DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

3.4.1 The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam driven pump capable of being driven from an operable steam supply system and two motor driven pumps) and associated manual initiation circuitry shall be operable.

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry associated with loss of main feedwater pumps as sensed by low hydraulic oil pressure shall be operable.

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

b. One turbine driven emergency feedwater pump or one emergency feedwater flow path may be inoperable for a period of up to 72 hours. If the inoperable pump or flow path is not restored to operable status within 72 hours the unit will be at hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

d. With three emergency feedwater pumps and/or both emergency feedwater flow paths inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump and associated emergency feedwater flowpath to operable status. The unit shall be at hot shutdown within 12 hours and below 250°F in another 12 hours if one emergency feedwater pump and associated flowpath are not restored to operable status.

e. If an emergency feedwater pump is inoperable due only to automatic initiation circuitry as specified by 3.4.2, the additional provisions of 3.4.3 a, b, c, and d which require cooldown of the RCS do not apply.

3.4.4 The 16 main steam safety relief valves shall be operable.

3.4.5 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell. A minimum of 6 ft. (=30,000 gal.) shall be available in the upper surge tank.

3.4.6 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump, condensate booster pump, and a main feedwater pump.

Operability of the Emergency Feedwater System (EFW) assures the capability to remove decay heat and cool down the Reactor Coolant System to the operating conditions for switch over to decay heat removal by the Decay Heat Removal System, in the event that the Main Feedwater System is inoperable. The EFW system consists of a turbine driven pump (880 gpm), two motor driven pumps (450 gpm each), and associated flow paths to the steam generators.

The limiting transient requiring maximum EFW flow is the loss of main feedwater with offsite power available. For this transient, a minimum EFW flow rate equivalent to 400 gpm at 1050 psia and no more than 130°F is adequate. Each of the three EFW pumps is capable of delivering this flow.

A 100% flowpath is defined as: The flowpath to either steam generator including associated valves and piping capable of being supplied by either the turbine driven pump or the associated motor driven pump.

One flow indicator or steam generator level indicator per steam generator is sufficient to provide indication of emergency feedwater flow to the steam generators and to confirm emergency feedwater system operation. In the event that at least one indicator per steam generator is not available, then the flowpath to this steam generator is considered to be inoperable.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as sensed by low hydraulic oil pressure. This specific automatic initiation logic is placed in service prior to criticality and may be bypassed when shutdown to present inadvertent actuation during startup and shutdown. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the Turbine Bypass System to the condenser. Decay heat also can be also removed from the steam generators by steam relief through the main steam safety relief valves. The total relief capacity of the 16 main steam safety relief valves is 13,105,000 lbs./hr. In this case the minimum amount of water in the upper surge tank, condensate storage tank, and hotwell is sufficient to remove decay heat for at least 4 hours at hot shutdown conditions. This provides adequate time to establish normal flow through the condenser by restarting a Condenser Cooling Water (CCW) pump in a loss of station power events. The turbine bypass valves can then be utilized to relieve steam to the condenser and commence a cooldown of the RCS.

A 6 foot level in the upper surge tank will ensure that 30,000 gallons of water are available to the EFW pumps from that source. The 6 foot level setpoint includes an allowance for instrument error and for the depletion of inventory while switching to an alternate suction source.

REFERENCE

1. FSAR, Section 10.
2. Selected Licensee Commitments, Section 16.7

Channel Description		Check	Test	Calibrate	Remarks
49.	Emergency Feedwater Flow Indicators	MO	NA	RF	
50.	PORV and Safety Valve Position Indicators	MO	NA	RF	
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	RF	
52.	RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
	a) Control Oil Pressure Switches	NA	45 Days STB	RF	
53.	Emergency Feedwater Initiation Circuits				
	a) Control Oil Pressure Switches	NA	MO	RF	
54.	Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II.F.1.3

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL SPECIFICATIONS
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3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

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3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry *associated*~~associated~~ with loss of main feedwater pumps as sensed by low hydraulic oil pressure ~~or low discharge pressure~~ shall be operable.

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

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c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

d. With three emergency feedwater pumps and/or both emergency feedwater flow paths inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump and associated emergency feedwater flowpath to operable status. The unit shall be at hot shutdown within 12 hours and below 250°F in another 12 hours if one emergency feedwater pump and associated flowpath are not restored to operable status.

e. If an emergency feedwater pump is inoperable due only to automatic initiation circuitry as specified by 3.4.2, the additional provisions of 3.4.3 a, b, c, and d which require cooldown of the RCS do not apply.

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The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as sensed by low hydraulic oil pressure~~ or low feedwater pump discharge pressure~~. This specific automatic initiation logic is placed in service prior to criticality and may be bypassed when shutdown to present inadvertent actuation during startup and shutdown. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the Turbine Bypass System to the condenser. Decay heat also can be also removed from the steam generators by steam relief through the main steam safety relief valves. The total relief capacity of the 16 main steam safety relief valves is 13,105,000 lbs./hr. In this case the minimum amount of water in the upper surge tank, condensate storage tank, and hotwell is sufficient to remove decay heat for at least 4 hours at hot shutdown conditions. This provides adequate time to establish normal flow through the condenser by restarting a Condenser Cooling Water (CCW) pump in a loss of station power events. The turbine bypass valves can then be utilized to relieve steam to the condenser and commence a cooldown of the RCS.

A 6 foot level in the upper surge tank will ensure that 30,000 gallons of water are available to the EFW pumps from that source. The 6 foot level setpoint includes an allowance for instrument error and for the depletion of inventory while switching to an alternate suction source.

REFERENCE

1. FSAR, Section 10.
2. Selected Licensee Commitments, Section 16.7

Channel Description			Check	Test	Calibrate	Remarks
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50.	PORV and Safety Valve Position Indicators		MO	NA	RF	
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches		NA	45 Days STB	RF	
52.	RPS Anticipatory Reactor Trip System Loss of Main Feedwater					
	a)	Control Oil Pressure Switches	NA	45 Days STB	RF	
	~~b)~~	~~Discharge Pressure Switches~~	~~NA~~	~~45 Days STB~~	~~RF~~	
53.	Emergency Feedwater Initiation Circuits					
	a)	Control Oil Pressure Switches	NA	MO	RF	
	~~b)~~	~~Discharge Pressure Switches~~	~~NA~~	~~MO~~	~~RF~~	
54.	Containment High Range Radiation Monitor (RIA-57, 58)		NA	MO	RF	TMI Item II.F.1.3

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OCONEE NUCLEAR STATION

ATTACHMENT 3

TECHNICAL JUSTIFICATION

TECHNICAL JUSTIFICATION

Proposed Technical Specification Change

The proposed changes to Technical Specification 3.4.2 and Table 4.1-1 of the Technical Specifications remove the requirement for EFDW and ARTS actuation on low Main Feedwater Pump discharge pressure. ARTS actuation would result from low Main Feedwater Pump hydraulic oil pressure only with EFDW actuation resulting from either low Main Feedwater Pump hydraulic oil pressure or low steam generator level. The surveillance requirements for a low Main Feedwater Pump discharge pressure actuation would also be eliminated.

Background Information

The ARTS sub-system of the Reactor Protective System (RPS) monitors main feedwater and main turbine parameters to provide anticipatory reactor shutdown following a loss of all main feedwater or a main turbine trip. These trip functions minimize challenges to the pressurizer Power Operated Relief Valve (PORV) following either of these two events.

The ARTS sub-system was originally added to the RPS in 1980 via NSM-1489 in response to NRC IE Bulletin 79-05B as well as in response to NRC commission orders. The ARTS sub-system consists of a safety-grade anticipatory reactor trip upon a loss of both main feedwater pumps or main turbine. The inputs to each of the four RPS channels of ARTS currently consist of the following monitored parameters:

Loss of Main Feedwater (as sensed by)

1. Main Feedwater Pumps Control Oil Pressure
2. Main Feedwater Pumps Discharge Pressure

Loss of Main Turbine (as sensed by)

1. Main Turbine Emergency Trip Header Pressure

Loss of a main feedwater pump (MFDWP) “A” or “B” is determined by a loss of discharge pressure (below 800 psig) or loss of main feedwater pump turbine control oil pressure (below 75 psig) on the respective pump. This in turn causes the RPS trip contact buffer to trip. Each individual channel (A,B,C,D) will trip if both the “A” MFDWP and the “B” MFDWP contact buffers are tripped and reactor power is above approximately 1.75% full power.

The loss of the main turbine is determined by a loss of emergency trip supply (ETS) oil pressure (below 800 psig). A loss of ETS pressure is indicative of a main turbine trip and will cause a reactor trip when reactor power is above approximately 30% full power.

Each of these two trips is automatically bypassed when reactor power decreases below 0.5% full power for loss of both main feedwater pumps logic and below 28% full power for loss of the main turbine .

The feedwater and associated systems are designed to control steam generator inventory during all normal plant modes involving heat removal via the steam generators. When main feedwater cannot be supplied to the steam generators, a backup supply of feedwater must be available. At Oconee Nuclear Station, this backup is supplied by the Emergency Feedwater System. After a reactor trip, decay heat is dissipated by boiling water in the steam generators and venting the steam to the condenser or to the atmosphere. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer area and a continuation of the decay heat removal process. The EFDW system is designed to start automatically in the event of a low water level in either steam generator for thirty seconds or a loss of both main feedwater pumps. Emergency feedwater maintains level in the steam generator for an extended period allowing time to restore normal feedwater or to cool down to the point at which decay heat can be removed by the Low Pressure Injection System. In the event of a loss of main feedwater, as sensed by low feedwater header pressure or low hydraulic control oil pressure on both main feedwater pump turbines, an auto-initiation of the EFDW system will result.

The proposed modification will delete the diverse feedwater pump discharge sensors from providing any inputs to the ARTS or EFDW initiation circuits.

The Anticipated Transient Without Scram (ATWS)/ATWS Mitigation System Actuation Circuitry (AMSAC) low MFDWP discharge pressure switches will be retained.

Benefits from Elimination of Pressure Switches

First, these pressure switches were the subject of LER 269/91-09, "One of Two Diverse Actuation Systems for Loss of Main Feedwater Mitigation Systems was Found Inoperable due to a Design Deficiency". Briefly summarizing this LER, these switches were found to have an interaction with the 'E' Heater Drain Pumps ('E' HDP's), and the 'D' Heater Drain Pumps ('D' HDP's). Insufficient operating margins were found between the discharge head of these pump combinations and the ARTS/EFDW initiation pressure switches when set at an actuation setpoint of 750 psig. Operating margin is the difference between the error-adjusted setpoint and the developed pump discharge head. The setpoint was increased to 800 psig until the pumps could be destaged to reduce the discharge head. During late 1991 and 1992, the pumps were destaged and the setpoint was reduced to 770 psig. Destaging the pumps did not reduce the discharge head sufficiently to return to the original 750 psig setpoint. The removal of the MFDWP discharge pressure switches is desired to eliminate the 'D' Heater Drain Pump interaction with the ARTS and EFDW systems.

Second, the original pressure switches are no longer available as a 10 CFR 50 Appendix B product from the vendor. Also, this switch type began to develop condensate fluid leakage through the diaphragm that necessitated replacement. Due to restrictions on diaphragm material (stainless steel) and required drift, only one vendor was identified which advertised a qualified pressure switch with suitable diaphragm material and the desired drift characteristics. These switches were procured and installed from March to November of 1994 on all three Oconee units.

In late 1994 and early 1995, periodic calibration activities on these replacement switches began to identify unacceptable drift of the actuation setpoint. One switch had drifted enough to render the switch inoperable due to being outside of the allowed uncertainty . The remedial activity was to increase the calibration frequency of these pressure switches and to contact the current vendor to determine why the pressure switches could not meet specified drift during Technical Specification calibration frequencies. Calibration techniques, installation, etc. were all reviewed. No definitive explanation was identified for the drift experienced at Oconee. The calibration intervals were changed to once every week on Unit 3 due to higher heater drain pump discharge pressure and to once every two weeks for Units 1 and 2.

Improved pressure switches from the vendor were installed on Unit 3 in early 1995. These new pressure switches have experienced statistically less drift than their predecessors, but not enough to extend the calibration frequency to the same interval as Unit 1 and 2 using the current setpoint.

Due to the frequent calibrations, an engineering analysis was initiated to justify increasing the actuation setpoint from 770 psig to 800 psig. The engineering analysis and supporting 10 CFR 50.59 evaluation were completed in September of 1995 and the Unit 3 setpoint for ARTS/EFDW actuation was changed to 800 psig. This same setpoint change was completed for Units 1 and 2 in October of 1995. This setpoint change allowed the calibration frequency to be extended.

Third, the original concept for the ARTS, as reviewed by the NRC in their preliminary design approval, accepted a generic design philosophy for the Oconee, Rancho Seco, Arkansas Nuclear One - Unit 1, and Crystal River -3 plants which used four pressure switches (one per RPS Channel) per main feedwater pump. While not explicitly identified, these switches were referred to in the October 5, 1979, Duke Power Company submittal as monitoring either main feedwater pump turbine control oil pressure <u>or</u> main feedwater pump discharge pressure. Individual utilities were given the latitude to select the process input for loss of feedwater. Duke at the time of final design development chose to use inputs from <u>both</u> parameters. Duke/Oconee now desires to revise the ARTS design to only utilize the Main Feedwater Pump Turbine control oil pressure inputs similar to all other B&W design plants.

A significant cost saving can be realized by the deletion of these pressure switches. This cost saving is based on the deletion of calibration requirements and replacement costs for these switches and is estimated to be $300,000 over the life of the plant.

In 1986, the B&W Safety and Performance Improvement Program (SPIP) was developed to, in part, reduce the number of reactor trips for B&W units in the industry. Several recommendations from the SPIP related to limiting the amount of trip initiators from the secondary side of the plant. Following an increase in reactor trips in 1994, Duke Power formed an ad hoc working group to investigate reactor trips from January 1, 1989 through October 10, 1994. The report from this group recommended that an evaluation be performed for all secondary side instrumentation that can cause a reactor trip. The goal of the evaluation was a reduction in the number of reactor trips. These main feedwater pump discharge pressure switches fit into this category.

Raising the pressure switch setpoint to 800 psig has decreased surveillance requirements resulting in decreased surveillance costs. Additionally, a lower calibration frequency reduces the possibility of spurious reactor trips during pressure switch calibration. However, with these switches installed, the possibility of spurious trips still remains. Elimination of these pressure switches will improve plant safety in that the design requirements for ARTS and EFDW actuation will still be met, and a source of inadvertent and unnecessary reactor trips will be removed.

Safety Review

The primary purpose of ARTS is to reduce the probability of challenging the PORV to lift and then reseat following a loss of main feedwater or turbine trip transient.

This can be accomplished in a number of ways. First, component setpoints can be selected which would separate the high Reactor Coolant System (RCS) pressure trip point of the RPS and the lift pressure of the PORV by an adequate range. This would prevent a challenge of the PORV when the RCS pressure increases after an RPS high pressure trip. When selecting the setpoint, piping considerations and operational maneuverability must be considered. This was the initial strategy after the TMI-2 incident. The NRC initially required all B&W plants to lower the RPS high RCS pressure trip to 2300 psig and increase the PORV actuation setpoint to > 2400 psig such that the likelihood of challenging the PORV was reduced . In addition, a control-grade anticipatory trip was installed to trip the reactor on main turbine trip and loss of main feedwater. The control grade ARTS was then to be upgraded to safety-grade.

All B&W plants installed control-grade ARTS on their plants in 1979 with the exception of TMI-1. By 1980-1981, all B&W plants had upgraded or installed safety-grade ARTS systems. TMI-1 activities regarding ARTS were delayed by their response to the TMI-2 accident. These systems are similar in design basis and purpose.

ARTS

The design basis of the RPS and NRC requirements for the ARTS is still met without the MFDWP discharge pressure switches. The design basis of ARTS/RPS is to generate a reactor trip immediately upon a turbine trip or a total loss of main feedwater event. These trips limit the extent of overheating of the RCS that could occur during a turbine trip or total loss of main feedwater. These trips are only activated above certain power levels. The anticipatory reactor trip from a turbine trip is active at greater than approximately 30% reactor power and the anticipatory reactor trip from a loss of main feedwater is active at greater than approximately 1.75% reactor power.

The loss of main feedwater inputs are in compliance with the NRC design approval because the main feedwater pump turbine control oil pressure switches still meet the requirements of being anticipatory in function. In addition, an anticipatory reactor trip due to loss of main feedwater pumps, as sensed by low control oil pressure, remains single failure proof. The anticipatory trip based on low control oil pressure will continue to provide additional protection and conservatism beyond that provided by the existing RPS. The existing diverse parameters of the RPS will continue to trip the reactor in the highly unlikely event that the ARTS system should fail to function.

In the power range from 20-65% full power, a condition potentially exists where a loss of feedwater does not generate an anticipatory reactor trip. This condition can occur only in a plant configuration where one main feedwater pump is running with the second main feedwater pump in a RESET condition with its RPS contact buffers also reset. Current procedural guidance precludes operation in this mode during plant startup, shutdown and normal power operation.

During a power reduction the second main feedwater pump is shut down and placed in a tripped condition as soon as it is not needed. By tripping the second main feedwater pump, the RPS contact buffers for that main feedwater pump are placed in a tripped state and an ARTS signal would be generated should the running main feedwater pump trip.

For normal power operation, both main feedwater pumps are in service and this condition is not a concern.

During power escalation, procedures provide guidance to reset the RPS contact buffers after each main feedwater pump has been placed in service. The contact buffers for the second main feedwater pump will be in a tripped state during the power increase. Thus, until the second main feedwater pump is placed in service a trip of the running main feedwater pump will initiate an ARTS signal. This signal will be initiated since the RPS will have an existing tripped signal for the non-running main feedwater pump and a trip signal for the running main feedwater pump will be initiated. Procedural control of the RPS contact buffers for the main feedwater pumps minimizes the time period during which one main feedwater pump is running and the other main feedwater pump is in the RESET mode with its RPS contact buffers reset.

Therefore, procedural guidance minimizes the time where an ARTS trip signal may not be generated and a challenge to the PORV is possible. Removal of the main feedwater pump discharge pressure switches is not expected to increase this time period or significantly increase the probability of challenging the PORV. PRA analysis addresses conditions where both the PORV and code safety valves lift. Therefore, any non-significant increase in the probability of challenging the PORV is bounded by the existing PRA analysis and will not result in an increased probability of core melt.

Emergency Feedwater

The design basis of the EFDW System is still met by the Main Feedwater Pump Turbine (MFWPT) control oil pressure loss of main feedwater initiation and the steam generator low-low level initiation after 30 seconds . When main feedwater cannot be supplied to the steam generators, as sensed by loss of main feedwater or low-low level in the steam generators, a backup supply of feedwater must be available. The backup supply is provided by the EFDW system. In addition, the Auxiliary Service Water (ASW) System, or the Standby Shutdown Facility Auxiliary Service Water (SSF ASW) provides a backup supply should the EFDW system become unavailable.

After a reactor trip, decay heat is dissipated by boiling water in the steam generators and venting the steam to the condenser or to the atmosphere. The EFDW system maintains adequate water level inventory to perform this heat transfer activity from the RCS. The EFDW system will start automatically upon loss of main feedwater or upon low-low water level in either steam generator for 30 seconds. This low-low water level start of EFDW is a QA-1 system and is controlled by SLC 16.7.3.

The EFDW system will respond to a loss of main feedwater as indicated solely by the control oil pressure switches with low-low steam generator level being a diverse backup actuation. This is acceptable based upon loss of feedwater conditions that are indicated solely by control oil pressure.

AMSAC

The design bases of the AMSAC system will continue to be met. The ATWS design is required by 10 CFR 50.62. During the ATWS scenarios, the AMSAC circuitry will automatically start EFDW and trip the main turbine.

All functions of AMSAC will still be performed as required. The 'D' & 'E' HDP's discharge pressure head is not of concern, because, under ATWS scenarios, all plant functions, other than RPS, continue to perform as designed.

The following criteria from 10CFR50.36 were reviewed to determine if a Technical Specification is necessary to address the actuation of ARTS and/or EFDW as sensed by pressure switches monitoring MFDWP discharge pressure:

1. Is the initiation circuitry for ARTS and EFDW, as sensed by low MFDWP discharge pressure installed instrumentation used to detect, and indicate in the Control Room a significant abnormal degradation of the reactor coolant pressure boundary?

   No. The above mentioned EFDW/ARTS initiation circuitry does not provide any information directly associated with the reactor coolant pressure boundary. This circuitry is associated with the FDW system and does not receive any input from Reactor Coolant System parameters.

2. Is the initiation circuitry for ARTS and EFDW, as sensed by low MFDWP discharge pressure, a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes failure of or presents a challenge to the integrity of a fission product barrier?

   No. ARTS currently receives input of a loss of MFDW event from either of two conditions: low hydraulic oil pressure on the MFDWPs or low discharge pressure on MFDWPs. EFDW currently receives input of a loss of MFDW from either low hydraulic oil pressure on the MFDWPs, low discharge pressure on the MFDWPs or low steam generator level. The deletion of the low MFDWP discharge pressure signal will not prevent ARTS/EFDW from receiving an actuation signal should a loss of MFDW occur. Consequently, a loss of main feedwater in any design basis accident will continue to result in an input being sent to both ARTS and EFDW. Since PORV challenges will not be increased, the removal of the main feedwater pump discharge pressure switches will not present a significant challenge to the integrity of a fission product barrier.

3. Is the initiation circuitry for ARTS and EFDW, as sensed by low MFDWP discharge pressure, a part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure or presents a challenge to the integrity of a fission product barrier?

   No. As discussed in #2 above, a loss of MFDW signal will still be provided to ARTS and EFDW by a condition of low control oil pressure on the MFDWPs, and, additionally, to EFDW by a condition of low steam generator level. Consequently, any events/transients, which assume that ARTS/EFDW will be actuated by a loss of MFDW, will continue to be mitigated as designed due to the continued presence of an initiation signal to ARTS and EFDW.

4. Has the initiation circuitry for ARTS and EFDW, as sensed by low MFDWP discharge pressure, been shown to be significant to public health and safety per operating experience or probabilistic safety assessment?

   No. Probabilistic analysis shows that the probability of ATWS damage is not noticeably increased by removing the Main Feedwater Pump discharge pressure ARTS initiation capability, nor is the probability of core damage due to a loss of feedwater initiator. For the loss of feedwater case, the reliability of other emergency feedwater initiation signals is sufficient to start emergency feedwater pumps providing secondary cooling in adequate time to prevent core damaging conditions. In the unlikely event that secondary cooling does fail, the reliability of primary feed and bleed capabilities with primary heat removed through either the PORV or (in the event the PORV is unavailable) safety relief valves is sufficient to prevent core damage. Thus, a review of Probabilistic Risk Assessment shows that core damage frequency and accident consequences are not significantly increased due to the removal of the ARTS and EFDW initiation input from main feedwater pump discharge pressure.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

No. The accidents addressed within the Oconee Final Safety Analysis Report (FSAR) have been reviewed with respect to this proposed Technical Specification amendment request. The probability or consequences of any accident previously evaluated is not significantly increased by the proposed amendment. Emergency Feedwater is required for the mitigation of some accidents and the availability of this system will be unaffected by this proposed revision. Both manual and automatic actuation of the EFDW system on a loss of main feedwater will remain.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

No. This amendment eliminates a portion of the automatic actuation circuitry for EFDW and ARTS. This circuitry removal does not create the possibility of a new or different kind of accident as the design of the circuitry is to sense a loss of main feedwater and supply a signal for the initiation of ARTS and EFDW. A loss of main feedwater signal will continue to be supplied to ARTS and EFDW; however, this loss will be sensed by low hydraulic oil pressure on the Main Feedwater Pumps (ARTS and EFDW) and low steam generator level (EFDW only) rather than by a low Main Feedwater Pump discharge pressure. Since a loss of Main Feedwater will continue to be recognized, the system will continue to function as before. Hence, no new or different accidents will be created.

(3) Involve a significant reduction in a margin of safety.

No. The margin of safety will not be significantly reduced as an actuation signal to ARTS and EFDW will continue to be generated by a loss of Main Feedwater. Consequently, ARTS and EFDW will continue to perform the safety function required for accident mitigation.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

This conclusion is supported by the No Significant Hazards Consideration evaluation which is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment will not change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

The proposed amendment will not increase the individual or cumulative occupational radiation exposure.

In summary, the proposed amendment request meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.