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SUBJECT: Forwards descriptions of Oconee facility changes completed between Jan-Dec 1993,per 10CFR50.59(b)(2).

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June 30, 1994

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

Subject: Oconee Nuclear Station
Docket Nos. 72-04, 50-269, 50-270, 50-287
10 CFR 50.59 Annual Report

Gentlemen:

Attached are descriptions of Oconee facility changes which were completed subject to the provisions of 10 CFR 50.59 between January 1, 1993 and December 31, 1993. This report is submitted pursuant to the requirement of 10 CFR 50.59 (b)(2).

Very truly yours,

J W Hampton

J. W. Hampton

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REPORT: OE 5777, 5202

DESCRIPTION: This modification will lower the actuation setpoint (feedwater discharge pressure switches) of the Anticipated Transient Without Scram (ATWS) Mitigation Systems Actuation Circuitry (AMSAC) and of the Emergency Feedwater (EFW) System. These pressure switches provide Loss of Main Feedwater (LMFW) signals for Reactor Protection System (RPS) initiated anticipatory reactor trip, EFW actuation for the post-shutdown Reactor Coolant System (RCS) decay heat and Reactor Coolant Pump heat removal, and for the actuation of the AMSAC initiated Turbine Trip and EFW actuation.

AMSAC, RPS and EFW are designed to actuate if both feedwater pumps can no longer supply water to the steam generators. This condition is sensed in two ways: the hydraulic oil pressure for both feedwater turbines drops below 75 psig or both main feedwater pump discharge pressures drop below 800 psig. This setpoint will be changed to 770 psig.

SUMMARY: During July 1991, the AMSAC/EFW initiation setpoint was raised from 750 psig to 800 psig in response to a Unit 3 trip which revealed that feedwater pump discharge pressure remained above the low pressure actuation setpoint (then 750 psig) until the D Heater Drain Pumps were secured. On May 8, 1992, Unit 1 tripped from 15 percent power due to an up spike in condensate flow and down spike in condensate flow and down spike in feedwater pump suction and discharge pressures. The Main Feedwater Pump did not trip, but the reactor experienced an anticipartory trip designed to sense a Loss of Main Feedwater Pumps. During the transient, the lowest apparent feedwater pressure reached was 802 psig, the pressure switch setpoint was 800. It was concluded that too much margin between equipment tranfer and the trip setpoint may have been removed.

Minor modification OE-5445 removed the tenth stage of the 2D1 Heater Drain Pump; and minor modification OE-5446 removed the tenth stage of the 2D2 Heater Drain Pump. These modifications were performed to lower the maximum (dead head) discharge pressure of the heater drain pumps so that their continued operation during a Loss of Main Feedwater would not inhibit the operation of the EFW and AMSAC systems. The destaging of the 2D2 Heater Drain Pump yielded an actual shutoff head pressure of 657.85 psid. With instrument and switch inaccuracies included, over 40.15 psig of margin exits between the new setpoint (770 psig) and the maximum destaged 2D2 Heater Drain Pump discharge pressure.

The difference between the claculated maximum discharge pressure and the actual discharge pressure as seen during the March 3, 1992 transient has been attributed to error from reading the pump curves. The setpoint implemented by this minor modification is based upon as measured discharge dead head tests for the pumps installed in the system. The destaged 2D1 Heater Drain Pump dead head test pressure measured was 673.05 psid. This value, with worst case switch inaccuracies and pump suction pressure conditions, provides a margin of over 24.95 psig based upon actual pump performance during the dead head test. The maximum E Heater Drain Pump discharge pressures are less than the maximum discharge pressure of either destaged D Heater Drain Pump and, therefore, are not limiting.

With the maximum discharge pressure from either the D or E Heater

Drain Pumps less than the actuation setpoint, the AMSAC and EFW systems will perform their intended functions when needed. Also, the lowering of the actuation setpoint will provide more margin between it and normal operational parameters of the feedwater system and between it and equipment transfers. Thus, more margin to spurious actuations of these systems in response to plant transients will result. AMSAC and EFW features will continue to be controlled by procedure during startup and shutdown, as before, at 800 psig main steam header pressure. RPS automatic actuation will remain unchanged.

Of the previously evaluated accidents in the FSAR, only the Main Steam Line Break (MSLB) accident is potentially affected by this setpoint change. The Main Steam Line Break analysis was not significantly affected when the setpoint was raised and therefore was not changed. Although credit is taken for the steam generator low level actuation setpoint, none is taken for the anticipatory loss of feedwater setpoint which is being changed in this minor modification. Thus, the MSLB analysis remains valid for the new AMSAC/EFW setpoint.

This modification involves no safety concerns or unreviewed safety questions.

REPORT: OE 5340

DESCRIPTION: The scope of the minor modification is to replace 3CC-77 CRD Cooling Penetration #44 Inside Check Valve from a 3 inch valve to a 2 1/2 inch valve.

SUMMARY: The modification will not change any features of the component or introduce any unwanted system interactions. Seismic and environmental qualifications will be maintained, as required. The modification will not have a significant affect on the steady state or transient characteristics of the system. The modification will not affect the other units and the station will be in compliance with all technical specifications after the modification. The modification will not affect the frequency of operation or increase any of the testing requirements. The modifications will not change or prevent any actions described in the FSAR nor will it alter any assumptions previously made in evaluating the radiological consequences of an accident. The modification will not affect any fission product barriers or hinder the access to accident mitigation equipment in post accident conditions.

REPORT: OE 5720, 5719

DESCRIPTION: Minor modification 5719 and 5720 will replace the valve operators on Low Pressure Service Water (LPSW) valves 2LPSW-565 and -566. These changes are in response to Generic Letter 89-10, and will improve the vlaves' ability to close during accident conditions. Normal operation of the valves will not be affected.

SUMMARY: Valve 2LPSW-565 is a containment isolation valve, which is required to close upon ES actuation to isolate LPSW supplied to the Auxiliary Reactor Building Coolers (RBCS). This would occur during certain accident scenarios when containment integrity was required. As the Auxiliary RBCs are not designed to function during an accident, this valve is also a system boundary. The valve's stroke time will be the same as the existing setup and will continue to be adequate for all design functions.
Valve 2LPSW-566 is also technically a containment isolation valve, but is not required to close upon ES actuation. In fact, it opens to provide LPSW cooling water to the "B" Reactor Building Cooler (RBCU) in the event of elevated Reactor Building pressure. The valve's stroke time will be the same as the existing setup, and will continue to be adequate for all design functions.

Environmental and seismic qualifications of the valves will not be affected. All new equipment installed will be QA Condition 1. Pressure boundaries will remain intact. The valves will continue to perform their design functions. No new failure modes or accidents will be introduced as a result of this change. The valves will satisfy the requirements of all accident analyses, and the stroke times will continue to be adequate.

No unreviewed safety questions are created by or involved with this modification.

REPORT: OE 5761

DESCRIPTION: This minor modification will replace 2HP-144 with a DMV-851. 2HP-144 is currently a stop check valve which is being replaced with a check valve and a block valve (the block valve will be added by Minor Modification OE-5762). The scope of the minor modification will include adding reducers to the line and revising the affected documents.

SUMMARY: 2HP-144 is a QA Condition 1 Velan Stop Check Valve which functions as the SEal Supply to Pump A2 Stop Check. It is a containment isolation valve which must be able to isolate Reactor Building penetration 23. 2HP-144 is in a 1 1/2" line with a design pressure of 3040/3120 psi and a design temperature of 200/150F. The replacement valves will be a 1" Anchor Darling Check Valve and a 1" Anderson Greenwood Globe Valve(OE-5762). Together they will serve the same function as the existing valve.

The minor modification is necessary because the existing valve has a cracked seat and cannot be repaired. This was discovered when the valve failed it's leak rate test.

This modification will not change any features of the system or introduce any unwanted system interactions. Seismic and environmental qualifications will be maintained, as required. The modification has no affect on the other units and the station will be in compliance with all technical specifications after the modification. The modification will require a revision to the FSAR because the line size is being reduced from 1 1/2" to 1".

The replacement valves will meet the design, material, and construction standards of the system. System operation and function will not be affected by the modification so the system will not operate outside of its design or testing limits. The combination of the check valve and globe valve will perform the same function as the existing stop check valve.

The modification will not change or prevent any actions described in the FSAR nor will it alter any assumptions previously made in evaluating the radiological consequences of an accident. 2HP-144 does not play a role in mitigating the radiological consequences of an accident. The modification will not affect any fission product barriers or hinder the access to accident mitigation equipment in post accident conditions. This modification does not involve any unreviewed safety questions.

REPORT: NSM ON-12564,BL1

DESCRIPTION: The Turbine Driven Emergency Feedwater Pump (TDEFWP) presently resets itself when a main feedwater pump resets. This will cause a termination of Emergency Feedwater if the TDEFWP is the only pump operating. This modification will modify the control circuit for the TDEFWP to provide a seal-in such that deliberate operator action is required to reset it from its control switch.

The existing control switch for each train of the Steam Generator Level Control System (SGLCS) for selection of Automatic or Manual does not provide the operator any indication of status other than physical position orientation. The selector switch will be replaced with an indicating-type pushbutton which will provide the operator with true automatic/manual selection capability. This also involves circuitry changes within the SGLCS cabinet.

Limit switches will be added to valves to provide inputs to indication lights which will be mounted on the main control board.

The power supplies for each train of SGLCS will be relocated within the cabinet to provide improved heat dissipation. Manul control connections for FDWML0046 and FDWML0047 will be upgraded with vinyl tubing.

SUMMARY: The TDEFWP start circuit is modified to seal-in when an auto-initiation signal is received. This will maintain the circuit de-energized following a restart of the main feedwater pumps. This contact can only be bypassed to reset the TDEFWP by the Main Control Board. This modification will require deliberate operator action to reset the pump.

The Auto/Manual switches for each train of the SGLCS are removed and replaced with pushbuttons. The existing switches supplied an automatic option which was not completely reliable in all conditions. The new pushbuttons can select between full auto and full manual at any time or plant condition. This will provide a more reliable manual loader and direct acting demand indication. The control pushbuttons for selection of PRIMARY and BACKUP control trains for valves FDW-315 and FDW-316 will be replaced with selector switches. Changes to the Main Control Board have been seismically reviewed and have no adverse impact.

The SGLCS cabinet will be changed to electrically isolate the existing analog computer output signal. This involves the addition of a signal isolator card. This card's failure will not affect the SGLCS operation; it will only cause the control room indication to fail in a way that is easily detected. The lead/lag card will be removed from each loop because it is not performing any required function. The only cabling changes will be within the cabinet.

Relocation of the power supplies for each train of the Steam Generator Level Control System will provide additional air circulation between the supplies and the card racks. This modification is needed to reduce the likelihood of heat-related card failures. A 10CFR50 Appendix R review has been performed and has concluded there is no impact on how the system will perform.

Manual control connections for FDWML0046 and FDWML0047 will be improved by replacing the existing tubing with vinyl tubing. The existing tubing has experienced periodic leaks. The new vinyl tubing is recommended by the manufacturer of the indicators.

The seismic evaluation for this modification concludes that the changes to the SGLCS cabinet (i.e. the addition of the new indicator card, deletion of the lead/lag card, and the relocation of the existing power supplies) has minimal impact on the seismic response of the SGLCS cabinet and the seismic qualifications of the components contained inside.

REPORT: NSM ON-12593

DESCRIPTION: This modification will change the indicating range of the quench tank temperature instrument loop from 0-250 F TO 50-350F. This modification will entail changing the scales on the control board meter, recorder, and re-calibrating the present instrument loop.

SUMMARY: The new quench tank temperature monitoring instrument will be calibrated to have a range of 50-350 F. The current instrument calibration covers a range that includes lower temperatures, but the NRC reviewed the quench tank temperature instrument range beginning at 50 F and found it acceptable. This instrument is for monitoring and performs no automatic functions. This modification does not physically remove or replace the instrument, meter, or recorder. The modification is only re-calibrating the instrument loop and replacing the meter and recorder scales. These changes do not adversely affect the components' quality, environmental qualification, or method of display. The mounting of the instrument, meter, and recorder is not adversely affected since the instrument loop calibration and new meter and recorder scales do not affect the components' weight or weight distribution. since the components' weight and weight distribution are not changed, no adverse seismic effects to the control board are created. The instrumentation's accuracy and sensitivity for normal operation monitoring is not degraded.

REPORT: NSM ON-12596,BL1

DESCRIPTION: Part A of NSM ON-12596 replaces non-safety related nuclear instrumentation (NI) recorders identified as R7 and R12 located on the safety-related main control board. The two existing recorders will be replaced with a single functionally equivalent recorder. The new recorder will be non-safety qualified. The orginal control board hole for recorder R12 will be enlarged to accommodate the new recorder. The existing signal connections for both R7 and R12 will be connected to the applicable terminals of the new recorder. The xisting power supply connection for R12 (a highly reliable, battery backed non-safety power supply) will be used for the new recorder and the power connection for R7 will be deleted. The new recorder will retain the R12 designation on the control board drawings and all original scales and ranges will be retained.

The replacement of the two NI recorders with a single recorder 1) provides additional space on the main control board for additional instrumentation displays, controls, etc., and 2) was indentified as an operational enhancement by the control room review team in response to NUREG-0737, Supplement 1.

Part B will add two full range channels of neutron flux instrumentation. In order to install two channels (intermediate rante) will be replaced with new fission chambers. Subsequently, NI-3 will be referred to as the "wide range" detector. The signal processor drawer contains diagnostic and self-testing circuitry. It will be housed in the RPS cabinet and receive power from the same source as the rest of the cabinet, but it will have its own isolation breaker at the source. New cabling, electronic panels, and qualified QA Condition 1 display devices will also be added.

SUMMARY: The NI system is designed to supply the reactor operaator with neutron flux information over the full operating range of the reactor and to supply reactor power information to the Reactor Protection System (RPS) and to the Integrated Control System (ICS).

The nuclear instrumentation consists of nine channels divided into three ranges of sensitivity: 2 redundant source ranges, 2 redundant intermediate ranges, and 5 power range channels. The three ranges combine to give a continuous measurement of reactor power level to approximately 125 percent of rated power. A minimum of one decade of overlapping information is provided between successive higher ranges of instrumentation. At any time, one channel each of the source and intermediate ranges provide input to the existing recorder R7. Each of the four source and intermediate range channels are provided with interlocks.

Four power range channels provide input signals to the RPS and the fifth provides an input signal to the ICS and to the existing recorder R12. A second linear power indication is provided to ICS by one of the four power range channels providing RPS input (NI-5). The two power range signals input to the (SASS) for detection of mismatches between the two input signals.

Replacing the NI recorders will not result in any changes to the RPS and ICS input signals or to the signal ranges of source, intermediate, or power range neutron flux monitoring. However, the addition of the new detectors and associated equipment will provide new safety and non-safety outputs to the Main Control Board and the OAC respectively. This new equipment meets our Regulatory Guide 1.97 commitment for neutron flux monitoring.

All equipment has been environmentally and seismically qualified. Class 1E isolator outputs to the Main Control Board are physically and electrically isolated from the Non-1E isolator outputs to the plant computer. One loop for each channel will contain a recorder, and power to all equipment is safety-related.

The Main Control Board layout, which is designed to minimize the time required for the operator to evaluate the system performance under accident conditions and assure safe and reliable operations, will be enhanced by part A of this modification by providing more control board space needed for adding additional indicators. The new indicators addes in Part B are qualified, QA-1 Dixon indicators with a scale of 1E-8% to 200% power. A control board seismic review has been completed for these new indicators.

Although not required to mitigate the consequences of any accident, the source and intermediate range channels provide control rod withdrawal "inhibit" interlock signals to limit the power ascension rate during reactor startup operations. Replacement of the two NI detectors and recorders will not affect this RPS "inhibit" function.

The power range channel provides one of the RPS signals necessary to mitigate overpower transients resulting from control rod withdrawal at rated power. Replacement of NI recorders or the source and intermediate range monitors will not affect the RPS to maintain reactor thermal power below technical specification safety limits.

REPORT: NSM ON-12892

DESCRIPTION: This modification installs auctioneer modules (1 per loop) in the ICS main feedwater valve differential pressure circuitry. This revised circuitry allows the operator to manually select either of the two transmitters in each loop or place the loop in automatic allowing the auctioneering modules to automatically select the highest of the two transmitter signals.

SUMMARY: The change in instrumentation logic will require less operator action and reduces the probability of a unit transient. The modification will not adversely affect any accident mitigation structure, system or component. The ICS does not perform any accident mitigation function. FSAR accident analysis involving feedwater flow ill not be affected by this change in instrumentation logic. The seismic qualification of the control board is not affected by this modification. The change in instrumentation logic will reduce the probability of a malfunction of the instrumentation's input to the ICS. This modification involves no unreviewed safety questions or safety concerns. No FSAR changes are required. FSAR Section 7.4.2.2.2 states the selected differential pressure signal is indicated. It is recommened some clarification be made in order to prevent confusion.

REPORT: NSM ON-22563

DESCRIPTION: This modification will modify the control circuits of the Emergency Feedwater Syste, the Auto/Manual selector switch of the Steam Generator Level Control System, and add limit switches to FDW-315 and -316 to indicate their position in the Control Room. These changes address three different Human Engineering Discrepancies identified in the Control Room.

SUMMARY: The TDEFWP start circuit is modified to seal-in when an auto-initiation signal is received. A normally open contact from time delay EFWPTX is inserted into the circuit ahead of relay SV74 to maintain the circuit de-energized following a restart of the main feedwater pumps. An automatic start of the TDEFWP turbine must be present for 15 seconds for the seal-in circuit to actuate. This will be changed from the existing system, which only indicated to the control room after 15 seconds. This will have no effect on safety analysis calculations because there will be no impact on EFW actuation time. This modification will require deliberate operator action to reset the pump.

The Auto/Manual switches for each train of the SGLCS will be removed and replaced with pushbuttons. The existing switches supplied an automatic option which was not completely reliable in all conditions. The new pushbuttons can select between fully automatic and fully manual control at any time or plant condition. This will provide a more reliable manual loader and direct acting demand indication. The control pushbuttons for selection of PRIMARY and BACUP control trains for valves FDW-315 and FDW-316 will be replaced with selector switches. The design documents reflect the correct selector switches. Changes to the Main Control Board have been seismically reviewed and will have no adverse impact.

The limit switches added to FDW-315 and -316 will be located in the Penetration Room, a potentially harsh environment. The new limit switches and the new cable run (from the switches to the SGLCS cabinet) will be environmentally qualified for their environment. The switches will be powered by a non-safety, highly reliable power source.

The SGLCS cabinet will be changed to electrically isolate the existing analog computer output signal. This involves the addition of a signal isolator card. This card's failure will not affect the SGLCS operation; it will only cause the computer indication to fail in a way that is easily detected. The lead/lag card will be removed from each loop because it is not performing any required function. The only cabling changes (beside the new limit switch cabling) will be within the cabinet.

Relocation of the power supplies for each train of the Steam Generator Level Control System will provide additional air ciruculation between the supplies and the card racks. This modification is needed to reduce the likelihood of head-related card failures. A 10CFR50 Appendix R review has been performed and has concluded there is no impact on how the system will perform.

REPORT: NSM ON-22593

DESCRIPTION: This modification will change the indicating range of the quench tank temperature instrument loop from 0-250 F to 50-350 F. This modification will entail changing the scales on the control board meter, recorder, and re-calibrating the present instrument loop.

SUMMARY: The new instrument will be calibrated to have a range of 50-350 F. The current instrument calibration covers a range that includes lower temperatures, but the NRC reviewed the quench tank temperature instrument range beginning at 50 F and found it acceptable. This instrument is for monitoring and performs no automatic functions. This modification does not physically remove or replace the instrument, meter, or recorder. The modification is only re-calibrating the instrument loop and replacing the meter and recorder scales. These changes do not adversely affect the components' quality, environmental qualification, or method of display. The mounting of the instrument, meter, and recorder is not adversely affected since the instrument loop calibration and new meter and recorder scales do not affect the components' weight or weight distribution. Since the components' weight and weight distribution are not changed, no adverse seismic effects to the control board are created.

The failure mode for the meter, on loss of power to the meter, will not change because of the scale change. The meter scale will not light up for this failure mode. The failure mode for loss of instrument signal to the meter is also not changed. The meter will still flash, indicating loss of instrument signal. The failure mode for the recorder, on loss of power to the recorder, will not be changed because of the scale change. The recorder will stop recording. The failure mode for loss of instrument signal to the recorder will not change. The recorder will still record a constant temperature. The instrumentation's accuracy and sensitivity for normal operation monitoring is not degraded. The instrumentation readout is within 0.5 F of the current instrumentation's accuracy. This increased error is considered acceptable for the quench tank temperature monitoring function.

REPORT: NSM ON-22596

DESCRIPTION: Part A of NSM ON-22596 replaces non-safety related nuclear instrumentation (NI) recorders identified as R7 and R12 located on the safety-related main control board (i.e., control board 2UB1). The two existing recorders will be replaced with a single functionally equivalent recorder. The new recorder will be non-safety qualified. The original control board hole for recorder R12 will be enlarged to accommodate the new recorder. The existing control board hole for recorder R7 will be patched to allow installation of indicators associated with NSMs ON-22447 and ON-22448. The existing signal connections for both R7 and R12 will be connected to the applicable terminals of the new recorder. The existing power supply connection for R12 (a highly reliable, battery backed non-safety power supply) will be used for the new recorder and the power connection for R7 will be deleted. The new recorder will retain the R12 designation on the control board drawings and all original scales and ranges will be retained.

The replacement of the two NI recorders with a single recorder 1) provides additional space on the main control board for additional instrumentation displays, controls, etc., and 2) was identified as an operational enhancement by the control room review team in Oconee's response to NUREG-0737, Supplement 1.

Part B of NSM ON-22596 will replace the two excore neutron detectors with two full range channels of neutron flux instrumentation. In order to install two channels of instrumentation, detectors NI-1 (source range) and NI-3 (intermediate range) will be replaced with new fission chambers. Subsequently, NI-3 will be referred to as the "wide range" detector. The signal processor drawer contains diagnostic and self-testing circuitry to continuously verify proper equipment and system operation. It will be housed in the RPS cabinet and receive power from the same source as the rest of the cabinet, but it will have its own isolation breaker at the source. New cabling, electronic panels, and qualified QA Condition 1 display devices will also be added.

SUMMARY: The NI system is designed to supply the Reactor Operator with neutron flux information over the full operating range of the reactor and to supply reactor power information to the Reactor

Protection System (RPS) and to the Integrated Control System (ICS).

Nuclear Instrumentation consists of nine channels divided into three ranges of sensitivity: 2 redundant source ranges, 2 redundant intermediate range, and 5 power range channels. The three ranges combine to give a continuous measurement of reactor power level to approximately 125 percent of rated power. A minimum of one decade of overlapping information is provided between successive higher ranges of instrumentation. At any time, one channel each of the source and intermediate ranges provides input to the existing recorder R7. Each of the four source and intermediate range channels are provided with interlocks (i.e., control rod withdrawal "inhibit" on high startup rates).

Four power range channels provide input signals to the RPS and the fifth provides an input signal to the ICS and to the existing recorder R12. A second linear power indication is provided to ICS by one of the four power range channels providing RPS input (NI-5). The two power range signals input to the ICS are monitored by the Smart Automatic Signal Selector (SASS) for detection of mismatches between the two input signals.

Replacing the NI recorders (Part A) will not result in any changes to the RPS and ICS input signals or to the signal ranges of source, intermediate, or power range neutron flux monitoring. However, the addition of the new detectors and associated equipment (Part B) will provide new safety and non-safety outputs to the Main Control board and the OAC, respectively.

Neutron flux monitoring is a Category 1, Type B variable. Regulatory Guide 1.97 delineates monitoring requirements including redundant instrumentation which is seismically qualified, environmentally qualified, electrically independent, physically separated, and powered from safety related power sources. At least one channel should be displayed on a direct reading or recording device. The instrumentation should also be capable of monitoring neutron flux in the range of 1E-6% - 100% full power. The new neutron flux monitoring channels will meet all of the above criteria. Class 1D isolator outputs to the Main Control Board are physically and electrically isolated from the non-1E isolator outputs to the plant computer. One loop for each channel will contain a recorder, and power to all equipment is safety-related. The base system equipment is QA-1 and qualified for post-accident monitoring. The signal amplifier mounting, RPS cabinets (signal processor and isolation/expansion assembly, which are rack mounted), instrument wells (shielding), and control boards modified by this change have been reviewed for seismic and anchorage requirements. All necessary equipment has been environmentally and seismically qualified. This new equipment meets Duke Power's Regulatory Guide 1.97 commitment for neutron flux monitoring.

The Main Control Board layout, which is designed to minimize the time required for the operator to evaluate system performance under accident conditions and assure safe and reliable operations, will be enhanced by Part A of this modification by providing more control board space needed for adding additional indicators (Reference NSMs ON-22447 and ON-22448). The new indicators added in Part B are qualified, QA-1 Dixon indicators with a scale of 1E-8% to 200% power. A control board seismic review has been completed for these new indicators.

The source range detectors are required to be operational during fuel loading and refueling operations (Technical Specification 3.8.2). Therefore, it is necessary that all fuel movement be completed prior to the replacement of source range detector NI-1.

Although not required to mitigate the consequences of any accident, the source and intermediate range channels provide control rod withdrawal "inhibit" interlock signals to limit the power ascension rate during reactor startup operations (i.e. interlocks can avoid Startup Accident analyzed in Oconee FSAR Section 15.2). Replacement of the two NI detectors and recorders will not affect this RPS "inhibit" function.

The power range channel provides one of the RPS signals necessary to mitigate overpower transients resulting from control rod withdrawal at rated power (i.e., Rod Withdrawal Accident at Rated Power analyzed in Section 15.3 of the Oconee FSAR). Replacement of NI recorders or the source and intermediate range monitors will not affect the RPS "high flux trip" function or the ability of the RPS to maintain reactor thermal power below technical specification safety limits (i.e., 112% of rated core power level).

Appendix R is not impacted for Part A of the modification and an Appendix R review has been initiated for Part B.

Based on the evaluation above, this modification involves no safety concerns of USQs. Changes to FSAR section 7.4.1 and table 7.4 will be required. These changes deal mainly with the discussions of the specifics of the detectors and their operation.

REPORT: NSM ON-22827,AL1,AM1

DESCRIPTION: This modification will replace radiation monitor 2RIA-50. Monitor 2RIA-50 is used to check the Component Cooling (CC) System for leakage from heat exchangers located in radioactive systems that are cooled by the CC System. Oconee is replacing analog monitors with "state-of-the-art" digital models.

SUMMARY: This monitor is not safety related. The purpose of the monitor is to identify the spread of contamination within the CC System, thereby addressing ALARA considerations. The range of detection for the new monitor will be better than the old monitor. The SCADS System to which the monitor will be connected, will continuously process the monitor signal indication. In addition, the SCADA System will control monitor alarms and functions.

The new monitor will be installed on a skid. The skid for the new monitor will be seismically mounted. The new monitor will be located in the same general location as the existing monitor. A flow switch is provided to indicate loss of flow to the monitor. There is no impact to any of the existing support/restraints for piping. Appendix R of 10CFR50 is not affected by this modification. There are no regulatory guide 1.97 requirements for this monitor.

REPORT: NSM ON-22892

DESCRIPTION: This modification provides the operator with the capability to choose between manual and automatic signal selection for Integrated Control System (ICS) circuitry. In particular, the differential pressure transmitter signal instrumentation for the main feedwater valves will be changed out. The auctioneer switches on the control board will be changed out. Two auctioneer modules and two relays will be added to the ICS cabinets. An additional cable will be run between the control board and the ICS cabinet.

This change will provide the operator with an automatic selection of pressure transmitter (two available) input to the ICS circuitry. The instrumentation input to the ICS circuitry is used to control the speed of the feedwater pumps based on the differential pressure generated by flow through the main feedwater valves. In the automatic mode, both transmitters are used for input. The auctioneer module will select the higher of the two transmitter signals. In the event of the failure of the selected transmitter, the other transmitter (higher signal) would be automatically selected. In the manual mode, the operator selects which transmitter is used. The manual mode provides a means for selecting a specific transmitter while performing maintenance on the opposite transmitter.

At present, the operator is required to manually select transmitter signal input. In the event of a transmitter failure, the operator would need to manually select the opposite transmitter before a unit runback or trip occurs. The unit runback or trip would occur due to changes in feedwater flows and pressures.

SUMMARY: The ICS provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator feedwater system, and turbine. The control system provides limiting actions to assure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power.

The ICS does not perform any accident mitigation function. The bulk of this modification is not safety related. Changing selector switches on the control board is the only portion affecting safety related components (i.e. control board). The entire modification will be performed in the control room. The control board switch changes will not affect the seismic qualification of the control board since the switches are the same type, and have the same weight and mounting as the existing switches. The new instrumentation logic requires less operator action, and reduces the probability for a unit transient. The control board selector switch will indicate which mode the instrumentation is placed in. The modification will not affect the control functions of the ICS System. An Appendix R fire review was performed.

This modification involves no USQs or safety concerns. No FSAR or technical specification changes are required. FSAR Section 7.4.2.2.2 states the selected differential pressure signal is indicated. It is recommended some clarification be made in order to prevent confusion.

REPORT: NSM ON-22923,AK1,AL1

DESCRIPTION: Install QA Condition 1 flow instrumentation to monitor Low Pressure Service Water (LPSW) flow to the Low Pressure Injection (LPI) coolers and modify the control circuitry for LPI cooler shell outlet valves 2LPSW-4 and 2LPSW-5 to provide throttling capability. To ensure LPSW flow can be controlled using 2SPSW-4 and 2LPSW-5 during design basis events, switch controlled solenoid valves will be installed to vent air from the valve operators of the LPI cooler flow control valves 2LPSW-251 and 2LPSW-252, ensuring that the valves open. The control switches will be relocated to a place near the new control room flow indicators. Root valves and high point vent valves for the impulse lines will be replaced with qualified valves so that the loops can be upgraded. A protective shield will be installed to prevent damage to the QA-1 flow rate transmitters from seismic induced interactions.

SUMMARY: The new flow instrumentation is proposed to be classified as a Regulatory Guide 1.97 Type A, Category 1 variable. The indicating range of the new instrument loops will be 0-8000 gpm. The indicating range of the current loops is 0-6000 gpm. The change of the range has been reviewed and determined not to adversely affect any existing engineering calculations.

The new flow instruments are installed in an existing instrument line that has been constructed to non-QA Condition Specifications. The existing instrument line is being upgraded to QA Condition 1 (e.g. seismic qualification, material qualification). The line is qualified as QA Condition 1 and the existing root valves and high point vent valves are being replaced with qualified valves. The existing instruments in the instrument line are not qualified to functionally operate, but are qualified for pressure boundary integrity. Thus, the instrument line is qualified as a pressure boundary. Seismic interaction of the upgraded instrument line and new instruments has been reviewed and determined to not be a concern. A shield is to be installed to protect the flow rate instrumentation from being damaged by non-seismically mounted equipment falling as a result of a seismic event. QA condition 1 power is provided to the new instruments from KVIA and KVIB in such a manner that if KVIA or KVIB fails, one indication (QA-1 or non-QA) will be present on each flow line.

Valves 2LPSW-251 and 2LPSW-252 which are safety related currently fail open to a travel-stopped position on loss of air to assure LPSW flow to certain components. The addition of safety related solenoids in the air lines of valves 2LPSW-251 and 2LPSW-252 provides the operator with added assurance that the valves will be open in the event air is lost to the valves. If the solenoid valve inadvertently vents the air supply, 2LPSW-251 or 2LPSW-252 would fail open. This potential currently exists with the existing non-safety controller. If the solenoid sticks closed, 2LPSW-251 or 2LPSW-252 would not fail open. This potential also exists with the current system. The flexible hose between the solenoid and the valve is being replaced with hosing that meets Commercial Grade requirements, to allow use in the safety related application. Switches in the control room allow the operator to bleed the air off valves 2LPSW-251 and 2LPSW-252 to assure they fail open.

Existing non-QA condition indicators will be replaced with QA-1 indicators. The control switches for valves 2LPSW-4 and 2LPSW-5 will be relocated to near the new indicator. This will provide easier positioning of the valves and throttling of flow. The control switches for the solenoid valves for 2LPSW-251 and 2LPSW-252 will also be relocated, providing a centralized location of control for LPSW flow to the LPSW coolers.

The possibility of a valve failure resulting from the operation of these valves in an intermediate position (throttled) during an accident has been investigated. Separation or breakage of the seat or disk of the valve should not occur under these conditions. Thus, neither the valve or the LPSW supply piping to the LPI coolers will become blocked by a valve failure and remove cooling water to the LPI coolers. Maintaining this position over an extended period of time will cause some small deformations and degrade the sliding friction factor which will increase the load on the motor operator. Some minor chipping of the disk and seat surfaces may occur. Thus, this modification will not have an adverse effect on system operation due to the potential for flow blockage or flow restriction. The valve will maintain its integrity in the as-throttled position indefinitely.

The function of the LPSW System has not changed for the evaluated conditions of operability. Either the air operated valves or the modified motor operated valves could be used to throttle flow during accident conditions. Operators will have safety related flow instrumentation available to throttle flow with valves 2LPSW-4 and 2LPSW-5. Throttling flow with 2LPSW-251 and 2LPSW-252 is the current operating mode. Valves 2LPSW-251 and 2LPSW-252 presently have non-safety related controllers and non-safety related air supplies. The controllers for valves 2LPSW-252 rely upon flow indication from non-safety related flow instruments. Equipment is not required to be environmentally qualified since it is located in a mild environment. A control board seismic review for the control board changes and additions has been performed and no adverse consequences are postulated.

Potential single failures are now examined in case throttling is used in the future as new conditions of operability are evaluated. Electrical failures were reviewed to assure that no single failure would result in loss of a pump and flow instrumentation. If one of the new flow instruments failed, an assured method of determining LPSW flow to the LPI coolers would not be available. This failure would be considered the single failure. However, all three LPSW pumps would be available. If all three pumps are available, the LPSW valves do not have to be throttled. Another potential single failure is examined. No single failure can elimninate the potential for the assured method of throttling and the other trains' safety related flow instrument.

REPORT: NSM ON-32827,AL1,AM1

DESCRIPTION: This modification will replace radiation monitors 3RIA-42 and 3RIA-50 with new monitors. 3RIA-42 is used to monitor Recirculating Cooling Water (RCW) for leakage from heat exchangers in contact with radioactive systems. 3RIA-50 monitors Component Cooling (CC) for leakage from heat exhangers in contact with radioactive systems.

Oconee is replacing analog monitors with "state-of-the-art" digital models. Replacement parts and vendor technical support for the analog monitors are becoming increasingly unavailable. The new digital monitors are compatible with the System Control and Data Acquisition (SCADA) System installed by a previous modification. By connecting the digital monitors to the SCADA system, the control room rate meter located on the control board will no longer be needed and will be removed in a later modification.

SUMMARY: These new monitors will be installed on skids. The skids for the new monitor will be seismically mounted. The new monitors will be located in the same general location as its existing counterpart. 3RIA-42 will have a dedicated skid mounted pump to provide sample flow. 3RIA-50 does not have a dedicated pump, but uses system pressure to provide a loss of flow indication. Ther is no impact to any of the existing support/restraints for piping. Appendix R of 10CFR50 is not affected by this modification. There are no Regulatory Guide 1.97 requirements for this monitor.

REPORT: NSM ON-32892

DESCRIPTION: This modification provides the operator with the capability to choose between manual and automatic signal selection for Integrated Control System (ICS) circuitry. In particular, the differential pressure transmitter signal instrumentation for the main feedwater valves will be modified. Control switches on the contro board will be changed out. Two auctioneer modules and two relays will be added to the ICS cabinets. An additional cable will be run between the control board and the ICS cabinet.

This change will provide the operator with an automatic selection of pressure transmitter (two available) input to the ICS circuitry. The instrumentation input to ICS circuitry is used to control the speed of the feedwater pumps based on the differential pressure generated by flow through the main feedwater valves. In the automatic mode, both transmitters are used for input. The auctioneer module will select the higher of the two transmitter signals. In the event of the failure of the selected transmitter, the other transmitter (higher signal) would be automatically selected. In the manual mode, the operator selects which transmitter is used. The manual mode provides a means for selecting a specific transmitter while the opposite transmitter is taken out of service for any reason.

At present, the operator is required to manually select transmitter signal input. In the event of a transmitter failure, the operator would need to manually select the opposite transmitter before a unit runback or trip occurs. The unit runback or trip would occur due to changes in feedwater flows and pressures.

SUMMARY: The ICS provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator feedwater syste, and turbine. The control system provides limiting actions to assure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power.

The ICS does not perform any accident mitigation function. The bulk of this modification is not safety related. Changing selector switches on the control board is the only portion affecting safety related components (i.e. control board). The entire modification will be performed in the control room. The control board switch changes will not affect the seismic qualification of the control board since the switches are of the same type, and have the same weight and mounting as the existing switches. The control board slector switch will indicate which mode the instrumentation is placed in. The modification will not affect the control functions of the ICS System. An Appendix R fire review was performed. No new accidents are created.

The feedwater valve D/P transmitters can reasonably be expected to fail in only three modes. Failing low is the most likely. Failure in the high position is slightly less probable than failing low. Failing in a stable intermediate position is expected to be very remote. If a transmitter fails in the low position, the potential for a trip is reduced since the higher transmitter signal will be selected. If a transmitter failed high, the failed transmitter would be selected and a unit transient would occur. This failure mode currently exists if the transmitter that failed high is the transmitter that was selected. The potential for a transmitter that has failed high to cause a transient is increased somewhat with the modification installed, but this increase is offset by the reduced possibility of a transient with the modification installed and one of the transmitters failing low.

REPORT: NSM ON-52795,DL1

DESCRIPTION: This modification will add and replace fire detection equipment in the Turbine, Reactor, and Auxiliary Buildings. The new detectors will utilize the existing fire detection system cabling. However, new cable will be spliced into the existing cable when an existing detector is relocated. Existing detectors will be relocated to improve fire detection coverage.

The Auxiliary Building fire detection system coverage will also be expanded. This expansion will be accomplished by adding fire detectors to locations currently provided with fire detection coverage and also by locating fire coverage and locating fire detectors in areas not previously provided with fire detection coverage.

All detectors that are part of this modification are to be installed but only curtain detectors are to be activated by the fire protection system software. The fire protection system software has the capability to ignore specific fire detectors. Technical Specification 3.17 list the number of detectors that are provided and the number of detectors required to be operable for certain areas of the plant. This modification is activating the new detectors that are being exhanged one for one for areas specified in the technical specification. The new activated detectors are being installed in the existing detector locations for the areas listed in the technical specifications. Additional detectors that are to be located in the areas listed in Technical Specification 3.17 are installed but are not activated. Detectors in areas not listed in Technical Specification 3.17 are installed and activated. A technical specification change is required before the additional detectors installed in the areas listed in Technical Specification 3.17 can be activated. Fire detection system control panels, located in the Unit 1 and 2 control room and the Unit 3 control room, will be removed and replaced with new control panels. The new control panels are also to be located in the control rooms in the existing fire detection cabinets. The new control panels will offer the ability to address individual fire detectors from the respective control rooms.

A test loop will be added to the Unit 3 Reactor Building. This loop will consist of different types of detectors and is being installed to see which type is best suited for that type of environment. The existing Reactor Building detectors will be left in place to provide the required detection capability during the testing of the new detectors.

SUMMARY: The purpose of the fire detection system is to provide adequate warning capability for prompt detection of fires.

This modification contains portions that are QA Condition 3 and non-QA Condition. The modification changes do not degrade the fire protection system for 10 CFR 50 Appendix R events or for plant equipment protection. Fire detection coverage is being increased in areas where technical specifications do not list the number of detectors and fire detection coverage is not decreased in areas where technical specifications list the number of detectors. Each area in the FSAR and Technical Specification has coverage.

The number of detectors required by the technical specification may need to be changed since the minimum number of detectors currently required, along with an increased number of detectors prvided as part of this modification, may allow situations in which the minimum number of detectors are in more localized areas than currently could exist. Activating only the detectors that are being exchanged one for one in the areas listed in the technical specification does not require a technical specification change. The NRC must review the technical specification change before the additional detectors are activated and used by the plant.

The existing fire protection control panels are QA condition 3 and are being replaced with QA condition 3 control panels. The existing fire detection cabinest are being used with the internals being replaced. The replacement of the control panels have no adverse impact on the cabinet mounting during a seismic event. A CRT and a cabinet for it are being added in the ventilation rooms. They will be mounted non-QA Condition since they will cause no seismic interaction concerns. The detectors are QA Condition 3 and are mounted non-QA Condition since a review determined that any failure of them would not reduce functioning of safety related equipment or cause incapacitating injury to occupants of the control room. The existing detectors are mounted non-QA condition.

The new and existing detectors are ionization type detectors. The FSAR lists deviations from National Fire Protection Association (NFPA) Fire Code 72D. The modification will not add any new deviations. One deviation will be eliminated. This current deviation involves the automatic alarm logging device device that will be provided with the new system. The new fire detection system is wired using the type of cable specified in the FSAR. The system uses the same power supply (battery backed) and will require less load on the power supply. The new system will provide an audible and visual alarm and annunciation in the control room. The new detection system will be able to alert the operators to which detector is alarming instead of a general alarm. The present system uses a fire detector alarm that notifies operators of a general, non-specific fire detector alarm that does not indicate which detector is alarming but only the fire zone is indicated.

REPORT: NSM ON-52827

DESCRIPTION: This modification will replace radiation monitors 1RIA-42 and 1RIA-50. Monitor 1RIA-42 is used to monitor the Recirculating Cooling Water (RCW) System for leakage from heat exchangers located in radioactive systems that are cooled by the RCW System. Monitor 1RIA-50 is used to check the Component Cooling (CC) System for leakage from heat exchangers located in radioactive systems that are cooled by the CC System.

This modification will also remove radiation monitor 0RIA-52. This monitor is used to check the recirculated cooling water system line that returns from the Interim Waste Evaporator heat exchangers. These heat exchangers are located in the IRW Building and are no longer used. The monitor sample lines will be left in place in the event sampling should be required in the future.

Oconee is replacing analog monitors with 'state-of-the-art' digital models. Replacement parts and vendor technical support for the analog monitors are becoming increasingly unavailable. The new digital monitors are compatible with the System Control and Data Acquisition (SCADA) System installed by a previous modification (ON-12422). By connecting the digital monitors to the SCADA system, the control room rate meters located on the control board will no longer be needed. These meters will be removed.

SUMMARY: These monitors are not safety related. The purpose of the monitors is to identify the spread of contamination within plant systems, thereby addressing ALARA considerations. The range of detection for the new monitors will be better than the old monitors. The SCADA System to which the monitors will be connected, will continuously process the monitor signal indications. In addition, the SCADA System will control monitor alarms and functions.

The new monitors will be installed by using a skid for each monitor. The skids for the new monitors will be seismically mounted. The new monitors will be located in the same general location as the existing monitors. A flow switch is provided for each monitor. The flow switch will indicate loss of flow to the monitors. There is no impact to any of the existing piping support/restraints. A seismic review of control board changes was done.

Appendix R of 10CFR50 is not affected by this modification.

There are no regulatory guide 1.97 requirements for these monitors.

REPORT: CP/0/B/3002/02 (Procedure for "Chemical Additions to Secondary Systems for Normal Operating Conditions")

Description: The purpose of this change is to allow addition of the chemical Ethanolamine or approved amine to the Secondary side for pH control.

Summary: Secondary side pH control additives do not play a significant role in the accident and malfunction of equipment scenarios outlined in the FSAR. They also do not create additional scenarios of this type not considered in the FSAR. Their purpose is to minimize the corrosion of metallic surfaces and therefore extend the life and improve the reliability of the components they come in contact with. Therefore, the safety margin as described per the FSAR is not changed due to the use of Ethanolamine or alternate amine for secondary side pH control.

REPORT: SLC 16.6-1

DESCRIPTION: This SLC revision is a new addition to the SLC manual. The list of containment penetrations in Technical Specification 4.4-1 have been transferred to the SLC manual for administrative consistency and will be located in a new section in the SLC manual. The technical content of the information will remain the same and the Technical Specification requirements will not change since the tables were for reference purposes.

SUMMARY: Since this change did not involve a change in technical content, it in no way will affect safety analyses. There is no reduction in the protection of the health and safety of the public.

REPORT: SLC 16.7-4

DESCRIPTION: SLC 16.7.4 "Source Range Neutron Flux Monitoring" requires that one source range nuclear instrument be in service whenever fuel is in the Reactor Vessel and Reactor Power level is below 10e-4 % (on new Gammametric NIs) or below 10e-9 amps (on old intermediate range NIs). This commitment formalizes station operating philosophy.

SUMMARY: This addition to the SLCs formalizes operating principles already in practice at Oconee Nuclear Station and as such does not reduce the level of protection of the health and safety of the public.

REPORT: SLC 16.7-5

DESCRIPTION: This SLC fulfills an NRC commitment to provide OTSG overfill protection operability and surveillance requirements by adding a section with these operability and surveillance requirements.

SUMMARY: This revision adds new surveillance and operability requirements to provide OTSG overfill protection. No safety analyses are degraded as a result of this change and the level of protection of the health and safety of the public is not reduced.

REPORT: SLC 16.8-1

DESCRIPTION: This SLC was originated to ensure that requirements of FSAR section 9.4.1.1 (Design Bases for Control Room Ventilation) are met by adding surveillance requirements and corrective actions for monitoring of control room temperature.

SUMMARY: Since this SLC adds requirements for monitoring control room temperature to ensure that FSAR requirements are more readily met, there is no reduction in the level of protection of the health and safety of the public.

REPORT: SLC 16.9-2

DESCRIPTION: This SLC is being changed to facilitate more efficient use of station manpower without reducing the effectiveness of the fire protection sprinkler and spray systems.

SUMMARY: These changes do not reduce the level of protection of health and safety to the public since coverage of fire protection in the SSF is not reduced by changes within this SLC.

REPORT: SLC 16.9-7

DESCRIPTION: This revision to the SLCs adds the absolute lake level requirements to teh various level limits in order to make that information more readily available to personnel.

SUMMARY: This revision does not affect any safety analyses in affect nor does it add any new limitations regarding safety analyses. This change does not reduce the level of protection of the health and safety of the public.

REPORT: SLC 16.9-7

DESCRIPTION: This SLC revision changes the format of this SLC in order to more readily determine the actions that are required for various lake levels. There is no significant change to the technical content of this SLC.

SUMMARY: Since there is no change in technical content of this SLC, there are no unreviewed safety questions.

REPORT: SLC 16.9-7

DESCRIPTION: Currently there are references and requirements concerning Keowee Lake level in various Specifications, directives, procedures, calculations and other documents. SLC 16.9-7 " Keowee Lake Level" identifies these references and compiles them in one location for utilization by site personnel for a quick guide to determine if level is being maintained within proper ranges to ensure operability of structures, systems, and components.

SUMMARY: This SLC does not change any present requirements which have been previously approved by site management nor does it affect any safety analyses. There is no reduction of the level of protection to the health and safety to the public as a result of this change.

REPORT: SLC 16.9-7

DESCRIPTION: This revision to SLC 16.9.7 (Keowee Lake Level) changes the required lake level for LPSW pump operability since a revision to a Duke Power Calculation has proven that adequate NPSH would be maintained to the LPSW pumps at this new required lake level.

SUMMARY: A Duke Power Calculation has shown that a lower minimum lake level is required for maintaining required NPSH to the LPSW pumps. This calculation was determined to contain overly conservative assumptions about LPSW pump strainer d/p. This revision does not change the assumptions in any previous safety analyses and does not reduce the level of protection to the health and safety of the public.

REPORT: SLC 16.11-1

DESCRIPTION: This change to the SLC manual is to revise the radiactive liquid effluent release rate limit in order to update the SLC to conform with revisions to 10 CFR 20.

SUMMARY: This change is in compliance with the applicable sections of 10 CFR 20 and 10 CFR 50, which delineate the basic requirements for Selected Licensee Commitments (SLCs) concerning effluent from nuclear power reactors. Compliance with effluent SLCs will ensure that average annual releases of radioactive material in effluents will be small percentages of the 10 CFR 20/50 requirements while continuing to provide operational flexibility. Remaining within these limits ensures the health and safety of the public is maintained.

REPORT: SLC 16.11-2

DESCRIPTION: This proposed change concerns the setting of radiation monitor setpoints for monitors which monitor gaseous effluent as the effluent is exiting the Unit Vents.

SUMMARY: The basic requirements for SLCs concerning effluents from nuclear power reactors are stated in 10 CFR 50.36a. These requirements indicate that compliance with effluent SLCs will keep average annual releases of radioactive material in effluents to small percentages of the limits specified in the old 10 CFR 20.106 (new 10 CFR 20.1301). These requirements further indicate that operational flexibility is allowed, compatible with considerations of health and safety, which may temporarily result in releases higher than such small percentages, but still within the limits specified in the old 10 CFR 20.106 which references Appendix B, Table II concentrations (MPCs). These referenced concentrations are specific values which relate to an annual dose of 500 mrem. It is further indicated in 10 CFR 50.36a that when using operational flexibility, best efforts shall be exerted to keep levels of radioactive materials in effluents as low as is reasonably achievable as set forth in 10 CFR 50, Appendix I. As stated in the Introduction to Appendix B of the new 10 CFR 20, the gaseous effluent concentration (EC) limits given in Appendix B, Table 2, Column 1, are based on an annual dose of 50 mrem for isotopes for which inhalation or ingestion is limiting or 100 mrem for isotopes for which submersion (noble gases) is limiting. Since release concentrations corresponding to limiting dose ratesless than or equal to 500 mream/year to the whole body, 3000 mrem/year to the skin from noble gases, and 1500 mrem/year to any organ from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than eight days at the site boundary has been acceptable as a SLC limit for gaseous effluents to assure that the limits of 10 CFR 50 Appendix I and 40 CFR 190 are not likely to be exceeded, it should not be necessary to restrict the operational flexibility by incorporating the dose rate associated with the EC value for isotopes based on inhalation/ingestion (50 mrem/year) or the dose rate associated with the EC value for isotopes based on submersion (100 mrem/year).

Operational history at Oconee has demonstrated that the use of the concentration values associated with the old 10 CFR 20.106 as SLC limits has resulted in calculated maximum individual doses to a member of the public that are small percentages of the limits of 10 CFR 50 Appendix I. Therefore, the use of concentration values which correspond to an annual dose of 500 mrem/year to the whole body (due to noble gas), 3000 mrem/year to the skin (due to noble gas), and 1500 mrem/year to any organ (due to Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than eight days) should not have a negative impact on the ability to continue to operate within the limits of 10 CFR 50, Appendix I and 40 CFR 190.

REPORT: SLC 16.11-3

DESCRIPTION: This Selected Licensee Commitment revision was completed to incorporate programmatic controls for radiological effluents and radiological monitoring in the Technical Specifications consistent with 10 CFR 20/50. Procedural details of the Radiological Effluent Technical Specifications were transferred from the Technical Specifications to Chapter 16 (SLCs) of the FSAR.

SUMMARY: This change does not result in any technical changes which would result in reduction of the level of health and safety to the public, as it is only editorial and relocational in nature.

REPORT: SLC 16.11-3

DESCRIPTION: This proposed change concerns the setting of radiation monitor setpoints for monitors which monitor gaseous effluent as the effluent is exiting the Unit Vents.

SUMMARY: The proposed change does not increase the probability of an accident previously evaluated in the FSAR. This proposed change will affect radiation monitors which assist in accounting for activity released via the Unit Vents. The subject proposal will help to ensure that dose limits to the public are not exceeded. The proposed change concerns an established monitoring system and associated setpoints. The change does not create any potential for the possibility of an accident which is different than any already evaluated in the FSAR. This proposed change does not increase the probability of a malfunction of equipment important to safety and previously evaluated in the FSAR. The change will continue to allow monitoring of normal as well as accident-related releases of radioactivity via the Unit Vent, and does not adversely affect other equipment. This proposed change does not increase the consequences of an equipment malfunction. With the use of the setpoints prescribed by this SLC change, dose to the public will continue to be maintained below regulatory limits. The possibility of malfunctions of equipment important to safety different than any evaluated in the FSAR will not be created. The proposed change affects only established radiation monitoring systems and not other systems.

REPORT: SLC 16.11-3,6,9

DESCRIPTION: This change revises the reporting frequency for selected Radiological Effluent Reports as an action that eliminates duplicate or inconsistent regulatory requirements without reducing the protection of the public health and safety.

SUMMARY: This change corrects inconsistencies in reporting requirements as identified by the NRC and does not reduce the protection of the public health and safety.

REPORT: SLC 16.11-4

DESCRIPTION: This revision relocates 2 surveillance requirements from table 16.11-2 to 16.11-4. This is only an editorial change.

SUMMARY: This is only an editorial change and does not reduce the level of protection of the public health and safety.

REPORT: SLC 16.11-7

DESCRIPTION: This proposed change has no effect on the technical content, but only an editorial change to update the applicable new 10 CFR 20 section references.

SUMMARY: This proposed change is an editorial change only. It has no effect on the technical content.

REPORT: TM 1076

DESCRIPTION: Modify control circuitry for Low Pressure Injection (LPI) cooler shell outlet valves 1LPSW-4 and 1LPSW-5 to provide throttling capability during an accident scenario.

SUMMARY: The LPSW System provides cooling water for normal and emergency services throughout the station. Two of the safety related functions served by this system are heat removal from the RBCUs and from the LPI coolers.

Mortor operated valves 1LPSW-4 and 1LPSW-5 are being modified to add throttling capability. LPSW floww is currently throttled with air operated valves. These air operated valves presently have non-safety related controllers and non-safety related air supplies. The controllers for these valves rely upon flow indication from non-safety related flow instruments. Operators will need to rely on existing non-safety related LPSW flow instrumentation to regulate LPSW flows while throttling with valves 1LPSW-4 and 1LPSW-5.

The possibility of a valve failure resulting from the operation of these valves in an intermediate position (throttled) during an accident has been investigated. Separation of breakage of the seat or disk of the valve is not expected to occur under these conditions. Thus, neither the valve or the LPSW supply piping to the LPI coolers will become blocked by a valve failure and isolate cooling water to the LPI coolers. Maintaining this position over an extended period of time will cause some small increase the load on the motor operator. Some minor chipping of the disk and seat surfaces may occur. Thus, this modification will not have an adverse effect on system operation with respect to the potential for flow blockage or flow restriction. The valve will indefinitely maintain its integrity in the throttled position. Throttling with these valves during non-accident conditions and at flow rates below 3000 gpm has not been evaluated.

The power and control for valves 1LPSW-4 and 1LPSW-5 are safety related. The control circuitry changes for these valves is QA Condition 1. The indicating circuitry for these valves is unchanged and will indicate full open, full closed, or intermediate positions. The position indication does not indicate the degree of travel between full open and full closed.

The function of the LPSW System has not changed for the evaluated conditions of operability. Either the air operated valves or the modified motor operated valves could be used to throttle flow during accident conditions and both would rely on the use of non-safety related instrumentation to adjust the flow.

REPORT: CALC 5363

DESCRIPTION: FSAR Section 9.1.3 is being updated to reflect the current Spent Fuel Cooling systems' designs and design bases. The current version of the FSAR contains incomplete and outdated descriptions of the design bases. The current design bases have been previously reviewed and approved by the NRC. No modification is being made to the systems' design, design bases, or mode of operation.

SUMMARY: The primary function of Spent Fuel Pool Cooling System for Units 1 and 2 is provide decay heat removal for the spent fuel stored in the Units 1 and 2 spent fuel pool. The current cooling system design requirements are the composite of two separate sets of criteria:

a) the rack licensing criteria prior to the 1980 re-racking, and
b) the additional criteria imposed by the 1980 re-racking modification, pursuant to Amendments 90, 90, and 87 for License Nos. DPR-38 and DPR-55.

Other system functions are to maintain the pool inventory, clarity and chemistry at acceptable levels.

Spent fuel pool heat removal is accomplished by recirculating spent duel coolant water through heat exchangers and then back to the pool. The spent fuel pumps take suction from the spent fuel pool and transport the flow through the coolers, which are arranged in parallel. The waste heat is removed from the shell side of the coolers by the Recirculated Cooling Water System. The cooled spent fuel pool water is then directed back to the spent fuel pool.

The spent fuel pool water temperature is a direct function of the decay heat load produced by the fuel in the racks, in conjunction with the heat removal capability of the spent fuel cooling syste. At the time that the Units 1 and 2 spent fuel pool was re-racked, its spent fuel cooling system was upgraded to handle the higher total heat load expected from the increased number of stored fuel assemblies. The heat removal capability of the upgraded spent fuel cooling system has been sized to meet the new design limits.

The Unit 3 Spent Fuel Pool Cooling System duplicates the equipment used for the Units 1 and 2 system. The Unit 3 system is designed to remove the decay heat from the stored fuel in the Unit 3 spent fuel pool. The cooling system design requirements are the composite of two separate sets of criteria:

a) the original rack licensing criteria, and
b) NRC Standard Review Plan Section SRP-9.1.3

Other system functions are to maintain the pool inventory, clarity and chemistry at acceptable levels.

At the time that the Unit 3 spent fuel pool was re-racked, its spent fuel cooling system was upgraded to handle the higher total heat load expected from the increased number of stored fuel assemblies. The heat removal capability of the upgraded spent fuel cooling system has been sized to meet the new design limits.

This modification involves no unreviewed safety questions or safety concerns.

REPORT: PIP 0-092-0535

DESCRIPTION: This PIR documents a degraded configuration for the plant involving the Low Pressure Service Water (LPSW) System pumps for certain lake levels. These pumps supply cooling water to safety related equipment. The area of concern involves maintaining the Net Positive Suction Head (NPSH) requirements of the LPSW pumps while simultaneously providing adequate flow for accident mitigation equipment and decay heat removal. Operability evaluations discuss the available NPSH for the LPSW System pumps and the ability of the pumps to meet the new flow requirements. The design bases NPSH requirements for the LPSW pumps is the area of concern.

The design basis accident scenario of concern involves a loss of coolant accident (LOCA) occurring in conjunction with a loss of offsite power (LOOP) event. The LOOP condition would cause a loss of instrument air and loss of power to operating Condenser Circulating Water (CCW) System pumps. Loss of instrument air implies pneumatically controlled LPSW flow control valves move to their "full open to travel stop" position, increasing flow demands on the LPSW System. The CCW pumps are located upstream of the LPSW pumps and therefore contribute to NPSH for the LPSW pumps. Loss of these pumps implies a reduction in available NPSH. The elevation of Lake Keowee also has an impact on the available NPSH for the pumps.

The LPSW pumps supply cooling water wo the Reactor Building Cooling Units (RBCU) and the Low Pressure Injection (LPI) coolers, as well as some other equipment that is not safety related. The initiation of this accident scenario would place more demanding operating conditions on operating LPSW pumps than normal operating conditions.

A calculation of LPSW System flow rates was performed for the alignments expected during the LOCA/LOOP accident, using information from LPSW System flow tests. Conclusions of this calculation resulted in several operator actions and operational restrictions that needed to be met for the LPSW System to be operable. Operability evaluations were performed which gave several conditions of operability. One of the conditions was that lake elevation could not drop below 795 feet. A new set of conditions of operability have now been determined for a lower lake elevation limit. In general, these conditions consist of 1) isolating the main turbine oil tank (MTOT) during certain conditions, 2) maintaining the lake elevation at or above 785 feet, and 3) for Units 1 and 2 only, throttling LPSW flow to all LPI coolers within the first 20 minutes during certain conditions.

SUMMARY: The postulated problem for the LPSW System involves ensuring the NPSH requirements of the LPSW pumps, and the flow requirements relative to design bases accident scenarios are able to be met. Adequate LPSW flow is available for mitigating design basis accidents provided that the specified conditions of operability are met.

The required lake level of 785 feet is within the limits specified in Technical Specification 3.7.1 (i) of 775. There are also commitments for operability requirements pertaining to lake level specified in the Selected Licensee Commitments (SLIC) Section 16.9.7.

If an MTOT oil cooler flow control valve is being bypassed or in manual at the time a LOCA occurs, additional demand for LPSW System flow will exist. This additional flow diversion requires consideration of additional operator action in order to ensure the required NPSH is capable of being supplied if specific scenarios exist. Isolation of the MTOT oil cooler bypass is required within 30 minutes if simultaneous loss of offsite power and loss of instrument air and a loss of one of the LPSW pumps occurs. The main turbine would have tripped after an accident and this flow would not be needed. The 30 minute action time is within NRC guidance for manual action in the plant.

The Units 1 and 2 conditions of operability require an additional action to be performed under certain conditions. If two LPI coolers are in operation on a shutdown unit at the same time a LOCA occurs on the operating unit, and a simultaneous loss of offsite power with loss of instrument air and loss of one of three LPSW pumps occurs, LPSW flow to all LPI coolers is to be throtteled within the first 20 minutes following an accident. Valves LPSW-251 and LPSW-252 are air operated valves that will normally be used for throttling this flow. These valves are air operated and have non-safety related controllers and non-safety related air. They fail open to their travel stop position on loss of air. Valves LPSW-4 and LPSW-5 are safety related gate valves that have been modified to have throttling capability. The power and control for these valves is safety related. The LPSW flow instrumentation used to adjust flow is QA Condition 1 for Unit 2. The flow instrumentation for Units 1 and 3 has been reviewed by the NRC and determined to be acceptable for use until QA instrumentation has been installed.

Valves LPSW-251 and LPSW-252, if available, or LPSW-4 and LPSW-5, if the air operated valves are not available, do not need to be throttled until 20 minutes have elapsed from accident initiation. The LPSW pumps can withstand short term operation (20 to 30 minutes) with inadequate NPSH. The NRC allows credit for some complex operator actions (such as throttling flow from the control room) no earlier than 20 minutes after the accident has occurred. The 20 minute time period for performing the operator action from the control room for this scenario is within the NRC guidance times that the NRC allows for complex control room operator actions.

If valves LPSW-251 and LPSW-252 can not be used for throttling, they should fail open to their travel stop position on loss of air. The operability evaluations do not make any changes to the operational requirements of valves LPSW-251 and LPSW-252.

Since LPSW-4 and LPSW-5 are gate valves that may be used as throttle valves, the possibility of a valve failure resulting from the operation of these valves in an intermediate position (throttled) during an accident was investigated. Separation or breakage of the seat or disk of the vlave should not occur under these conditions. Thus, neither the valve nor the LPSW supply piping to the LPI coolers will become blocked by a vlave failure and prevent cooling water from being provided to the LPI coolers.

Maintaining this position over an extended period of time will cause some small deformations and degrade the sliding friction factor which will increase the load on the motor operator. Some minor chipping of the disk and seat surfaces may occur. The damage to the valve if throttled at one throttled position for extended periods of time (several days) would not prevent the valve from being able to be further opened from its long term throttled position. Damage could possibility prevent the valve from being able to be further closed from its long term throttled position if the valve had been kept in one throttle position for several days. The valve will maintain its integrity in the as-throttled position indefinitely.

REPORT: PIP 0-093-0637

DESCRIPTION: The concern raised by PIP 0-093-0637 is that the letdown line orifice is not large enough to pass the amount of flow required in certain situations. To ensure the pressurizer does not become water-solid (completely filled with liquid water), additional conditions and/or compensatory actions must be added to station procedures. The conditions of operability proposed as the resolution of will establish the following conditions: Provide an alternative means of RCS inventory letdown before the pressurizer becomes water-solid by use of the Reactor Vessel head vent. This will be done by using the head vent to intermittently vent RCS coolant to discharge side of RBCU's. This requires pulling cables and equipment installation in order to provide power and control for the reactor vessel head vent valves, from the SSF.

SUMMARY: The above compensatory actions ensure that additional RCS letdown capability will be added following an SSF event before the time that the pressurizer becomes water-solid. The onsite manpower requirements pertains to getting the SSF operational within the allotted 10 minutes after the initiation of an SSF event. Since pulling cables (in order to power the head vent valves) is a compensatory action, and it is not required to be done within the first few minutes, the manpower necessary to accomplish this task caan be on callout. The power supply which will be used is the same which is designed such is adequate and compatible with the head vent solenoid valves.

Two SSF emergency procedures will be revised before the SSF is declared technically operable. Emergency response procedure RP/0/B/1000/22 will instruct the Emergency Coordinator to direct I&E personnel to immediately start connecting cables to provide SSF power to the head vent soldenoid valves (for Units 2 & 3) while the fire is being extinguished. Also, Operation's Abnormal Procedure AP/0/A/1700/25 will instruct SSF Operation's personnel how to actually perform the RCS letdown using the head vent line, thus preventing the pressurizer from going solid. This Abnormal Procedure will also direct the SSF personnel to request from the Operational Support Center (OSC) that the head vent valves on Unit 1 be powered from the SSF if it is determined that it is needed. These procedure changes will take into consideration the ability and timing to perform these additional manual actions, training of personnel, occupational hazards that might be incurred (such as radiation, temperature, chemical, sound or visibility hazards) and available personnel resources. After installation, the newly-powered valves will be able to letdown enough RCS inventory to compensate for SSF seal injection and prevent overfilling the RCS.

Since the letdown is vented to the Reactor Building, it was considered that the vented RCS inventory could heat of humidify the building beyond the environmental qualification limits or necessary equipment. It was determined that with the expected letdown rates through the head vent line that the added heat and humidity to containment does not exceed the environmental qualification limits of necessary equipment.

The vent line has two soleniod valves to which SSF power will be connected. These two valves are in series, ensuring against single failure of not being able to re-close. Since the SSF is not designed to meet the single failure criterion, the failure mode of not being able to open able to open one of the vent valves is not applicable. However, since the valves are in series, a failure of one of the vent valves being stuck open will not cause a SBLOCA. The head vent valves will be able to perform the necessary function required of them by this SSF scenario. The vent line meets all the requirements of NUREG 0737 Item II.B.1, NUREG-0800 Section 5.4.12 and 10 CFR 50.44. The vent line is also QA-1 line and designed to provide post-accident venting of non-condensible gases from the RCS in order to prevent the defeating of natural circulation.

The power cables for the vent valves will be run after the postulated Appendix R fire, therefore, spurious operation of the vent valves is not a concern. The controlling of the RCS letdown through the head vent will depend soley upon the pressurizer level indication in the SSF. Since the RCS will be kept at or near, system pressure, the occurance of voids in the reactor vessel head is not expected. If there were localized voiding would occur only if the RCS pressure was significantly reduced. Pressure fluctuations will be small, so there is no adverse impact on the SSF RC Makeup System pulsation damperners is postulated.

REPORT: PIP 0-093-0738

DESCRIPTION: The Keowee Main Transformer Mulsifyre System was tested to determine its operability. This system requires a minimum supply pressure of 54 psig to be fully operable. Based on the results of the test, a lake leve of 795.5 feet is now required to meet the minimum supply pressure of 54 psig. This calculation evaluates the conditionally operable recommendation for requiring a certain lake level to assure the Keowee Main Transformer Mulsifyre System remains operable.

SUMMARY: The emergency power source for Oconee is the Keowee Hydro Station. The main transformer at Keowee has an automatic spray system to confine and extinguish fires. Operability requirements for this Mulsifyre system are addressed in the Selected Licensee Commitments (SLC) Section 16.9.2. Current lake level requirements for the Mulsifyre system are in SLC Section 16.9.7(3a) and are listed as greater than or equal to 787.9 feet.

The lake water is the supply for the Mulsifyre system. Based on the results of a recent performance test for the Keowee fire protection system, a higher lake level than is currently required is now needed to maintain the Mulsifyre system operability. This change does not create any conditions or events which lead to accidents previously evaluated in the SAR. No new equipment is added. The potential for increased fire damage to the Keowee main transformer, due to the Mulsifyre system's failure, is not increased since a higher minimum lake level is administratively controlled to maintain fire protection and mitigation equipment operability.