety as defined in any Tech Spec bases are related to the operation of the RWF. Therefore, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52648

DESCRIPTION: This modification involves the addition of an instrument loop off an existing loop (OVRFS1820) to provide indication of Dry Product Air Blower flow. A flow transmitter will be installed near the existing gage and a receiver gage will be installed on the S.E. Co. control panel. Tubing, valves, fittings and cabling will be installed as required to complete the loop.

SAFETY EVALUATION: The subject modification is to a QA Condition 2 sub-system of the Oconee Radwaste facility (RWF). The facility and all of its associated systems are non-nuclear safety and non-seismic (except for a portion of the building's foundation which is designed to prevent the runoff of its liquid inventory following a seismic event). The facility does contain radioactive materials which are available for release to the environment. However, analyses using conservative, worst-case assumptions have determined that the maximum hypothetical release would result in doses below 10CFR20 limits. The subject modification is an operational enhancement which does not increase the probability nor the consequences of such release, does not create the possibility for any new type of accident, and does not adversely affect the design bases of the plant or the RWF.

The RWF is a remote facility which does not contain any equipment important to safety. Also, the potential for interactions, either direct or secondary, with equipment which is important to safety does not exist. The subject NSM, therefore, does not increase the probability nor the consequences of malfunctions of such equipment and does not create the possibility for any new type of malfunction of such equipment.

The Technical Specifications applicable to radioactive effluents and waste handling are not affected by the NSM, and no margins of safety as defined in any Tech Spec bases are related to the operation of the RWF. Therefore, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52650

DESCRIPTION: This modification provides redundant level indication on the scrubber preconcentrator tank and modifies scrubber loop return feed, density, and level control systems to allow on-line return of feed.

SAFETY EVALUATION: The subject modification is to a QA Condition 2 sub-system of the Oconee Radwaste facility (RWF). The facility and all of its associated systems are non-nuclear safety and non-seismic (except for a portion of the building's foundation which is designed to prevent the runoff of its liquid inventory following a seismic event). The facility does contain radioactive materials which are available for release to the environment. However, analyses using conservative, worst-case assumptions have determined that the maximum hypothetical release would result in doses below 10CFR20 limits. The subject modification is an operational enhancement which does not create the possibility for any new type of accident, and does not adversely affect the design bases of the plant or the RWF.

The RWF is a remote facility which does not contain any equipment important to safety. Also, the potential for interactions, either direct or secondary, with equipment which is important to safety does not exist. The subject NSM, therefore, does not increase the probability nor the consequences of malfunctions of such equipment and does not create the possibility for any new type of malfunction of such equipment.

The Technical Specifications applicable to radioactive effluents and waste handling are not affected by the NSM, and no margins of safety as defined in any Tech Spec bases are related to the operation of the RWF. Therefore, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52655

<u>DESCRIPTION:</u> This Radwaste Facility (RWF) modification provides exhaust gas condenser level instrumentation. The modification includes remote indication, a statalarm, current alarm, power source, and a level transmitter.

<u>SAFETY EVALUATION:</u> The RWF, along with all of its associated systems, is non Nuclear Safety Related. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so the probabilities of FSAR accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 limits and the design basis of the RWF is not adversely affected, the creation of a new accident scenario not previously evaluated is not possible.

The remote RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not affect the probability of any malfunction of such equipment.

The proposed modification does not degrade any plant safety system. Thus, the consequences of any accident or equipment malfunction are not increased.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

<u>STATUS:</u>	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52671

DESCRIPTION: This Radwaste Facility (RWF) modification installed a perforated deflector plate in the Resin Batch tank. This modification prevents resin carryover out of the tank discharge.

SAFETY EVALUATION: The RWF, along with all of its associated systems, is non Nuclear Safety Related. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so the probabilities of FSAR accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 limits and the design basis of the RWF is not adversely affected, the creation of a new accident scenario not previously evaluated is not possible.

The remote RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not affect the probability of any malfunction of such equipment.

The proposed modification does not degrade any plant safety sytsem. Thus, the consequences of any accident or equipment malfunction are not increased.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety/design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52702

DESCRIPTION: Six variable speed pumps are affected by this modification.

1) Contaminated Resin Pump A (RR)
2) Contaminated Resin Pump B (RR)
3) Waste Feed Pump A (LW)
4) Waste Feed Pump B (LW)
5) Resin recirculation Pump (VR)
6) Backflush Storage Tank Transfer Pump (VR)

For each affected pump an auxiliary relay will be added to the electronic trip relay output of the variable speed drive unit to cause the indicating lights in the RWF control room to change to green when the electronic trip operates. An annunciator will alarm on electric trip. Also, an auxiliary relay will be added to monitor power failure to the variable speed drive. This relay will initiate an annunciator alarm in the RWF control room on power failure to the variable speed drive.

The on-off speed control of the pumps will continue to operate as before. However, in the event of an electronic trip in the variable speed drive, the indicating lights in the RWF control room will now show the pump is "off", and an annunciator alarm will indicate the electronic trip has occurred. Also, if a power failure to the drive occurs, this will now be indicated by an annunciator alarm. No changes to the normal functional operation of the pump have been made.

SAFETY EVALUATION: Personnel dose exposure has been considered. All cabling is located within the Radwaste Facility, appendix R review has been considered. No part of this modification involves safety related equipment, systems or structures. No civil equipment or components are affected. No mechanical systems or components are affected. The RWF, along with all of its associated systems, is not Nuclear Safety Related. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so the probabilities of FSAR accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 Limits and the design basis of the RWF is not adversely affected, the creation of a new accident scenario not previously evaluated is not possible.

The remote RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not affect the probability of any malfunction of such equipment.

ON - 52702
(cont'd)

The proposed modification does not degrade any plant safety system. Thus, the consequences of any accident or equipment malfunction are not increased.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety/design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52712

DESCRIPTION: The subject NSM adds statalarm windows in the Radwaste Control Room for the process and area radiation monitors in the Radwaste Facility. A total of nine annunciator windows will be installed: seven for the area monitors (RIA-61 thru -67), one for the gas monitors (RIA-45 & 46), and one for the liquid effluent monitor (RIA-33A). Each annunciator will indicate when its corresponding radiation monitor has either a high radiation or a monitor fail alarm. The alarm signals will be taken from vendor-supplied contacts and wired to the RCR annunciator panel.

SAFETY EVALUATION: The proposed modification involves the Radiation Monitoring System for the Radwaste Facility (RWF). The RWF, along with all of its associated systems, is non Nuclear Safety Related. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so the probabilities of FSAR accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 limits and the design basis of the RWF is not adversely affected, the creation of a new accident scenario not previously evaluated is not possible.

The remote RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not affect the probability of any malfunction of such equipment.

The proposed modification does not degrade any plant safety system. Thus, the consequences of any accident or equipment malfunction are not increased.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety/design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52717

DESCRIPTION: Radiation monitor RIA-33 is used to sample a portion of the flow being released to the Keowee tailrace from a monitor tank. In order to obtain the desired release and sampling flow rates in the main 3" and sampling lines respectively, a new orifice replaced the existing 1.00" orifice in the 3" line and a flow control valve was installed in the Duke portion of the 1/2" sampling tubing.

SAFETY EVALUATION: The NSM involves the Liquid Waste and Recycle System (LW) located in the Oconee Radwaste Facility (RWF). The RWF, along with all of its associated systems, is non Nuclear Safety Related. The LW system has been designated QA Condition 2. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF. The purpose of the orifice and valve is to assure that, for all possible release rates, the sample flow rate to the radiation monitor is adequate for the instrument to accurately monitor the radioactive liquid effluent. This effluent monitoring is a Tech Spec requirement. Based on system flow calculations, an operational procedure will provide guidance for the adjustment of the position of the throttling valve as a function of the release flow rate in order to maintain proper sample flow.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so both the probability and the consequences of these accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 limits, the creation of a new accident not previously evaluated is not possible.

The RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not impact either the probability or the consequences of any malfunction of such equipment.

The Technical Specifications applicable to radioactive waste treatment and disposal are not adversely affected by the modification. Also, no plant parameter that has any effect on the Safety/Design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52746

DESCRIPTION: This modification installs additional pressure taps and thermocouples on the fluid bed incinerator and fluid bed dryer in the Radwaste Facility.

SAFETY EVALUATION: The proposed modification is to the non-Nuclear Safety Related Radwaste Solidification system (VR). With the exception of portions of the building itself none of the structures, systems and components of the Radwaste Facility (RWF) are required to be seismically qualified. The subject NSM is an operational enhancement and will not affect the functional operation of either the Dryer or Incinerator. Therefore this NSM does not introduce any new failure modes or adverse operational characteristics. The new pressure tap, thermocouple and well designs will be the same as those currently being used so as not to degrade either the Fluid Bed Incinerator or Dryer.

There are no accidents evaluated in the FSAR involving the RWF or associated systems, so the probability and consequences of these accidents are unaffected by any RWF modification. Also, since the worst-case RWF release is bounded by 10CFR20 limits, the creation of a new accident not previously evaluated is not possible.

The RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety related equipment. Therefore, the NSM does not impact either the probability or the consequences of any malfunction of such equipment.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety/design analyses is impacted. Thus, no margin of safety is reduced. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52750

DESCRIPTION: Controls for resin and condensate feed to the fluid bed incinerator contained many timers and interlocks which made the control system very complicated and prohibited proper system operation. This modification simplifies the control logic to allow the operator greater flexibility while maintaining proper safeguards for equipment and personnel.

SAFETY EVALUATION: This modification is an enhancement of the controls for the Volume Reduction (VR) system.

The Radwaste Facility (RWF), along with all of its associated systems, is non-Nuclear Safety Related. Also, with the exception of portions of the building itself, none of the structures, systems and components of the RWF are required to be seismically qualified. The subject NSM is merely an operational enhancement which does not adversely affect the design bases of the plant or the RWF.

There are no accidents evaluated in the FSAR involving the RWF or any of its associated systems, so the probabilities of FSAR accidents are unaffected by any radwaste modification. In addition, since the worst-case RWF release is bounded by 10CFR20 limits and the design basis of the RWF release is adversely affected, the creation of a new accident scenario not previously evaluated is not possible.

The remote RWF does not contain any equipment that is important to safety and there is no means of interaction, either direct or secondary, with safety-related equipment. Therefore, the NSM does not affect the probability of any malfunction of such equipment.

The proposed modification does not degrade any plant safety system. Thus, the consequences of any accident or equipment malfunction are not increased.

None of the Technical Specifications applicable to radioactive waste management are adversely affected by the modification. Also, no plant parameter that has any effect on the plant's safety/design analyses is impacted. Thus, no margin of safety is reduced.

Therefore, based on the above analysis, there are no unreviewed safety questions involved with the NSM.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

ON - 52753

DESCRIPTION: The purpose of the subject NSM, which involves the Keowee Hydro Station, is to revise the 3" connection between a 12" service water header and the oil room deluge strainer, in order to increase the available pressure at the strainer. The existing nozzle on the service water line was capped and an elbow was in place of the tee at the strainer; a long run of 3" pipe connected the strainer/elbow to a point on the 12" service water line downstream from the chlorine mixer and contributed significant head losses in the oil room fire protection supply. The existing run of 3" pipe will not be removed or modified, except at the strainer connection as described above. The modification is QA Condition 3.

SAFETY EVALUATION: The modification described above is an enhancement to the fire protection capabilities at Keowee, providing increased system pressure when it is needed without degrading the Asiatic clam protection provided by chlorination. Seismic design is not required due to the shortness of the added line and the lack of any significant interaction concerns in the affected area. The new line can be isolated when necessary for testing, maintenance, etc.

No accidents which are analyzed in the FSAR are initiated by failures, malfunctions, or misoperations of Keowee service water systems, nor does that potential exist. Therefore, the probability of previously evaluated accidents is not increased by the proposed modification and the possibility of a different type of accident is not created. The performance of the fire protection system is improved by the increase in supply pressure and no functions of the system including clam protection are degraded by the modification. Therefore, the probability of a previously evaluated malfunction of equipment important to safety is not increased and the possibility of a different malfunction is not created.

The improvement in fire protection system performance helps to assure that Keowee will be available to provide emergency power to Oconee when needed. Therefore, the consequences of accidents or equipment malfunctions are not increased by the modification, and the margin of safety is not reduced. As such, there are no unreviewed safety questions involved with this modification.

STATUS:	Keowee Unit 1	Keowee Unit 2
	Complete	Complete

OE - 962

DESCRIPTION: This modification adds a reinforcing fillet weld to the OTSG 1A MK-25 drain nozzle. This weld was required to demonstrate the adequacy of the structural integrity of the nozzle.

SAFETY EVALUATION: This modification was performed in response to IE Bulletin 79-14 and in no way affects the function of the OTSG. No unreviewed safety questions are posed by this exempt change.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

OE - 1227
1228
1527
1528

DESCRIPTION: This modification permanently installs hold-down rings in the Once Through Steam Generator (OTSG) outlet nozzles. In addition, nozzle dams are installed in both cold legs of each OTSG for the purpose of isolating the primary side of the OTSGs from the rest of the reactor coolant system during a refueling shutdown.

SAFETY EVALUATION: The nozzle dams are installed only during a refueling shutdown. Catastrophic failure of the dams has been shown by analysis to be highly improbable. The dams will not affect the ability to mitigate a design basis accident. No unreviewed safety questions are judged to be created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	N/A	N/A

OE - 1236

DESCRIPTION: This exempt change enlarges the Unit 1 OTSG "B" support skirt manway opening and provides bolt holes for support blocks for temporary shielding during maintenance and repair.

SAFETY EVALUATION: The enlargement of the skirt opening will not reduce the margin of safety for the support structure. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

OE - 1279
OE - 1711

DESCRIPTION: This exempt change installs new thermowells for the decay heat coolers such that reliable test temperature readings can be obtained to determine actual cooler efficiency.

SAFETY EVALUATION: Installation of thermowells provides the ability to determine actual cooler efficiency. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

OE - 1402

DESCRIPTION: This exempt change enlarges the Unit 3 OTSG "A" support skirt manway opening and provides bolt holes for support blocks for temporary shielding during maintenance and repair.

SAFETY EVALUATION: The enlargement of the skirt opening will not reduce the margin of safety for the support structure. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

OE - 1403

DESCRIPTION: This exempt change enlarges the Unit 3 OTSG "B" support skirt manway opening and provides bolt holes for support blocks for temporary shielding during maintenance and repair.

SAFETY EVALUATION: The enlargement of the skirt opening will not reduce the margin of safety for the support structure. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

OE - 1437

DESCRIPTION: This exempt change removes reactor coolant system instrument root valves IRC-21, -28, -32, -36, and -40, as well as the associated instruments and instrument tubing. This instrumentation was originally installed for first of a kind testing. Pipe caps are welded where the root valves were.

SAFETY EVALUATION: This exempt change removes unused equipment in order to remove potential leak paths from the reactor coolant system. No unreviewed safety questions are judged to be created as a result of this exempt change.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	N/A	N/A	N/A

OE - 1480
OE - 1481
OE - 1482
OE - 1483

DESCRIPTION: This exempt change modifies the shaft leak-off cover and pump coupling on Unit 2 reactor coolant pumps for easier seal installation.

SAFETY EVALUATION: This modification will increase the reliability of the reactor coolant pump seals. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1507

DESCRIPTION: This exempt change enlarges the Unit 2 OTSG "A" support skirt manway opening and provides bolt holes for support blocks for temporary shielding during maintenance and repair.

SAFETY EVALUATION: The enlargement of the skirt opening will not reduce the margin of safety for the support structure. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1508

DESCRIPTION: This exempt change enlarges the Unit 2 OTSG "B" support skirt manway opening and provides bolt holes for support blocks for temporary shielding during maintenance and repair.

SAFETY EVALUATION: The enlargement of the skirt opening will not reduce the margin of safety for the support structure. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1641

DESCRIPTION: This exempt change applies to the 2A1 reactor coolant pump and upgrades the suction piece to casing volute capscrew material, and the wear ring to suction capscrew material.

SAFETY EVALUATION: This exempt change will improve reactor coolant pump reliability. No unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1642

<u>DESCRIPTION:</u> This exempt change applies to the 2A2 reactor coolant pump and upgrades the pump bearing, suction piece to casing volute capscrew material, wear ring capscrew material, and the stuffing box bearing.

<u>SAFETY EVALUATION:</u> This exempt change will improve reactor coolant pump reliability. No unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1643

DESCRIPTION: This exempt change applies to the 2B1 reactor coolant pump and upgrades the pump bearing, suction piece to casing volute capscrew material, wear ring capscrew material, and the stuffing box bearing.

SAFETY EVALUATION: This exempt change will improve reactor coolant pump reliability. No unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1644

DESCRIPTION: This exempt change applies to the 2B2 reactor coolant pump and upgrades the suction piece to casing volute capscrew material and the wear ring to suction capscrew material.

SAFETY EVALUATION: This exempt change will improve reactor coolant pump reliability. No unreviewed safety questions are created.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 1648
OE - 1649
OE - 1650
OE - 1651

DESCRIPTION: These exempt changes apply to the Unit 3 reactor coolant pumps and modify the shaft, leakoff cover, and pump coupling to allow easier seal installation.

SAFETY EVALUATION: These exempt changes will improve reactor coolant pump reliability. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

OE - 1656

DESCRIPTION: This modification installs the necessary piping, valves, etc. from the condensate polishing demineralizer backwash sump pump discharge to the security boundary. In addition, necessary piping, valves, etc. to the chemical treatment ponds is installed.

SAFETY EVALUATION: This modification provides Radwaste Facility interface with existing plant piping. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

OE - 1720

DESCRIPTION: The External Grid Protection System tripped 230kV Breaker 23, as well as other 230kV breakers. This tripping of Breaker 23 would cause the loss of Unit 2. The opening of Breaker 23 from the External Grid Protection System is unnecessary due to the tripping of Breaker 24 which will clear Unit 2 Bus from the Yellow Bus. This exempt change removed the Grid Isolation Trip Circuits from 230kV Switchyard Breaker 23 and allowed Unit 2 not to be tripped due to the operation of the External Grid Protection System. Breaker 23 will be tripped by its own relaying in the event of overload or breaker failure.

SAFETY EVALUATION: The Yellow Bus will be cleared by 230kV Breaker 24, thus enabling Keowee to supply power through the Yellow Bus. Unit 2 will remain operable following a grid separation and provide an alternate source of station power. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 2072

DESCRIPTION: This modification rewired the SSF controllable group 3B pressurizer heater terminal boxes to use 9 elements that have hardwired connections. Heater elements controlled by the SSF had ceramic connectors. There is no problem with the ceramic connectors, but the NRC prefers that the SSF controlled elements be hardwired. All cabling changes took place between terminal boxes in the Reactor Building.

SAFETY EVALUATION: This modification will affect pressurizer heaters in banks 1, 2 and 3. These heaters are not safety related and perform no accident mitigation function for any FSAR chapter 15.0 design basis accident. The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during Reactor Coolant (RC) System heatup, and restore system pressure following a transient. The heaters are grouped in four banks. For events requiring operation of the SSF, power and control of 9 heaters is transferred to the SSF to insure a steam bubble is maintained in the pressurizer.

FSAR section 5.4.6.2 states that a minimum of 126kW of pressurizer heaters, corresponding to 9 elements should be available from an assured power source within two hours after loss of offsite power in order to establish and maintain natural circulation at hot standby conditions. Although they should be available, credit is not taken for the pressurizer heaters in the mitigation of loss of electric power accidents (FSAR section 15.8). The pressurizer heaters are supplied from non-safety-related motor control centers which are powered from the 4160V engineered safeguard busses.

This modification replaced the current SSF controllable pressurizer heater elements with similar heater elements. This will not change the way any of the heaters operate during normal operation. All cabling changes for this modification took place between terminal boxes located near each other in the Reactor Building. The same cables from the terminal boxes to the SSF, which fed the SSF controllable elements, feed the replacement elements and remain in the same fire area. Therefore, there are no 10 CFR 50 Appendix R concerns. Since heater elements were replaced by similar elements, the circuit breaker protection is properly sized and no new failure modes will be introduced.

The pressurizer heaters are not the initiators of and are not taken credit for in any of the FSAR chapter 15.0 accidents. The minimum heater capacity will be met for RC natural circulation cooldown and for SSF scenarios, the heaters are similar, and normal power operation will be unaffected. Therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. This modification will not create any new failure modes. Therefore, a possibility for an accident or equipment malfunction of a different type than any evaluated previously in the FSAR will not be created. No key safety parameters are affected by this modification. Therefore, the margin of safety as defined in the basis for any technical specification will not be reduced. No unreviewed safety questions are created by this modification.

OE - 2091

DESCRIPTION: This exempt change revised civil design specifications regarding edge distance criteria for concrete expansion anchors. This change was in response to NRC Information Notice 88-25.

SAFETY EVALUATION: The revised edge distances will ensure that concrete expansion anchor installations will perform as design. This change will create no unreviewed safety questions.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	Complete

OE - 1821
OE - 1822

DESCRIPTION: Nozzle Dams are installed in both cold legs of each OTSG for the purpose of isolating the primary side of the OTSGs from the rest of the reactor coolant system (RCS) during a refueling shutdown. With the nozzle dams in place, the RCS and the refueling canal can be flooded to allow fuel movement while the primary side of the OTSGs remain dry, allowing access through the manway for such activities as tube sleeving. The ability to perform OTSG maintenance, repair and testing concurrent with refueling operations can significantly reduce the length of the outage.

A major difference between the McGuire/Catawba steam generators and the OTSGs at Oconee, however, is that the OTSGs do not have nozzle dam hold-down rings already installed in the outlet nozzles. Therefore, hold-down rings must be installed before nozzle dams can be used. The rings are supplied by the dam manufacturer and are made from Inconel 600 per ASTM B564. They will be TIG (Tungsten Inert Gas) welded in place by a remote automatic process and thus will be permanent fixtures within each OTSG.

SAFETY EVALUATION: The safety concern with the use of OTSG nozzle dams during refueling is that their failure could result in the loss of water from the refueling canal through the open OTSG manway. If the failure was catastrophic and was to occur during fuel movement, it is possible that spent fuel assemblies would be uncovered, creating a severe radiation hazard at the operating floor as well as a personnel hazard due to flooding in the lower containment.

Another concern involves the permanent installation of hold-down rings in the OTSG outlet nozzles. The rings have been designated non-QA but should be installed in such a way as not to degrade the RCS pressure boundary and to assure that they are unlikely to become loose parts in the RCS.

The criteria of 10CFR50.59 are applied as follows:

1) May the probability of an accident previously evaluated in the FSAR be increased?

   No. The nozzle dams are installed only during a refueling shutdown and serve as a boundary between the refueling canal and the primary side of the OTSGs. It has been shown by analysis that a catastrophic failure of the dams is highly improbable so the most severe failure mode would be leakage, which would result in drainage of the refueling canal and flooding of the lower containment only if no actions were taken to stop the leakage. The worst leakage would be so small that it would take days for the canal to drain so there is sufficient time for corrective actions to be taken. No design basis accidents are impacted by the use of nozzle dams, even in a failed condition. They cannot cause a fuel handling accident nor lead to the release of radiation.

   The nozzle hold-down ring remains in place during all modes of plant operation. The welding process will be controlled such that the depth of penetration does not reach the base metal of the OTSG shell. The weld penetration depth will be no more than .062", which is less than half the

thickness of the shell cladding, so the RCS pressure boundary is unaffected by the installation of the ring. A catastrophic failure of the weld would result in a large loose part within the RCS which could cause failure of an RC pump. However, failure of the weld has been shown by analysis to be a highly improbable occurrence due to the strength of the weld and the number of stress cycles it would experience. Therefore, the probability of a loss of flow accident is not increased.

No other design basis accidents might be affected by the use of nozzle dams or the installation of nozzle hold-down rings. Therefore, the probability of previously analyzed accidents is not increased.

2) May the consequences of an accident previously evaluated in the FSAR be increased?

No. As described above, the nozzle dams are used only during a refueling shutdown to prevent drainage of the refueling canal. The dams will not affect the ability of any mitigating system to perform as intended following an accident. If fuel is being moved, containment integrity will be maintained and will not be degraded by the failure of a nozzle dam.

During other modes of operation, as described in 1) above, the nozzle hold-down rings will remain intact and thus will not affect the design function of any mitigating system. The rings will not significantly affect the flow characteristics in the OTSG outlet nozzles, so the RCS will behave as analyzed under both normal and transient conditions.

Therefore, the consequences of previously evaluated accidents are not increased by the use of nozzle dams or the installation of nozzle hold-down rings.

3) May the possibility of an accident which is different than any already evaluated in the FSAR be created?

No. The failure of a nozzle dam during refueling or the failure of a nozzle hold-down ring during any mode of operation will not result in any accidents which are outside the design basis of the plant or which have not already been analyzed. Other mechanisms for drainage of the refueling canal and spent fuel pool have been previously identified and the consequences analyzed. Also, the consequences of loose parts of various sizes have been analyzed and means for detecting them have been provided.

4) May the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

No. During the refueling shutdown mode, the nozzle dams are secured inside the OTSG outlet nozzles, with control panels located just outside. Interactions with equipment important to safety, either directly or indirectly, are not likely. During other modes of operation the nozzle

OE - 1821
OE - 1822
(cont'd)

hold-down rings remain intact in the OTSG outlet nozzles, even in the unlikely event of weld failure, the loose parts would remain in the RCS and could not interact with any equipment important to safety. Therefore, the probability of a malfunction of such equipment is not increased.

5) May the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

No. As described in 2) above, the use of nozzle dams and the installation of nozzle hold-down rings do not affect the function of any system so as to limit its ability to perform any mitigative actions required in the event of an accident or a malfunction of any safety component.

6) May the possibility of malfunctions of equipment important to safety which are different than any already evaluated in the FSAR be created?

No. As described in 4) above, the nozzle dams and the nozzle hold-down rings cannot interact with any equipment important to safety because they are within the OTSG when in use, or in case of ring failure, confined within the RCS.

7) Will the margin of safety as defined in the bases to any technical specifications be reduced?

No. The nozzle dams will be in use only during a refueling shutdown and will not adversely affect any system required to maintain proper shutdown conditions. During other modes of operation the nozzle hold-down rings do not adversely affect the performance of the RCS. Therefore, the margin of safety is not reduced.

According to the criteria of 10CFR50.59a (2), there are no unreviewed safety questions involved with the use of OTSG nozzle dams or with the permanent installation of OTSG outlet nozzle hold-down rings. There are also no applicable Tech Specs which might be affected.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	Complete	N/A

OE - 2040
OE - 2047

DESCRIPTION: The Integrated Control System (ICS) Unit Load Demand (ULD) runback rate for loss of one main feedwater pump (MFDWP) or one reactor coolant pump (RCP) was set at 50% per minute. This modification changed this runback rate to 25% per minute. All necessary changes will be made within the ICS cabinets. No external wiring will be required.

SAFETY EVALUATION: The ICS ULD runback rate for loss of one MFDWP or RCP is the only function affected by this change. The ICS is a non-safety related system. The applicable FSAR accident is discussed in Section 15.6 (Loss of Coolant Flow Accident) and describes the loss of one or more RCP's. Section 15.6.3 states "The reactor is protected against reactor coolant pump failure(s) by the RPS and the ICS. The ICS initiates a power reduction on pump failure to prevent reactor power from exceeding that permissible for the available flow. The reactor is tripped if insufficient reactor coolant flow exists for the power level." This change will not affect this accident analysis since the Reactor Protection System (RPS) safety trips will not be affected and the ULD runback rate is not taken credit for in the analysis. This change will help the unit withstand a loss of one MFDWP without a reactor trip and in so doing will decrease challenges to plant systems. Equipment on the unit secondary side will not be adversely affected by a slower unit runback rate.

This change affects the ULD runback rate for loss of one MFDWP or RCP and does not affect RCP or MFDWP operation. Therefore, this will not increase the probability of a loss of main feedwater accident or a loss of coolant flow accident. This change will not affect RPS safety trip and therefore will not increase the consequences of a loss of main feedwater accident or loss of coolant flow accident. Changing the ULD runback rate will not create any new equipment failure modes and, therefore, this change will not create the possibility of a different accident or equipment malfunction than previously evaluated. This will not affect the RPS safety trips or any key safety parameters and will reduce challenges to plant systems on loss of one MFDWP. Therefore, this change will not reduce the margin of safety as defined in the basis to any Technical Specifications. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	Complete	N/A

OE - 2060

DESCRIPTION: Main Steam Pipe support S/R 2-01A-0-1401B-R13 is overstressed when the Unit 2 main steam line is in the "cold" position, this exempt change provides for a revised S/R to replace existing 2-01A-0-1401B-R13 and which will adequately support the main steam line for all design conditions.

SAFETY EVALUATION: S/R 2-01A-0-1401B-R13 is located in the turbine building at El. 808+2 (approx). It supports main steam piping between the S/Gs and the Turbine Stop Valves and is QA condition 1. No valving, pipe configuration or physical location is changed and no pipe stresses are increased as a result of this VN. Therefore, no accident or malfunction of equipment important to safety previously evaluated in the FSAR is increased in probability or in consequences. No other system important to safety is affected nor its operability threatened as a result of this VN. Therefore, no possibility of accidents or malfunctions of equipment important to safety different from any already evaluated in the FSAR is created. No assumptions in the station FSAR are invalidated by this VN and no plant parameters or set points are altered. Therefore, there is no reduction in the margin of safety as defined in the bases to any Tech Spec. There are no unreviewed safety questions created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	Complete	N/A	N/A

OE - 2072

DESCRIPTION: This modification rewired the SSf controllable group 3B pressurizer heater terminal boxes to use 9 elements that have hardwired connections. Heater elements controlled by the SSF had ceramic connectors. There is no problem with the ceramic connectors, but the NRC prefers that the SSF controlled elements be hardwired. All cabling changes took place between terminal boxes in the Reactor Building.

SAFETY EVALUATION: This modification will affect pressurizer heaters in banks 1, 2 and 3. These heaters are not safety related and perform no accident mitigation function for any FSAR chapter 15.0 design basis accident. The pressurizer heaters replaces heat lost during normal steady state operation, raise the pressure to normal operating pressure during Reactor Coolant (RC) System heatup, and restore system pressure following a transient. The heaters are grouped in four banks. For events requiring operation of the SSF, power and control of 9 heaters is transferred to the SSF to insure a steam bubble is maintained in the pressurizer.

FSAR section 5.4.6.2 states that a minimum of 126kW of pressurizer heaters, corresponding to 9 elements should be available from an assured power source with in two hours after loss of offsite power in order to establish and maintain natural circulation at hot standby conditions. Although they should be available, credit is not taken for the pressurizer heaters in the mitigation of loss of electric power accidents (FSAR section 15.8). the pressurizer heaters are supplied from non-safety-related motor control centers which are powered from the 4160V engineered safeguard busses.

This modification replaced the current SSF controllable pressurizer heater elements with similar heater elements. This will not change the way any of the heaters operate during normal operation. All cabling changes for this modification took place between terminal boxes located near each other in the Reactor Building. The same cables from the terminal boxes to the SSF, which fed the SSF controllable elements, feed the replacement elements and remain in the same fire area. Therefore, there are no CFR 50 Appendix R concerns. Since heater elements were replaced by similar elements, the circuit breaker protection is properly sized and no new failure modes will be introduced.

The pressurizer heaters are not the initiators of and are not taken credit for in any of the FSAR chapter 15.0 accidents. The minimum heater capacity will be met for RC natural circulation cooldown and for SSF scenarios, the heaters are similar, and normal power operation will be unaffected. Therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. This modification will not create any new failure modes. Therefore, a possibility for an accident or equipment malfunction of a different type than any evaluated previously in the FSAR will not be created. No key safety parameters are affected by this modification. Therefore, the margin of safety as defined in the basis for any technical specification will not be reduced. No unreviewed safety questions are created by this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	N/A	N/A	N/A	Complete

OE - 2091

DESCRIPTION: This exempt change revised Civil design specifications regarding edge distance criteria for concrete expansion anchors. This change was in response to NRC Information Notice 88-25.

SAFETY EVALUATION: The revised edge distances will ensure that concrete expansion anchor installations will perform as design. This change will create no unreviewed safety questions.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	Complete

OE - 2359
OE - 2360
OE - 2361

DESCRIPTION: The magnetic setting for the circuit breakers of Reactor Building cooling unit fans (RBCUF) A and C was set at 1400 Amp. This exempt change raised the magnetic settings for these breakers to maximum.

SAFETY EVALUATION: Each of the above RBCUF's is fed directly from a load center breaker. The magnetic setting of a breaker determines the speed at which it responds to a high surge of current, such as a fault. This setting should be set to provide protection to the equipment, but should not affect upstream breaker coordination.

Breaker coordination between the circuit breakers of RBCUF's A and C and the load center breakers is not required as long as upstream coordination is maintained. Upstream breaker coordination is not affected by the change. This is true because the load center breaker only feeds one RBCUF.

Overload protection is provided by the thermal element of the breaker. Therefore, overload protection is maintained.

This change will not affect operation or protection of RBCUF motors or the operation of any other electric loads. Therefore, this modification will not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

There will still be breaker coordination and proper protection for all affected loads, therefore no new failure modes will be created. This change will not create the possibility of a different accident or equipment malfunction than previously evaluated.

This modification will not affect any key safety parameters or setpoints. Therefore, this modification will not reduce the margin of safety as defined in the bases of any Technical Specification. No unreviewed safety questions are created as a result of this modification.

STATUS:	Unit 1	Unit 2	Unit 3	Station
	Complete	Complete	Complete	N/A

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

Summary of Procedure Changes
Completed under 10 CFR 50.59

OCONEE NUCLEAR STATION

Summary of Issued Procedures and Procedure Changes Completed Under 10CFR50.59 and Reviewed by the Nuclear Safety Review Board for 1988

CP/0/A/2002/10

"Addition of Hydrogen Peroxide to the Reactor Coolant System" was issued to describe the addition of unstabilized Hydrogen Peroxide to the RCS to cause the release of activated corrosion products at a time when removal of these products by demineralization can be optimized. The FSAR only provides for injection of boron, lithium hydroxide and hydrazine into the reactor coolant system. No additional impact on corrosion of the primary system has been identified by the use of hydrogen peroxide; therefore, an unreviewed safety question is judged not to exist.

TT/0/A/128/5

"Extraction and Examination of Failed Fuel Rods Controlling Procedure for Test Operations" was issued to control testing for specialized operations necessary to extract and examine selected fuel rods from three Oconee-2 Batch 7 fuel assemblies with suspect leaking rods. Only one fuel pin is affected at a time. FSAR analyses assume 56 pins will fail. The probability and consequences of the FSAR analyses are not increased; therefore, an unreviewed safety question is judged not to exist.

TT/3/A/0124/06

"Spent Fuel Assembly Inspection by Ultrasonic Testing" was issued to ultrasonical test fuel assemblies in the Unit 3 spent fuel pool for defective cladding. The test will not be performed over spent fuel assemblies and the loads imposed on the fuel storage racks are acceptable. An unreviewed safety question is judged not to exist.

OP/2/A/1104/04

This procedure, "Low Pressure Injection System," describes the proper operation of the Low Pressure Injection System. Change 37 to this procedure allows the continued use of purification or deborating demineralizers to enhance RCS cleanup prior to Unit refueling. The results of the safety analyses indicate that if either a purification demineralizer or a deborating demineralizer with fresh resins is inadvertently valved in, given 2200 ppm initial boron concentration and maximum boron loading is achieved, there is no criticality concern. An unreviewed safety question is judged not to exist.

TT/1/A/0711/11

"Unit 1 Cycle 11 Zero Power Physics Test" was issued to verify that the core physics behavior is consistent with the core design predictions upon which the Unit 1

cycle 11 safety analysis and Technical Specifications are based. All testing is in accordance with the Oconee Generic Startup Physics Test Program as approved by the NRC. An unreviewed safety question is judged not to exist.

CP/3/A/2002/04D

Change 8 to this procedure, "Test Procedure for Operation of the Post Accident Liquid Sampling," was issued to initiate sampling and allow valve manipulation (to find the optimum means of operating the panel) and to correct for typing errors that mis-identified certain valves. Since the "new" PALSS works basically the same as the previously evaluated PALSS worked, (it has only been improved upon - made easier to use), neither the probability, consequences, possibility of an accident already evaluated in the FSAR be increased nor the possibility, consequences, probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased. An unreviewed safety question is judged not to exist.

OP/0/A/1102/23

"Operation of Containment Hydrogen Recombiner System," was reissued to properly operate the preferred method of maintaining the post-accident containment atmosphere hydrogen concentration below its lower flammability limit of 4% by volume. Included in this reissue are changes to the FSAR which changed hydrogen purge initiation setpoint from 3.5% to 2.8% volume hydrogen. Alarm setpoints and the procedure have been changed to reflect this. An unreviewed safety question is judged not to exist.

OP/2/A/1102/01

Change 136 was incorporated into this procedure, "Controlling Procedure for Unit Startup," which was reissued to add changes necessary because of modifications to the pressurizer auxiliary spray and the feedwater pump minimum recirculation valves. The temperature at which the fourth RCP can be started was raised from 390°F to 482°F. An unreviewed safety question is judged not to exist.

OP/2/A/1102/04

Change 11 was incorporated into this procedure, "Operation at Power," which was reissued to show the changes to "Special Instruction for <4 RCP Operation," (Nuclear Instruments will be calibrated to Thermal Power Best instead of Secondary) and to "Power Escalation 15 to 100% FP," (changes necessary because of modification to the main feedwater pump minimum recirculation valves and at 70% FP the Steam Extraction Check Valve Test will be performed). These changes did not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did it change the margin of safety. An unreviewed safety question is judged not to exist.

OP/2/A/1102/10

Change 71 to 72 was incorporated into this procedure, "Controlling Procedure for Unit Shutdown," which was reissued to incorporate changes necessary because of modifications to pressurizer auxiliary spray. This reissue did not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did it change the margin of safety. An unreviewed safety question is judged not to exist.

OP/2/A/1104/02

Change 46 was incorporated into this procedure, "High Pressure Injection System," which was reissued to include the change to the checklist to lock the component cooling system's isolation valve's engage lever in the AUTO position. This change did not change the level of safety or increase the consequences of malfunction as evaluated in the FSAR, nor did it change the margin of safety. An unreviewed safety question is judged not to exist.

OP/2/A/1104/08

Change 15 was incorporated into this procedure, "Component Cooling System," which was reissued to include changes necessary because of a modification to the pressurizer auxiliary spray allowing spray line-up without a Reactor Building entry. This change did not change the level of safety or increase the consequences of malfunction as evaluated in the FSAR, nor did it change the margin of safety. An unreviewed safety question is judged not to exist.

OP/2/A/1106/01

Change 25 was incorporated into this procedure, "Turbine-Generator" (which describes the proper operation of the turbine-generator). The procedure was reissued to include changes necessary for resetting main turbine trip contact buffers and properly operating main transformer oil cooler fans. This change did not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did it change the margin of safety. An unreviewed safety question is judged not to exist.

OP/2/A/1106/02

Change 65 was incorporated into this procedure, "Condensate and Feedwater System," which was reissued to include changes necessary because of modifications to feedwater pump recirculation valves, moisture separator-reheater drain sample and moisture separator tube bundle replacement. These changes did not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did they change the margin of safety. An unreviewed safety question is judged not to exist.

OP/3/A/1107/10

Change 1 was incorporated into this procedure, "Operation of the Batteries and Battery Chargers," which was reissued to include changes necessary for proposed

Technical Specification change of 3.7.1 (f) (I & C Batteries) which will be more conservative than the existing one. These changes did not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did they change the margin of safety. An unreviewed safety question is judged not to exist.

<u>PT/2/A/115/08</u>

Change 7 was incorporated into this procedure, "Reactor Building Containment Isolation and Verification," which was reissued to include additional sign-offs for new valves providing double isolation of the potential intersystem LOCA leak path from reactor coolant to the low pressure injection system via the auxiliary spray line. An unreviewed safety question is judged not to exist.

<u>PT/1/A/600/01</u>

Change 90 was incorporated into this procedure, "Periodic Instrument Surveillance," which was reissued to change the length of surveillance from a 3-shift format to a 2-shift format bringing the procedure in-line with actual shift schedules while still meeting Technical Specification requirements. An unreviewed safety question is judged not to exist.

<u>TT/0/A/0400/18</u>

"Diesel Air Check Valve Operability Test" was issued to demonstrate the Standby Shutdown Facility's diesel generators starting air check valves are operable. Testing these valves demonstrate their ability to perform their intended function. The diesel air system has two redundant trains each of which is capable of performing the system design function; therefore, removing one train at a time for testing will not increase the consequences or probability of an accident. An unreviewed safety is judged not to exist.

<u>MP/0/A/1150/02</u>

This procedure, "Reactor Vessel Closure Head Removal," was reissued (with approved changes 17-21 incorporated) to provide instruction necessary to remove and store reactor vessel closure head and install indexing fixture. The FSAR and Technical Specifications both address this activity which is performed when the head performs no safety-related function. Duke's response to NUREG-0612 ensured there are no unreviewed safety questions involved.

<u>MP/0/A/1150/02A</u>

This procedure, "Reactor Vessel Closure Head Installation, " was reissued (with approved changes 11-15 incorporated) to provide guidelines for installing the reactor vessel closure head onto the reactor vessel. The FSAR and Technical Specifications both address this activity which is performed when the head performs no safety-related function. Duke's response to NUREG-0612 ensured there are no unreviewed safety questions involved.

MP/0/A/1150/03

This procedure, "Reactor Vessel Closure Head Stud Detensioning and Removal," was reissued (with approved changes 19-20 incorporated) to provide instructions for detensioning and removing reactor vessel closure head studs and installing guide studs. These activities are addressed in FSAR chapters 5, 9, and 13 in a general manner. The procedural instructions were obtained from the approved vendor manuals. All activities lie within the bounds of the procedures mentioned in the FSAR. An unreviewed safety question is judged not to exist.

MP/0/A/3009/11

This procedure, "Motor - Reactor Building Cooling Unit - Disassembly, Inspection, and Reassembly," was reissued (with approved changes 1-2 incorporated) to provide instructions for disassembly, inspection, and reassembly of any reactor building cooling unit motor which has been removed from service, Oconee Technical Specifications 3.3.5 and 4.5.2 address the limiting condition of operation and the surveillance requirements for the Reactor Building Cooling Unit system and allows for the removal from service one of the RBCU motors. This procedure is performed when the motor is not in operating service and cannot be involved in an accident evaluated in the FSAR. The procedure is used to ensure quality maintenance; therefore, upon returning the motor to service, the probability and consequences of equipment malfunction should not be increased. No unreviewed safety question is judged to exist.

OP/0/A/1102/22

This procedure, "Reactor Building Hydrogen Analyzer System," change 3-4 incorporated, was reissued to instruct how to place the hydrogen analyzer in service when indication of hydrogen concentration is needed. Included in this reissue are changes necessary for changes made to FSAR 15.16.3.3.4 which changed hydrogen purge initiation setpoint from 3.5% to 2.8% volume. This reissue does not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR, nor did it change the margin of safety. No unreviewed safety question is judged to exist.

PT/0/A/201/04

This procedure, "PORV Operability Test," change 1 incorporated, was reissued to verify operability of the pressurizer PORV prior to the reactor coolant exceeding 200°F. Changes to this procedure were required because of revisions to Technical Specification 3.1.12 which lowered the operability requirement from 250°F to 200°F. This is a conservative change and will not affect the probability or consequences of accidents previously evaluated in the FSAR. No unreviewed safety question is judged to exist.

PT/1/A/202/13

This procedure, "High Pressure Injection and Low Pressure Injection Pump Venting," provides instructions

on how to vent the casings of the non-operating HPI and LPI pumps for Unit 1 to insure that these pumps can fulfill their intended function in case of a LOCA. The changes to the procedure did not change the method of venting the LPI or HPI pumps, but the procedure steps were rearranged to prevent the operator from having to dress out as many times as before. No unreviewed safety question is judged to exist.

<u>PT/2/A/600/01</u>

Change 74 was incorporated into this procedure, "Periodic Instrument Surveillance," which periodically verifies the proper operation of various instruments and systems. The changes created a separate enclosure for monthly signoffs, altered the limits of operability checks to meet I & E recommendations, and altered the frequency of surveillances from a 3-shift format to a 2-shift format which brings the procedure in-line with actual shift schedules while still meeting Technical Specification requirements. No unreviewed safety question is judged to exist.

<u>PT/0/A/600/15</u>

Change 11 was incorporated into this procedure, "Control Rod Movement," which periodically tests control rod drive operation under actual operating conditions. The changes provided specific guidance on how to perform a step already in the procedure (checking fuses) and added a step to select sequence override before testing Groups 5 thru 7 (this action has been required for past performance of this test, but was not included as a signoff). No unreviewed safety question is judged to exist.

<u>PT/0/A/600/19</u>

Change 3 was incorporated into this procedure, "Surveillance of Safety-related 4160v and 600v Breakers," which periodically determined by visual examination that safety-related 4160v switchgear and 600v load centers are properly racked in. The change altered the location of a signoff for the 'B' Low Pressure Service Water Pump (LPSW) within the procedure. The 'B' LPSW Pump breaker checks were moved to the enclosure for the station rather than Unit 1 or Unit 2. With Unit 1 or Unit 2 below Hot Shutdown and 'B' LPSW Pump being powered from that unit, there was no provision for checking the pump breaker. No unreviewed safety question is judged to exist.

<u>PT/2/A/202/13</u>

This procedure, "High Pressure Injection and Low Pressure Injection Pump Venting," provides instructions on how to vent the casings of the non-operating HPI and LPI pumps for Unit 2 to insure that these pumps can fulfill their intended function in case of a LOCA. The changes to the procedure did not change the method of renting the LPI or HPI pumps, but the procedure steps were rearranged to prevent the operator from having to

dress out as many times as before. No unreviewed safety question is judged to exist.

PT/3/A/202/13

This procedure, "High Pressure Injection and Low Pressure Injection Pump Venting," provides instructions on how to vent the casings of the non-operating HPI and LPI pumps for Unit 3 to insure that these pumps can fulfill their intended function in case of a LOCA. The changes to the procedure did not change the method of venting the LPI or HPI pumps, but the procedure steps were rearranged to prevent the operator from dressing out as many times as before. No unreviewed safety question is judged to exist.

TT/2/A/0150/20

"Coreflood Check Valve Test" was issued to measure the differential pressure at which the coreflood check valves will open during refueling outage EOC-9. The FSAR does not describe this test which will be conducted with no fuel in the core. The test will be performed using demineralized water and instrument air. No unreviewed safety question is judged to exist.

CP/0/B/5200/45

"Liquid Waste Release from Radwaste Facility" was issued to provide the procedure for releasing all processed radioactive liquid waste from the Radwaste Facility to the Keowee Tailrace. FSAR Chapter 11.5.2 specifically addresses RIA 33 for monitoring liquid releases. Although the monitor is being replaced, the new RIA has the same specifications as the old one. The new RIA will also perform all of the required Tech. Spec. surveillances. No unreviewed safety question is judged to exist.

CP/0/B/5200/45

Change 1 was made to this procedure, "Liquid Waste Release from Radwaste Facility," in order to incorporate operating experience on the new equipment and label "RIA-33" as "1RIA-33". FSAR Chapter 11.5.2 specifically addresses RIA-33 for monitoring liquid releases. This change will not affect anything addressed in this section. No unreviewed safety question is judged to exist.

MP/0/A/1100/16

This procedure, "Reactor Building Cooling Units - Coils - Rodding, Back Flushing, Chemical Cleaning, and Visual Inspection," was reissued (with approved changes 2-3 incorporated) to provide guidelines on the above maintenance activities for the RBCUs. This procedure is performed while the RBCUs are out of service and the procedure changes only serve to improve the accuracy and completeness of the procedure steps. Performance of the procedure does not in any way alter the function of the coils from their design as original evaluated in the FSAR. No unreviewed safety question is judged to exist.

MP/0/A/1130/36

This procedure, "OTSG - Tube - Mechanical Plugging/Sleeving," was reissued (with approved change 1 incorporated) to provide guidance on the plugging and sleeving of OTSG tubes. The process takes place while the unit is in refueling shutdown. The potential does not exist for interactions with any equipment important to safety and containment integrity is not degraded by the process. The probability and consequences of accidents and malfunctions of equipment important to safety are not increased. The margin of safety is not reduced. No unreviewed safety question is judged to exist.

MP/0/A/3009/14

"RBCU - Fusible Patches - Preventive Maintenance Inspection" was issued to provide guidelines for preventive maintenance inspection of fusible patches on the Reactor Building Cooling Units after all other RBCU maintenance has been performed. The inspection procedure does not modify the patches in any way except to place them in peak condition. The purpose of the procedure is to ensure that the dropout plates will function as originally designed. No unreviewed safety question is judged to exist.

OP/1 & 2/A/1107/10

This procedure, "Operation of the Batteries and Battery Chargers," was reissued (with change 1 incorporated) to describe the steps necessary for starting up, shutting down, and operating the battery chargers. This change was necessary for proposed revision of Technical Specification 3.7.1 (f) (I & C Batteries) which will be more conservative than the existing one. This reissue does not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR. No unreviewed safety question is judged to exist.

OP/1/A/1107/2

This procedure, "Normal Power," was reissued (with change 32 incorporated) to list the in-plant electrical system lineup for Oconee Unit 1 and provide direction for various normal power operations. This change was necessary for the proposed revision of Technical Specification 3.7.1 (f) (I & C Batteries) which will be more conservative than the existing one. This reissue does not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR nor does it change the margin of safety. No unreviewed safety question is judged to exist.

OP/2/A/1107/2

This procedure, "Normal Power," was reissued (with change 25 incorporated) to list the in-plant electrical system lineup for Oconee Unit 2 and provide direction for various normal power operations. This change was necessary for the proposed revision of Technical Specification 3.7.1 (f) (I & C Batteries) which will be more conservative than the existing one. This reissue does not change the level of safety or increase the

consequences of malfunctions as evaluated in the FSAR, nor does it change the margin of safety. No unreviewed safety question is judged to exist.

OP/3/A/1102/10

This procedure, "Controlling Procedure for Unit Shutdown," was reissued (with changes 90-92 incorporated) to outline the steps necessary to take the plant from an operating condition of about 15% reactor power to a cold shutdown condition. Most of the changes were to add clarity to the procedure, but the major change was the addition of an enclosure to add hydrogen peroxide to the reactor coolant system for crud burst on shutdown. During the performance of the enclosure, the unit will exit the cooldown mode, go into a stabilization period and then enter the heatup mode for the controlled crud burst. Reactor coolant pump operation is not restricted for the RCS heatup mode; therefore, the possibility of an accident or the consequences of an accident previously evaluated in the FSAR will not be increased. The probability or consequences of malfunctions of equipment that have been previously evaluated in the FSAR will not be increased. No unreviewed safety question is judged to exist.

OP/3/A/1103/11

This procedure, "Draining and Nitrogen Purging of Reactor Coolant System," was reissued with change 15 incorporated. This rewrite was made to add more detailed information concerning draining levels, improve the clarity of the procedure and to incorporate approved changes. This rewrite does not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR nor does it change the margin of safety. No unreviewed safety question is judged to exist.

OP/3/A/1107/02

This procedure, "Normal Power," was reissued (with change 25 incorporated) to list the in-plant electrical system lineup for Oconee Unit 3 and provide direction for various normal power operations. This change was necessary for the proposed revision of Technical Specification 3.7.1 (f) (I & C Batteries) which will be more conservative than the existing one. This reissue does not change the level of safety or increase the consequences of malfunctions as evaluated in the FSAR nor does it change the margin of safety. No unreviewed safety question is judged to exist.

PT/0/A/230/01

Change 73 was made to this procedure, "Radiation Monitor Check," to reflect changes to the original Design Engineering Department calculations for the correlation factors. Alarm setpoints were changed as a result of changes to the meteorology tower and the setpoints were based on NUREG 0654. No unreviewed safety question is judged to exist.

PT/1 & 2/A/150/23

This procedure, "High Pressure Injection System Leakage, " was reissued (with change 1 incorporated) to periodically test the High Pressure Injection System outside containment for leakage. The changes to this procedure were editorial in nature and changed the names of some valves to agree with the valve book and current procedures, and changed the location listing of some valves because the locations were incorrect. These changes do not affect the operation or availability of the HPI system.

PT/1/A/600/01

This procedure, "Periodic Instrument Surveillance," was reissued (with change 90 incorporated) to periodically verify the proper operation of various instruments and systems. The changes created a separate enclosure for monthly signoffs and altered the limits for operability checks to meet I & E recommendations. No unreviewed safety question is judged to exist.

AP/1/A/1700/07

"Loss of Low Pressure Injection System" was issued to provide the necessary actions following the loss of the low pressure injection system during Emergency Core Coolant System operation mode or decay heat removal mode. The procedure gives guidance on how to reestablish LPI flow in the event normal LPI pump suction is lost by providing an alternate LPI pump suction from the RCS. This action reduces the consequences of accidents and malfunctions without increasing the probability and possibility. The margin of safety is considered to be increased. No unreviewed safety question is judged to exist.

AP/1/A/1700/11

Change 4 was made to this Unit 1 procedure, "Loss of Power," to provide a method of powering the standby buses from transformer CT-5 to ensure that undervoltage conditions do not damage plant equipment. This action is in response to if a blackout should occur and reduces the consequences of such an accident without increasing the probability or possibility. The margin of safety is considered to increase. No unreviewed safety question is judged to exist.

AP/2/A/1700/11

Change 5 was made to this Unit 2 procedure, "Loss of Power," to provide a method of powering the standby buses from transformer CT-5 to ensure that undervoltage conditions do not damage plant equipment. This action is in response to if a blackout should occur and reduces the consequences of such an accident without increasing the probability or possibility. The margin of safety is considered to increase. No unreviewed safety question is judged to exist.

AP/3/A/1700/11

Change 4 was made to this Unit 3 procedure, "Loss of Power," to provide a method of powering the standby buses from transformer CT-5 to ensure that undervoltage

conditions do not damage plant equipment. This action is in response to if a blackout should occur and reduces the consequences of such an accident without increasing the probability or possibility. The margin of safety is considered to increase. No unreviewed safety question is judged to exist.

MP/0/A/1150/05B

This procedure, "Reactor Vessel - Canal Seal Plate - Removal," was reissued (with changes 7-8 incorporated) to provide guidelines for removal and storage of canal seal plate. The canal seal plate is analyzed as a missile during a postulated LOCA within the reactor vessel annulus area. No specific placement of the seal plate around the vessel head is required to prevent a safety concern. The minor change in the placement of the seal plate specified in this procedure revision to provide consistency between units does not affect the analysis performed. The change in placement of the canal seal plate, although not exactly as specified in the FSAR, is in agreement with the manufacturer's instruction. The placement, as specified in the FSAR, is apparently in error, as the seal plate cannot be supported or installed as stated. No unreviewed safety question is judged to exist.

MP/0/A/1150/12

This procedure, "Reactor Vessel - Closure Head - Stud Handling for Inspection and Cleaning," was reissued with previously approved changes 10-13 incorporated. Cleaning of head studs occur during refueling shutdown when the studs serve no safety-related function and occur in the Hot Machine Shop or adjacent to the head storage stand where no safety-related equipment is located. Handling of studs has been addressed by Duke Power's response to NUREG-0602 in so for as the handling could affect any safety-related equipment. No unreviewed safety question is judged to exist.

MP/0/A/1200/27

This procedure, "Valve - Fisher - YY Type - Disassembly and Reassembly," was reissued with previously approved changes 8-9 incorporated. The changes to this procedure are all of a minor nature and are only to improve the format and to satisfy Administrative Policy Manual requirements. No unreviewed safety question is judged to exist.

MP/0/A/2000/32

This procedure, "Keowee Hydro's Mulsifyre System Semiannual Check," was reissued with change 9 incorporated. This procedure will verify the proper operation of the mulsifyre systems. In addition, steps are included for the test of the Keowee Main Step-up Transformer lock-out circuit, the Keowee Fire Protection Pump, and the automatic closure circuit of the liquid waste dilution valve during a mulsifyre trip of the fire protection circuits. This procedure will not

change the margin of safety as determined in the Technical Specifications or change accident mitigation as described in the FSAR. This procedure will be performed within the limitations of the Tech. Specs. and FSAR.

MP/0/A/3007/37

This procedure, "Air Handling - Reactor Building Cooling Unit Fan - Inspection, Adjustment of Blade Pitch and Torquing of Blade Nuts," was reissued with previously approved changes 1-2 incorporated. This procedure involves the inspection of the RBCU fan blades to verify that they are positioned at the proper pitch angle. The inspection and readjustment of pitch does not adversely affect the operating characteristics of the fans. Verifying that the pitch of the blades are at the originally specified angle ensures that the fans will operate as designed. Performance of this procedure does not affect the possibility, probability, or increase the consequences of accidents or malfunctions evaluated in the FSAR. No unreviewed safety question is judged to exist.

MP/0/B/1800/47

This procedure, "Condenser - Main - Tubes - Testing and Plugging," was reissued with previously approved change 3. FSAR Section 10.4.1.3 states that the main condenser is not assigned a safety class and is not required for safe reactor shutdown. This procedure only addresses condenser inspection and plugging. Plugging will not affect the possibility, probability, or consequences of an accident or equipment malfunction. At most it may cause a slight decrease in condenser efficiency. No unreviewed safety question is judged to exist.

MP/0/B/1800/49

This procedure, "Heater - Low Pressure Feedwater - Tube Plugging and Repair," was reissued with previously approved change 3. The feedwater heaters at Oconee are addressed in FSAR Section 10.4.6.; however, no specific mention is made of the feedwater heaters with regard to accident analysis. The heaters are not safety-related components. The performance of this procedure to open, repair, and restore the low pressure feedwater heaters will allow us to maintain the heaters in a state of efficient operation. No unreviewed safety question is judged to exist.

MP/1/A/2200/03

This procedure, "Keowee Hydro's Governor Actuator No. 1 Inspection and Maintenance," was reissued with change 4 incorporated. A description of the Keowee Hydro Station is given in FSAR Section 8.3 and Technical Specification Section 3.7 gives time restrictions for the operation of Keowee for test or maintenance of one emergency power path, 72 hours, or unit maintenance in excess of the 72 hour time restriction of up to 45

days. A detailed description is given in the "Bases" of Section 3.7 of operation with one unit out of service and limits of operation as associated with the outage. This procedure will not change accident mitigation. No unreviewed safety question is judged to exist.

<u>MP/1/A/2200/04</u>

This procedure, "Keowee Hydro's Replacement of Unit No. 1 Turbine Shaft Packing," was reissued with change 1 incorporated. Maintenance performed in this procedure will be performed per Technical Specifications which give time restrictions for the operation of Keowee for test or maintenance of one emergency power path. A detailed description is given in the "Bases" of Tech. Spec. 3.7 for operation of Keowee with one unit unavailable and limits of operation as associated with the outage. This procedure will not change the margin of safety nor change accident mitigation. No unreviewed safety question is judged to exist.

<u>MP/1/A/2200/10</u>

This procedure, "Keowee Hydro's Unit 1 Servomotor Packing Maintenance," was reissued with change 1 incorporated. Maintenance performed in this procedure will be performed per Technical Specifications which give time restrictions for the operation of Keowee for test or maintenance of one emergency power path. A detailed description is given in the "Bases" of Section 3.7 of operation of Keowee with one unit unavailable and limits of operation as associated with the outage. This procedure will not change the margin of safety nor change accident mitigation. No unreviewed safety question is judged to exist.

<u>MP/1/A/2200/09</u>

This procedure, "Keowee Hydro's Unit No. 1 Turbine Bearing Inspection or Replacement," was issued as a new procedure. Maintenance performed in this procedure will be performed per Technical Specifications which give time restrictions for the operation of Keowee for test or maintenance of one emergency power path. A detailed description is given in the "Bases" of Section 3.7 of operation of Keowee with one unit unavailable, both units unavailable, and limits of operation as associated with the outage. This procedure will not change the margin of safety nor change accident mitigation. No unreviewed safety question is judged to exist.

<u>MP/2/A/2200/03</u>

This procedure, "Keowee Hydro's Governor Actuator No. 2 Inspection and Maintenance," was issued as a new procedure. A description of the operation of Keowee is given in the FSAR. Technical Specifications give time restrictions for the operation of Keowee for test or maintenance of one emergency power path. A detailed description is given in the "Bases" of Section 3.7 of

operation with one unit out of service and limits of operation as associated with the outage. This procedure will not change the margin of safety nor change accident mitigation. No unreviewed safety question is judged to exist.

OP/0/A/1600/11

Change 4 was incorporated into the "Standby Shutdown Facility Emergency Operating Procedure." The change involved the operation of the SSF auxiliary service water pump throttling control valves which was necessary because plant performance tests showed the pump operates well below its design head-capacity curve. With the changes, the pump is capable of fulfilling its design function. These changes do not affect the possibilities, probabilities, or consequences of accidents or malfunctions as evaluated in the FSAR. No unreviewed safety question is judged to exist.

OP/0/A/2000/01

This procedure, "Keowee Hydro Main Step-up Transformer," was reissued (with change 1 incorporated) to outline the steps for the removal from service and Restoration to service of the Keowee Main Step-up transformer. A description of the main step-up transformer is given in the FSAR. Technical Specifications give time restrictions for the operation of Keowee for test or maintenance of one emergency power path. A detailed description is given in the "Bases" of Section 3.7 of operation with or without the transformer and limits of operation as associated with the outage of the transformer. This procedure will not change the margin of safety nor change accident mitigation. No unreviewed safety question is judged to exist.

OP/3/A/1102/10

This procedure, "Controlling Procedure for Unit Shutdown," was reissued (with changes 93 to 95 incorporated) to outline the steps necessary to take the unit from an operating condition of 15% power to cold shutdown. Changes made to Enclosure 4.2 concerning the RCS Low Range Pressure Indicator and the PORV Setpoint Selector were made to ensure that low RCS temperature overpressure protection is available. Changes made to Enclosure 4.3 and 4.8 for Auxiliary Pressurizer Spray line-up was due to a modification that changed the High Pressure Injection System to allow Auxiliary Pressurizer Spray valve alignment without Reactor Building entry. These changes did not affect the possibility, probability, or consequences of accidents or malfunctions evaluated in the FSAR nor change the margin of safety. No unreviewed safety question is judged to exist.

OP/3/A/1104/02

This procedure, "High Pressure Injection," was reissued (with change 35 incorporated) to define the operation of the High Pressure Injection System. Changes were

made to the valve checklist and enclosures because of a modification to allow auxiliary pressurizer spray valve alignment without Reactor Building entry. Also, valve names were changed to agree with the Control Room switches. This procedure does not affect the possibility, probability, or consequences of an accident or malfunction nor does it affect the margin of safety. No unreviewed safety question is judged to exist.

OP/3/A/1104/04

Change 36 was made to this procedure, "Low Pressure Injection System," to delete an auxiliary pressurizer spray valve and add two isolation valves because of a modification to allow auxiliary pressurizer spray valve alignment without Reactor Building entry. This procedure change does not affect the possibility, probability, or consequences of an accident or malfunction nor does it change the margin of safety. No unreviewed safety question is judged to exist.

OP/3/A/1104/18

Change 32 was made to this procedure, "Gaseous Waste Disposal System," to allow Unit 3 Reactor Building depressurization via the spent fuel pool ventilation system. This change does not affect the possibility, probability, or consequences of accidents or malfunctions nor does it change the margin of safety. No unreviewed safety question is judged to exist.

PT/0/A/610/05B

This procedure, "Electro-Mechanical Relay Breaker Trip Test," was reissued with change 14 incorporated. The changes were to correct several errors associated with relay numbers, reworded prerequisite statements for testing startup transformers for clarity and changed the frequency of this test from seven times per year to quarterly. The Operating Department conducted a study and recommended that the frequency of testing of these relays could be changed to quarterly per the Methods and Procedures Manual. Based on past experience, the Operating Department felt these changes will not affect the reliability of off-site power. No unreviewed safety question is judged to exist.

MP/0/A/1130/39

This procedure, "OTSG - Nozzle Dam - Installation and Removal," was issued as a new procedure. The nozzle dams are installed only during a refueling shutdown and serve as a boundary between the refueling canal and the primary side of the OTSGs. It has been shown by analysis that a catastrophic failure of the dams is highly improbable so the most severe failure mode would be leakage. The worst leakage would be so small that it would take days for the canal to drain so there is sufficient time for corrective actions to be taken. No design basis accidents are impacted by the use of nozzle dams, even in a failed condition. They cannot cause a fuel handling accident nor lead to the release

of radiation. This procedure does not affect the possibility, probability, or consequences of accidents or malfunctions nor does it change the margin of safety. No unreviewed safety question is judged to exist.