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DENTON, H. R. Office of Nuclear Reactor Regulation

SUBJECT: Ack receipt of NRC 790917 ltr & submits evaluation of potentially adverse environ effects on nonsafety-grade control sys.

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
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October 5, 1979

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

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

Dear Mr. Denton:

Attached is an evaluation of potential non-safety-grade control system interactions during design basis high energy line break accidents for Oconee Nuclear Station. This evaluation was performed to determine whether such systems could generate interactions (as a consequence of high energy line break environment) adversely affecting the consequences of licensing basis accidents evaluated in the safety analysis reports. This letter and the attached report are submitted in response to your letter of September 17, 1979.

Briefly, in this evaluation those non-safety-grade control systems which were judged to have the potential for influencing the course of the steam line break accident, main feedwater line break accident, and the loss of coolant accident as evaluated in the safety analysis reports were identified and then further examined for their potential of being subjected to high temperature-humidity conditions for selected break locations both inside and outside the containment. Control systems with components located in such adverse environment and without environmental qualifications were assumed to produce adverse interactions during the course of the accident, unless it was determined (1) that the failure mode of the affected component leads to a safe position for the actuated device, (2) that the effect of the failure on the assumed behavior is conservative, or (3) that the required action assumed in the safety analysis occurs in a short time. An assessment of the impact on safety analysis was made for those situations in which adverse control system interactions could not be precluded.

The results of this evaluation are summarized as follows:

1. Most of the non-safety-grade equipment examined is seen to be capable of performing its functions during the accidents considered—by virtue of its location and/or environmental qualification.

2. The reactor trip functions can be considered to be completed before the gradual environmental effects on the rod drive control could significantly change the transient behavior.

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3. The interactions of certain non-safety-grade control systems during certain accidents are such that the system performance would be in the conservative direction with respect to the safety analyses or that the system would fail in the safe direction.

4. Malfunctions in the turbine bypass valve control and main feedwater flow control systems during a steamline break could not be ruled out since these systems include environmentally unqualified components. However, the effect of independent malfunctions in these systems during a steam line break were previously evaluated in the safety analysis reports.

5. Adverse interactions by the pressurizer PORV control system during a steamline break inside the containment could not be ruled out since the contactor in this equipment string is not environmentally qualified. This situation could represent a potential inconsistency with the steam line break safety analysis.

Therefore, within the scope of this investigation only the steam line break inside containment—pressurizer PORV control combination was identified as potentially involving a variation from the safety analysis basis. Although this scenario has not been quantitatively analyzed, the conclusions of the existing safety analysis are not expected to be significantly altered since the opening of the PORV would have a negating effect during the uncontrolled cooldown of the primary system due to the steamline break and vice versa. Furthermore, the probability of the severence of a large steam line at the specified location at the worst core burnup time, with the most reactive control rod stuck out to result in the worst case calculated results combined with the probability for the PORV control system to interact in the adverse direction would be extremely small. On the basis of these results and considerations, it is concluded that the Oconee Units 1, 2 and 3 can continue to operate without undue risk to the health and safety of the public and that no modification of the operating licenses for these facilities is warranted.

The potential pressurizer PORV interaction will be further investigated and appropriate remedial actions taken. We will also continue our investigation into the turbine bypass valve and main feedwater control system for possible qualification of the few unqualified components. Because of the short schedule made available to address this issue, we have not been able to investigate the long-term availability of needed operator indications. We believe that the scope of the investigation required in this area is closely related to the Abnormal Transient Operating Guidelines (ATOG) program currently underway. We will, therefore, integrate these additional investigations into the planned ATOG program.

Very truly yours,

William O. Parker, Jr.

PMA:scs
Attachment

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this statement concerning Oconee Nuclear Station Facility Operating Licenses DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this the 5th day of October, 1979.

Notary Public

My Commission Expires:

December 8, 1982

OCONEE NUCLEAR STATION

# EVALUATION OF POTENTIALLY ADVERSE ENVIRONMENTAL EFFECTS ON NON-SAFETY GRADE CONTROL SYSTEMS

Prepared By

Duke Power Company

&

Babcock & Wilcox Company

October 5, 1979

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TABLES

## I. Introduction

This report consists of the evaluation for Oconee Nuclear Station of the performance of pertinent non-safety-grade control systems during certain high energy break conditions to determine if such systems could generate adverse interactions as a consequence of the adverse environment during design basis high energy break accidents. This evaluation has been performed in response to the September 17, 1979 letter from H. R. Denton of the NRC to Duke Power Company. This evaluation includes consideration of the four items listed in the IE information Notice 79-22 and covers the scope of the evaluation discussed with the NRC Staff on September 20, 1979: i.e., the evaluation of the impact on licensing basis accident analyses due to consequential environmental effects on non-safety-grade control systems by (1) identifying the impact on licensing basis accidents which cause an adverse environment, (2) defining safety analysis inputs and responses used during licensing basis accidents, and (3) verifying safety analysis conclusions or recommending actions justifying continued operation.

In performing this evaluation, first, the licensing basis high energy line break accidents which had been evaluated in the safety analysis reports were reviewed, and the control system responses considered in the analysis of each of these accidents were identified (Section II). Using this as the basis, those control systems which were judged to have the potential for influencing the course of the accident were identified, and their susceptibility to a high energy break environment for accidents inside and outside the containment was evaluated (Section III). For each candidate non-safety-grade control system thus identified to have the potential of being affected by a particular accident, a detailed survey of associated components was then performed to determine the degree to which the components would be adversely affected by the high temperature-humidity environment for break locations specified by the past high energy line break analysis (Section IV).

An assessment of the impact on the safety analysis is also made in Section IV for each non-safety-grade control system with components environmentally unqualified and susceptible to high temperature-humidity conditions. A summary of the safety assessment is presented in Section V. Finally, items for additional investigation in the future are described in Section VI.

## II. Discussion of Safety Analysis Functions and Parameters for High Energy Line Break Accidents

The purpose of this section of the report is to summarize the input parameters and control system responses considered in the licensing basis safety analysis of high energy line break (HELB) accidents.

The plant licensing basis analyses, as presented in the FSAR, were reviewed to define the inputs, assumptions and responses used for non-safety grade control systems. This information is summarized in Table I, which lists typical equipment actions and actuation times used in the safety analyses for B&W 177 fuel assembly plants. The data has been categorized to reflect the functional requirements as follows:

A. Reactor Power Control and Shutdown

B. Reactor Pressure Control

C. Steam System Isolation and Pressure Control

D. Feedwater System Isolation and Control

This categorization has been developed to focus upon those primary functions which have a potential for control system interaction.

This table identifies the range of equipment actions and actuation times used in the plant safety analysis for steam line break, feedwater line break and large and small LOCA.

## III. Assessment of Potential Environmental Interactions with Non-Safety-Grade Control Systems

The non-safety-grade control systems have been reviewed to determine if an accident environment could adversely affect the analyzed course of the event. Specifically, the approach taken was to use the safety analysis functions and parameters from Table I as a basis to identify where potential control system effects could have an impact. The result of this evaluation is summarized in Table II, Potential Environmental Effects on Non-Safety-Grade Control Systems. The matrix identified, for six accident types, the non-safety-grade control systems which could be adversely affected by the environment caused by the event. Where no entry is made in the matrix, no potential for environmental effects exists due to the physical location of the equipment with respect to the high energy line break: i.e., breaks inside containment do not affect equipment outside containment and vice versa. If an entry is made (X or Y), a potential effect exists as follows:

X - The adverse environment caused by the break could affect the equipment, and equipment malfunction could affect safety analysis functions identified in Table I.

Y - The adverse environment caused by the break could interact with the equipment, but the equipment malfunction would not affect safety analysis functions identified in Table I.

This structuring of the potential effects matrix provides a focus on those non-safety-grade control systems which are important and identifies areas for further evaluation of the impact on the safety analysis (i.e., X's).

## IV. System Evaluation of Potential Control System Interactions and Safety Assessment of Adverse Interactions

For each control system (except for the rod drive control) identified in Section III to be a candidate for possible adverse interactions in the course of an accident (and designated with an X in Table II), a system evaluation was performed by surveying the components and examining their locations relative to specified break locations for each accident. This system evaluation was performed utilizing the Unit 1 equipment layout (Units 2 and 3

arrangement is similar). For each interaction that was determined to have the potential for affecting the course of an accident, the impact of this interaction upon the safety analysis is discussed.

The following bases were used in surveying systems and evaluating environmental consequences.

(a) Duke report, Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station (MDS Report OS73.2 dated April 25, 1973) was used to identify main feedwater and steam line break locations outside containment. These breaks were located in the Turbine Building and the penetration room of the Auxiliary Building. Other areas outside containment (e.g. control room, cable room, equipment room) have no postulated breaks. LOCA's, feedwater line breaks, and steam line breaks were also considered in containment.

(b) The evaluation of the non-safety-grade control systems performance was limited to the consequences of environmental conditions resulting from postulated pipe breaks.

(c) The following environmental conditions[1] were considered:

| | | |
|---|---|---|
| Zone 1 - Turbine Building (SLB, FWLB) | - Temperature - | 212°F for 30 minutes |
| | | Ramping Down to 120°F within an additional 90 minutes. |
| | | Humidity - 100% non-condensing |
| Zone 2 - Auxiliary Building Penetration Room (FWLB) | - Same as above | |
| Zone 2 - Auxiliary Building[2] Penetration Room (SLB) | - Temperature - | 330°F for 4 seconds, |
| | | 212°F for 30 minutes ramping down to 120°F within an additional 90 minutes. |
| | | Humidity - 100% non-condensing |
| Zone 3 - Containment[3] (LOCA, FWLB, SLB) | - Considered LOCA environment (Oconee Nuclear Station FSAR Ch. 14) | |

NOTE 1: Each break was assumed to raise the entire zone to the specified environment.

NOTE 2: Due to this rapid transient (330°F for 4 seconds) components were assumed to only reach the 212°F temperature).

NOTE 3: LOCA radiation levels do not apply to the FWLB & SLB conditions.

Using the above environmental conditions and criteria, the following steps were carried out in this survey for each non-safety-grade control system of interest:

(a) Identified control components necessary to perform intended function.

(b) Identified cables necessary to perform intended function.

(c) Identified locations of components and routes of cables/connections.

(d) Compared these locations with environmental zones.

(e) Where components/cables were located within a break zone, component/cable performance was evaluated considering:

1) Qualification data
2) Failure modes
3) Effects on system function
4) Impact of safety analysis

The results of this survey and safety assessment are given below.

1. Control Rod Drive Control System Interaction with all Postulated Adverse Environments

Postulated failure of the control rod drive control system is considered as the spurious withdrawal of the control rods prior to a reactor trip. A significant increase in initial power level as a result of spurious rod withdrawal prior to reactor trip has not been included in the SLB, FWLB or LOCA analysis. While it is likely that such an increase in power would be offset by the reduction in the time-to-trip for each of these accidents, confirmatory analysis has not been performed. The following summarizes the liklihood of significant rod withdrawal for each case.

1) For steam and feedwater line breaks, the time-to-trip is very short (up to 8 seconds for SLB and 13.4 seconds for FWLB). Adverse environmental effects on any equipment, e.g., out-of-core detectors, which could result in apurious rod withdrawal, is considered extremely unlikely.

2) The same rationale applies to all but the very smallest LOCA's: i.e., time to low RC pressure trip is short for the majority of small breaks. Conversely, "leaks" (breaks too small to result in a low pressure trip) are not expected to generate a severe environment.

From the above, it is concluded that adverse interaction resulting in significant reactor power increases is extremely unlikely.

2. Pressurizer PORV Interactions

System Evaluation

The PORV is a normally closed relief valve which is solenoid operated. The solenoid is deenergized in the closed position. The solenoid can be manually energized from the control room via a switch in auxiliary control cabinet 13 or automatically opened by high reactor coolant pressure. This automatic pressure signal comes from either reactor coolant narrow range pressure transmitters RC3A-PT1 or RC3B-PT4 via the protection system racks and the ICS racks in the control room. The system employs no pneumatics and uses the "energized-to-open" philosophy.

The principal components of the PORV system are the RC pressure transmitters (inside containment), bistables (outside containment), cabling, the PORV solenoid (inside containment), the PORV itself (inside containment), and the power supply contactor (inside containment). These components have been examined for potential environmental effects resulting from breaks inside the containment, in the penetration room, and in the turbine building. As a result of this review, it has been determined that only the power supply contactor could be susceptible to adverse interactions for breaks occurring in the containment.

Power for valve operation is provided to a contactor located in a terminal box located within containment. Control signals are fed to this contactor for valve operation. Spurious opening of the valve could be hypothesized if the contactor had an electrical failure which caused the contactor to energize. An analysis of the circuit and components required for the above action indicates the fault must be an electrical short across two specific terminals. If the fault is shown in any other combination of terminal to terminal or terminal to ground, the valve will not open.

Safety Assessment

The consequences of spurious opening due to adverse environments has not been specifically analyzed in the SAR. However, the following summarizes the conclusions for each case:

1) Large LOCA - spurious opening of the PORV would have an insignificant effect on the course of the accident.

2) Small LOCA - spurious opening of the PORV would be expected to improve the results of this analysis in that it would aid in depressurization, increase HPI cooling flow, and provide an additional path for heat removal.

3) FWLB inside containment - spurious opening of the PORV or failure to close if opened is not specifically analyzed in the SAR. However, as a result of the TMI-2 incident, analysis and operator guidelines have been developed for the case of LOFW concurrent with stuck open PORV. Furthermore, it should be noted that if the valve were to spuriously open early in the transient, it would aid in reducing the pressure transient. Therefore, the consequences are acceptable.

4) SLB inside containment - spurious opening of the PORV is judged to have an effect on the analysis. The extent of the adverse effect has not been quantitatively evaluated.

Justification for Continued Plant Operation

Adverse interactions of the PORV during a steam line break are considered unlikely since:

a. A specific electrical short is required; otherwise fuses will blow, protecting the valve.

b. The contactor is in an enclosed terminal box (not directly exposed to the break environment).

c. The operator has PORV energized (open) indication.

d. The block valve can be closed.

Furthermore, since the existing plant operating conditions have significant margins in core parameters (shutdown margin, moderator temperature coefficient, etc.), RCS parameters (coolant activity), and meteriological conditions compared to the values assumed in the safety analysis, even if unanalyzed effects were to occur, the basic conclusions of the FSAR safety analysis (that the steam line break event does not constitute an undue risk to the health and safety of the public) would remain unchanged. Therefore, the Oconee units can continue to operate without undue risk to the health and safety of the public.

Long range upgrade to eliminate the hypothesized failure would be to move the contactor to a protected area or replace components with qualified hardware or rework the control scheme to incorporate another contactor in series with the existing one but in a protected area.

3. Turbine Trip/Turbine Stop Valve Interactions

For the turbine trip/stop valve closure function, fifteen components were reviewed to determine the consequences of pipe break environment. These components consisted of relays, terminal blocks, solenoid valves, and interconnecting cabling. All of these were found to be unaffected by Main Steam Line Break or Feedwater Line Break in the penetration room. Similarly a Main Steam Line Break, Feedwater Line Break or a LOCA inside containment had no effect on the turbine trip/stop valve closure components. For a Main Steam Line Break or Feedwater Line Break in the Turbine Building all affected components are rated for the pipe break environment with the exception of the trip solenoid valves. At present time environmental data is unavailable for these solenoid valves.

These turbine trip solenoids relieve hydraulic fluid pressure which holds the turbine stop valves open. There are three trip solenoids arranged so as to provide two independent means of dumping hydraulic fluid pressure, thus closing the stop valves. Two of these solenoids are normally energized and actuate the closing of the stop valves when both deenergize. The third solenoid is energized to actuate stop valve closure.

It is not expected that the environment resulting from a Turbine Building pipe break could prevent a turbine trip/stop valve closure for the following reasons:

1. Two of the solenoid valves are normally energized and thus fail to the turbine trip mode.

2. The two normally energized solenoid valves and the normally de-energized solenoid valve are of different manufacturers thus providing equipment diversity.

3. The turbine trip/stop valve closure function actuates very quickly after a pipe break occurs (see Table No. 1). It is unlikely that a pipe break environment could elevate the temperature of the solenoids appreciably before the trip action occurs. Any subsequent failures of the solenoids after initial trip has occurred cannot reopen the stop valves.

Hence it is felt that the risk of a failure of the turbine trip/stop valve closure due to a pipe break environment is acceptably low.

4. Turbine Bypass Valves/Atmospheric Relief Valves Interactions

System Evaluation

The atmospheric relief valves in Oconee units are manual valves, and hence there is no potential for the environmental interactions of interest.

The evaluation of steam or feedwater line breaks or LOCA effects on the turbine bypass valves involved reviewing the environmental data of approximately 51 components. These components consisted of equipment and inter-connecting cabling necessary for proper turbine bypass valve operation. Environmental data for these components was found to be acceptable during steam or feedwater line breaks in the penetration room. Enviornmental data was also acceptable for breaks in containment and the Turbine Building, except for three types of devices. The devices, possible failures and their environmental data are provided below:

Turbine Building (SLB/FWLB):

1. The Bailey R/P type E/P converter is rated to 180°F with ±5% accuracy. Once above 180°F the converter's output may drift. The ±5% accuracy value is assumed to increase moderately as the temperature increases from 180°F to 212°F. Bypass valve movement depends on whether the converter's output drifts high or low. If it drifts low the valve will be closed, which is the valve's fail safe mode. If it drifts high, then the valve may slightly open; however, it is not expected that this would result in a significant amount of steam being bypassed.

2. The Motorola 56 PM type pressure transmitter is rated at 200°F with an accuracy of ±2%. The 56 PM employs similar electronics to the 56 PH. This transmitter extends its range from 200°F to 250°F while incurring an additional ±2% accuracy value. The accuracy value at 212°F

is not expected to cause the bypass valves to open. Even if slight opening did occur, it should not result in a significant amount of steam being bypassed.

3. The Bailey RU6000B hand-auto station is environmentally rated at 140°F. However the hand-auto station is located on the auxiliary shutdown panel, which is only in operation when a control room evacuation has occurred. During normal operation the hand-auto station is placed in the automatic mode, which switches all electronics out of the circuit leaving only a mechanical switch, which passes the signal through the hand-auto station without any processing being performed. Since this is the case, it is not expected that a line break environment would cause the hand-auto station to effect the signal being passed through it.

Containment (SLB/FWLB/LOCA):

1. Motorola 56 PM type pressure transmitter is rated at 200°F with an accuracy of ±2%. The 56 PM employs similar electronics to the 56 PH. This transmitter extends its range from 200°F to 250°F while incurring an additional ±2% accuracy value. Above the rated temperature the transmitter may drift high or low. If it drifts low the valve will be closed, which is the valve's fail safe mode. If it drifts high then the valve may slightly open; however, it is not expected that this would result in a significant amount of steam being bypassed.

   Analysis of the components with environmental data below the postulated environmental requirements will continue to be reviewed.

Safety Assessment

The safety analysis of loss-of-coolant accident relies upon safety-grade equipment for mitigation. The potential effects presented in Table II indicate that the control system functions, though considered in the analysis, are modeled conservatively such that postulated malfunctions of this system will not invalidate the analytical results. The reactor shutdown and pressure control during the blowdown and reflood phases do not rely upon non-safety-grade control systems. The secondary steam system is conservatively assumed to remain intact (bottled up) to provide a large heat source during the later stages of blowdown. The steam safety valves are used to maintain a conservatively high steam pressure. Potential control system effects which provide more steam relief would tend to improve the analytical results.

The main feedwater line break involves an overheating of the RCS. The steam safety valves provide the necessary steam relief and, therefore, operation of the TBV is not necessary. Inadvertent opening of the TBV would result in additional heat removal from the RCS, thus, improving the results of the accident. The inadvertent opening of the turbine bypass valve (20% capacity) in the unaffected steam generator during a steamline break accident would result in additional blowdown of the steam system. A scenario involving the blowdown of both steam generators during a steam line break in the turbine building was previously analyzed for the Oconee units in the High Energy Line Break Analysis Report. The

conclusions of this analysis were that the core would remain within its thermal limits and that the reactor damage criteria were met. The postulated inadvertent opening of the turbine bypass valve as a result of potential control system interactions during a steam line break accident is enveloped by the aforementioned safety analysis. The capability to provide the minimum required auxiliary feedwater flow is maintained because of the two trains (each with 65% capacity) of motor driven pump system.

5. Main Feedwater Flow Control Interactions

System Evaluation

During the evaluation of this system, 60 separate components which comprise the active, control and interconnecting portions of the main feedwater system were considered.

These components include cables, transmitters, valves, controllers and control cabinets which are necessary for the system to function as designed. All affected components studied were environmentally qualified to operate satisfactorily in the postulated line break areas with exception of the following five types of devices:

1. TURBINE BUILDING (Feedwater Line Break, Steam Line Break)

   a) Bailey Type R/P E/P converters

      These devices are rated to 180°F with ±5% accuracy up to this temperature. The pneumatic outputs from these E/Ps operate the startup and main feedwater control valves. Because the postulated pipe break environment is 212°F, some valve positioning outside of the specified accuracy range can be expected to occur. The resulting SG level deviation is not expected to be significant enough to cause serious system transients.

   b) Fisher Type VFJ-664A Control Valve Operators.

      These operators control the startup control valves and have a rated temperature of 180°F. The most probable failure of these valves above rated temperature is closed, which is the safe position.

   c) Motor Control Center 1XGB

      Due to the temperature rating of 104°F and the use of thermally operated protective devices enclosed in motor control centers, power availability following these events is not assured. This motor control center supplies the startup and main control motor operated block valves which are normally open (as desired) and will remain so on loss of power.

d) Barton 296 Feedwater Control Valve ΔP Transmitters

These transmitters are rated at 160°F. The output of these transmitters is expected to experience some drift when subjected to the postulated line break environment. This drift will cause some variance in FWPT speed with a resultant change in SG level. These changes however, are not expected to result in significant level changes in the affected steam generator. If these unexpected level changes were to occur, level limiting and FWPT overspeed features are provided by the Steam Generator Water Level Control System.

2. REACTOR BUILDING (Feedwater Line Break, Steam Line Break and LOCA)

a) Steam Generator Startup Level Transmitters.

These transmitters (Bailey BY Type) are qualified for the environment considered, however, the terminations have not received the splice procedure that similar devices in safety related circuits have.

3. PENETRATION ROOM (Feedwater Line Break, Steam Line Break)

a) No devices for this system are in this area.

Further study and investigation of environmental impacts and control interactions will continue until necessary data has been attained to determine the need for any necessary corrective actions.

Safety Assessment

In the traditional safety analyses of loss-of-coolant and feedwater line break accidents, the main feedwater system is conservatively modeled. Therefore, potential interactions in the main feedwater flow control system do not have any adverse impact on the safety analyses of loss-of-coolant and feedwater line break accidents.

For a steam line break accident the adverse interaction of the main feedwater flow control system of concern is the failure to automatically close the main and startup valves of the affected steam generator following the reactor trip. However, this situation has been previously analyzed in the FSAR (p. 14-17 and Supplement 3). Therefore, potential adverse interactions in the main feedwater flow control system do not involve an unanalyzed steam line break sequence.

6. Auxiliary Feedwater Initiation and Flow Control

The Oconee safety grade/safety related auxiliary feedwater system (initiation and flow control) consists of two trains (each with 65% capacity) of motor driven pump system and one train of turbine driven pump system. The appropriate environmental qualification requirements of this system for pipe break conditions outside the containment need to be established and further investigated.

## V. Summary of System Evaluation and Safety Assessment Results

The results of the system evaluation and safety assessment are summarized in Table III, Impact of Control System Effects on the Safety Analysis. These potential effects, due to an adverse environment, have been placed into several categories as follows:

1. Equipment Performance

   The identified non-safety-grade equipment can be shown to perform its function, consistent with the safety analyses, in the adverse environment.

1a. The auxiliary feedwater system, including initiation circuitry and level control, is safety grade/safety related equipment.

2. Period of Operability

   The required period of operability for the equipment (i.e., time frame in which the equipment must function) is considerably shorter than the time it takes for an adverse environment to have an impact.

3. Conservative Impact

   The effect of the adverse environment on the equipment is such that the equipment performance (or failure) is negligible or in a conservative direction with respect to the safety analyses.

4. Previously Analyzed Effects

   The effects of the adverse environment on the equipment are bounded by existing analyses of similar equipment failure or maloperation.

5. Potential Problem

   The effect of the adverse environment on the equipment is such that a potential problem exists. The evaluation performed to date has not shown that the safety analysis inputs and responses are consistent with the non-safety control system performance in an adverse environment.

## VI. Identification of Followup Actions

This evaluation in response to Mr. Denton's letter focused upon the confirmation that the plant's actual equipment actuation and performance are consistent with that used in the licensing basis analyses. The approach taken was to define potential effects of non-safety-grade control systems in an adverse environment and prepare an assessment to confirm the conclusions reached in the original safety analyses. Justification for continued operation was then based upon the results of this evaluation.

The scope of this investigation did not include long-term availability of

needed operator indications. A complete assessment of environmental effects on non-safety-grade control systems should include an evaluation of equipment required to maintain a safe shutdown following accidents which cause an adverse environment. To address this issue, a future program is being considered. This future program will include evaluations for (1) defining instrumentation and control functions required for safe shutdown, (2) identifying applicable equipment errors and responses in an adverse environment, and (3) preparing safety assessment and developing corrective actions, if required.

This effort will be closely coupled to the Abnormal Transient Operating Guidelines Program currently underway and will focus upon additional operator awareness to recognize and respond to the impact of an adverse environment on non-safety-grade control systems. The schedule for submittal of the Safety Assessment will be consistent with the current schedule for the Abnormal Transient Operating Guidelines Program (i.e., mid-1980).

Also, investigations into appropriate remedial actions for the potential problem area identified in the safety assessment will continue.

TABLE I

TYPICAL EQUIPMENT RESPONSE DURING HIGH ENERGY LINE BREAKS

B&W 177 FA PLANTS

| | | Steam Line Break | Feedwater Line Break | Large LOCA | Small LOCA |
|---|---|---|---|---|---|
| I. | Reactor Power Control and Shutdown | | | | |
| | Trip Function Utilized | High $\phi$ or Low RC Pressure | High RC Pressure | Reactor Trip Not Used | Low RC Pressure |
| | Time of Reactor Trip | 1.1-8.0 sec. | 8.2-13.4 sec. | | |
| II. | Reactor Pressure Control | | | | |
| | Time to PORV Actuation | PORV Not Actuated for Steam Line Break | 4-8 sec. | PORV Response Not Important | PORV not assumed to open |
| | Time at which PORV Closes | | ∿20 sec. | | |
| III. | Steam System Isolation and Pressure Control | | | | |
| (1) | Steam Line Isolation Time | 1.6-8.5 sec. | 6.0-12.0 sec. | Code Safety Valves are Used in the Analyses for Conservatism | Code Safety Valves are Used in the Analyses for Conservatism |
| (2) | Time to Steam Relief Valve Opening | 7.0-16.0 sec. | 7.0-7.5 sec. | | |
| (2) | Time for Steam Relief Valve Closure | 20-30 sec. | 25-30 sec. | | |
| IV. | Feedwater System Isolation and Control | | | | |
| (1) | Main Feedwater Isolation Time | 19-34 sec. | ∿18 sec. | Analysis Conservatively Assumes a Loss of All Feedwater | Not required |
| (1) | Auxiliary Feedwater Isolation Time | 19-34 sec. | ∿18 sec. | | Not required |
| (2) | Auxiliary Feedwater Initiation Time | ∿40 sec. | ∿40 sec. | | ∿40 sec. |

| | Steam Line Break | Feedwater Line Break | Large LOCA | Small LOCA |
|---|---|---|---|---|
| (2) Main or Auxiliary Feedwater Control | Maintain Minimum OTSG Level | Maintain Minimum OTSG Level | | Maintain preset OTSG level |

(1) Affected Steam Generator (2) Unaffected Steam Generator

TABLE II

POTENTIAL ENVIRONMENTAL EFFECTS ON NON-SAFETY GRADE CONTROL SYSTEMS

| Non-Safety Grade Control Systems | Licensing Basis Accidents | | | | | |
|---|---|---|---|---|---|---|
| | SLB Inside Containment | SLB Outside Containment | FWLB Inside Containment | FWLB Outside Containment | Large LOCA | Small LOCA |
| **I. Reactor Power Control and Shutdown** | | | | | | |
| Control Rod Drive Control System | X | X | X | X | X | X |
| **II. Reactor Pressure Control** | | | | | | |
| Power Operated Relief Valve | X | X | X | X | X | X |
| Pressurizer Heaters | Y | - | Y | - | Y | Y |
| Pressurizer Spray | Y | - | Y | - | Y | Y |
| **III. Steam System Isolation and Pressure Control** | | | | | | |
| Turbine Trip/Turbine Stop Valves | - | X | - | X | - | - |
| Steam Line Isolation Valves* | N | N | N | N | N | N |
| Turbine Bypass/Atm Relief Valves** | X | X | X | X | X | X |
| **IV. Feedwater System Isolation and Control** | | | | | | |
| Main Feedwater Control* | X | X | X | X | X | X |
| Main Feedwater Isolation Valves* | N | N | N | N | N | N |
| Auxiliary Feedwater Isolation Valves* | N | N | N | N | N | N |
| Auxiliary Feedwater Initiation** | - | X | - | X | - | - |
| Auxiliary Feedwater Level Control** | X | X | X | X | X | X |

* Affected Steam Generator
** Unaffected Steam Generator

- Environmental Effects Cannot Occur Due to Location of Equipment (Inside containment vs. outside containment)
Y Environment will not affect Safety Analysis Results
X Environment could affect Safety Analysis Results
N Not applicable to Oconee Units

TABLE III

IMPACT OF CONTROL SYSTEM EFFECTS ON SAFETY ANALYSIS

| | Licensing Basis Accidents | | | | | |
|---|---|---|---|---|---|---|
| | SLB Inside Containment | SLB Outside Containment | FWLB Inside Containment | FWLB Outside Containment | Large LOCA | Small LOCA |
| I. Reactor Power Control and Shutdown | | | | | | |
| Control Rod Drive Control System | (2) | (2) | (2) | (2) | (2) | (2) |
| II. Reactor Pressure Control | | | | | | |
| Power Operated Relief Valve | (5) | (1) | (3) | (1) | (3) | (3) |
| Pressurizer Heaters | | | | | | |
| Pressurizer Spray | | | | | | |
| III. Steam System Isolation and Pressure Control | | | | | | |
| Turbine Trip/Turbine Stop Valves | | (2) | | (2) | | |
| Steam Line Isolation Valves | | | | | | |
| Turbine Bypass/Atm Relief Valves | (4) | (4) | (3) | (3) | (3) | (3) |
| IV. Feedwater System Isolation and Control | | | | | | |
| Main Feedwater Control | (4)* | (4)* | (3) | (3) | (3) | (3) |
| Main Feedwater Isolation Valves | | | | | | |
| Auxiliary Feedwater Isolation Valves | | | | | | |
| Auxiliary Feedwater Initiation | | (1a) | | (1a) | | |
| Auxiliary Feedwater Level Control | (1) | (1a) | (1) | (1a) | (1) | (1) |

(1) Equipment Can Be Shown To Perform Intended Function
(1a) See text page 11
(2) Required Period of Operability Is Short
(3) Equipment Performance is Negligible or Conservative in Adverse Environment
(4) Covered by Existing Safety Analysis
(5) Potential Inconsistency With Safety Analysis Inputs and Responses

* Affected Steam Generator

Note: All Open Entries are Either a Dash (-), a N, or a Y on Table II.