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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation
REID,R.W. Operating Reactors Branch 4

SUBJECT: Forwards response to NRC 791107 ltr requesting addl info re intercommunication sys reliability analysis.Schedule for implementation of identified mods contingent upon completion of final design & availability of equipment.

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

December 21, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactor Branch No. 4

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

With regard to your letter dated November 7, 1979, which requested additional information regarding the ICS Reliability Analysis, please find attached our response to the recommendations identified by this analysis.

The schedule for implementation of the items identified has not been established. We intend to implement the identified modifications in a timely manner consistent with availability of equipment, completion of final design work, and unit availability.

Very truly yours,

William O. Parker, Jr.

RLG/sch
Attachment

ELECTRICITY FOR A PROGRESSIVE PIEDMONT
5th Anniversary
DUKE POWER 1904-1979

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P 395

DUKE POWER COMPANY
Response to NRC Letter of November 7, 1979
ICS Reliability Analysis

NNI/ICS POWER SUPPLY RELIABILITY

The power supply for the ICS is normally supplied from the station batteries through static inverters. An alternate source is provided from the AC regulated power system. A static transfer switch is provided to automatically transfer the ICS panelboard to the regulated power source within 1/4 cycle following loss of inverter power supply. The system will automatically re-transfer from regulated power supply to inverter supply two seconds after the inverter returns to a normal output condition. This scheme will be upgraded to include another transfer switch, down stream of the static switch, which will automatically transfer power back to the regulated power supply if the transfer back to the inverter supply fails. This modification will significantly increase the reliability of the NNI/ICS power supply.

The inverter or regulated AC feeds a panelboard which supplies five feeds to the ICS/NNI - auto, hand and three emergencies. Basically, the original design philosophy was to enable the system to ride through a loss of auto power and not trip the unit. However, subsequent design reviews revealed that the system probably wouldn't ride out an auto power loss and that it would not ride out a hand power loss. Therefore, the solution was to supply the ICS with a highly reliable power source since splitting the loads increased the probability of power loss.

RELIABILITY OF RC FLOW SIGNAL TO ICS

Presently each loop inputs a DP transmitter signal into the ICS. The signal is conditioned by a square root extractor, temperature compensated and then used in BTU limit, ΔTC and load limit control. The capability exists to transfer either or both inputs to buffered RPS system DP transmitter outputs. This is accomplished by manually transferring the cable from one jack to another. To increase the reliability of this signal, we are investigating the feasibility of revising the manual transfer scheme to automatically select each loop's highest flow.

ICS/BOP SYSTEM TUNING - particularly feedwater condensate systems and the ICS controls.

(1) Any particular operational (startup, etc.) problems experienced at your plant with respect to the ICS.

We have had some problems with the Main Feedwater Valves (MFWV) during startup due to excess leakage. Because of this, we have a maintenance program underway to test the valves for leakage and proper operation every refueling. Also, we intend to modify the controls for the MFWVs so that they won't close once opened above 10% power. In addition, we feel that some of our feedwater oscillating are caused by the Heater Drain system level controls. A modification is currently underway to replace the existing level controls with expanded range controllers. The above modifications should make the feedwater/condensate system more reliable and thus increase ICS reliability.

(2) Bases for operational intervention in place of automatic ICS action (including start-up, power operation and shutdown activities).

The operator will not intervene if he feels that the ICS is doing its job. However, if the operator has had a problem previously, chances are he will react in the same manner as before. We do not intend to instruct our operators not to over-ride ICS controls since to do so could lead to additional RPS challenges. We feel that this area is dependent on operator training and have committed ourselves to increased ICS training to further insure optimum use of automatic ICS controls. The plant simulator which is currently being designed will significantly improve the operators understanding of the ICS.

(3) Procedures used by the operator to perform the operation described in (2) above.

There are no specific procedures which tell the operator when to intervene with automatic ICS controls. We feel that this concern is best handled by increased operator training instead of additional procedures. However, there are procedures which deal with possible consequences of ICS failure, e.g., loss of KI Buss, Loss of SG Feedwater, Condensate and Feedwater, Main Steam Line Break, etc.

(4) Additional training provided to the operator.

See Item (2) above.

(5) <u>Balance of Plant</u>

A. Main feedwater pump turbine drive minimum speed control to prevent loss of main feedwater or indication of main feedwater.

A modification is underway to increase the oil pressure to the main feedwater pump to prevent loss during minimum speed control.

B. A means to prevent or mitigate the consequences of a stuck open main feedwater startup valve.

These valves will be tested for leakage and proper operation during every refueling outage. However, if the valve does stick open, the operator will close the block valve to mitigate the consequences.

C. A means to prevent or mitigate the consequences of a stuck open turbine bypass valve.

The operator will recognize a stuck open valve as a steam line break and react to it as that. He will see low S.G. pressure in one loop, large increase in reactor power and low Tave. Upon recognition, he will mitigate the consequences by closing the block valve.

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RLG/sch
Attachment

bcc: Mr. H. B. Tucker
Mr. K. S. Canady
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Master File OS 801.01
Section File OS 801.01
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Mr. J. E. Cole
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B&W Owners Group

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DUKE POWER COMPANY
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ATTACHMENT 4

OCONEE NUCLEAR STATION

PROCEDURES AND OPERATOR TRAINING

OCONEE NUCLEAR STATION

PROCEDURES AND OPERATOR TRAINING

1.0 INTRODUCTION

NUREG-0667, "Transient Response of Babcock and Wilcox Designed Reactors" draws several conclusions in the area of emergency procedures and operator training. Based on a limited review of emergency procedures at operating plants, the NRC Staff concluded that generally, B&W plants require more manual immediate actions on the part of the operator than other vendor types.

This specific area of concern appears to be based solely on the Staff's knowledge of the reactor trip and LOCA emergency procedures for B&W plants. No discussions were presented of the manual actions required by similar emergency procedures of other NSSS designs. Yet, based on this review, the Staff concludes the plant should be modified to reduce or eliminate manual immediate action for emergency procedures. This conclusion is not valid in the case of Oconee. Section 3.0 provides our basis for these conclusions.

In the area of operator training, the Task Force concluded, based on a survey of B&W training coordinators, that little formal instruction has been given to date. In the case of Oconee and Duke Power, this conclusion is erroneous.

Duke Power trains its operators, and in instances where they need to be promptly made familiar with an event at a similar plant as it affects Oconee, B&W would not be doing the training. It is inappropriate that the NRC base its conclusion on a survey of individuals who would not be involved in the training aspect of interest.

Duke did train its operators on the vital aspects of the Crystal River transient shortly after it occurred. The operators were made aware of the key events of the transient and how similar events would affect Oconee. They were briefed on the hardware and procedural changes that were made as a result of the transient as they are for all changes made to the plant. It is considered that these actions are sufficient for the operators involved and are consistent with other training demands. As with all significant events, this event will be incorporated in the appropriate lecture segment of the requalification program.

2.0 EVALUATION OF EMERGENCY PROCEDURES

A thorough review of the operator actions required in the immediate actions section of Oconee Emergency Procedures was conducted. It is considered that those actions which are manual, versus automatic, should remain so and that those actions are necessary yet their result is to minimize and/or reduce adverse transients and are not absolutely essential to the safe operation of the plant. Many of the items listed as manual actions are verifications of various automatic actions. The "actions" simply assure the correct functioning of automated equipment and often involve simply monitoring meters and gauges. These verifications are necessary to

identify problems which are usually simply corrected by merely realigning pumps or establishing predetermined alternate flow paths. The following is a list of other actions which may be required in certain of the procedures. The possibility of automating the function and the need for the action is discussed in the commentary.

(a) Increase Letdown Flow (open HP-7)	In various loss of heat sink events RCS swell should be corrected if possible without PORV or code safety lifting. Action not readily automated since improper actuation would worsen certain other transients.
(b) Manual Trip Reactor	In many instances it is appropriate to shut the reactor down thereby reducing the heat being put into the RCS. Action is taken in circumstances where automatic trips do not occur. Most of the situations are of an anticipatory nature thereby reduce the severity of the transient.
(c) Isolate Letdown (close HP-5)	The letdown flowpath is isolated to minimize the effect of RCS shrinkage following reactor trips. Automatic actions will eventually restore RCS volume. Automation would present undue risk during other transient conditions. (Additional discussion of the effect of this action is provided in Attachment 1.)
(d) Maintain LDST Level or align HPI pump suction to BWST (open HP-24)	Action assures source of borated water to the HPI pumps. Automatic switchover in the absence of ES actuation is not thought to be necessary or prudent.
(e) Start HPI Flow through injection loops (throttle open HP-26 and align/start additional HPI pump to BWST)	Action not essential to safe operation but allows normal monitoring of RCS conditions. (Additional discussion of the effect of this action is provided in Attachment 1.)
(f) Shutoff Bleed Transfer Pumps	Required during boron dilution events to correct possible source of dilution automation would be impractical. Unambiguous indications of dilution readily observable and actions are not burdensome.
(g) Achievable Natural Circulation (Primarily Verification)	Achieving natural circulation is automatic if several parameters are in acceptable ranges. Verification and corrective actions are more practical than attempting to predict every possible set of conditions.

(h)	Establish conditions whereby unit(s) can be controlled from Auxiliary Shutdown	In events requiring evacuation of the Control Room, certain actions are desired but not necessary. Automation would be highly impractical and unnecessary.
(i)	Stop All Reactor Coolant Pumps	In certain scenarios (to prevent damage to RCP's or to prevent void transfer into core) stopping RCP's may be desirable. Subject is being discussed between utilities/vendors/staff.
(j)	Minimize Flow to Turbine Building (Close turbine by-pass, stop CCW flow, etc.)	In case of flooding, it may be necessary to minimize water sources to the turbine building. Potential for inadvertent actuation at inappropriate times with adverse effects prohibits automation.
(k)	Line up CT-5 to Lee Direct	During certain loss of power scenarios manual lineup to Lee is required. This is a backup to the normal sources of emergency power. Automation is not appropriate nor required.
(l)	Isolate Faulted Battery	Certain battery failures may require isolation subsequent to operator review. Automation not practical.
(m)	Trip Service Water Pumps and secure other non-essential CCW cooling	In loss of CCW canal scenarios, manual action must be taken to minimize CCW usage. Potential hazard of inappropriate automatic action precludes automation.
(n)	Manually initiate HPI and restart RCP's	In certain conditions, anticipatory manual initiation of HPI may prevent or minimize severity of transient conditions (such as loss of FDW while in natural circulation). Automatic action is not prudent.
(o)	Trip main FDWP, and operate EFDW valves with nitrogen supply	During loss of instrument air, manual action required to correct resulting events. Automation for very narrow scenario is imprudent.
(p)	Manually transfer power source from KI to "AC Line"	This action is required in the event that automatic transfer has failed. (This type of occurrence is discussed in more detail in Attachment 3.)

3.0 EVALUATION OF TASK FORCE RECOMMENDATIONS

Recommendation 11

Modifications should be made to the plant, to the extent feasible, to reduce or eliminate manual immediate action for emergency procedures.

Evaluation

Based on our review, as discussed in the preceding section, the actions currently required of the operators are necessary and appropriate and no further action in response to this recommendation is necessary.

Recommendation 13

Lectures should be developed and given promptly to all licensed personnel concerning the Crystal River 3 event as well as their plant-specific loss of NNI/ICS analysis. A means to evaluate the training (e.g., quizzes) should be included. This training should be audited by the Office of Inspection and Enforcement.

Evaluation

Based on our review of the training program in effect at Oconee, it is considered that the intent of this recommendation has been met and that no further action is necessary.

Recommendation 14

Licensees should develop and implement promptly plant procedure concerning the loss of NNI/ICS power.

Evaluation

Contrary to the Task Force report, Rancho Seco was <u>not</u> the only facility having procedures that included the effect of the loss of power supply on the total plant. Our letter of March 12, 1980 discussed the emergency procedures which were implemented following the Oconee 3 transient November 10, 1979. Subsequently, these procedures have been revised and confirmed to be valid by the performance of a test on all Oconee units. No further action in response to this recommendation is deemed necessary.

Recommendation 15

Mandatory one-week simulator training should be required for all licensed B&W operators. The training should be oriented toward or include undercooling and overcooling events, solid system operation, and natural circulation cooling. Upgrading of simulator performance in accordance with the recommendations of the TMI-2 Action Plan (NUREG-0660) should be expedited.

Evaluation

As discussed in the report, Duke currently includes simulator training in the operator requalification program. Duke has discussed implementation of

the requirements in NRC letter of March 28, 1980 concerning qualifications of reactor operators. As a result of such discussions, it is considered that the Duke program as implemented meets the intent of the requirements. As such, no further action is deemed necessary in response to this recommendation.

ATTACHMENT 5

OCONEE NUCLEAR STATION

CONTAINMENT PURGE ISOLATION SYSTEM

OCONEE NUCLEAR STATION

INVESTIGATION OF THE NEED FOR
SAFETY-GRADE CONTAINMENT PURGE ISOLATION ON HIGH RADIATION

1.0 INTRODUCTION

NUREG-0667, entitled "Transient Response of Babcock & Wilcox - Designed Reactors", recommended that safety-grade containment high radiation signals be provided to initiate containment purge isolation in addition to the present signals. Currently, the purge system is isolated upon receipt of an Engineered Safeguards (ES) high containment pressure signal at 4.0 psig or low reactor coolant system pressure signal at the ES setpoint of 1500 psig. In addition, the purge isolation and control valves are closed upon receipt of a high radiation signal from unit vent gaseous monitor RIA-45. The purpose of this review is to 1) identify possible scenarios which might require purge isolation on high radiation and 2) evaluate the qualification of the currently installed radiation monitor interlock.

2.0 EVALUATION

2.1 Scenarios of Potential Concern

There are two classes of transients which could result in leakage of reactor coolant but which would not necessarily result in timely automatic containment isolation: 1) overpressure events due to extended loss of feedwater followed by cycling of the pilot-operated relief valve and/or pressurizer safety valves until the quench tank is overpressurized and the rupture disk blows out; and 2) very small loss-of-coolant accidents which do not depressurize the RCS to the low pressure setpoint and which require a long period to increase containment pressure to the high pressure setpoint. Core damage does not result from either of these two types of events. Thus, only normal RCS activity is available for release. Since these events result in relatively small quantities of lost reactor coolant, and since the contamination levels are low, ample time is permitted for assessment of the conditions by the operator and subsequent action to isolate the purge system, should it be in operation.

2.2 Current Isolation System

The containment purge system consists of an inlet line and an outlet line which can be isolated by closure of one of three valves in each line. Each line has an electric motor-operated valve inside containment and two pneumatically operated valves outside containment. Closure of all six valves is initiated by actuation of ES Channels 1 and 2 on high containment pressure or low reactor coolant pressure, providing diverse isolation signals as required by NUREG-0578. In addition, the pneumatically operated valves, valves PR-2, PR-3, PR-4

and PR-5, all receive a closure signal on high radiation. Upon reaching a high radiation setpoint, the unit vent gaseous monitor, RIA-45, generates a signal which causes the solenoid valves which supply air to open the purge valves to deenergize, resulting in closure of the valves. This signal is sealed in, so that resetting the radiation monitor will not result in reopening of the valves. The procedure for startup of the purge system requires a verification that RIA-45 and the unit vent particulate and iodine monitors are operable. The procedure contains additional instructions to assure that the unit vent monitoring is observed closely so that alarm limits are not reached. Assured power to the monitor is provided from a 120 VAC non-load shed panelboard. Thus, although the monitor is not safety-grade, there is reasonable assurance that it will fulfill its function.

3.0 CONCLUSION

A safety-grade high radiation containment isolation signal would effect isolation of the purge system only for classes of transients which would otherwise have very minimal consequences. The delay in achieving isolation by operator action results in only very small releases. This is supported by the discussion in Section 7 of NUREG-0667, which concludes that implementation of such a system would have very little impact on risk reduction. This already small risk is further reduced by the control-grade high radiation signal for purge isolation which is currently provided. Additionally, Duke Power Company has already taken steps to minimize purging while at power operation. The probability of the occurrence of one of the transients discussed during operation of the purge system is therefore very small. Thus, the recommendation does not appear to be justified, and no further action in response to this item is necessary.

ATTACHMENT 6

OCONEE NUCLEAR STATION

MISCELLANEOUS ITEMS

OCONEE NUCLEAR STATION

MISCELLANEOUS ITEMS

1.0 INTRODUCTION

Within NUREG-0667, "Transient Response of B&W Designed Reactors", recommendations were made on various plant systems. The function of this report is to address the validity and appropriateness of these recommendations to Oconee Nuclear Station.

2.0 EVALUATION OF RECOMMENDATIONS

Recommendation 6

A minimum set of parameters should be established to enable the operator to assess plant status. The set recommended by the Task Force follows:

(a) Wide range reactor coolant system pressure,

(b) Wide range pressurizer level,

(c) Wide range reactor coolant system temperatures: hot leg (each loop), cold leg (each loop), and core outlet (two or selectable),

(d) Makeup tank level,

(e) Reactor building pressure,

(f) Wide range steam generator level (both OTSG's),

(g) Wide range steam generator pressure (both OTSG's),

(h) Source range nuclear instrumentation, and

(i) Intermediate range nuclear instrumentation,

(j) Borated-water storage tank (BWST) level.

The instrumentation for the selected parameters must meet the following requirements:

(a) The instrumentation must be reliable and redundant and should meet all applicable codes and standards for protection system instrumentation; and,

(b) In accordance with safety standards, these require a minimum of two redundant channels of all designated information. At least one channel of which shall be recorded automatically on a timely basis for use in trending, instant recall, and post-event evaluation.

Evaluation

Duke Power agrees that a minimum set of parameters should be established to enable the operator to assess plant status. To this end, Duke engineers have actively participated in industry efforts in this area, particularly the AIF Subcommittee on Safety Parameter Integration. Duke endorses this subcommittee's efforts and considers that results, upon implementation by Duke, will effectively meet the intent of this item.

Recommendation 7

All B&W plants should provide the flexibility to substitute appropriate combinations of incore thermocouples for the loop resistance temperature detectors (RTDs) presently used for primary temperature input to the subcooling meter. All B&W plants should provide the capability of having a continuous or trending display of incore thermocouples. This display need not be indicated in the control room at all times but may be called up on demand from the computer.

Evaluation

The existing process computer system is used for saturation calculations at Oconee. The saturation calculations function was incorporated shortly after TMI occurred and provides saturation temperature and pressure margin calculations for each loop utilizing the loop resistance temperature detectors (RTDs) and loop wide range (WR) pressure reading. In addition, a saturation temperature and pressure margin is calculated for the core using the 52 incore thermocouples and core pressure reading. The hot leg RTDs providing input to the computer system from the ICS receive their source of power for signal conversion from a high reliability static inverter system which has several levels of backup power from a non-load shed supply. However, the loop resistance temperature detectors (RTDs) -- hot leg and outlet -- are validity checked and substitutive action is taken if either is found to be invalid. Upon determination of loss of ICS power, the hot leg temperature for each loop is set to a value of zero (0), thereby forcing the calculation to use the non-ICS outlet temperature associated with each loop. This was necessary since the hot leg temperatures fail to a value of 350°F upon loss of ICS power. Each incore thermocouple signal is range checked against the weighted average of the incore thermocouple signals and any invalid signal is discarded and a new average is calculated. Substitution of combinations of incore thermocouples for loop resistance temperature detectors (RTDs) is not provided due to the fact that three (3) individual saturation temperature margins are calculated, one for each loop as well as the incore thermocouples.

The system at Oconee permits the operator to obtain incore thermocouple readings in three forms:

a. A hardcopy digital trend capability is available through the output typers.

b. Incore thermocouples can be displayed on the CRTs mounted on the main control board.

c. The operator may select incore thermocouples to be trended on computer driven trend recorders on the main control board.

Output of the saturation condition values is provided by display on the CRTs located in the control room, hardcopy digital trend capability, and continuous margin value output (provided by 5 second updating of performance indicators located on the main control board).

Based on our review of the design of Oconee, the intent of this recommendation is met and no further modifications are necessary.

Recommendation 17

In order to provide an alternative solution to PORV unreliability and safety system challenge rate concerns, the following proposal (submitted by Consumers Power Company) should receive expenditious staff review for possible consideration and backfit on all B&W operating plants:

(a) Provide a fully qualified safety-grade PORV;

(b) Provide reliable safety-grade indication of PORV position;

(c) Provide dual safety-grade PORV block valves, capable of being automatically closed if a PORV malfunction occurs;

(d) Complete a test program to demonstrate PORV operability;

(e) Install safety-grade anticipatory reactor trip on total loss of feedwater; and

(f) Reset the PORV and high-pressure trip setpoints to their original values of 2255 psig and 2355 psig, respectively.

Evaluation

Duke agrees that the staff should expeditiously review the Consumers proposal. However, this solution is not necessarily suitable for backfit on operating B&W plants, particularly Oconee.

In response to the individual concerns included in this recommendation, the following statements are provided:

(a) Duke has supplied a description of Oconee PORV power supplies in response to NUREG-0578. This has been reviewed by the NRC staff as found in accordance with the requirements. As far as the fully qualified safety-grade PORV is concerned, Duke has committed to participate in the current EPRI program. When this program is complete, Duke will view its applicability to Oconee and make any necessary changes at that time.

(b) An acoustical monitoring system has been installed on each Oconee unit to monitor the position of PORV and safety valves. This acoustical monitoring system is similar to those found acceptable by the staff for this purpose for other pressurized water reactors. It is a reliable, single channel system, powered from a battery backed vital bus. It will provide the operator with positive indication of valve position and an annunciation of an open valve in the control room. The valve position indication components have been seismically and environmentally qualified as appropriate for conditions applicable to their location.

Backup valve position indication is provided by temperature sensors located downstream of the PORV and safety valves and by the quench tank level indicator.

The staff has reviewed the design and concluded that Oconee is in compliance with requirements for direct indication of PORV and safety valve positions.

(c) In response to an NRC letter dated May 7, 1980, which requested commitments to complete five additional TMI-2 related requirements, Duke committed in a letter dated June 13, 1980, to provide a report on overall safety effect of the PORV isolation system (NUREG-0660, Item II, k.3.2) and, if deemed necessary, modify the system appropriately.

(d) As pointed out in item 17 (a), Duke Power is participating in the EPRI program to demonstrate PORV operability.

(e) Duke Power presently has a control grade anticipatory reactor trip on total loss of feedwater installed on all three Oconee units. This control grade trip will be upgraded to safety grade.

(f) Duke Power is in agreement with resetting the PORV and high pressure trip setpoints to their original values.

Based on our review of the design of Oconee and on efforts currently underway in the areas of relief valve testing and PORV isolation system review, no additional actions are necessary.

Recommendation 21

The need to introduce auxiliary feedwater through the top spray sparger during expected transients should be reevaluated by licensees. This reevaluation should consider the reduced depressurization response if auxiliary feedwater could be introduced through the main feedwater nozzle and enter the tube region from the bottom of the unit.

Response

The original design considerations for high injection of AFW, and the benefits and potential problems associated with injecting AFW through the main feedwater nozzles have been reviewed. Based on this review, it has been concluded that this change should not be pursued. The NUREG-0667 recommendation was made to address a concern of potential overcooling.

One function of the auxiliary feed header is to inject feedwater into the steam generator upon loss of all four reactor coolant pumps. This elevated injection enhances the capability to establish natural circulation of the reactor coolant by providing a high effective thermal center in the OTSG. In addition, in the event that both main feed pumps are lost, auxiliary feedwater is used to provide secondary heat removal capability. If a steam generator dryout condition precedes AFW initiation, this means of adding AFW minimizes thermal shock of the steam generator vessel wall and lower tube sheet by providing some heating of the feedwater.

Injection of AFW through the main feedwater nozzles will require a relatively larger steam generator inventory to establish natural circulation due to the lower thermal center. Therefore, for the same AFW flowrate, it will require a longer time to establish a natural circulation condition. Potential overcooling effects with lower AFW injection will still exist unless automatic or manual control action is taken.

There are also structural concerns associated with injecting AFW through the main feedwater nozzles during expected transients. The service life of the main feedwater nozzles would be shortened due to thermal fatigue effects associated with cold auxiliary feedwater and increased usage. The thermal shock effects associated with the introduction of cold auxiliary feedwater to the shell and lower tube sheet would also be significant and may be unacceptable. A dry steam generator condition prior to AFW initiation would aggravate this concern due to the lack of aspirating steam for feedwater heating in the OTSG downcomer. In addition, use of the main feedwater nozzles for AFW injection would increase the potential for water hammer damage to the main feedwater lines and header.

In summary, any potential reduction in overcooling through the use of the main feedwater nozzles for AFW injection are far outweighed by the potential problems, both operational and structural, associated with this change. No further action in response to this recommendation is deemed necessary.