Nuclear Station Modification #ON-22422 (Unit 2)

DESCRIPTION: This modification involves replacement of the Reactor Building and unit vent monitors. The current monitors are troublesome and spare parts are becoming unavailable. A new detector, RIA-49A, will be added to the RB monitor for high range gas detection. In addition, a computer based system control and data acquisition system will be added for use by these and other monitors to be replaced in future plant modifications. Another portion of this modification will lock closed instrument air valve IA-1225 and designate IA-1226 normally closed. This will insure the prevention of an over pressure condition for the RB monitor.

SAFETY EVALUATION: 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 sample lines will be field routed and supported. The monitors will be seismically anchored. 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.

No adverse impact to any safety system will result. No unreviewed safety questions are created by this modification.

Nuclear Station Modifications:
ON-12484 (Unit 1)
ON-22484 (Unit 2)
ON-32484 (Unit 3)

DESCRIPTION:

Motor-operators will be added to valves 1,2,3CCW-268 and 1,2,3CCW-287 in the Standby Shutdown Facility Auxiliary Service Water (ASW) system to enable Operations to rapidly open these valves during SSF start-up. Valve controls for CCW-268 will be provided to allow remote throttling from the SSF control room. New instrumentation and alarms are provided to allow proper control of SSF ASW.

Present operation of this system depends on manual valve operation. Valve throttling requirements are based on a pressure gage readout comparison to a throttling curve. This NSM will significantly enhance Operations' ability to quickly start-up the SSF (within the 10 minute Appendix R requirement) and will remove uncertainties related to flow measurement and valve throttling. Details of the modification are an follows:

- Six new motor operators are being added to valves 1, 2, 3 CCW-268 and -287. These operators are equipped with handwheels.

- Flow instrumentation is to be added to each of the three SSF ASW lines supplying each unit's steam generators. Flow readout, a high flow alarm, and a low flow alarm are to be provided for each unit in the SSF Control Room. The low flow alarm indicates that initial minimum SSF ASW flow requirements are not being met and the high flow alarm indicates potential for SG tube flow-induced vibration.

- An SSF ASW Pump suction pressure switch is also to be added to alarm low suction pressure when conditions create potential for pump runout. Monitoring the pump for cavitation noise following this alarm will determine the need for reducing flow.

- Valves 1,2,3CCW-268 controls will allow for precise control of valve stem movement for precise throttling. Open, closed, and throttled position indication will be provided in the SSF control room. Valve 1,2,3CCW-287 are to be open/close only, and do not need throttling control capability.

SAFETY EVALUATION:

The valves that these modifications affect are in the SSF ASW System. This system provides diesel generator-powered emergency feedwater to the steam generators if main feedwater and emergency feedwater from the normal plant powered systems are lost.

The modifications are QA Condition I and safety related. The valves

ON-12484 (Continued)
ON-22484 (Continued)
ON-32484 (Continued)

will be controlled from the SSF control room and have local handwheels on them. The valves will fail "as is" on loss of power. Valve CCW-268 will be throttleable to prevent pump runout. The motor torque loads of the operators will change the valve position for the maximum pressure differential case. A pipe stress analysis was performed due to the valve motor additions and valve re-orientations. The valves are powered and controlled from the SSF. An Appendix R fire separation review was performed. The SSF control board already has spare buttons on it so no seismic review of the board is required. The SSF has sufficient power capacity for the addition of the motor operators. The new cables for the valve motors and instrumentations will be located in the SSF and west penetration room. The failure modes of the flow and pressure indications are low. Alarms indicate low readings. A high flow alarm is also incorporated into the design. The part of the platform that holds the grating is QA Condition 4. The grating and removable handrails are non-QA Condition. The grating will have grating clips to hold it in place and the handrails will be inserted into sleeves. Since vertical seismic accelerations are less than 1g, the grating and handrails should not become dislodged. The lifting supports are QA Condition 4. The box in the Cask Decontamination Room is QA Condition 1 since QA instruments are to be attached to it.

The SSF is discussed in Section 9.6 of the FSAR and the SSF ASW System is briefly discussed under the EFW FSAR Section (Section 10.4.7). Section 15.8 discusses the Loss of Electric Power Accidents, which this modification could affect. This modification will enhance the operability of the SSF so the Loss of Electric Power Accidents' effect would be reduced. Seismic interaction and Appendix R fire separation have also been addressed. Therefore, the consequences of accidents previously evaluated in the FSAR will not be increased.

These NSM changes are not related to any of the conditions or events that cause FSAR accidents. Therefore. the probability of any accidents previously evaluated in the FSAR are not increased. This modification does not degrade the SSF since design criteria and FSAR requirements for structures, systems, and components of the SSF are met. The new valves fail as is on loss of power. In both the present and proposed cases, if both valves were opened inadvertently, then the EFW System could possibly be short circuited to the CCW line. Presently, this situation would require 2 operator errors in which two normally closed manual valves would have to be opened. With the NSM, 2 operator errors would still be required. The errors would involve inadvertently pushing control room buttons. Thus, these NSMs are not considered to significantly increase the probability or consequence of malfunctions of equipment important to safety addressed in the FSAR. The new instruments fail in a position that would indicate an instrument error. Therefore, no new

ON-12484 (Continued)
ON-22484 (Continued)
ON-32484 (Continued)

accidents or malfunctions of equipment not addressed in the FSAR are postulated. No Technical Specifications are affected by this modification and no safety/design limits are adversely affected. Therefore, margins of safety as defined in the bases to Technical Specifications are not reduced.

There are no unreviewed safety questions associated with these NSMs.

Nuclear Station Modification #
ON-32587 (Unit 3)

Description:

This modification replaces the non-qualified channels of Decay Heat Removal Flow instrumentation with two qualified QA Condition 1 channels of instrumentation. This modification will not change any alarm functions, or the way the LPI system operates. The new instrument loop will not be as accurate as the existing loop, but the impact of the additional error is negligible.

All of the existing non-qualified LPI Flow instrumentation will be deleted.

Transmitters 3LPIFT0004A & 5A will be replaced with Rosemount transmitters which will directly feed to the OAC instead of feeding through the ICCM Cabinets.

The flow transmitters 3LPIFT004P & 5P will be replaced with Rosemount transmitters. The location of the new transmitters will be the same as the existing transmitters in the East Penetration Room. These transmitters will provide inputs for the ICCM cabinets.

The ICCM Cabinets will provide square root extraction and safety outputs to qualified indicators and recorders in the control room. The existing spare L&N recorders located on 3VB1 will be utilized, 3LPIRC0424 for Train A and 3LPIRC0421 for Train B. Pneumatic indicators 3LPIP0037 (Train A) and 3LPIP0030 (Train B) will be replaced with solid state Dixson Bargraph indicators Model SA101AXTX. 3LPIP0030 will be moved from 3VB3 and located beside of 3LPIP0037 on 3UB2. Welding and cutting will be required on 3UB2.

Instrument tubing for the two indicators beside 3LPIP0037 and 3LPIP0030 will have to be rerouted to allow the new indicators to be installed. The two indicators affected are 3LPIP0017 (Decay Heat Removal Disch Header A & B Press) and 3LPIP0018 (RB Emergency Sump Discharge Pressure).

A safety related power source will be provided to both trains for the new indicators and recorders. Train A will be supplied from 3SKJ Bkr 17. This power is also used for 3CCRO018 (Upper Surge Tank Level B) recorder which is located on 3VB1. Sliding links will provide points for individual isolation. This recorder will be out of service to allow the wiring for LPI Flow instrumentation to be added to this circuit.

ON-32587 (Continued)

Train B will be supplied from 3SKK Bkr 17. This power is also used for 3LPILT0132 current loop. Sliding links will be provide points for individual isolation. This will affect the following devices:

3LPICR0401	- Borated Water Storage Tank Level 3 recorder located on 3VB1
3SA3-35	- LP BWST Level High/Low Annunciator located on 3UB2
3ALPI00	- LP BWST LVL #3 Train B (OAC Input A2097)

The ICCM cabinets will also use the inputs from 3LPIFT005P and 3LPIFT004P to supply non-safety outputs to the following devices:

Train A

3ALP024	(A1310) LPI Loop A Flow
3DLP069	(D0733) LPI Loop A Flow High
3DLP068	(D0732) LPI Loop A Flow Low
3SA3-26	LPI Loop A Flow High/Low
3DLP062	(D2217) LP DH Loop A Flow Low
3SA3-8	LP Decay Heat Loop A Flow Low

Train B

3ALP026	(A1311) LPI Loop B Flow
3DLP071	(D0735) LPI Loop B Flow High
3DLP070	(D0734) LPI Loop B Flow Low
3SA3-27	LPI Loop B Flow High/Low
3DLP064	(D2218) LP DH Loop B Flow Low
3SA3-9	LP Decay Heat Loop B Flow Low

3KF Bkr 26 will be used for all of the above listed Train A & B non-safety loads. This power is also used for one of the two space heaters in the enclosure for 3LPILT0002A (LP BWST Level - Train A) which will be required to be de-energized to allow the wiring modification. The heaters are designed to be on when the internal enclosure temperature is below 35°F. Sliding links will provide points for individual isolation.

Six Cutler-Hammer relays and 3 terminal blocks will be added to Misc Term Cab 3MTC4 to provide the contact multiplication required for the non-safety outputs listed above.

SAFETY EVALUATION:

The Low Pressure Injection (LPI) System removes the decay heat from the core and sensible heat from the Reactor Coolant (RC) System during the later stages of a cooldown. The system can also provide auxiliary spray to the pressurizer for cooling, maintain the reactor coolant temperature during refueling, and provide a means for filling and draining the fuel transfer canal. In the event of a loss of coolant accident, the system injects borated water into the reactor vessel for long term emergency cooling. The LPI System,

ON-32587 (Continued)

the Inadequate Core Cooling Monitoring (ICCM) System, and the control room control boards are affected by this modification and are QA Condition 1. The relevant design basis accidents are found in FSAR Sections 15.13 (Steamline Break Accident) and 15.14 (Loss of Coolant Accidents)

The existing flow element and instrument lines leading to the existing transmitters will be used for the new transmitters and are qualified seismically, for materials, and for function, and are environmentally qualified. The new transmitters will be seismically mounted and located in the Penetration Room. The new transmitters are QA 1 and will receive safety-related power from the ICCM cabinets. The new indicators and recorders are QA 1 and will receive safety-related power. A control board seismic review has been performed for the control board changes and the new instrumentation has been seismically qualified. Cables will be run between the transmitters and the ICCM cabinets, from the ICCM cabinets to the control room records and indicators, from the safety-related power source to the recorders and indicators, and from the ICCM cabinets through nonsafety-related outputs to the computer and annunciator points. A 10 CFR 50 Appendix R review has been performed. Low flow from the LPI pumps during decay heat removal 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 technical specifications will not be affected. This instrumentation has no control functions, only a monitoring and alarm function. The new instrument loop is not as accurate as the existing loop, but the impact of the additional error is negligible.

This NSM involves no unreviewed safety questions.

Nuclear Station Modification #
ON-32588 (Unit 3)
ON-12588 (Unit 1)

DESCRIPTION:

This modification will replace the existing non-QA channel of reactor building spray flow with two qualified QA condition I channels of instrumentation in the control room on QA1 indicators. The existing reactor building spray flow instrumentation consists of instrument impulse tubing for A and B train, each train employing two non-QA transmitters. One transmitter in each train provides input directly to a Computer point for the Operator Aid Computer (OAC), and the other transmitter provides input through a square root extractor to a non-QA dual-indicating receiver gauge on the Main Control Board (MCB). The existing design also includes non-QA pressure switches, one per train, which provide input to OAC digital computer points and a control room annunciator. The modification will delete the existing non-QA transmitters and pressure switches and install new QA Condition 1 transmitters, one per train, to receive input from the instrument impulse tubing. The transmitters will then provide a 4-20mA signal to the respective A and B train Inadequate Core Cooling Monitor (ICCM) cabinets. Each ICCM cabinet will provide input to a QA1 electronic indicator on the MCB, a QA1 chart recorder on the MCB, a non-QA receiver gauge (replacement for the existing dual-indicating receiver gauge) on the MCB, non-QA OAC analog and digital computer points, and a control room annunciator.

This modification was initiated as a part of the control room review conducted by Duke Power Company in response to NUREG-0737, and will place Oconee Nuclear Station in compliance with Regulatory Guide 1.97, Rev. 2, with respect to post-accident monitoring of reactor building spray flow. Reactor Building spray flow instrumentation is not discussed in the FSAR or Technical specifications; therefore, the design basis for this modification is Regulatory Guide 1.97 and the subsequent correspondence between the NRC and Duke Power Company.

SAFETY EVALUATION:

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 system consists of redundant transmitters, which provide input to redundant and physically and electrically separate ICCM cabinets, which provide input to redundant MCB indications of reactor building spray flow. Each train is supplied with redundant class 1E power. No single failure within the system can cause the inability of the operators to ascertain the status of reactor building spray flow following an accident.

Regulatory Guide 1.97 provides latitude in the chosen method of

display of various parameters, by stating that the method of display may be dial, digital, CRT, or stripchart recorder indication. This modification exceeds the requirement by providing two methods of display for each train. Reactor building spray flow is a type "A" variable (variable which provides information required to permit the control room operator to take specific manually controlled safety actions), and the instrumentation is required to have a span of 0-110% of the design flow. Design flow is approximately 1500 gpm, and the instrument span is 0-2000 gpm. Therefore, the design of the instrument and indication is in conformity to Regulatory Guide 1.97, Rev. 2.

The ICCM cabinets internal circuitry accomplishes the required electrical and physical separation of the QA portions of the ICCM output and the non-QA portions of the ICCM output (MCB receiver gauge, annunciator, and computer points). An appendix R fire review has been performed for the new cabling. No penetrations will be utilized, as the transmitters are located in the auxiliary building. The transmitters are qualified for the environment they will be exposed to, and will be seismically mounted. The new instruments will be seismically mounted on the MCB, and a control board review for seismic effects was performed.

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, this instrument line and other Regulatory Guide 1.97 instrument line applications were reviewed in Design Study ONSD-0217. It was concluded that the instrument lines for control room indication are satisfactory for seismic loadings and safe from non-seismic pipe interactions.

As a result of this modification:

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

   No. No equipment associated with or affected by this modification is an accident initiator.

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

   No. This modification will upgrade reactor building spray flow instrumentation and indication to Regulatory Guide 1.97 standards so as to ensure availability for operator indication following a DBA.

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

ON-32588 (Continued)
ON-12588 (Continued)

No. No new failure modes are postulated.

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

No. The configuration installed by this modification is a is a Regulatory Guide 1.97 upgrade, and is therefore able to be relied upon for important operator information. The system is not susceptible to a single failure which would cause inoperability of both trains. Additionally, no adverse system interaction between equipment installed by this modification and already existing equipment is deemed to exist.

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

No. The upgrade will ensure that no single failure can cause the inability of the instrumentation to provide indication of important parameters to the operator.

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

No. No new failure modes are postulated.

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

No. No setpoints, limiting safety system settings, or accident mitigation equipment is affected by this modification.

CONCLUSIONS:

This modification does not involve any USQs or safety concerns. No Technical Specification changes are required.

Nuclear Station Modification #
ON-32589 (Unit 3)

DESCRIPTION:

Upgraded the High Pressure Injection (HPI) flow instrumentation, Loops A and B to QA condition 1. Both loops are displayed and recorded in the control room. Add parallel controls for valves 3HP-25 and 3HP-27 to the control boards. Upgrading the HPI flow instrumentation to QA-1 meets the requirements of Regulatory Guide 1.97 type "A" variable. The addition of parallel controls for valves 3HP-25 and 3HP-27 meet the intent of HED 0-3-0002.

Existing instrumentation, two DP transmitters one square root extractor, two pressure switches and one indicator per loop of HPI flow were removed. One qualified transmitter, indicator and recorder point was added to the HPI flow instrumentation loop. Additional controls for valves HP-25 and HP-27 were added to the control boards. This was accomplished by paralleling existing controls on the control boards. Replace flow transmitters 3HPIFT0159 and 3HPIFT0160.

Upgrading the HPI flow instrumentation to QA-1 with qualified components and the use of safety power, provides the operators with instrumentation that is qualified for post-accident monitoring.

The addition of controls for valves 3HP-25 and 3HP-27 to 3UB1 provides the operator control of the valves at or position near the HPI flow indication.

SAFETY REVIEW:

The Hiqh Pressure Injection (HPI) System. is designed to accommodate the following function (s) during normal operation:

a) Supply the Reactor Coolant System (RCS) with fill and operational makeup water.

b) Provide seal injection water for the reactor coolant pumps.

c) Provide for purification of the reactor coolant to remove corrosion and fission products.

d) Control the boric acid concentration in the reactor coolant.

e) In conjunction with the pressurizer, the system will accommodate temporary changes in rector coolant volume due to small temperature changes.

f) Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the RCS.

ON-32589 (Continued)

g) Provides continuous flow for cooling the normal HPI nozzles to minimize thermal shock.

h) Provides auxiliary pressurizer spray control for cooldown when normal pressurizer spray is unavailable.

The HPI also provides RCS makeup under abnormal loss Of RCS inventory conditions.

The HPI System, the Inadequate Core Cooling Monitorinq (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 existing flow element and instrument lines leading to the existing transmitters will be used for the new transmitters and are qualified seismically, for materials, and for function, and are environmentally qualified. The new transmitters will be seismically mounted and located in the Penetration Room. The new transmitters are QA 1, environmentally qualified for the harsh environment of the penetration room, and will receive safety-related power from the ICCM cabinets. The new indicators and recorders are QA 1, located in a mild environment and will receive safety-related power. A control board seismic review has been performed for the control board changes and the new instrumentation has been seismically qualified. Cables will be run between the 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 nonsafety-related outputs to the Operator Aid Computer and annunciator points. A 10 CFR 50 Appendix R review has been performed. Under emergency conditions, HPI high or low flow conditions are alarmed by control room annunciators. This modification will not change the function or operation of the HPI flow instrumentation or the operation of the HPI System. Technical specifications will not be affected. This instrumentation has no control functions, only monitoring and alarm functions. The new instrumentation loop accuracy is consistent with the existing system requirements.

May the modification:

1) Increase probability of an accident evaluated in the SAR?

No. The HPI flow instrumentation is not the initiator of the accidents described in FSAR Sections 15.9, 15.13 or 15.14.

2) Increase the consequences of an accident evaluated in the SAR?

NO- HPI high flow and other alarms are not changed by this

ON-32589 (Continued)

modification and the new HPI flow instrument loop has been verified to ensure adequate indication of HPI flow.

3) Create the possibility for an accident of a different type than any evaluated in the SAR?

No. This instrumentation has only monitoring and alarm functions and does not input to any controls. The new instrumentation loop is qualified QA 1.

4) Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?

No. The HPI flow instrumentation is not the initiator of any FSAR evaluated equipment malfunction. The new instruments are QA 1 and have been seismically qualified and the control board changes have been seismically reviewed.

5) Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR?

No. Refer to the response to Question 2.

6) Create the possibility of a malfunction of a different type than any evaluated in the SAR?

No. Refer to the response to Question 3.

7) Reduce the margin of safety as defined in the basis for any technical specification?

No. This modification does not involve any plant safety limits, set points, or design parameters.

Nuclear Station Modification #
ON-32592 (Unit 3)

DESCRIPTION:

This modification will replace the existing two thermocouples (TE-209 & TE-210) on decay heat cooler discharge with once which are environmentally qualified for their location. The indicating range will also be changed from 0-300 F to 0-400 F.

SAFETY REVIEW:

The Low Pressure injection (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 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 decay heat cooler discharge temperature monitoring 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 the decay heat cooler discharge temperature monitoring instrumentation as a Type D variable and category 2, with a specified range of 32-350 P. This variable is used to monitor operation and for analysis. A Type D variable provides information to indicate the operation of individual safety systems and other systems important to safety. Category 2 generally applies to instrumentation designated for indicating system operating status. The design criteria for Category 2 (Type D variable) includes environmental qualification; seismic qualification if the instrumentation is part of a safety related system; energized from a high-reliability power source; display on an individual instrument or processed for display an a CRT or other appropriate means; and method of display by dial, digital, CRT or stripchart recorder indication. Redundancy for Category 2 is not required. Instrumentation being upgraded for accident monitoring in not to degrade the accuracy and sensitivity required for normal operation.

The new instrument will be calibrated to have a range of 0-400 F. The currently non-qualified instruments are being replaced with QA Condition 1, environmentally qualified instruments with a non-safety related battery packed power source. This instrumentation is for monitoring and analysis, and performs no automatic functions. The modification does not physically remove or replace the meters. The modification is re-calibrating the instrument loop and replacing the meters' scales. These scale changes do not adversely affect the components' quality, environmental qualification, or method of display. The display is from two control board indications having

dual instrument readout. The method of display is dial and digital. The mounting for the meters is not adversely affected since the instrument loop calibration and new meters' scales do not affect the components' weight or weight distribution. Since the control room components' weight and weight distribution are not changed, no adverse seismic effects to the control board are created. The failure modes for the instruments and meters are not changed. The instrumentation's accuracy and sensitivity for normal operation monitoring is not degraded. The instrumentation readout is within 1 F of the current instrumentation's accuracy. This increased error is considered acceptable for the decay heat cooler discharge temperature monitoring function. The decay heat cooler discharge temperature instrumentation meets the criteria of Regulatory Guide 1.97.

Currently only one instrument output goes to a control room annunciator. Variation Notice OC-4823 will connect both instruments' output to this annunciator. Regulatory Guide 1.97 does not require the annunciation of this instrumentation. A 10 CFR 50 Appendix R review for electrical separation vas performed for this NSN.

As a result of this modification:

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

   No. This change does not create any conditions or events which lead to accidents previously evaluated in the SAR.

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

   No. The instrumentation and meters' ranges include those specified and reviewed by the NRC. The meters perform no automatic functions. The instrumentation is for monitoring and for analysis only.

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

   No. No accidents different than already evaluated in the SAR are postulated.

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

   No. The instrumentation and meters' ranges include those specified and reviewed by the NRC. The instrument is OA Condition 1 and environmentally qualified. This modification does not physically remove or replace the meters. The modification is re-calibrating the instrument loop and replacing the meters' scales. These changes do not adversely

ON-32592 (Continued)

affect the components quality, environmental qualification, or method of display. The mounting of the meter is not adversely affected since the instrument loop calibration and new meters' scales do not affect the components' weight or weight distribution. Since the components, weight and weight distribution are not changed, no adverse seismic affects to the control board are created. The failure modes for the meters are not changed. The instrumentation's accuracy and sensitivity for normal operation is not degraded. The method of display and the display meet the requirements of Regulatory Guide 1.97. A 10 CFR 50 Appendix R review for electrical separation was performed.

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

   No. No additional operating components are added. Components are only being replaced. The failure modes of the instruments and meters are not changed.

6) May the possibility of malfunctions of equipment important to safety different than any already evaluated in the SAR be increased?

   No. No new failure modes are postulated.

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

   No. This modification does not adversely affect any plant safety limits, set points, or design parameters.

This modification involves no USQs or safety concerns. No technical specification changes are required.

Nuclear Station Modification #
ON-32665 (Unit 3)

DESCRIPTION:

This modification will install several new motor control centers (MCCS) and power panelboards and then transfer existing Unit 3 loads on to them. The new MCCs and panelboards will be powered from the new auxiliary power system to be installed by NSM ON-52665. The loads to be moved are all nonsafety. The new MCCs and panelboards will be nonsafety but will be mounted QA4. One MCC and the panelboard will be installed in the Reactor Building and will be fed through electrical penetrations. Other than the removal of loads, the existing unit power systems will be unaffected by this modification.

SAFETY REVIEW:

This modification will affect the existing unit power systems, the new auxiliary power system installed by NSM ON-52665, and several nonsafety unit loads. The existing power systems will have nonsafety loads removed and will be otherwise unaffected. The new auxiliary power system is nonsafety and has the capacity to feed the transferred loads. The auxiliary power system has normal and alternate feeds from the switchyard and, therefore, will serve as a reliable source of power for the nonsafety loads to be transferred. The circuit breaker protection will be coordinated. Since this does not degrade plant loads, the accidents discussed in FSAR Section 15.8 are not adversely affected.

The new panelboards and MCCs will be mounted QA4 to prevent seismic interaction with surrounding equipment. The MCC and panelboard to be mounted in the reactor building will be fed through existing electrical penetrations. These will only be used during an outage and will be deenergized at all other times. The circuit breaker feeding the MCC will be opened and locked in that position so that the penetration cables will be deenergized during unit operation. At the end of each outage, these penetrations will be inspected for damage. A 10 CFR 50 Appendix R review has been initiated for the cabling required in this modification.

USO Evaluation:

As a result of this modification:

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

   No. All of the loads to be transferred are nonsafety. The circuit breaker protection will be coordinated and the new auxiliary power system is a reliable source of power.

ON-32665 (Continued)

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

   No. All loads to be transferred are nonsafety. Removing loads from existing load centers will have a positive effect on the existing system voltages. The electrical penetrations will be inspected after each outage and will be deenergized during unit operation.

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

   No. This will change only the power source to several nonsafety loads. The new auxiliary power system is a reliable source of power.

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

   No. All loads to be transferred are nonsafety. Circuit breaker protection will be coordinated and MCCs and panelboards will be mounted QA4. A 10 CFR 50 Appendix R review has been initiated. Safety-related equipment may benefit from increased voltages due to the load transfers.

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

   No. Refer to the answer to question 4.

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

   No. Refer to the answer to question 4.

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

   No. No key safety parameters or set points will be affected.

<u>Conclusion</u>:

This NSM involves no safety concerns or USQs.

Nuclear Station Modification #
ON-52665

DESCRIPTION:

This modification will install a new auxiliary power system that will be powered from the switchyard and will feed nonsafety station and unit loads. Several loads will be taken off existing buses and placed on this new auxiliary power system. NSM ON-52665 will install the power system major components (circuit breakers, transformers, 4.16kV switchgear and 600V load centers (LCs) and move existing loads to the new 4.16kV switchgear. This NSM will also install the bases and mountings for the Motor Control Centers (MCCs) to be installed under NSMs ON-12665 and ON-32665. MSMs ON-12665, ON-22665, and ON-32665 will move existing 600V loads, and install and correct new motor control centers to the new LCs.

This modification will install two transformers in the switchyard. One transformer will be fed from the 203kV switchyard red bus through PCB No. 4. PCB 4 was purchased under another NSM and will be installed by NSM ON-52665. The other transformer will be fed from the 230/525kV autotransformer tertiary winding through a circuit breaker. These transformers will feed separate 4.16kV switchgear in a common enclosure to be located outdoors near the Unit 3 transformer yard. The switchgear feeder cables and circuit breaker control cables will be in a common trench under the security fence. Each transformer will feed a separate 4.16 kV bus. Loss of one of these transformers will cause the connected switchgear to automatically transfer to the remaining transformer. New control switches will be mounted on the electric boards in the Unit 1 and 2 control room. The two 4.16 kV buses will each feed two 600V load centers (LCs). Three of these LCs will be for unit specific loads and each will be mounted on the mezzanine floor near the respective unit 4.16kV switchgear. The fourth LC will be mounted outside on the west side of the plant near Unit 3. Nonsafety unit and station loads will be moved from existing buses to these new buses. Other than the removal of loads, the existing station and unit power systems will be unaffected by this modification. PCB 4 controls and protective relaying will receive 125Vdc power from the safety-related 205kV switchyard batteries. The X-phase of the safety related Keowee overhead line will be moved several feet to obtain adequate clearance between the line of transformer 4T related equipment.

SAFETY REVIEW:

This modification will affect the 230kV switchyard red bus and 125Vdc power system, Keowee overhead transmission line, 230/525kV autotransformer tertiary winding, existing unit power systems, control room electric boards, and several nonsafety-related unit and station loads. Some safety-related buses will have nonsafety loads removed but will be otherwise unaffected and no other safety-related equipment will be affected. All of the cable tray used to support the LC feeders will be mounted over nonsafety-related

equipment only so there are no seismic interaction concerns. The unit LCs which will be mounted near the respective unit 4.16kV safety-related switchgear, will be mounted QA4. The MCC bases design and mounting will be QA4. The addition of control switches on the electric boards in the control room has been reviewed with regard to the electric board seismic mounting. The 230kV switchyard 125Vdc power system has enough capacity for the additional loads associated with the PCB 4 controls and protective relaying. The nonsafety PCB 4 control and protective relaying power will be separated from safety related dc powered equipment by a dc circuit breaker which will serve as the isolation device. Cable separation requirements will be maintained. Strain relief supports for the Keowee line X-phase conductor will be attached to the existing switchyard towers. Any additional loads on the towers due to this conductor adjustment will be insignificant and this will not affect the operation of the Keowee line. There are no applicable design basis accidents.

The double transformer and 4.16kV bus arrangement will prevent a single failure from deenergizing either bus. To prevent security concerns, the switchgear feeder cables will be direct buried where it passes under the security fence. A 10 CFR 50 Appendix R review has been performed and the fire protection of the new equipment to be installed has been reviewed.

USQ Evaluation:

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

   No. A malfunction of this new system would not initiate either of the loss of electric power accidents evaluated in FSAR section 15.8. Supplying power to station loads from two additional switchyard sources decreases the probability that all station loads will be separated from the transmission system.

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

   No. Refer to the answer for question 1. The three unit LCs will be mounted QA4. The MCC bases design and mounting will be QA4. Operation of the Keowee overhead line will not be affected by the movement of the X-phase conductor. The electric board seismic mounting will be adequate after the installation of the control switches.

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

   No. This will decrease the loading on existing buses, thereby reducing the potential for an equipment malfunction due to low voltage. The double bus arrangement of the new system prevents

a single failure from deenergizing either of the new 4.16kV bus.

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

   No. The loading on some existing safety buses will be decreased, thereby reducing the potential for a malfunction of equipment due to low voltage. Safety-related equipment will be otherwise unaffected. The new load centers to be installed near the 4.16kV switchgear will be mounted QA4 to prevent seismic interaction and a 10 CFR 50 Appendix R review has been performed. The operation of the 230kV switchyard 125Vdc power system will not be adversely affected by the addition of the PCB 4 loads. The nonsafety-related dc loads associated with PCB 4 will be isolated from the safety-related dc loads and cable separation requirements will be met. All cable tray will be mounted over nonsafety-related equipment.

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

   No. Refer to the answer to question 4.

6. May the possibility of malfunctions of equipment important to safety different than any already evaluated in the FSAR be created.

   No. Refer to the answer to question 3.

7. Will the margin of safety as defined in the bases to any Technical Specification be reduced?

   No. This will not adversely affect any key safety parameters or design limits.

<u>Conclusion</u>:

This NSM involves no safety concerns or USQs. No Technical Specification changes are required.

Nuclear Station Modification #
ON-12817 (Unit 1)

DESCRIPTION: NSM ON-12817 will add the ATWS Mitigation Systems Actuation Circuitry (AMSAC) and the Diverse Scram System (DSS).Existing pressure switches for the Main Feedwater pump discharge pressure and the Main Feedwater pump turbine control oil pressure will be used as input for the ASMAC logic. Wide range RCS pressure transmitter loops RCPT0244 and RCPT0245 will be modified to supply a signal to the appropriate channels of the DSS. All instruments to be added are non-safety related, and all other equipment is nonsafety with the exception of the isolation devices used to separate safety and non-safety signals. Two channels of automatic control will be established with individual programmable logic controllers (PLCs). Status and alarm indications will be added to the operator control board.

SAFETY EVALUATION: The AMSAC and the DSS are being added to meet the requirements of 10CFR50.62 on ATWS mitigation. The system is installed as a non-safety backup to the Reactor Protection System (RPS) and is not required to mitigate any of the FSAR, Chapter 15 accidents. The ATWS system is separated both physically and electrically from any safety-related system, and meets the requirements of the B&W Owners Group Generic ATWS Design Basis Document (10-09-85) approved by the NRC in 1988.

With the present design of the AMSAC and the DSS, the possibility of spurious actuation does exist. It is possible to actuate Emergency Feed Water (EFW) while main feed water is still available, and the DSS design could allow control rod groups to drop while the reactor is at full power. While this does not constitute an unsafe condition, it may create an unnecessary challenge to the Reactor Protection System (RPS). In both of the cases mentioned above, a spurious actuation would require a combination of extremely unlikely events to occur. The logic for the AMSAC and DSS is "two out of two", so in order to actuate, it must receive positive signals from two totally separate trains, and the probability of proceeding to trip or EFW actuation due to normal occurrences is minute.

The possibility for inadvertent actuation of the ATWS Mitigation System does exist. However the probability of inadvertent actuation is small due to the two-out-of-two logic needed for initiation. Also, the NRC has reviewed the design of the system and has issued its approval in the form of a Safety Analysis Report on Oconee's plant-specific submittal. Based on this prior NRC approval and the fact that the appropriate analyses have been performed for additions to the plant (i.e., seismic analysis and Appendix R review), it is determined that this modification does not involve any unreviewed safety questions.

Nuclear Station Modification #
ON-32849 (Unit 3)

DESCRIPTION: These urgent modifications replace flex hoses on the Auxiliary Service Water (ASW) portion of the Emergency Feedwater system with hard pipe. The flex hoses have not been reliable in service. They have leaked and blown completely on several occasions. The replacement piping will be carbon steel, which is compatible with existing ASW piping.

SAFETY EVALUATION: The low head Auxiliary Service Water (ASW) System is a backup system used to provide steam generator decay heat removal. This system is of particular importance during a tornado scenario, where the Auxiliary Service Water pump provides decay heat removal for all three units. Redundant trains of ASW for each steam generator are achieved by routing one train through the East Penetration room and one train through the West Penetration room. Also, the flex hose being replaced by this urgent modification serves no active.protection function during a tornado (such as the decoupling of connected piping to prevent further damage).

None of the Technical Specifications applicable to secondary side decay heat removal are adversely affected and the modification does not adversely affect any plant safety limits, design parameters, or setpoints.

Based on the above discussion, it is determined that this modification involves no safety concerns or unreviewed safety questions and no FSAR or Technical Specification changes are required.

Nuclear Station Modification #
ON-52855

Description:

A potential problem has been identified in which overloading of the Keowee generator could occur due to the presence of operating reactor coolant pumps during a postulated Loss Of Coolant Accident coincident with Loss of Offsite Power (LOCA/LOOP). Per the as-built Keowee emergency power configuration, either of the two Keowee generators may supply power to the 230 KV switchyard. If a Keowee emergency start signal is initiated, the Keowee air circuit breakers (ACBI for generator #1 and ACB2 for generator #2) automatically trip to ensure separation of the applicable Keowee Unit from the switchyard. Positive indication of switchyard isolation provides a close permissive to the tripped ACB, and after a 0.5 second time delay, one ACB closes to align-power to the overhead path. Closure of the ACB to the overhead path associated with the Keowee Unit aligned to the underground feeder is blocked. If the Keowee Unit not aligned to the underground feeder supplies power to the startup (CT) transformers through the 230 KV switchyard due to closure of its associated ACB, the reactor coolant pumps will not have tripped (due to bus undervoltage condition), as the total reactor coolant pump undervoltage relay time delay is longer in duration than the ACB time delay prior to breaker reclosure. The reactor coolant pumps trip when both normal and startup breakers are open for approximately 3.6 seconds. The LOCA loads on one Unit plus the hot shutdown loads on the other two Units is equal to 20.628 MVA, while one Keowee Unit is capable of 87.5 MVA. However, the reactor coolant pumps constitute a load of approximately 30 MVA for each Unit, and the combined LOCA, hot shutdown, and RCP loads exceed the capacity of a single Keowee Unit.

Upon failure of the ACBI (ACB2) to open when an ES signal is initiated, an already running Keowee Unit would fail to separate from the 230 KV switchyard (and consequently the CT transformers), and power would be supplied to the 6.9 Kv buses, causing the reactor coolant pumps to continue to operate and overload the operating Keowee unit.

This modification provides logic such that if the Keowee Units are generating power to the 230 KV switchyard, and a LOCA/LOOP occurs, the Keowee Unit realigned to the overhead path will not be overloaded. During a LOOP, both ACBs and PCB-9 will trip on switchyard isolate. Two timers will be added to the PCB-9 reclose logic, and PCB-9 will reclose after approximately 4 seconds. The ACB for the Keowee Unit not aligned to the underground feeder will reclose after approximately 4 seconds contingent upon a switchyard isolate signal being present. This will allow time for the operating reactor coolant pumps to trip, and prevent the potential for overloading of a Keowee Unit.

SAFETY REVIEW:

An Appendix R review has been initiated. There are no AppendiX R concerns because all cabling and all necessary electrical interlocks will be in the relay house. No penetrations are necessary and no core drills will be performed. All components either installed as a part of this modification or directly affected by it are located in a mild environment area. The portions which interface with QA1 components are required to function during seismic events. For Oconee Nuclear Station, a LOCA and a MHE (Maximum Hypothetical Earthquake) are assumed to occur simultaneously. Therefore, it is necessary for Keowee to provide emergency power to the Oconee 4160V buses during a seismic event and while withstanding the most limiting single active failure.

Since a LOOP event will cause both ACBs and PCB-9 to open, redundancy will be provided if either of the ACBs or PCB-9 fail to open. PCB-9 has two redundant trip coils, and two redundant signals of switchyard isolate will be used in the trip circuit of PCB-9. This will prevent a single failure of a switchyard isolation signal from affecting both the ACBS and PCB-9. If either the ACB for the Keowee Unit aligned to the overhead path or PCB-9 fails to reclose, this would constitute the most limiting single active failure, and emergency power would be available to the 4160V bus via the underground feeder. If the underground feeder fails to supply power to the 4160V bus due to failure of any component in its path (e.g., Keowee Unit, ACB3 (ACB4), 4160V bus incoming breaker, etc.) this would constitute the most limiting single active failure, and power would be supplied to the 4160V bus via the overhead path.

The time delays associated With ACBI and ACB2 are being changed from 0.5 seconds to 4 seconds, and two timers are being added to PCB-9 which will provide a permissive to reclose PCB-9 after 4 seconds. Since this is longer in duration than the reactor coolant pump undervoltage relay time delay, adequate time will be provided for the reactor coolant pumps to trip (the combined undervoltage relay/timer times is equal to 3.64 seconds including tolerances). One Keowee Unit will then be capable of supplying the plant loads. During a LOCA-only accident, this modification does not affect the alignment of power or time delays associated with energizing the yellow bus.

PCB-9, 18 (Unit 1), 27 (Unit 2), and 30 (Unit 3), closure was previously required for switchyard isolation confirmed. An item identified in the 230 KV Design Basis Documentation effort was that since all three Unit's PCBs (which supply power from the yellow bus to the CT transformers and the 4160V buses) must close to confirm switchyard isolation (switchyard isolation is a reclose permissive for the Keowee ACBs), if a single breaker fails, the other two units are being unnecessarily penalized for a single unit breaker's failure. The correction of this item is not critical to the Keowee

emergency power problem created if the reactor coolant pumps continue to operate; but this item is being corrected during and as a part of this modification. Since the confirmation of the closing of PCB-9 is being removed by this modification, the other contacts associated with breakers which are supposed to close (PCB-18, 27, and 30) in the switchyard isolate complete circuitry are also being removed. Since the yellow bus is isolated from the grid when the PCBs trip (8, 12, 15, 17, 21, 24, 26, 28, and 31), switchyard isolate complete signal can be generated regardless of the position of PCB-9, 18, 27, and 30.

The failure of some of the reactor coolant pumps to trip is postulated to ascertain the effects on one Keowee Unit to supply power to plant loads. Since a Keowee Unit is rated at 87.5 MVA, it is not essential that all reactor coolant pumps trip to prevent overloading of the Keowee unit. LOCA loads for one Oconee Unit and hot shutdown loads for the other two Oconee Units comprise a total of approximately 21 MVA. The reactor coolant pumps comprise a total of 30 MVA Per Oconee Unit. Since a load of approximately 66 MVA remains available, 8 reactor coolant pumps could fail to trip and Keowee would still be able to supply adequate power to the plant. Only 4 reactor coolant pumps must successfully trip to accomplish the desired configuration.

Sensitivity studies have shown that for a large break LOCA, the with the highest peak clad temperature results from the cases with the reactor coolant pumps running. This is so because with the reactor coolant pumps operating, reactor coolant system inventory depletion is increased. Therefore, this modification will have a positive effect on plant response to a LOCA/LOOP transient by tripping the reactor coolant pumps. The LOCA-only transient will remain the same, i.e., the reactor coolant pumps will continue to operate, and no fuel damage occurs. Additionally, the worst case small break LOCA scenario involves continued operation of the reactor coolant pumps. This modification would have a positive affect on the small break LOCA response. If a LOOP does not occur coincident with a small break LOCA, the current analysis remains bounding, in which we assume credit for operator action to trip the reactor coolant pumps in the small break LOCA-only transient upon loss of subcooling.

For non-LOCA Unit response, there are two possibilities: 1) the Unit sustains a runback to 15% reactor thermal power, or 2) the Unit trips due to exceeding limiting safety system settings. Although it is highly unlikely that the unit would sustain a load rejection on a LOOP event, it is analyzed and fully bounded. For this transient it is assumed that the reactor coolant pumps arefed from the unit generator, and the loss of load transient does not result in any fuel damage or excessive reactor coolant system pressure. if a unit trip occurs on the non-LOCA Unit, it is analyzed and fully bounded by the Loss of All Station Power scenario. For this transient, the reactor coolant pumps are not

operating, and reactor coolant system flow decays without fuel damage occurring.

Oconee Technical Specifications requires that PCB-9 be capable of supplying power to the 230 KV yellow bus. This modification does not require a change to the current requirements for operability contained in the Technical Specifications. The current requirements are correct and adequate.

USO EVALUATION:

As a result of this modification:

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

   No. No equipment associated with the Keowee Emergency Power System is an accident initiator.

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

   No. This modification will ensure that the reactor coolant pumps trip upon LOCA/LOOP, and thus have a Positive impact on large break and small break LOCA response.

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

   No. This modification involves accident mitigation equipment only. See response to question #1.

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

   No. This modification will remove the susceptibility of the Keowee Emergency Power system to overload conditions, and will add logic to account for single failure considerations. All components installed under this modification are in a mild environment, so no EQ concerns exist.

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

   No. The failure modes and effects analysis contained in this calculation yields the conclusion that no single active failure can cause the inability of the Keowee Emergency Power System to perform its intended function.

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

ON-52855 (Continued)

No. No new failure modes are postulated.

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

No. Unit large break and small break LOCA/LOOP response is improved under this modification.

Conclusions:

This modification does not involve any USQs or safety concerns. No Technical Specification changes are required.

Exempt Change OE-2873

DESCRIPTION:

This modification involves the following circuit breakers in the indicated load centers (LCs) and motor control centers (MCCs):

LC		Breaker	MCC		Breakers
LC	1X8	6A	MCC	1XS1	F3A
LC	1X9	6A	MCC	1XS2	F3A
LC	IX9	6B	MCC	1XS3	3A
LC	2X8	6A	MCC	2XS1	F3A
LC	2X9	6A	MCC	2XS2	F3A
LC	2X9	6B	MCC	2XS3	3A
LC	3X8	6A	MCC	3XS1	F3A
LC	3X9	6A	MCC	3XS2	F3A
LC	3X9	6B	MCC	3XS3	3A

These LCs and MCCs are safety-related. The circuit breakers involved are those which would supply backup power for the involved MCCs if the MCC normal power source is lost. The swapover from normal source to backup source is a manual operation. In order to prevent a single failure from damaging both MCC incoming breakers and to prevent both sources from being closed in at the same time, these MCC breakers are locked in the open position and the LC breakers are racked out. This is the condition of these components during normal plant operation.

Operations at the plant considers these separation measures excessive and they put an unnecessary burden on operators trying to restore power to a MCC.

This modification will replace the requirement that the LC breakers be racked out and the MCC breakers be locked open with the requirement that all of these breakers be placed in the open position. This modification will affect only these specified breakers and the associated 1-line drawings.

SAFETY REVIEW:

This modification will affect the breakers listed in the description section of this evaluation. All involved breakers are safety-related (QA-1). The function of these breakers is to provide backup power to the associated safety-related MCC upon a failure of the normal power source. These breakers are not used during normal operation, but only if there has been a failure of the normal power source.

This modification will not change the function of the breakers involved. Both the LC breaker and MCC breaker on a backup feeder will be kept open to prevent a fault or other failure from affecting the backup power source. In order for the LC to be connected to the associated MCC, it will take two independent actions.

Exempt Change OE-2873 (Continued)

USO EVALUATION:

As a result of this modification:

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

   No. The breakers involved with this modification are not accident initiators. These breakers are used only after a failure of a MCC normal power source.

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

   No. This modification will facilitate restoring power to a MCC if the normal power source is lost.

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

   No. This equipment is not an accident initiator. No new failure modes are postulated.

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

   No. This modification maintains the needed separation between a MCC and the backup power source. Thus, system design against single failures is maintained.

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

   No. See the response to question 2.

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

   No. No new failure modes are postulated.

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

   No. No plant safety limits, set points, or design parameters will be affected.

CONCLUSIONS:

No Technical Specifications changes are required. This modification does not involve any USQs or safety concerns.

Problem Investigation Report (PIR) #4-091-03

Purpose:

The purpose of this calculation is to determine if plant operation with the manual disconnect switches for one Keowee emergency power underground feeder ACB open will create any unreviewed safety questions (USQs) using the criteria of 10CFR50.59, paragraph (a) (2). This calculation is QA condition 1 because it determines the presence or absence of a USQ.

Background Information:

INPO (Institute of Nuclear Power Operations) Operations and Maintenance Reminder 371, "Single Failure Potential in Safeguards Switchgear Tie Breaker", has been evaluated and deemed to be applicable to Oconee Nuclear Station Emergency Power System. The O&MR describes a situation in which a postulated single failure of a breaker could cause two emergency power sources to be connected out of phase, possibly resulting in the damage of the two power sources. Keowee circuitry is such that if one Keowee Unit is aligned to the underground feeder through its respective ACB (ACB-3 for Keowee Unit 1 and ACB-4 for Keowee Unit 2), the associated unit's ACB to the overhead path (ACB-1 for Keowee Unit 1 and ACB-2 for Keowee Unit 2) cannot automatically close. Aligning one Unit to the underground feeder also blocks the other Unit from aligning to the underground feeder. Therefore, one Keowee Unit is aligned to the underground feeder and the other Unit will automatically align to the overhead path upon completion of switchyard isolate. However, the specific scenario for Oconee is that with one of the Air Circuit Breakers (ACBs) to the underground feeder closed, the other ACB may close from a failure and tie the two Keowee generators together while out of phase. In this scenario, a single failure could cause the inability to supply emergency power to Oconee during accident sequences by possibly damaging the Keowee generators.

The recommended condition of operability is that the manual disconnect switches (operated from Keowee) for the ACB which is not aligned to the underground feeder be opened to prevent the possibility of a malfunction causing the Keowee units to tie together while not in phase.

Safety Review:

As required by Oconee Technical Specifications, one Keowee Unit is to be aligned to the underground feeder by its associated ACB being closed and the other Keowee unit is to be operable and capable of supplying power through the overhead path. The Keowee Unit not aligned to the underground feeder will have its manual disconnect switches in the open position for the underground feeder ACB. If the Unit aligned to the overhead path (either during power supply to the grid or by automatically closing following a LOCA/LOOP accident) experiences a failure the other Keowee Unit will supply emergency power through the underground feeder. If the Unit aligned to the

underground feeder experiences a failure, the other Keowee Unit will automatically supply emergency power through the overhead path (Keowee main step-up transformer, PCB-9, the yellow bus, and each Oconee Unit's start-up transformer). With the underground feeder ACB manual disconnect switches open for the Keowee Unit which will align to the overhead path, no single failure of the ACB can prevent emergency power supply to Oconee. The Oconee FSAR contains a single failure analysis for the Keowee emergency power system which sates that upon loss of the 13.8 kV underground feeder circuit breaker (ACB), cables, or transformer (CT4), "One circuit of emergency power would be lost; however, both units could supply emergency power over the 230 kV overhead line". This operating alignment does not defeat any automatic functions or alter the intended operation of the system, because electrical interlocks currently exist to prevent the ACBs from both Keowee Units from aligning to the underground feeder. The scenario in which both Keowee Units are aligned to the underground feeder due to inadvertent operator action closing the ACB for the second Keowee Unit is also prevented by the manual disconnect switches for one Keowee Unit being in the open position. The disconnect switches would not be inadvertently closed by the operator, as they can only be operated from Keowee. Therefore, Keowee would be capable of fulfilling its design function in a LOCA/LOOP event or LOOP only event.

It is not necessary to consider a tornado concurrent with a LOCA, in which the switchyard is rendered unavailable. Also, while loss of all AC power to Oconee is an analyzed transient in the Oconee FSAR, it is not a design basis accident, and therefore a single failure need not be postulated concurrent with a tornado or any other design basis event which would render the switchyard unavailable. The station can sustain a loss of all AC power for 106 minutes before core uncovery begins, and this transient has been reviewed by the NRC and found acceptable.

USQ Evaluation:

As a result of this conditional operability evaluation:

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

   No. The disconnect switches for the underground feeder ACBs are not accident initiators.

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

   No. The Keowee generators (and associated ACBs and cabling) will be available to provide power to accident mitigation equipment. This condition of operability is an enhancement of system operation by precluding an existing failure mode.

PIR #4-091-03 (Continued)

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

   No. This equipment is not an accident initiator. No new failure modes are postulated.

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

   No. The disconnect switches (for the underground ACB) being open for the Keowee unit not aligned to the underground feeder neither adversely affects a) the Keowee unit which is aligned to the underground feeder to supply power through the overhead path. Thus, system design against single failures is maintained.

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

   No. In the case of any single failure in the Keowee emergency power system, the opposite Keowee unit will still supply power through its respective path.

6) May the possibility of malfunctions of equipment important to safety different than already evaluated in the SAR be created.

   No. No new failure modes are postulated.

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

   No. Emergency power will be provided to accident mitigation equipment in the design basis accidents with a concurrent single failure. The operation of accident mitigation equipment is not altered. This conditional operability evaluation does not adversely affect any safety limits, limiting safety system settings, or design parameters.

Conclusions:

This conditional operability does not involve any USQs or safety concerns. No Technical Specifications changes are required. Table 8-3 of the Oconee FSAR will require revision, in which it will not be true that "the underground feeder could be transferred by the Oconee operator to the running unit". This action will not require the involvement of Keowee personnel.

Nuclear Station Modification #
ON-12447 (Unit 1)

Description:

This modification adds 4 channels of QA1 instrumentation for the steam generator outlet steam pressure, consisting of two channels per steam generator. The new transmitters will tee off existing instrument impulse lines for steam pressure transmitters, and provide input to the Inadequate Core Cooling Monitoring (ICCM) cabinets. The ICCM cabinets then provide input to the control room indications. The indication for both channels of instrumentation on each steam generator will be displayed on the main control board, and one channel per steam generator will also provide input to a QA1 recorder to be located in the main control room.

This modification was initiated as a part of the control room review conducted by Duke Power Company in response to NUREG-0737, and will place Oconee Nuclear Station in compliance with Regulatory Guide 1.97, Rev. 2, with respect to post accident monitoring of steam generator pressure. Steam generator pressure indication is not discussed in the FSAR or Technical Specifications; therefore, the design basis for this modification is Regulatory Guide 1.97 and the subsequent correspondence between the NRC and Duke Power Company.

Safety Review:

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, which is separated by ICCM cabinets A and B. Each of these cabinets provide power to one of the two transmitters for each steam generator. The cabinets are electrically and physically separate, and provide indication to the redundant indicators on the main control board. A single ICCM cabinet also provides input for a recorder on the main control board and both cabinets provide input to OAC computer points. Therefore, no single failure within the system can cause the inability of the operators to ascertain the appropriate plant status following an accident.

Regulatory Guide 1.97 provides latitude in the chosen method of display, by stating that the method of display may be by dial, digital, CRT, or stripchart recorder indication. This modification exceeds the requirement by providing various methods of display, including dial and stripchart recorder, both of which are QA1 instruments, in addition to OAC computer points (non-OA). Regulatory Guide 1.97 stipulates that steam pressure indication should have a range from atmospheric pressure to 20% above the

ON-12447 (Continued)

lowest safety valve setting. The range of 0-1200 psig corresponds to 14%. above the lowest safety valve setting and 8% above the highest safety valve setting. While this range does not comply with the requirements of Regulatory Guide 1.97, Duke Power Company submitted an analysis which demonstrated that this range would adequately indicate the highest pressure postulated to occur based on safety valve setpoint and accumulation. The USNRC has accepted this design as being adequate.

An appendix R review has been performed and demonstrated that there are no fire related concerns. This modification will utilize existing electrical penetrations. The power supply for the ICCM cabinets is adequately sized to handle the loads from these transmitters as well as all of the other loads being placed on them due to Regulatory Guide 1.97 upgrades.

This instrument line and other Regulatory Guide 1.97 instrument line applications were reviewed in Design Study ONDS-0217. It was concluded that the use of the current instrument lines to tee off for control room indication is satisfactory for seismic loadings and safe from non-seismic pipe interactions. The control room indications and recorders are also seismically qualified. The type and model transmitters being used are qualified for post-accident containment conditions. If the mounting of the existing non-OA steam generator pressure transmitters does not meet current QA-4 mounting practices, the failure mode may be postulated in which, upon a seismic event, the transmitter dismounts, breaking the instrument impulse line connection to the transmitter, venting the line to containment atmosphere, and thus rendering inoperable the new QA-1 instrumentation installed under this NSM. To prevent the possibility of the existence of this failure mode, engineering instructions were written to evaluate the current mounting of the existing non-QA transmitters, and upgrade the mounting if it does not meet or exceed current QA-4 mounting practices.

USO Evaluation:

As a result of this modification:

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

   No. No equipment associated with or affected by this modification is an accident initiator.

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

   No. This modification will upgrade steam generator pressure instrumentation and indication to Regulatory Guide 1.97 standards so as to ensure availability for operator indication following a DBA.

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

   No. No new failure modes are postulated.

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

   No. The configuration installed by this modification is a Regulatory Guide 1.97 upgrade, and is therefore able to be relied upon for important operator information. Additionally, no adverse system interactions between the equipment installed by this modification and already existing equipment is deemed to exist.

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

   No. The upgrade will ensure that no single failure can cause the inability of the instrumentation to provide indication of important parameters to the operator.

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

   No. No new failure modes are postulated.

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

   No. No setpoints, limiting safety system settings, or accident mitigation equipment is affected by this modification.

<u>Conclusions</u>:

This modification does not involve any USQs or safety concerns. No Technical Specification changes are required.

Nuclear Station Modification #
ON-12448 (Unit 1)

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 pressurizer level instrumentation consists of three level transmitters, which provide individual 0-10VDC input to the Integrated Control System (ICS) for signal processing. Any of the three transmitters may be selected by the operator (by operation of a selector switch on the MCB) for processing. The ICS presently provides input to:

1) A non-QA chart recorder on the MCB.

2) Hi-Hi and Lo-Lo pressurizer level control room annunciators.

3) High and low pressurizer level annunciators.

4) Lo-Lo pressurizer level interlock for heater cutoff.

5) Level indication to the Auxiliary Shutdown Panel (ASP) indication as well as selector station for makeup control at the ASP.

The existing Pressurizer water temperature instrumentation consists of two elements in a single resistance temperature detector (RTD) which provide input to the ICS for:

1) Operator Aid Computer (OAC) alarms.

2) Temperature compensation signal.

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 Signal

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

ON-12448 (Continued)

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).

Contact outputs (for annunciator use)

Pressurizer level:

Pressurizer level and temperature instrumentation is not controlled by Technical Specifications, and is briefly discussed in the Oconee Final Safety Analysis Report, Section 7.4.2.2.3, under non-nuclear process instrumentation. The design basis for this modification is Regulatory Guide 1.97 and subsequent correspondence between the USNRC and Duke Power Company.

Safety Review:

This modification was initiated as a part of the control room review conducted by Duke Power Company in response to NUREG-0737, and will place Oconee Nuclear Station in compliance with Regulatory Guide 1.97, Rev. 2, with respect to post-accident monitoring of pressurizer level and temperature.

Pressurizer level is a Category 1 Type A variable. 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 signals. 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

ON-12448 (Continued)

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.

Regulatory Guide 1.97 provides latitude in the chosen method of display, by stating that the method of display may be dial, digital, CRT, or stripchart recorder indication. Regulatory Guide 1.97 also states that at least one channel should be displayed on a direct-indicating or recording device. This modification exceeds the requirement 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 the 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, Duke has performed analyses 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 Duke 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 and 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 was performed.

A loop accuracy calculation for the temperature compensation loop has been completed. Regulatory Guide 1.97 requires that upgraded instrumentation does not "degrade the accuracy and sensitivity required for normal operation". The accuracies of all of the pressurizer level signals used by the control room operator and the ICS have been reviewed and found acceptable.

For the Regulatory Guide 1.97 upgrades at Oconee, the lack of written criteria did not provide enough historical information to verify that instrument impulse lines were installed properly. However, this instrument line, 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. This modification has been reviewed against the existing description of pressurizer level and temperature controls in the Oconee FSAR. 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 this modification which varies in characteristics from the original, the RTD drawing, Vendor Seismic Analysis, and Vendor Similarity Report were reviewed by Duke Power Company Oconee Civil Design Engineering and found to be acceptable with no further action required.

<u>USQ Evaluation</u>:

As a result of this modification:

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

   No. The pressurizer level and temperature instrumentation is not an accident initiator.

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

   No. Pressurizer level and temperature is an input to the ICS, and a postulated MSLG 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.

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

   No. See question #1; no new failure modes are created.

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

   No. The modified pressurizer level and temperature

instrumentation is being upgraded to meet Regulatory Guide 1.97 standards. The modified instrumentation is supplied with class 1E power, is redundant, and has been qualified for the environment in which it will be located.

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

No. The effects of failure of the instrumentation are the same as with the previous design.

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

No. No new failure modes are created.

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

No. Process setpoints and interlocks (for HP-120 for normal makeup control, and for pressurizer heaters) will remain the same. No Limiting Safety System Settings are affected by this change.

Conclusions:

There are no USQs or safety concerns involved with this modification.

Nuclear Station Modification #
ON-12589 (Unit 1)

Description Safety Review:

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 specified 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. 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 1B crossovers to main headers A and B on the RCS side of valves IHP-26 and 1HP-27, respectively).

Main headers A and B each have one pneumatic transmitter (1HPIFT0007A and IHPIFT0008A, respectively) and electronic transmitter (IHPIFT0007P and 1HPIFT0008P, respectively) 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 <u>new</u> <u>non</u>-qualified electronic transmitters. The new non-qualified transmitters will <u>not</u> 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 safety related transmitters use safety related power.

HPI crossover flow paths (i.e., HPI flow paths designed to bypass failed-closed/blocked 1HP-26 or 1HP-27 valves) are currently fitted with non-qualified flow instrumentation (1HPIFT0159 and 1HPIFT0160). Although not addressed as a commitment, this modification will also replace the instrumentation with qualified electronic transmitters and control board indicators.

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 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 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 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. 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 in Reference 10 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

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the thermal hydraulic calculations for the various ECCS LOCA models 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 from the vertical board to the main control board in the control room. The controls on the vertical board are not to be eliminates 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.

USQ EVALUATION:

May the modification:

1) Increase probability of an accident evaluated in the SAR?

   No. The HPI flow instrumentation is not the initiator of the accidents described in FSAR Sections 15.9, 15.13, or 15.14.

2) Increase the consequences of an accident evaluated in the SAR?

   No. The new HPI flow instrument loop has been verified to ensure adequate indication of HPI flow. The new high flow alarm setpoint has been verified to provide adequate pump protection and allow the system to provide Adequate ECCS injection flow under all postulated small break LOCA conditions.

3) Create the possibility for an accident of a different type than any evaluated in the SAR?

   No. This instrumentation has only monitoring and alarm functions and does not input to any controls. The new instrumentation loop is qualified QA 1.

4) Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?

   No. The new transmitters that are used to meet Regulatory Guide 1.97 commitments are QA Condition 1 and receive safety related power from the ICCM cabinets. The 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 I, and will 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 non-qualified transmitters not postulated to add a new failure mode that would allow erroneous readings to occur from the safety related instrument.

The failure mode of the current and new transmitters is low flow on loss of air or power. The current transmitters use non-safety related air and power. The new transmitters use safety related power. Separation between safety related and non-safety related components is provided using isolation devices. Electrical separation and effects on equipment for Appendix R mitigation are not adversely affected. Each new channel of instrumentation for HPI main header flow indication 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 HPI flow instrumentation has only monitoring and alarm f functions and does not have any control functions. The new instrument loops' accuracy, due to instrument error, has been determined to be acceptable for monitoring required accident flows.

5) Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR?

No. No additional monitoring equipment is added. The failure of the current pneumatic and electronic instrumentation and the new electronic instrumentation, on loss of air or power respectively, will indicate low on the HPI flow scale. Thus, these failures will all result in the same failure indication.

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

No. No failure modes, other that those addressed in questions 4 and 5 are postulated.

7) Reduce the margin of safety as defined in the basis for any technical specification?

ON-12589 (Continued)

No. The required 450 GPM design basis minimum ECCS injection flow capability is not affected by the new instrumentation loop and high flow alarm setpoint. This modification involves no other plant safety limits, set points, or design parameters.

CONCLUSION:

This modification involves no safety concern or USQS. No technical specification changes are required.

Nuclear Station Modification #
ON-32593 (Unit 3)

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.

Safety Review:

The quench tank is a tank that is located in the Reactor Building. This tank collects effluent from the pressurizer safety valves and various other vents and condenses the effluent. The quench tank temperature monitoring 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 the quench tank temperature monitoring instrumentation as a Type D variable and Category 3, with a specified range of 50-750 F. This variable is used to monitor operation. A Type D variable provides information to indicate the operation of individual safety systems and other systems important to safety. Category 3 is intended to provide requirements that will ensure that high-quality off-the-shelf instrumentation is obtained and it applies to backup and diagnostic instrumentation. It is also used where the state of the art will not support requirements for higher qualified instrumentation. The design criteria for Category 3 (Type D variable) states that the instrument should be of high-quality commercial grade, able to withstand the specified service environment, and the method of display may be by dial, digital, CRT, or stripchart recorder indication. Instrumentation being upgraded for accident monitoring is not to degrade the accuracy and sensitivity required for normal operation.

Duke pursued an exception to the above listed range for this instrument and stated that the instrument range that was to be used would be in the range of 50-325 F. The NRC reviewed this exception to the guidance given in Regulatory Guide 1.97 and found it acceptable.

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.

USO Evaluation:

As a result of this modification:

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

   No. This change does not create any conditions or events which lead to accidents previously evaluated in the SAR.

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

   No. The meter and recorder monitoring ranges include those reviewed and accepted by the NRC. The meter and recorder perform no automatic functions. The instrumentation is for monitoring only.

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

   No. No accidents different than already evaluated in the SAR are postulated.

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

   No. The NRC reviewed the range of 50-325 F and found it acceptable. The new instrument will be calibrated to have a

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range of 50-350 F. 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 affects to the control board are created. The failure modes for the meter and recorder are not changed. The instrumentation's accuracy and sensitivity for normal operation is not degraded.

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

No. No additional operating components are added. Only the scales are replaced. The failure modes of the recorder and meter are not changed.

6) May the possibility of malfunctions of equipment important to safety different than any already evaluated in the SAR be increased?

No. No new failure modes are postulated.

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

No. This modification does not adversely affect any plant safety limits, set points, or design parameters.

Conclusions:

This modification involves no USQs or safety concerns. No technical specification changes are required.

Nuclear Station Modification #
ON-22614 (Unit 2)

Description:

This modification is to install a new Post Accident Liquid Sample (PALS) Panel that will take reliable, accurate samples after an accident. The existing PALS Panel will be removed. The new System is called PALS II+. The PALS System is required per Regulatory Guide 1.97.

The PALS II+ System includes the following features:

1) simplified layout, reduced number of valves, fittings,, etc.

2) a normal drainage return line to the High Activity waste Tank (HAWT)

3) an internal sump to collect minor leakage

4) a large sump pump, and

5) a mimic (flow diagram) on the control panel with valve controls and position indication.

Modifications external to the panel include the following:

1) Deletion of the Reactor Building (RB) normal sump sample point.

2) Addition of a Low Pressure Injection (LPI) discharge sample routed to PALS.

3) Addition of a letdown sample routed to PALS.

4) Addition-of an Appendix R local sample point (Unit 1 only). Unit 2 will utilize an existing sample point for Appendix R purposes.

5) Addition of a return line to HAWT.

6) Addition of a y-strainer in the sample inlet line. The strainer will be equipped with a drain line for flushing.

7) Install thermocouples on the sample cooler cooling water lines.

8) Provide improved-flow metering instrumentation on the sample panel inlet line so as to measure flow at both high and low pressures.

9) Route Recirculated Cooling Water (RCW) to cool this sample cooler (rather than demineralized water).

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10) The new sample panel will be permanently located in the same location as the old sample panel. The new junction box will be wall mounted in the Auxiliary Building corridor. The old control panel mounting pad will be deleted. The new control panel is portable and is to be stored in the corridor just outside the sample panel room with the wheels locked in place while not in use.

11) Relocate high point vents and low point drains from deleted tubing to the appropriate corresponding locations on the new field-routed tubing.

12) Relocated flow element from the deleted tubing to a location on the new tubing.

13) Route existing nitrogen and instrument air to their corresponding connections on the new panel.

14) Install limit switches on existing valves, as required, to provide position indication at the control panel.

Safety Review:

The PALS System provides the capability to promptly obtain and analyze a liquid Reactor Coolant System (RCS) or containment sump sample under accident conditions without incurring excessive radiation exposure. Exposure to any individual is not to be in excess of five rems whole body dose or 75 rems to extremities. The diluted liquid and dissolved gas samples obtained from the system have the capability to:

1) Provide information related to the extent of core damage that has occurred or may be occurring during an accident.

2) Determine the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment.

3) Provide information on coolant chemistry.

The present and new PALS Systems are non-QA Condition and non safety related. The PALS System can be non QA Condition based on Regulatory Guide 1.97 requirements. The new PALS System meets Regulatory Guide 1.97 requirements and the design basis of the PALS System as specified in the FSAR. The PALS System performs no safety function. Piping changes made are non QA Condition and QA Condition 1 and 2. The QA condition 1 changes are made to the letdown line that is to be used to supply a sample to the PALS System. The new control panel is to be mounted at the present sample panel's location. The new junction box is to be wall mounted in the Auxiliary Building corridor. The PALS sample panel and new junction box are mounted QA Condition 4 to prevent seismic interaction. The new PALS control

panel is portable and is not secured seismically. The new control panel is in an area where no equipment important to safety is located and therefore poses no seismic interaction concerns. The non seismically analyzed class pipe is small (less than or equal to 1 inch) and should not cause any greater seismic interaction concerns than presently exist with non seismic pipe already in the area.

The nitrogen and instrument air supplies have approximately the same supply requirements with the new PALS System as with the existing PALS System. Adequate sample amounts will be attainable and RCW flow for sample cooling is sufficient. System flows to the PALS sample panel do not degrade the systems' other functions. The primary samples are cooled in a heat exchanger arrangement in both the existing and new PALS System.

Pipe class breaks are at normally closed valves that are deenergized during normal operation. These valves are deenergized at the PALS sample panel when the sample panel is not in use. The valves that form the pipe class breaks and also act as pressure boundary valves are solenoid valves that fail closed on loss of power. The PALS System is not normally in service, with sample lines and service lines isolated, and is placed in service manually at some time following an accident. The system is not provided with any automatic features and is not required to have safety grade power. Mechanical system changes were reviewed and determined to have no adverse impact on Appendix R shutdown methods. Electrical changes were also reviewed for meeting Appendix R separation criteria. Pipe stress analysis and support/restraint design has been completed.

The PALS sample panel and connecting pipe can be exposed to high pressures. The adjoining pipe with a lower design pressure is protected from these high pressures by check valves for incoming flow and by an internal manually set PALS sample panel valve that keeps a low discharge pressure for flow leaving the sample panel.

The modifications are entirely within the Auxiliary Building and involve no new containment penetrations. Containment isolation valve operation is not affected. A return line is provided to the HAWT and sample cooling is provided by the closed RCW System. Changing the cooling water to RCW from demineralized water reduces the possibility of a PALS sample panel tube leak release of radioactivity to the environment since the RCW System is a closed system. Demineralized water is used only for sample dilution and strainer backflush. Therefore, there is no-concern for unmonitored releases of radioactivity. In the present and now PALS, the demineralized water release to the emergence sump is possible, but the possibility is controlled through procedures.

<u>USQ Evaluation</u>:

ON-22614 (Continued)

As a result of this modification:

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

   No. The potential for a boron dilution accident due to demineralized water addition to the emergency sump is not increased.

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

   No. The PALS System will still have the capability to obtain and analyze a liquid RCS or containment sump sample under - accident conditions without incurring excessive radiation exposure.

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

   No. No accidents different than previously evaluated in the FSAR are postulated.

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

   No. The existing and new PALS Systems are non QA Condition and non safety related. The new sample panel and junction box are mounted QA Condition 4 and the new control panel is in an area where no equipment important to safety is located. Thus no seismic interaction concerns are posed by the sample panel, junction box, or control panel. The non seismically analyzed pipe is small (less than or equal to 1 inch) and should not cause any greater seismic interaction concerns than presently exist with non seismic pipe already in the area.

   Adequate sample and cooling flow is available and the systems that go to the PALS sample panel do not degrade the systems' other functions. The pipe class break and pressure boundary locations are at normally closed valves that are deenergized during normal plant operation. These valves are deenergized at the PALS sample panel when the sample panel is not in use. The pipe class break and pressure boundary valves are fail closed on loss of power. The lower pressure pipe will not be exposed to the PALS System high pressure because of check valves and an internal PALS sample panel valve that keeps a low discharge pressure for flow leaving the sample panel. Using RCW for sample cooling reduces the potential for radioactive release to the environment. Pipe stress analysis and support/restraint design was performed.

5) May the consequences of a malfunction of equipment important

to safety previously evaluated in the FSAR be increased?

No. The PALS System performs no safety function. No new containment penetrations are made and no new unmonitored release paths for radioactivity are created.

6) May the possibility of malfunctions of equipment important to safety different than previously evaluated in the FSAR be increased?

No. Appropriate Appendix R reviews were performed which determined that shutdown methods were not adversely affected and separation criteria is not degraded.

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

No. This modification does not adversely affect any plant safety limits, setpoints, or design parameters.

Conclusions:

These NSMs involve no safety concerns or USQS. No Technical Specification changes are required.

Nuclear Station Modification #
ON-12820 (Unit 1)

Description:

This modification will replace existing monitors 1RIA-57 and 1RIA-58 as a part of the overall monitor replacement program at Oconee. Instead of locating the monitors inside containment, the new monitors will be located in spare electrical penetrations. The sensitive portion of the detector protrudes into containment and is covered by a thin cap. Also, since the new monitors are compatible with the System Control and Data Acquisition (SCADA) system installed by NSM ON-12422, the old control room rate meters will be replaced with new digital rate meters. Finally, backup power will be provided to the monitors to ensure post-accident operation.

Safety Review:

The High Range Containment monitors 1RIA-57 and IRIA-58 are designed to provide station operators with an indication of the gross gamma activity in the Reactor Building atmosphere following an accident. The monitors are used to provide early detection of fission product barrier breach (Type C) and determination Of the magnitude of releases (Type E). The monitors are specified as Category 3 (commercial grade) and category I (safety related) for the two types referenced above, respectively. Category 1 requires that these monitors be mounted seismically and be qualified for a harsh environment. In addition, the new monitors will provide for maintenance post-accident. The seismic mounting for the new monitors has been completed, and the monitors are EQ qualified for post-accident operation. An Appendix R review for the new cabling has been completed and a seismic review has been done for modifications to the control board. The monitor sensitivity and range also meet the guidance of Reference 4 which is the version adopted for Oconee. Post modification testing will require that the penetrations which house the monitors be subjected to local leakrate tests following installation to assure containment integrity.

USQ Evaluation:

As a result of this modification:

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

   No. The failure of the High Range Containment monitors is not assumed to cause an accident, and the newly installed endpiece does not pose a missile hazard.

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

   No. The new monitors are designed in accordance with Reference

4 to assure proper emergency response, and the containment penetrations which house the monitors will be tested in accordance with Reference 2.

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

No. No new failure modes are postulated.

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

No. The appropriate seismic analyses have been performed for the mounting of the monitors and the control board modifications. An Appendix R review has been completed, and the monitors are powered from an essential power supply system with battery backup.

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

No. See the response to question 4.

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

No. The system design is maintained for this modification, and no new failure modes are postulated.

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

No. This modification does not adversely affect any plant safety limits, setpoints, or design parameters.

Conclusion:

Based upon the discussion above, there are no unreviewed safety questions associated with this modification.

Nuclear Station Modification #
ON-12821 (Unit 1)

Description

This modification will replace radiation monitor IRIA-40 (Air Ejector Vent monitor) with a new monitor. The existing monitor is obsolete, and its replacement is part of a larger program to replace all of the radiation monitors at Oconee with a digital radiation monitoring system. The new monitor will be compatible with the system control and data acquisition (SCADA) system installed by NSM ON-12422. By connecting the monitor with the SCADA system, there is no longer a need for the existing control room rate meter, so the rate meter will be removed by this NSM. The new monitor is an off-line monitoring system which has a pump to pull sample flow from the vent line, through a refrigerated air dryer and the monitor, then discharge it back to the vent line. The general location of the monitor will be in an open space in the Purge Equipment Room.

In addition, this NSM also included some power cable routing which involves adjacent monitors installed by NSM ON-12422. When the Reactor Building and Unit Vent Monitors were installed under NSM ON-12422, voltage and cabling problems were experienced. As a quick fix to get the system up and operating, a VN was written to temporarily switch the power cables for those monitors from the KM panel to the KG panelboard. Then new cables for the Unit Vent and Reactor Building monitors could be pulled from the KM panel at the same time as those for IRIA-40. These cabling changes were incorporated into the design package for the current NSM (NSM ON-12821).

Safety Review:

The replacement monitor will use an off-line system to reduce background problems experienced with the old system. The new monitor is non-safety related and has no safety function. It and the dryer are to be anchored seismically (QA-4), and the control board rate meter modification has been reviewed for any seismic impact. Also, an Appendix R review been completed for the new cabling associated with the modification. Finally, no support/restraint calculations are affected by the modification to the air ejector vent piping.

USQ Evaluation:

As a result of this modification:

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

   No. The failure of the Air Ejector Vent monitor is not assumed to initiate any Chapter 15 accident.

ON-12821 (Continued)

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

   No. The new monitor is non-safety related and does not provide any accident mitigation function.

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

   No. No new failure modes are postulated.

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

   No. The appropriate seismic analyses have been performed for the mounting of the monitor dryer, and the vertical board modifications. An Appendix R review has been completed, and no support/restraint calculations involving the air ejector vent piping are affected.

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

   No. See the response for Question 4.

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

   No. The system design is maintained for this modification and no new failure modes are postulated.

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

   No. This modification does not adversely affect any plant safety limits, setpoints, or design parameters.

Conclusions:

Based on the discussion presented above, it is determined that this NSM does not involve any unreviewed safety questions.

Nuclear Station Modification #
ON-12852 (Unit 1)

DESCRIPTION:

This modification implements the negotiated commitments between Duke and the NRC as detailed in Generic Letter (GL). 89-19 "Request for Action Related to Resolution of Unresolved Safety Issue A-47 'Safety Implication of Control Systems in LWR Nuclear Power Plants pursuant to 10 CFR 50.54(f)." It 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 I level or upon loss of both MFPS Loss of both MFPs is the only condition which presently results in MDEFWP starts.

SAFETY REVIEW:

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 described in NUREG-1217.

Overfill protection is normally provided by ICS which terminates feedwater to the OTSGL 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 MVP 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 overfill non-safety related. The hardware for existing protection circuits is comprised of proven, reliable, nonsafety 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 (Rosemount 1154). 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 on 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 SGLCS 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. Upon further design review, the noise filtering circuits mentioned in reference 9 were evaluated to be unnecessary, and were not installed. If it becomes apparent later that they are needed, the circuits can be installed.

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). These modifications do not affect the probabilities or consequences of the accidents listed above but they do enhance plant response to conditions and transients discussed in references 12 and 13. The purpose of these modifications is to provide increased plant

ON-12852 (Continued)

protection against these types of events.

A four position switch for control of the MDEFWPs automatic initiation (either low, OTSG level or, low hydraulic control oil pressure or low MFP discharge pressure) is being installed to support operating conditions (startup and shutdown) where the station needs to bypass the low MFP start. The possibility does exist for the four position switch to be placed in the position which allows initiation of the MDEFWPs on low OTSG level only during plant operation. The position of this switch will be controlled by procedure.

An Appendix R review has been initiated. The new components have been seismically qualified. The new components have been environmentally qualified.

USQ EVALUATION

May the modification:

1) Increase the probability of an accident evaluated in the SAR?

No. Neither steam generator control nor instrumentation is a postulated initiator of the accidents described in FSAR Sections 15.5 and 15.13.

2) Increase the consequences of an accident evaluated in the SAR?

No. The current control related setpoints are not being changed. The setpoints being established are for the purpose of providing additional automatic control capability to accomplish corrective actions in response to abnormal conditions. Also, there is no change to the control of OTSG, feed rate. Improper positioning of the four position switch (most probable in startup or shutdown mode) will not prohibit EFW initiation but could delay it. However, this is equivalent to the current EFW automatic initiation potential since this feature must also be overridden during startups and shutdowns to prevent inadvertent initiation. Reference 15 discusses the proposed changes to Technical Specification 3.4 to support this mode of operation (when turbine header pressure is less than 800 psig). The position of this switch will be controlled by procedure. The response of the EFW System to a Loss of Feedwater accident (as sensed by feed pump discharge pressure or turbine control oil pressure) is not affected. On a steam line break, operators will continue to be required to secure feed to the affected OTSG manually.

Also, NUREG-1218, section 4, addresses the safety benefit of the modifications and assigns a quantitative value for the reduction of core melt frequency per reactor-year, and

estimated risk reduction, in man-rem, over the life of the plant. NUREG-1217 states that B&W PWR plants which provide control system designs which initiate auxiliary feedwater on a low steam generator water level condition to prevent steam generator dryout on a loss of main feedwater are an improvement over those without this feature. Later, it also states that PWR designs which include automatic initiation of auxiliary feedwater upon the condition of low steam generator water level, are adequate to preclude core overheating. The modification resolution to USI A-47 was selected "after considering the safety benefits derived in terms of risk reduction".

3) Create the possibility for an accident of a different type than any evaluated in the SAR?

No. The purpose of the modification is to reduce the potential for steam generator dryout (overheat) or overfill occurrence. Although the possibility exists that a malfunction of the new control and actuation equipment could result in either worst case scenario of EFW initiation at the high level point or termination of feed at the low level, this possibility exists with the present design.

4) Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?

No. The existing transmitters, control systems, and mechanical systems components are used. Redundant components are used for inputs to both the ICS and SGLCS. Both allow the operator the ability to manually select sensors. In addition, the more important inputs to ICS are monitored by SASS. Failed instruments are automatically switched for the redundant normally operating sensor. The manual selection of a failed component is prohibited by SASS. The SGLCS uses two separate and redundant trains, with separate power supplies. The EFW System motor and steam driven pumps are designed to act as two separate and redundant full capacity systems.

The possibility exists that the plant could experience spurious EFW initiations, but the possibility of these initiations is minimized by providing (1) sufficient margin between the ICS main feedwater low level control setpoint (25 inches) and the low OTSG level MDEFWP initiation setpoint (21 inches) and (2) a thirty second delay before initiation. For overfill protection the requirement for testing of the feedwater pump overspeed test circuit should reduce the likelihood of undetected failures.The additional trip device should ensure that the pump trips when needed.

The two out of two logic configuration should preclude

spurious electrical trips or trips due to instrument failure. Failure of a single instrument would negate the protective function, essentially reverting the plant to the overheat and overfill protection that it had before the implementation of the modification. Jumper connections are provided which can be used to reconfigure the system logic to one out of one, should an instrument fail. These connections are normally used for testing (reference 4). The two out of two configuration has been reviewed by the NRC. The NRC has evaluated and requested the modification pursuant to NUREG-1217, NUREG-1218 and Generic Letter 89-19.

5) Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR?

No. No additional monitoring equipment or transmitters are added. The current failure position of the ICS and SGLCS, and the solenoid and pneumatically or hydraulically controlled valves, instrumentation and components upon a loss of hydraulics, air or power will result in the same failure condition and indication as presently exists. Upon a loss of Auto Power and/or Hand Power, the OTSG high level circuits will fail de-energized and trip the main turbine and the MFW pumps, causing a reactor trip due to the Anticipatory Reactor Trip System and subsequent initiation of EFW.

6) May the possibility of malfunctions of equipment important to safety different than any evaluated in the SAR be created?

No. See question 3. Also, the method of control (pneumatic hydraulic, or electrical) of the mechanical and electrical system components is not changed by this modification.

7) Reduce the margin of safety as defined in the basis for any technical specification?

No. The EFW system will still be used as the back up means to provide water to steam generators and level control during EFW operation will continue to be independent of ICS. Periodic pump and valve testing will continue be required reference 6). Existing flow rates and setpoints will not be changed.

<u>CONCLUSION:</u>

This modification involves safety concerns which have been previously reviewed by the NRC. The NRC has determined that automatic control systems, such as those added by this modification, enhance overall plant safety. Also, based upon the safety evaluation presented in this calculation, this modification has been determined to involve no USQS. No Technical specifications changes are required.

Problem Investigation Report (PIR) #3-091-0069

DESCRIPTION:

In order to declare each unit Conditionally Operable, a change to Feedwater Pump (FWP) discharge pressure switch setpoints must be implemented. The pressure switches affected provide Loss-Of-MFW (LMFW) signals to initiate the following functions: 1) Reactor Protective System (RPS) initiated reactor trip, 2) Emergency Feedwater Pump (EFWP) actuation to provide RCS Post-shutdown heat removal, and 3) ATWS Mitigation Systems Actuation Circuitry (AMSAC) initiated Turbine Trip and EFWP actuation.

Pressure switch setpoints are changed from 750 psig to 800 psig to account for: 1) Feedwater Heater Drain Pump pressurization of the Feedwater System downstream of the FWPs above 750 psig (maximum of approximately 773 psig) and 2) instrument uncertainties (limited to 2.14% or +/- 17.2 psig). Feedwater Heater Drain Pump pressurization and instrument uncertainties were not addressed when the original 750 psig setpoints were established.

SAFETY REVIEW:

The effects of as-built component operational characteristics on system pressure (i.e., Feedwater Heater Drain Pumps) and pressure switch accuracy have been considered in the development of a new setpoint to indicate a LMFW event. A new setpoint for the affected pressure switches (800 psig) has been established as adequate to assure that the Emergency Feedwater, RPS and AMSAC receive the necessary input signals to initiate required automatic control functions (i.e., minimum actuation pressure is 800-17.2 or 782.8 psig, which is above the maximum FW Heater Drain Pump capacity of 773 psig).

The increase in the pressure setpoint from 750 psig to 800 psig does not increase the potential for inadvertent reactor trip or EFWP actuations. The affected EFWP and AMSAC automatic actuation features are switched "off" by procedure during plant startup and shutdown when the main steam header pressure is below 800 psig. The RPS automatic actuation function is automatically disabled below 0.5% full power when decreasing reactor power and below 1.75% when increasing reactor power; both RPS disable levels correspond to a steam line pressure in excess of 800 psig. The corresponding pressure in the FWP discharge piping will be higher due to pressure drops across the SG, startup flow control valves (FDW-34 and FDW-35), an associated piping. The pressure drop across these components will assure greater than the 817.2 psig maximum actuation pressure (includes 17.2 psig instrument uncertainty).

Of the previously evaluated accidents in Chapter 15 of the Oconee FSAR, only the Main Steam Line Break (MSLB) accident is potentially affected by this setpoint change. The MSLB represents the limiting plant cooldown transient. Raising of the EFWP actuation setpoint from 750 psig to 800 psig will not significantly affect the MSLB

core cooling response since 1) the setpoint change is only 50 psi, 2) the EFWP flow capacity is much lower than the flow provided by the Main FW System (i.e., initiating EFW flow slightly sooner will have only a second order effect since main feedwater flow is the major source of cooling water). and 3) the rate of pressure change following a MSLB is so rapid, a change of 50 psig in the EFWP actuation will contribute negligible additional cooling water. As described in Section 15.13.4.1, it is conservative not to include any emergency feedwater flow in the evaluation of containment pressure response to the MSLB. Therefore, raising the setpoint level for EFWP actuation to provide EFWP flow slightly sooner would enhance the containment pressure response following a MSLB.

USQ EVALUATION:

May the modification:

1) Increase the probability of an accident evaluated in the SAR?

   No. The main feedwater instrumentation is not the initiator of the accidents described in FSAR Sections 10.4 or Chapter 15 of the Oconee FSAR.

2) Increase the consequences of an accident evaluated in the SAR?

   No. The new setpoints have been verified to assure adequate indication of LMFW. The LMFW automatic action setpoint has been verified to provide adequate feedwater flow under all conditions including MSLB accident conditions. The core cooling and containment pressure responses for MSLB accident were reviewed and found not to be negatively affected by the setpoint changes.

3) Create the possibility for an accident of a different type other than any evaluated in the SAR?

   No. Automatic control features which receive input from the affected pressure switches are not functionally affected by this modification. The slightly higher setpoint pressure has been reviewed and determined not to negatively impact any automatic control features.

4) Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?

   No. Automatic control features which receive input from the affected pressure switches are not functionally affected by this modification. The slightly higher setpoint pressure has been reviewed and determined not to negatively impact any automatic control feature.

5) Increase the consequences of a malfunction of equipment

important to safety evaluated in the SAR?

No. The setpoint changes were reviewed and determined not to affect any previously evaluated equipment malfunction or accident.

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

   No. The setpoint changes do not introduce any new failure modes.

7) Reduce the margin of safety as defined in the basis for any technical specification?

   No. The setpoint changes do not relate to safety margins defined in any Technical Specification. The requirement for operable automatic EFW actuation circuitry with turbine header pressure above 800 psig is not affected (Proposed Tech Spec 3.4 - Reference 6). A turbine header pressure of 800 psig continues to be high enough to assure no inadvertent actuations of EFWPs or reactor trips occur.

<u>Conclusion</u>:

This modification involves no safety concern or USQs. No technical specification changes are required. FSAR Chapter 7 and 10 should be revised to reflect revised setpoint values.