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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -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, 1992 and December 31, 1992. This report is submitted pursuant to the requirement of 10 CFR 50.59(b)(2).

Very truly yours,

J.W. Hampton

J.W. Hampton

MTM/mtm

Attachment

xc: W/Attachment
S.D. Ebneter
Regional Administrator, Region II

L.A. Wiens, ONRR

NRC Resident Inspector
Oconee Nuclear Station

IE47

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Exempt Change #OE-3841, (Unit 3)
OE-4970, (Unit 3)

DESCRIPTION:

This modification lowered 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.

SAFETY EVALUATION:

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. At about 3:00 am on May 8, 1992, Unit 1 tripped from 15 percent power due to an up spike in condensate flow and down spike in feedwater pump suction and discharge pressures. The Main Feedwater Pump did not rip, but the reactor experienced an anticipatory 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 transfer and the trip setpoint may have been removed.

Exempt change #OE-4855 removed the tenth stage of the 3D1 Heater Drain Pump; and exempt change #OE-4802 removed the tenth stage of the 3D2 Heater Drain Pump. Modifications were performed to destage 3D1 and 3D2 Heater Drain Pump 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 3D2 Heater Drain Pump yielded an actual shutoff head pressure of 662 psid. With instrument and switch inaccuracies included, over 36 psig of margin exists between the new setpoint (770 psig) and the maximum destaged 3D2 Heater Drain Pump discharge pressures. The maximum E Heater Drain Pump discharge pressures are less than the destaged 3D2 Heater Drain Pump maximum discharge pressure and, therefore, are not limiting.

The setpoint implemented by this exempt change is based upon as measured discharge dead head tests for the pumps installed in the system. The destaged 3D1 Heater Drain Pump dead head test pressure measured was 696 psid. This value is less than the revised Test Acceptance Criteria and, with worst case switch inaccuracies and pump suction pressure conditions, provides a margin of 2 psig based upon actual pump performance during the dead head test.

With the maximum discharge pressure from either the D or E Heater Drain Pumps less than the actuation setpoint, the AMSAC and EFW systems should perform their intended functions when needed. Also, the lowering of the actuation setpoint should provide more margin between it and normal operational parameters of the feedwater system and between it and equipment transfers. Thus, more margin to spurious actuation of these systems in response to plant transients should 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. The Main Steam Line Break analysis was not significantly affected when the setpoint was raised and therefore was not changed. Thus, the analysis remains valid for restoring the AMSAC/EFW setpoint to near its previous value.

The modification will result in the AMSAC and EFW systems initiating when required and providing their designed functions since the Heater Drain Pumps on line will not be able to provide a pressure in excess of the new setpoint due to their recent destaging. The automatic control functions which monitor feed pump discharge pressure are not affected. The possibility of spurious initiations of AMSAC or EFW will decrease with more margin between the normal plant operational condition and the actuation setpoint. The automatic control functions which monitor feed pump discharge pressure are not affected. AMSAC will operate as designed to insure mitigation of a Loss of Main Feedwater. No new failure modes are postulated. The setpoint changes do not relate to safety margins defined in any technical specification. This modification involves no safety concerns of unreviewed safety questions.

Exempt Change #OE-4360, (Unit 1)
OE-4361, (Unit 2)
OE-4362, (Unit 3)

DESCRIPTION:

Reduce the Integrated Control System (ICS) Asymmetric Rod run back rate from 30% full power (FP) per minute to 5% FP per minute. The asymmetric rod run back rate change involves 1) tuning the ICS rate voltage divider (basically a rheostat/resistance circuit adjustment), and 2) configuring/rewiring ICS logic circuitry to change limiting run back rate priority for the Asymmetric Rod, Loss of Feedwater Pump and Loss of Reactor Coolant Pump conditions.

SAFETY EVALUATION:

An analysis that demonstrates the acceptability of changing the ICS asymmetric rod run back rate from 30% FP per minute to 5% FP per minute has been performed. The conclusion of the analysis is that changing the run back rate to 5% FP per minute will only impact the degree of additional protection provided by ICS under asymmetric rod conditions. This is because ICS actions taken upon detection of an asymmetric rod condition are not required to prevent plant protection criteria from being exceeded. The evaluations described in Section 15.7 of the Oconee FSAR continue to bound the plant response to an asymmetric rod run back event assuming a 5% FP per minute run back rate.

The electrical modifications required to change the asymmetric rod run back rate involves tuning the ICS rate voltage divider and reconfiguring/rewiring the ICS logic circuitry to change the limiting run back rate priority for the Asymmetric Rod, Loss of Feedwater Pump and Loss of Reactor Coolant Pump conditions. The change only affects the rate of SG power demand and reactor power reduction upon ICS detection of an asymmetric rod condition and does not affect how the ICS accomplishes these run back functions (i.e., the same secondary side valve operators are actuated, the same control rod groups are inserted, etc.) An Appendix R review has been performed and there is no seismic impact on the electrical panels involved. Changing the asymmetric rod run back rate does not degrade the ability of control rods to drop during a reactor trip and the "No ICS Protective Action" rod misalignment accident case evaluated in FSAR continues to bound the ICS run back case (assuming the new 5% FP per minute rate). The same failure mode exists and is evaluated with or without the run back rate change. The method used by the ICS to accomplish its function is not changed. The priority of run back rate requirements are properly changed to assure loss of Reactor

#OE-4360 (Continued)
OE-4361 (Continued)
OE-4362 (Continued)

Coolant Pump or Feedwater Pump indications induce the specified run back rates for those conditions. This modification does not adversely affect any plant safety limits, setpoints, or design parameters. This change involves no safety concerns or unreviewed safety questions.

Exempt Change #OE-4907, (Unit 3)

DESCRIPTION:

This change installed 16 new R4C Position Indication Tubes and Amplifier Cards. These replaced the ones which were installed on groups 6 and 7. The R4C Position Indication Tubes area a definite upgrade from what is currently being used. They provide dual indication for redundancy where the current tube only has single indication. The changeout of the tubes and amplifier cards are a one for one replacement. Nothing has to be modified for the installation.

SAFETY EVALUATION:

Evaluation of the new R4C tubes reveals that it is an improvement over the existing tubes with better accuracy and redundancy. The accuracy of the existing position indication tube is 1.48 inches. However, the accuracy of the R4C Position Indication Tubes is 1 inch with both channels in operation. This improved accuracy however is only attained when both channels are in operation. When one of the channels of the new R4C is eliminated the accuracy drops from 1 inch to two inches which is not as good as what is currently installed. The point here is however that although it drops slightly you still have indication where if you have a failure with the present tube you do not have any indication. The only place that this drop in accuracy will be realized is on the computer point printout in the control room. The PI meters which are used by operators will not be affected due to their accuracy. Therefore, the R4C Position Indication Tube is a definite improvement over the existing configuration.

System operation will not be degraded as a result of this change. In fact it should be improved with redundant and more accurate equipment. No unreviewed safety questions are involved as a part of this exempt change.

Nuclear Station Modification #ON-12245, (Unit 2)

DESCRIPTION:

The controls pertaining to MS-87 and MS-93 will be modified to prevent pressure build-up in the steam line to the Turbine-Driven Emergency Feedwater Pump (TDEFWP) turbine. Additionally, safety valve MS-140 will be deleted and the turbine discharge lines altered to ensure a permanently open flow path.

SAFETY EVALUATION:

The piping and valve changes associated with this modification are QA Condition 1. The new valve controller is non-QA Condition. The existing pneumatic controller is also non-QA Condition. Valve MS-140 is presently in the line that provides the assured safety related TDEFWP exhaust path for the steam. The modification will replace this assured exhaust path with a safety-related path with no valves in the line. Presently exhaust steam goes through the line containing valve MS-96 since this valve is kept open. The line with valve MS-96 is not safety related. A new drain line is to be located on the pump discharge path to allow condensate and rain water to be removed from the line. The drain is located in the Turbine Building to prevent freezing and blockage of the drain's contents.

The replacement valve for MS-92 will have the same pressure setpoint and will be class F (safety related). The control system changes for valves MS-87, MS-93, MS-94 will reduce the probability of pressure surges which challenge relief valve MS-92. The controls for valve MS-87 are presently backed up with nitrogen bottles. The new control system will be backed up by a UPS. There is a 2 hour supply of nitrogen to valve MS-87 in the event of loss of air or power. The UPS will supply DC power to valve MS-87, independent of any AC power source, for a minimum of 2 hours. Valves MS-93 and MS-94 input a signal to the valve MS-87 controller but do not receive a signal from it. The nitrogen bottle will still be used on valve MS-87 for changing the valve position based on the signal it received from the DC power backed control system. Valves MS-87 and MS-93 still fail open on loss of air or power. Stress analysis has been performed on the new pipe routing and changes for Units, 1, 2, an 3. If a steam generator tube rupture accident occurs, the affected steam line will be isolated at valves MS-82 (header A) or MS-84 (header B). Thus, the new exhaust line to the atmosphere, which contains no valves, will not constitute a new radiological release path. The replacement of the control systems from pneumatic to electronic does not degrade the operation of control valve MS-87 during a loss of all AC power. The UPS provides at least 2 hours of power

independent of any AC power source. No turbine back pressure changes will occur since in the present and proposed new state, the piping is open to the atmosphere. The proposed new discharge path is a safety related class F line which assures a method of steam relief from the turbine. The replacement of relief valve MS-92 is with a valve of the same class and it opens and closes at the same setpoints. This modification does not degrade the operation of the TDEFW pump. The TDEFWP still performs its intended design function and the pump is able to mitigate loss of main feedwater events. There are no unreviewed safety questions associated with this modification.

Nuclear Station Modification #ON-12587/0, (Unit 1)
#ON-22587/0, (Unit 2)

DESCRIPTION:

This modification replaces the non-qualified pneumatic channels of Low Pressure Injection (LPI) flow instrumentation with two qualified QA Condition 1 channels of instrumentation. Each channel will be powered by safety power sources. In the control room each channel will be displayed by a qualified indicator and continuously recorded by a qualified recorder. The LPI flow instrumentation is being upgraded to comply with Regulatory Guide 1.97, Revision 2. Two non-qualified transmitters, one per train, will provide analog points to the Operator Aid Computer (OAC).

The existing instrument lines routed from the process tap connections to the transmitters for each channel were reviewed to ensure that the instrument lines are acceptable for QA Condition 1 application. The pneumatic instrumentation associated with the existing flow elements will be deleted. Two qualified differential pressure transmitters are added to the instrument loops. The new transmitters provide inputs to the Inadequate Core Cooling Monitoring System (ICCM) cabinets. The ICCM cabinets provide safety outputs to qualified indicators and recorders located in the control room. The indicators are to be replaced with qualified indicators. The indicators and recorders are labeled to indicate that they are required for post accident monitoring. The ICCM cabinets also provide non-safety outputs for OAC and annunciator points. Alarms generated in the ICCM cabinets provide high and low LPI flow and low Decay Heat Removal flow for each train.

SAFETY EVALUATION:

The LPI System removes the decay heat from the core and sensible heat from the Reactor Coolant System during the later stages of a cooldown. The system also maintains the reactor coolant temperature during refueling, and provides a means for filling and draining the fuel transfer canal. In the event of a loss of coolant accident (LOCA), the system injects borated water into the reactor vessel for long term emergency cooling. The LPI and decay heat removal flow monitoring instrumentation are addressed under Regulatory Guide 1.97. Regulatory Guide 1.97 describes a method acceptable to the NRC staff for complying with their regulations to provide instrumentation to monitor plant variables and systems during and following an accident in light-water-cooled nuclear power plant.

Regulatory Guide 1.97 classifies the LPI and decay heat removal flow monitoring instrumentation as Type D variables and Category 2, with a specified range of 0 to 110% design flow. These variables, as Type D, are used to monitor operation.This instrumentation was also selected as a Type A variable and Category 1, and given a range of 0 to 110% design flow. A Type A variable provides primary information needed to permit the control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. A Type D variable provides information to indicate the operation of individual safety system and other systems important to safety. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. The design criteria for Category 1 (Type A and Type D variables) includes environmental qualification, seismic qualification, design against single failure, at least one channel displayed on a direct indicating or recording device, instrumentation energized from station standby power sources, continuous indication display, recording of instrumentation readout information, separation between safety related and non-safety related equipment using isolation devices and instruments identified on control panels so that the operator knows that they are for use under accident conditions. Instrumentation being upgraded for accident monitoring is not to degrade the accuracy and sensitivity required for normal operation.

The existing flow elements and instrument lines leading to the existing transmitters will be used for the new transmitters. For the Regulatory Guide 1.97 upgrades at Oconee, the lack of written criteria did not provide enough historical information to verify that the instrument impulse lines were installed properly. However, these instrument lines and other Regulatory Guide 1.97 instrument line applications were reviewed in Design Study ONDS-0217/00. The instrument lines for Regulatory Guide 1.97 instrumentation were determined to be satisfactory for seismic loadings and safe from non-seismic interactions. The instrument line valves and materials utilized on the applications were also found to be acceptable from a material compatibility standpoint (piping, tubing, and fittings) and from an ASME code standpoint (valves).

The current LPI flow instruments consist of two transmitters for each train. Each train has one pneumatic transmitter and one electronic transmitter connected to a common line that gets input from a flow element. The existing pneumatic transmitter goes to the flow gage on the main control board. The existing electronic transmitter goes to the non-safety related OAC in the control room. One safety related electronic transmitter replaces the current pneumatic transmitters for each train. The current non-qualified electronic transmitter is being replaced with a new non-qualified electronic transmitter. The new non-qualified transmitter will not provide a signal to the control boards or be mounted on the control boards. The failure mode of the current transmitters is low flow on loss of air or power. The failure mode of the new transmitter is low flow on loss of power (the new transmitter does not use air). Failure of one loop will not adversely affect the control room reading of the other loop's instrumentation. The current transmitters use non-safety related air and power. The new safety related transmitter use safety related power.

The new transmitters that are used to meet Regulatory Guide 1.97 commitments are QA Condition 1 and will receive safety related power from the ICCM cabinets. These new transmitters are environmentally qualified, seismically qualified, and are to be seismically mounted. The new indicators and recorders are seismically qualified, seismically mounted, QA Condition 1 and are to receive safety related power. The new indicators and recorders are located in a mild environment and require no special environmental qualification. The recorders are already installed. The new QA Condition 1 components were reviewed for the potential of seismic interaction from non-safety related equipment and were found to have no interaction concerns. A control board seismic review has been performed for the control board changes. The OAC and annunciators already exist. The instrumentation exposed to system pressure is designed to appropriate design conditions. The range for the new instruments covers the Regulatory Guide 1.97 listed range of 0 to 110% design flow. The design flow is 3000 gpm and the instrument range is 0 to 6000 gpm. The current instrumentation range is 0 to 6000 gpm.

The new non-qualified transmitters are non-QA Condition (non-safety related) for operational function. The instrumentation loop, including the non-qualified transmitters, will maintain the pressure boundary integrity during design basis events. Thus the non-qualified transmitter is not postulated to add a new failure mode that would allow erroneous readings to occur from the safety related instrument. The new non-qualified transmitters are mounted seismically to prevent seismic interaction.

Each new channel of safety related instrumentation is to be displayed by a qualified indicator and continuously recorded by a qualified recorder. The indicators and recorders are to be labeled to indicate they are required for post accident monitoring.

Cables will be run between the new electronic transmitters and the ICCM cabinets, from the ICCM cabinets to the control room recorders and indicators, from the safety-related power source to the recorders and indicators, and from the ICCM cabinets through non-safety related outputs to the OAC and annunciator points. Separation between safety related and non-safety related components is provided using isolation devices. A 10CFR50 Appendix R review for electrical separation has been performed and mitigation of an Appendix R scenario is not adversely affected. The instrumentation being modified is not required to be replaced in the Appendix R scenario during damage control measures so its replacement in this NSM does not adversely affect the Appendix R shutdown procedures.

Low flow from the LPI pumps during the decay heat removal mode of operation is alarmed to signify a reduction or stoppage of flow and cooling to the core. This modification will not change the way this or any other alarm functions and will not change the function or operation of the LPI flow instrumentation or the operation of the LPI system. The LPI flow instrumentation has only monitoring and alarm functions and does not have any control functions. The new safety related instrument loops' accuracy, due to instrument error, has been determined to be acceptable for monitoring required accident flow. The new safety related and non-safety related instrumentation has as good or better accuracy compared to the existing instrumentation for normal operation. This modification involves no unreviewed safety questions or safety concerns.

Nuclear Station Modification #ON-22448, (Unit 2)

DESCRIPTION:

This modification will replace the existing non-QA channels of pressurizer level and temperature instrumentation with qualified QA Condition 1 channels of instrumentation. The existing level transmitters, which are not rated to ensure post-accident environment availability, will be replaced with new QA1 transmitters, qualified for a harsh environment, which will utilize the existing level taps to detect pressurizer level. The new transmitters will provide as input to both the Train A and Train B Inadequate Core Cooling Monitor (ICCM) Reactor Vessel Level Instrumentation (RVLIS) cabinets. The ICCM RVLIS cabinets will provide the following outputs:

4-20mA signals

1) Individual temperature compensated pressurizer level channels to new QA1 chart recorders on the main control board (MCB).

2) Individual temperature compensated pressurizer level channels to new QA1 digital indicators on the MCB.

3) Individual pressurizer temperature channels to new QA1 digital indicators on the MCB, and one channel to the new QA chart recorder mentioned in 1) above.

4) Temperature and pressure compensated level as well as uncompensated pressurizer level and pressurizer temperature non-safety related Operator Aid Computer (OAC) analog computer points.

0 to -10VDC

Three temperature compensated pressurizer level signals to the Bailey ICS. The operator can manually select between two Train A and one Train B signal(s).

SAFETY EVALUATION:

This modification will be in compliance with Regulatory Guide 1.97, Rev. 2, with respect to post-accident monitoring of pressurizer level and temperature. Pressurizer temperature is not enveloped under the requirements of Regulatory Guide 1.97. However, Safety related pressurizer temperature RTDs will be installed for use in the temperature compensation of the pressurizer level signal. Type A variables are defined as those which are monitored to provide the primary information required

to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents. As a Type A variable, Regulatory Guide 1.97 requires that no single failure of the instrumentation or supporting power sources prevent the operators from being presented the information necessary for them to determine the status of the plant and to bring and maintain the plant in a safe condition following a design basis accident. In accordance with this requirement, the ICCM system is supplied with redundant class 1E power to ICCM cabinets A and B. These cabinets are electrically independent and physically separated, and provide indication to redundant indicators on the MCB. Therefore, no single failure within the ICCM system can cause the inability of the operators to ascertain the appropriate plant status following an accident.

This modification exceeds the requirement as stated in Regulatory Guide 1.97 by providing various methods of display, including dial or stripchart recorder, both of which are QA1 instruments, as well as OAC analog computer points (non-QA).

Regulatory Guide 1.97 requires that pressurizer level indication range be from bottom to top of the pressurizer. The indicating range (with existing level taps) is 0-400 inches, which corresponds to 11% to 84% pressurizer level as a percentage of volume. Therefore, the instrumentation range does not fully comply with Regulatory Guide 1.97 requirements. However, an analyses has been performed to demonstrate that the range of the existing level taps is adequate for both normal operational transients and accident conditions. The NRC Staff has concluded, based upon a review of the analyses that the existing range is an acceptable deviation from Regulatory Guide 1.97. An Appendix R review was performed for this modification. Power supply from ICCM cabinets to transmitters and various alarms and indications is adequately sized to handle new loads. The transmitters in containment are seismically mounted and qualified for post-accident conditions. The new pressurizer temperature RTDs are qualified for post accident containment conditions. The new indicators will be seismically mounted on the MCB, and a control board review for seismic effects were performed.

A loop accuracy calculation for the temperature compensation loops has been completed. The accuracies of all of the pressurizer level signals used by the control room operator and the ICS have been reviewed and found acceptable.

#ON-22448 (Continued)

The instrument pulse lines and other Regulatory Guide 1.97 instrument line applications were reviewed and found to be acceptable for use with safety related indications.

Pressurizer level and temperature is an input to the ICS, and a postulated MSLB accident scenario is evaluated both with and without operator action and ICS. While availability of the ICS moderates plant response, the Unit can successfully mitigate the transient without taking credit for ICS. Pressurizer level and temperature is also an input to HP-120 position, but this function is for normal makeup flow control, and is not required for accident mitigation. These modification has been reviewed against the existing description of pressurizer level and temperature controls. Failure high or low of the instrumentation does not create any effects not previously existing with the instrumentation being replaced. The pressurizer was originally constructed with the RTD thermowell in place, and B&W performed the analysis to demonstrate acceptable stresses. Since an RTD was specified in these modifications which varies in characteristics from the original, the RTD drawing, Vendor Seismic Analysis, and Vendor Similarity Report were reviewed and found to be acceptable with no further action required.

Pressurizer level and temperature is an input to the ICS, and a postulated MSLB accident scenario is evaluated both with and without operator action and ICS. While availability of the ICS moderates plant response, the unit can successfully mitigate the transient without taking credit for ICS and operator response. The role of the pressurizer level and temperature instrumentation in the mitigation of accidents has not changed. This instrumentation is not accident mitigation equipment. With this upgrade, it will serve the role of post-accident operator indication per Regulatory Guide 1.97.

Nuclear Station Modification #ON-22589, (Unit 2)

DESCRIPTION:

Add two qualified QA Condition 1 Channels of high injection flow instrumentation powered by safety sources. Display both channels in the Control Room on QA1 indicators, and record them in the Control Room on QA1 recorders. Add parallel controls for valves 2HP-25 and 2HP-27 to the main control boards. Existing high pressure injection flow instrumentation is not QA1. This QA Condition instrumentation is a requirement for Regulatory Guide 1 97 Type "A" variables. The addition of parallel controls for valves 2HP-25 and 2HP-27 is in response to HED 0-2-61, part A.

SAFETY EVALUATION:

The HPI System, the Inadequate Core Cooling Monitoring (ICCM) System, and the control room control boards are affected by this modification and are QA Condition 1. Relevant design basis accidents are addressed in the FSAR Sections 15.9 (Steam Generator Tube Rupture Accident), 15.13 (Steam Line Break Accident), and 15.14 (Loss of Coolant Accidents).

The HPI flow monitoring instrumentation is addressed under Regulatory Guide 1.97. Regulatory Guide 1.97 describes a method acceptable to the NRC staff for complying with their regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.

Regulatory Guide 1.97 classifies HPI flow monitoring as a Type D variable and Category 2, with a specific range of 0 to 110% design flow. This variable, as Type D, is used to monitor operation. This instrumentation is also selected as a Type A variable and Category 1, and given a range of 0 to 110% design flow. A Type A Variable provides primary information needed to permit control room operating personnel to take the specific manual control actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents events. A Type D variable provides information to indicate the operation of individual safety systems and other systems important to safety. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. The design criteria for Category 1 (Type A and Type D variables) includes environmental qualification, seismic qualification, design against single failure, at least one channel displayed on a direct indicating or

recording device, instrumentation energized from station standby power sources, continuous indication display, recording of instrumentation readout information, separation between safety related and non-safety related equipment using isolation devices, and instruments identified on control panels so that the operator knows that they are for use under accident conditions. Instrumentation being upgraded for accident monitoring is not to degrade the accuracy and sensitivity required for normal operation.

The existing flow elements and instrument lines leading to the existing transmitters will be used for the new transmitters. For the Regulatory Guide 1.97 upgrades at Oconee, the lack of written criteria did not provide enough historical information to verify that the instrument impulse lines were installed properly. However, these instrument lines and other Regulatory Guide 1.97 instrument line applications were reviewed in Design Study ONDS-0217/00. The instrument lines associated with Regulatory Guide 1.97 instrumentation were determined to be satisfactory with respect to seismic loadings and safe from non-seismic interactions. The instrument line valves and materials utilized on the applications were also found to be acceptable from a material compatibility standpoint (piping, tubing, and fittings) and from an ASME code standpoint (valves).

The current HPI flow instruments consist of two transmitters for each main header and one transmitter for each HPI crossover flow path to each main header (i.e., HPI Pump 2B crossovers to main headers A and B on the RCS side of valves 2HP-26 and 2HP-27, respectively).

Main headers A and B each have one pneumatic transmitter and one electronic transmitter connected to a common line that gets input from a flow element. The existing pneumatic transmitter provides the input signal to the main control board. The existing electronic transmitter provides the input signal for the non-safety related OAC in the control room. One safety related electronic transmitter replaces the current pneumatic transmitter on each train. The new safety related electronic transmitters will provide input signals to both the control board and the non-safety related OAC. Each new channel of safety related instrumentation is to be displayed by a qualified indicator and continuously recorded by a qualified recorder. The indicators and recorders are to be labeled to indicate they are required for post accident monitoring. The current non-qualified electronic transmitters used to monitor HPI main header A and B flow are being replaced with new non-qualified electronic transmitters. The new non-qualified transmitters will not provide control board signals or be mounted on the control boards. The failure mode of

the current and new transmitters is low flow on loss of air or power. Failure of one loop will not adversely affect the control room reading of the other loop's instrumentation.

The current transmitters use non-safety related air and power. The new non-safety related transmitters use non-safety related power and the new safety related transmitters and recorders/indicators use safety related power. Power supplies have been reviewed for adequacy in supplying the new loads.

HPI crossover flow paths (i.e., HPI flow paths designed to bypass failed-closed/blocked 2HP-26 or 2HP-27 valves) are currently fitted with non-qualified flow instrumentation. Although not addressed as a commitment, this modification will also replace the instrumentation with qualified electronic transmitters and control board indicators.

The new safety related transmitters that are used to meet Regulatory Guide 1.97 commitments are QA Condition 1 and will receive safety related power from the ICCM cabinets. These new transmitters are environmentally qualified, seismically qualified, and are seismically mounted. The new indicators and recorders are seismically qualified, seismically mounted, QA Condition 1 and receive safety related power. The new indicators and recorders are located in a mild environment and require no special environmental qualification. The recorders are already installed. The new QA Condition 1 components were reviewed for the potential of seismic interaction from non-safety related equipment and were found to have no interaction concerns. A control board seismic review has been performed for the control bard changes. The OAC and annunciators already exist. The instrumentation exposed to system pressure is designed to appropriate design conditions. The range for the new instruments covers the Regulatory Guide 1.97 listed range of 0 to 110% design flow. The design flow for HPI headers A and B is 500 gpm and the instrument range is 0 to 750 gpm. The current instrumentation range is 0 to 1000 gpm.

The new non-qualified transmitters are non-QA Condition (non-safety related) for operational function but will maintain instrument loop pressure boundary under design basis conditions. The new non-qualified transmitters are also seismically mounted prevent seismic interaction. Thus the non-qualified transmitter is not postulated to add a new failure mode that would allow erroneous readings to occur from the safety related instrument.

Cables will be run between the new electronic transmitters and the ICCM cabinets, from the ICCM cabinets to the control room recorders and indicators, from the safety-related power source to

the recorders and indicators, and from the ICCM cabinets through non-safety related outputs to the OAC and annunciator points. Separation between safety related and non-safety related components is provided using isolation devices.

A 10 CFR 50 Appendix R review for electrical separation has been performed and mitigation of an Appendix R scenario is not adversely affected.

HPI pump discharge header low and high flow rate conditions are alarmed in the control room. The low flow rate condition (< 150 GPM) signifies a reduction or stoppage of flow from a single header to the core. The accuracy of the new instrumentation is adequate to perform the low flow indication/alarm function and this modification will not change the low flow alarm setpoint or affect the way this function is accomplished. The high flow rate condition (currently defined as a main header flow rate > 500 GPM) signifies HPI pump runout flow. The high flow rate alarm setpoint is reduced from 500 GPM to 475 GPM as part of this modification. The reduction in the setpoint level is necessary to account for instrument loop inaccuracy. The new instrument loop inaccuracy has been established to be less than +/- 25 GPM at a measured flow rate of 475 GPM. Since instrument loop inaccuracy was not considered in the establishment of the old 500 GPM high flow alarm setpoint, this change in setpoint enhances pump integrity protection (i.e., run-out) and is not considered to be a reduction in high flow protection. The new 475 GPM high flow alarm setpoint also provides the necessary margin to assure HPI/ECCS minimum flow requirements are satisfied for all small break LOCA conditions. A minimum of 450 GPM (at an RCS pressure of 600 psig) of HPI supplied flow is required to satisfy all assumptions made in the thermal hydraulic calculations for the various ECCS LOCA [illegible] performed by B&W. A setpoint of 475 GPM with a total instrument loop uncertainty of less than +/- 25 GPM satisfies both the maximum and minimum flow limits and requirements of the system.

The addition of parallel controls for HP-25 and HP-27 on the main control boards completes implementation of a Human Engineering Review Team recommendation regarding HPI system control functions being moved rom the vertical board to the main control board in the control room. The controls on the vertical board are not to be eliminate or altered, but "parallel" controls are being added to the main control board. The operation of the motor operated valves HP-25 and/or HP-27 is not affected in any way by this modification and Appendix R and seismic reviews of the control board changes have been performed.

Nuclear Station Modification #ON-22806, (Unit 2)

DESCRIPTION:

This Modification involves the replacement of Valves 2LPSW-18, 2LPSW-21, and 2LPSW-24. These valves are located on the discharge side of RBCUs '2A', '2B', and '2C'. They serve as the main throttling valve for LPSW through the RBCUs. Presently these valves are 8 in. Walworth glove valves that require frequent maintenance and have been proven to be unreliable. Replacement Valves are DMV-618, Jamesberry, High Performance Butterfly Valves. These valves have flanged connections. The operators for these valves will also be replaced and must be EQ. Limitorque SMB-0002 EMO's will be installed for all three valves. Two hangers will also be modified in the turbine building.

SAFETY EVALUATION:

LPSW 18, 21, 24 are the outlet valves for the Reactor Building Cooling Units located in the East Penetration room. These valves perform two major safety functions. They maintain and regulate flow through the RBCU's that provide Reactor Building cooling post-Loca and provide system isolation when cooler leakage is detected by RIA-31 or 35. Valves 18, 21, 24 are QA condition one components; EQ qualified for post-LOCA environments and backed by essential power supply. The proposed modification will maintain the current system design criteria. The selection of butterfly valves for this application will reduce the probability of valve leakage when the system is isolated. Throttling of the valves, not normally done with butterfly valves is only a concern during normal operation testing. During post accident operation these valves will fully open. The new valve operators are analyzed for seismic conditions and support/restraints are redesigned. The bolting details for the new valves will be different and a stress analysis is performed for the connection, valve, and operator. Electrical power requirements are the same for the new valves and operators. Power will be connected with the same cable off of the same power bus. There is no significant safety impact of this modification or no unreviewed safety questions.

Nuclear Station Modification #ON-22818, (Unit 2)

DESCRIPTION:

This modification involves the replacement of the following monitors:

2RIA-2 Main Fuel Handling Bridge Monitor
2RIA-3 Auxiliary Fuel Handling Bridge Monitor
2RIA-4 Reactor Building Personnel Hatch Monitor
2RIA-5 Incore Instrument Tank Monitor
2RIA-10 Primary Sample Hood Area Monitor

Replacement of these monitors is part of a larger program to replace all radiation monitors at Oconee with new monitors. The new monitors will have new digital technology and when necessary be tied to a new computer system installed by NSM 22422/01. As a result the existing control room rate meters for these monitors are not needed and will be removed by this NSM.

SAFETY EVALUATION:

The purpose of 2RIA-2 is to monitor radiation levels in the reactor building refueling area. Technical Specifications allow for optional monitoring in the event that 2RIA-2 and -3 become inoperable. The backup method is to use portable monitoring having the appropriate ranges and sensitivity to fully protect individuals involved in the refueling operation. A Portable Bridge Monitor will replace the old 2RIA-3 and will monitor the Auxiliary Fuel Handling Bridge. The Portable Bridge Monitor will include an analog rate meter and audible alarm. These monitors are needed to protect personnel working in the area and it will not be necessary to provide a signal to the control room or SCADA System. The new 2RIA-3 will be permanently relocated on the west wall beside the Fuel Transfer Canal. This monitor will be connected to the control room and be cable of operating during fuel movement. 2RIA-3 is not required for reactor power operations and may be removed from containment during reactor power operations. 2RIA-4 monitors the Reactor Building Personnel Hatch Area. The existing monitor is used as a backup indication of containment activity. A high range portable survey meter may be used to provide a backup if 2RIA-4 becomes inoperable. Existing alarm features will be retained for the new monitor including its connection to the Reactor building Alarm System. The range for this monitor remains the same. 2RIA-5 which monitors the Incore Instrument Tank will maintain the same location and the same alarm functions as the existing monitor. The range of this monitor will be decreased from that of the existing monitor; however the monitor will still perform its intended function.

#ON-22818 (Continued)

Radiation level monitor 2RIA-10 has been relocated to better monitor the Primary sample hood area. The alarm setpoint has been raised. 2RIA-10 will be provided with a local readout and horn.

All the new radiation monitors are QA Condition 4 and will be seismically anchored. There are no special power requirements for these monitors. Double fuse protection for the electrical penetration is not applicable. The monitors are not nuclear safety related and have no impact on the function of any system. The new monitors are fully capable of performing their functions. Therefore, the consequences of an accident evaluated in the FSAR will not be increased. No new failure modes are created by this modification. A 10CFR50 Appendix R review has been performed. Based on the discussion presented above, it is determined that this NSM does not involve any unreviewed safety questions.

Nuclear Station Modification #ON-22852/00, (Unit 2)

DESCRIPTION:

This modification implements commitments detailed in Generic Letter (GL) 89-19 that will modify the Steam Generator Level Control System (SGLCS) to start the Motor Driven Emergency Feedwater Pumps (MDEFWPs) upon low-low Once Through Steam Generator (OTSG) level and the Integrated Control System (ICS) to provide a backup trip of the Main Feedwater Pumps (MFPs) on high-high OTSG level. In addition, this modification will also make two changes to the Emergency Feedwater (EFW) System not required by GL 89-19. It will activate the SGLCS automatic level control features upon Turbine Driven Emergency Feedwater Pump (TDEFWP) starts, allowing automatic control of level if only the turbine is operating due to a loss of feedwater. It will also install a four position switch for control of the MDEFWPs providing two automatic initiation positions: (1) start the MDEFWPs on low steam generator level only, or (2) start the motor driven pumps on low steam generator level or upon loss of both MFPs.

SAFETY EVALUATION:

Steam generator level is affected by various plant factors, the most important being feedwater flow rate, feedwater temperature, steam pressure, reactor power, and the programmed setpoints. Normally, the Steam Generator Control portion of ICS attempts to match feedwater flow to power level (demand). ICS provides control signals for changing MFP speeds and control valve positions accordingly.

The consequences of not maintaining control over OTSG level are (1) overfill: carryover, or the entraining of water droplets into the steam system, which could result in catastrophic and irreparable damage to turbine generators or their blading, or (2) dryout (overheat): a loss or degradation of the reactor's heat sink which could result in a loss of cooling. The OTSG level response system being added is designed to protect against the overfill and dryout (overheat) events.

Overfill protection is normally provided by ICS which terminates feedwater to the OTSG, at the high level setpoint (upon receipt of two out of two signals), by tripping the MFPs. ICS is designed to limit the potential for overfill through the control of pump speed and feed control valve position. This NSM will wire in a back-up MFP trip device to assure termination of feed flow upon indication of a high level condition. Solenoid valve SV6, in the hydraulic control oil system, will be connected to existing steam generator level monitors by existing contacts.

These contacts currently are used to trip the main turbine, and are separate from the existing ICS MFP trip initiation device. The current MFP trip device is solenoid valve SV12, also part of the hydraulic control oil system. An auxiliary relay will be added to trip both pieces of equipment. SV6 and SV12 will both be powered from the 125VDC station batteries (non-safety). The SV6 auxiliary relay will be powered from the 125VDC station instrumentation and control batteries. Existing instrumentation will be used. Initiation of overfill protection will continue to require satisfaction of the existing two out of two logic configuration. The overfill portion of this modification is non-safety related. The hardware for existing overfill protection circuits is comprised of proven, reliable, non-safety grade components. Presently, dryout protection is provided by the initiation of EFW on anticipation of a possible Loss of Feedwater Accident by monitoring hydraulic control oil pressure and MFP discharge pressure. The dryout protection portion of this modification will also use existing transmitters. The setpoint of 21 inches has been chosen to provide a high level of confidence that an EFW start signal will be generated prior to dryout, taking into account the uncertainty in the OTSG level instrument string (train) and the lack of density compensation for the instruments. The setpoint has also been chosen, along with the thirty second initiation delay, to minimize spurious EFW initiations from normal plant transients. It (the 21 inch setpoint) has also been chosen to prevent inadvertent EFW starts by providing sufficient margin to the ICS main feedwater low level control setpoint (25 inches). Adding the low level setpoint to the OTSG level transmitter instrument loops will provide a diverse means of actuating EFW as the OTSG approaches dryout. The GFLCS is a redundant two train system, which is located in a mild environment. The logic configuration tests each OTSG for a dryout condition by using two signals from an OTSG. One signal is routed through train A, and one through train B. Failure of one train would result in one remaining input for each OTSG. Each train is powered from the Vital Instrumentation and Control Power panelboards. Upon initiation of EFW, computer alarms will alert the operator to the initiation due to low OTSG level and serve as a record for transient event evaluation. The low level start portion of this modification is QA-1.

The three accidents which are affected by steam generator level are Steam Line Break, Loss of Condensate/Main Feedwater (LMFW), and Loss of all Onsite and/or Offsite AC Power (resulting in LMFW).

Nuclear Station Modification #ON-32422, (Unit 3)

DESCRIPTION:

This modification involves replacement of the Reactor Building (RB) and Unit Vent monitors. The current monitors are troublesome and spare parts are becoming unavailable. A new detector, 3RIA-49A, will be added to the RB monitor for high range gas detection. In addition, a computer based system control and data acquisition (SCADA) system will be added for use by these and other monitors. The new monitors will be anchored seismically. A new isokinetic sampler designed to meet ANSI N13.1-1969 criteria will be installed in the Unit Vent. The sample lines will be rerouted to fit the new monitors. New lighting fixtures and power outlets will be provided where necessary. Another portion of this modification will lock close instrument air valve IA-1265 and designate IA-1264 as normally closed. This will insure the prevention of an over pressure condition for the reactor building monitor. This instrument air line is used for the purpose of makeup air during RB hydrogen purge and piping through a penetration is shared with the sampling lines for the RB monitor.

SAFETY EVALUATION:

The purpose of the monitors is to monitor the Unit Vent for radioactive effluent and the RB atmosphere for indication of equipment malfunctions and personnel access limitations. These monitors are supplied with a check source of sufficient half-life which is used to automatically verify monitor operability on a periodic basis. Control Room annunciation of high radiation level is provided for each detector channel. A high radiation level reading on the Unit Vent Monitor will terminate normal operation of the RB purge system. A high radiation level signal on the Reactor Building Monitor automatically isolates the RB normal sump line. Vent Monitor fail, back-up actions are specified in the Technical Specifications.

The modification has no impact on the function of any system. These replacement monitors are as good as or better than the monitors being replaced. The monitors are used to insure that radioactive effluent releases are maintained within acceptable limits. As such, the probability of an accident or malfunction of equipment important to safety which were previously evaluated in the FSAR will not be increased.

No new failure modes or operating characteristics are created by this modification. The rerouted monitor sample lines will be design routed and field supported. The monitors will be seismically anchored (QA Condition 4) to provide protection to class C piping adjacent to the monitor.

#ON-32422 (Continued)

A seismic interaction review was performed to show that the added lighting fixtures could not affect any safety related equipment. Therefore, the possibility of an accident or malfunction of equipment important to safety which is different than any already evaluated in the FSAR will not be created.

The modification has no adverse impact on any safety systems. The instrument air valves that are being locked closed would only be used in the event the hydrogen recombiner were unavailable. Should this situation arise, the valves could be opened to address removal of hydrogen. Therefore, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

Technical Specification 3.5.5 requires that particulate and iodine sampling must be performed for the Unit Vent. The new Unit Vent monitor package will provide both particulate filter and iodine cartridge sampling equipment to meet this Technical Specification. No safety limits or reactor coolant system parameters are affected by this modification. Also, the bases to any technical specifications is not affected. Therefore, the margin of safety as defined in the bases of any Technical Specification is not reduced. Based on this discussion, no unreviewed safety questions are created by or involved with this modification.

Nuclear Sta n Modification #ON-32597/00, (Unit 3)

DESCRIPTION:

The Main Steam Line Monitors (3RIA-16 and -17) are area radiation monitors which are located next to the Main Steam line piping to measure the presence of radioactivity in the process stream. 3RIA-16 monitors the "A" main steam header and 3RIA-17 monitors the "B" main steam header. These existing monitors have become obsolete and difficult to maintain. This NSM will replace them with new monitors which comply with the requirements of Regulatory Guide 1.97 and are compatible with the System Control and Data Acquisition (SCADA) system. This new computer system allows operators to perform system control and data acquisition functions from the control room computer terminals. As a result, the existing control room rate meters for these monitors will become obsolete and will be removed from service.

The Stack Flow Radiation Monitor (3RIA-56) is an accident range radiation monitor in the Unit Vent Stack which was specifically installed to cover the upper (high-high) requirements of Regulatory Guide 1.97. RIA-45 and RIA-46 cover the normal and high ranges, respectively. The three detectors provide overlapping coverage of the range specified for common plant vent radioactive discharge. This NSM will upgrade RIA-56 to be compatible with the SCADA system (RIA-45 and RIA-46 which were updated previously).

SCADA system software will be updated to include the above monitors. Analog signals from each monitor will be sent to the Operator Aid Computer (OAC).

SAFETY EVALUATION:

The original purpose of monitors 3RIA-16 and -17 was to detect significant steam generator tube leakage. As a result of TMI, new accident monitoring requirements were issued through NUREG-0737 and Regulatory Guide 1.97 including releases from the main steam safety valves. This modification addresses the three major items of regulatory compliance: monitor range, location, and environmental qualification. The upper end of the monitor's range is being extended to meet Regulatory Guide 1.97 requirements. The monitors are being relocated from downstream of the main steam safety valves to upstream of them. They will be environmentally qualified to Regulatory Guide 1.97 standards. The associated microprocessor will be located in a "mild" environment. A check source is provided to verify detector response.

A new support structure will be required to mount the detector assembly. The detectors and shield assemblies will be designed to facilitate seismic mounting and outside weather conditions. The mountings are QA condition 4 due to their close proximity to the main steam line; the RM-80s are mounted non-QA. Concern over the possibility of potential support structure vibration was investigated. The mounting of the modification would not cause any vibration problems for the monitors. The movement characteristics of the main steam piping have been reviewed. Adequate clearance is maintained for the monitors.

The range of the 3RIA-56 detector will remain the same as it is now. The old detector will be used with the new system. Detector operation is continuously verified by a "keep-alive" source. The electronic response of the monitor is periodically checked by a "checkcurrent". Regulatory guide 1.97 environmental qualification requirements will be met for operation in post accident conditions. The associated microprocessor will be located in a "mild" environment. The detector is mounted QA Condition 1 and the RM-80 is mounted non-QA.

3RIA-16, -17, and -56 are classified as Type E, Category 2, key variables. They are non-safety related. The monitors will be powered from a high reliability, non-load shed power source consistent with Regulatory Guide 1.97 requirements. This modification involves no safety concerns or unreviewed safety questions.

A 10CFR 50 Appendix R review of the changes has been performed. No support/restraint or pipe stress calculations are affected by this modification.

This modification will have no adverse impact on any safety systems. The steam line detector/shield assembly mountings are QA-4 due to their close proximity to the main steam lines. The system design is maintained for this modification and no new failure modes are postulated.

Nuclear Station Modification #ON-32820, (Unit 3)

DESCRIPTION:

This modification as proposed will replace existing monitors 3RIA-57 and -58 as per the overall radiation monitor change-out task. The monitors are used for emergency planning and perform no safety-related function as per 10CFR50 Appendix A. However, Regulatory Guide 1.97 proposes that these monitors be seismically mounted and be EQ qualified. This NSM intends on satisfying these and other Regulatory Guide 1.97 design criteria.

Instead of locating monitors inside of containment, the monitors will be located in an electrical penetration. The sensitive portion of the detector is to protrude into containment, covered by a thin cap. The sensitivity and range of the monitor will satisfy guidance stated in Regulatory Guide 1.97, Rev. 2, which is the adopted version for Oconee.

SAFETY EVALUATION:

RIA-57 and 58 are the radiation monitors for the Reactor Building. These monitors measure Type C and Type E variables as described in Regulatory Guide 1.97, Revision 2, December 1980. The Type C variable gives early indication of Reactor Coolant system breach. This variable is listed as a category 3 variable. A category 3 variable as defined 1.97-5 is to be designed for high quality, commercial grade equipment. The Type E variable on page 1.97-22 is considered category 1. The detector provides detection of significant releases to containment, release assessment, long-term surveillance, and emergency plan actuation. The high range monitors are designed to category 1 design criteria to be EQ as per Regulatory Guide 1.89 for class 1E equipment. The monitors are designed seismically as per Regulatory Guide 1.100, and are single failure proof. The equipment is powered from the essential power supply system and is battery backed. An Appendix R fire protection review has been performed. A seismic review has been performed for the equipment on the control room board. Containment penetration testing will be performed as per Technical Specification 4.4.1.3. In conclusion, there is no significant safety impact of this modification, ie. no changes in the Technical Specifications, no unreviewed safety questions.

Nuclear Station Modification #ON-52825, (Unit 1 & 2)

DESCRIPTION:

This modification involves the replacement of the following radiation monitors:

1RIA-37	Gaseous Waste Disposal (GWD) Normal Range Gas Monitor
1RIA-38	GWD High Range Gas Monitor
1RIA-39	Control Room Gas Monitor
1RIA-41	Spent Fuel Pool Building Gas Monitor

RIA-37 and -38 monitor the discharges of the Gaseous Waste Disposal System to the Unit Vent to help insure that releases do not exceed Technical Specification limits. RIA-39 and RIA-41 pull air samples from their respective ventilation systems to detect abnormal concentrations of radioactive gases in those areas. The new monitors will have new digital technology and be tied to a new computer system installed by NSM ON-12422/01 and ON-22422/01. This new computer system allows operators to perform system control and data acquisition functions from control room computer terminals. As a result, the existing control room rate meters for these monitors are not needed and will be removed by this NSM.

SAFETY REVIEW:

The purpose of RIA-37 and RIA-38 is to detect a wide range of activity in the waste gas effluent due to batch releases and to automatically terminate discharge at a preset concentration within prescribed release limits. At a preset radiation level, interlocks from this monitor automatically stop the waste gas exhauster (if it is running) and close the waste gas tank discharge valves and waste gas exhauster isolation valves. The new monitor will be in an on-line configuration like the existing one. The monitor has an Instrument Air purge connection just upstream of the monitor to remove residual waste gas from the line following a Waste Gas Tank release. The installation of this Instrument Air purge line was never documented. This NSM will permanently install this line and resolve the documentation deficiencies.

The purpose of RIA-39 is to monitor for the presence of radioactive gases in the return air duct of the Control Room Ventilation System. RIA-41 performs the same function for the exhaust duct from the Spent Fuel Pool and Fuel Loading areas. Neither monitor has automatic interlock functions or any safety functions. However, the RIA-39 high radiation alarm alerts the operators to energize the outside air booster fans and filter

system. RIA-41 sample lines are located so that it monitors exhaust from the Spent Fuel Pool Area whether the air is directed through a filtered or unfiltered path. The replacement monitors for RIA-39 and -41 will be in an off-line configuration like the existing ones.

The monitors associated with this NSM will have Control Room annunciation of high radiation level and equipment failure including loss of power and loss of sample flow. They are supplied with check sources or a "keep-alive" source, as appropriate, which will be used to verify monitor operability. Each monitor will also be calibrated periodically to assure that the desired detector sensitivities are maintained.

Monitor locations have been reviewed for seismic anchorage; it was determined that none of the monitors required QA Condition 4 anchorage. Changes made to the Control Room Vertical Boards have been reviewed to ensure that the removal of the rate meters will not have any seismic impact. The total range of RIA-37 and -38 is being decreased slightly by this modification to obtain better overlap between the two detectors. As a result, the monitors will better perform their intended functions. No unreviewed safety questions are created by or involved with this modification.

Nuclear Station Modification #ON-52828, (Unit 1 & 2)

DESCRIPTION:

This modification involves the replacement of area radiation monitors 1RIA-11, -12 and -13 with new monitors. 1RIA-11 monitors the Auxiliary Building Corridor 3rd Floor, 1RIA-12 monitors the chemical addition area in the Auxiliary Building 2nd Floor Corridor, 1RIA-13 monitors the Waste Disposal Control Area (1st Floor Auxiliary Building). These monitors have become obsolete and require increasing amounts of maintenance to keep them operating while spare parts and vendor technical support are becoming increasingly unavailable. The new monitors will have new digital technology and be tied to a new computer system installed by NSM #ON-12422/00. Appropriate SCADA system computer software will be updated. By connecting the new monitors to the computerized SCADA system, there will no longer be a need for the existing control room rate meters. As a result, the existing control room rate meters will be removed by this NSM.

SAFETY EVALUATION:

Area radiation monitors provide gamma radiation indication for various areas of the station to control room operators and station personnel. In general, these monitor readings are used primarily for personnel protection to sound an alarm to nearby workers and alert Control Room operators if preset alarm setpoints are exceeded. They can also be used to detect equipment malfunctions as well as to assess plant radiological conditions following an accident. The new monitors will maintain the same general location with their detectors, connection boxes, and microprocessor components being mounted QA Condition 4. Modifications to the control board will be reviewed for seismic impact.

The monitors associated with this NSM will have Control Room annunciation of high radiation level and monitor failure. They are supplied with check sources which will be used to automatically verify monitor operability on a periodic basis. Each monitor will also be calibrated periodically.

The range of 3RIA-11, 3RIA-12, and 3RIA-13 will be changed from 1E-1 - 1E7 mR/hr to 1E-1 - 1E4 mR/hr. This range has been reviewed and found adequate to monitor the respective areas for personnel protection under all plant conditions and also meets the recommendation of ANSI/ANS-HPSSC-6.8.1-1981 for area monitoring.

#ON-52828 (Continued)

The monitors are not nuclear safety related and have no impact on the function of any system. No unreviewed safety questions are created by or involved with this modification.

Nuclear Station Modification #ON-52829/00, (Unit 1 & 2)

DESCRIPTION:

This modification involves the replacement of area radiation monitors ORIA-7, and -8, with new monitors. ORIA-9, 14, 18 will be deleted from the RIA system. ORIA-7 monitors the Hot Machine Shop Area, ORIA-8 monitors the Hot Lab Area, ORIA-9 monitors the Low Level Drumming Area, ORIA-14 monitors the Waste Drumming Area, and ORIA-18 monitors the World of Energy Roof. These monitors have become obsolete and require increasing amounts of maintenance to keep them operating while spare parts and vendor technical support are becoming increasingly unavailable. The new monitors will have new digital technology and be tied to a new computer system installed by NSM #ON-12422/00. This new computer system allows operators to perform system control and data acquisition functions from a control room computer terminal. As a result, the existing control room rate meters for these monitors are not needed and will be removed by this NSM.

SAFETY EVALUATION:

Area Radiation Monitors provide gamma radiation indication for various areas of the station to control room operators and station personnel. In general these monitor readings are used for personnel protection to sound an alarm to nearby workers and alert Control Room operators if alarm setpoints are exceeded. The new monitors will maintain the same general location. The monitors associated with this NSM will have Control Room annunciation of high radiation level and monitor failure. They are supplied with check sources which will be used to automatically verify monitor operability on a periodic basis. Each monitor will also be calibrated periodically. The range of ORIA-7 and -8 will be changed from 1E-1 - 1E7 mR/hr to 1E-1 - 1E4 mR/hr. The new range was reviewed and found to be adequate for monitoring these areas. ORIA-9, and -14 will be deleted from the system. The original function of these monitors has changed and therefore are no longer needed. ORIA-18 has never functioned properly and was replaced by a portable instrument years ago. All area radiation monitors are QA Condition 4. These monitors will be seismically anchored. There are no special power requirements for these monitors. Double fuse protection for the electrical penetration is not applicable.

There are no unreviewed safety questions involved with this NSM. The monitors are not nuclear safety related and have no impact on the function of any system. No new failure modes are created by this modification. A 10 CFR 50 Appendix R review has been performed.

Nuclear Station Modification #ON-52878, (Unit 1,2,3)

DESCRIPTION:

This modification will provide degraded voltage protection to the Oconee 4160V essential auxiliary power system when the Standby Buses and Main Feeder Buses are being supplied by transformer CT-5. Under voltage relays, time delay relays and interconnecting cable will be installed in the Unit 1 and 2 switchgear blockhouse in order to accomplish this modification. Some connections will be made from the blockhouse to equipment in the cable rooms, and two SL breaker trip interlock "defeat" switches will be added to control board 2AB3.

SAFETY EVALUATION:

Oconee Nuclear Station has an undervoltage relaying system for the 230 KV Switchyard which separates essential auxiliaries from offsite power supplies if those supplies are experiencing degraded voltage conditions. When a degraded voltage condition is sensed, the system isolates the switchyard from the system grid and automatically provides power to the Main Feeder Buses via the Keowee Hydro Units. The standby Buses supply power to the Main Feeder Buses when power from the respective unit's startup transformers is not available. The Standby Buses may receive power from either auxiliary transformer CT-4 or CT-5. Transformer CT-4 gets power from a Keowee Hydro unit, and CT-5 can receive power from the Central Switchyard or a dedicated 100 KV transmission line connected directly to the Lee Gas Turbines.

During refueling outages it is desirable to power the Standby Buses and thus the Main Feeder Buses (of the shutdown unit) from the Central Tie Switchyard via auxiliary transformer CT-5. This line-up is particularly convenient when maintenance is necessary to the shutdown unit's startup transformer. Without the 100 KV degraded voltage protection being provided by this modification, a postulated LOCA/LOOP Design Basis Event could expose essential equipment to less than adequate voltage if an automatic transfer to the Standby Buses occurs when supplied from transformer CT-5.

For an offsite power source to satisfy NRC requirements, special relay protection must be provided to protect essential plant auxiliaries in the event of system grid voltage degradation. This will allow the offsite power source to trip and the onsite emergency power sources to supply power. The tie to the Central Switchyard via transformer CT-5 has no such protection.

The alignment of the 4160V essential auxiliary power system to the Standby Buses via transformer CT-5 may not be used at present due to lack of degraded voltage protection. By installing the required undervoltage protection for transformer CT-5, this modification will give operations personnel the flexibility of using this alignment. Also, Duke committed to a planned corrective action to ensure that any time the Standby Buses are energized, that they will be energized from an appropriate emergency power source. Installation of this modification meets this commitment for the case when the Standby Buses are energized from the Central Switchyard via CT-5, and the design of the modification meets the criteria for offsite power sources. An Appendix R review and a control board seismic review have been initiated for the required cabling and breaker trip interlock "defeat" switch additions. Mounting of the equipment in the blockhouse will be done QA-4. The modification does not adversely affect any of the essential auxiliaries potentially supplied by CT-5 in case of an accident, or the separation and redundancy criteria for these transformers and cabling.

The modification will provide a reliable source of power to the Standby Buses. If the Standby Buses are aligned to transformer CT-5, and a LOCA/LOOP Design Basis Event occurs, the logic provided by this modification will ensure that essential equipment, which becomes automatically connected to transformer CT-5, will not be exposed to less than adequate voltage.

Appendix R and control board seismic reviews have been initiated, and all equipment is mounted QA-4. Separation criteria for cabling and components is not adversely affected and circuit diversity is maintained. Redundancy is assured by separate, independent relays in a two-out-of-three scheme receiving power from redundant DC sources. Also, the design of the modification meets the criteria for offsite power sources found in the NRC Generic Letter on the adequacy of station electric power distribution system voltages.

This modification does not involve any unreviewed safety questions or safety concerns, and no Technical Specification changes are required.

Operating Procedure #HP/3/A/1009/17, (Unit 3)
#HP/2/A/1009/17, (Unit 2)

DESCRIPTION:

The change to this procedure replaces the thiosulfate solution with a particulate filter and silver zeolite cartridge in the Post-Accident Containment Air Sampling System.

SAFETY EVALUATION:

The operation of the Post-Accident Containment Air Sampling System is described in FSAR 9.3.6.2.3. The description states in part; "The Post-Accident Containment Sampling System isolates a known quantity of containment atmosphere, moves this quantity through a thiosulfate solution for separation of iodine and particulate from the noble gases, and provides a sample of the diluted gas for analysis." Thiosulfate solution is a highly efficient collection media for soluble molecular species of iodine; however, soluble forms of iodine typically precipitate from the containment atmosphere and therefore the collection efficiency of the thiosulfate solution is greatly diminished. Efforts to quantify the iodine collection efficiency of the Post-Accident Containment Air Sampling system were unsuccessful. Replacing the thiosulfate solution with a particulate filter and silver zeolite cartridge provides a highly efficient previously quantified collection media. In addition eliminating the corrosive thiosulfate solution should reduce maintenance, increase equipment reliability, and simplify sampling and analysis. This change serves to improve the Post-Accident Containment Air Sampling System. No physical changes to the sampling system are required and this change does not create any unreviewed safety questions.

Selected Licensee Commitments (SLC) 16.5.4, 16.5.5, and 16.5.6 Shutdown Cooling Requirements

DESCRIPTION:

Selected Licensee Commitments 16.5.4, 16.5.5, and 16.5.6 specify the required methods and equipment needed to ensure that adequate decay heat removal capacity is maintained when the RCS average temperature is below 250°F.

Each commitment applies to a different plant condition as listed below:

16.5.4 RCS Loops filled and average temperature below 250°F.
16.5.5 RCS Loops not full and Fuel Transfer Canal not full
16.5.6 Fuel Transfer Canal full

Acceptable combinations of in service and standby DHR systems for each condition are specified. Required actions and completion times are provided to give guidance when acceptable combination are not available.

The addition of these commitments to the Selected Licensee Commitment Manual formalizes the station's operating philosophy and is being done to fulfill a commitment made to the NRC on October 9, 1991 by Duke Power management at a meeting at NRC NRR headquarters concerning requirements for shutdown operations.

SAFETY EVALUATION:

These commitments do not create any changes to the operation or configuration of structures systems or components addressed in the FSAR that differ from the FSAR evaluated conditions. No additional changes to the FSAR will be necessary as a result of incorporating these SLCs into Chapter 16 of the FSAR. The requirements of these SLCs will add controls to the operation of systems used to remove decay heat to ensure adequate capacity and redundancy are available during shutdown operations. No tests or experiments are needed to implement the requirements imposed by these SLCs.

These SLCs are concerned only with the use of systems used to remove decay heat with the RCS average temperature below 250°F. Loss of decay heat removal accidents are not analyzed in the FSAR, therefore the probability or consequences of an accident previously analyzed in the FSAR will not be increased by these SLCs. Likewise, no increase in the probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will occur as a result of these SLCs.

SLC 16.5.4, 16.5.5, and 16.5.6 (Continued)

The Shutdown Cooling SLCs do not create configurations or operating conditions not already incorporated into station procedures. No possibilities of accidents or malfunctions of equipment important to safety different than those already evaluated in the FSAR will be created by placing these SLCs in effect.

The margin of safety defined in the bases of the Technical Specifications is unaffected by this change. An unreviewed safety question does not exist.