NRC FORM 195
(2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER: 50-269/270/287
FILE NUMBER:

TO: B.C. Rusche

FROM: Duke Power Co.
Charlotte, N.C.
W.O. Parker,Jr.

DATE OF DOCUMENT: 8-13-76
DATE RECEIVED: 8-17-76

☒ LETTER
☒ ORIGINAL
☐ COPY
☐ NOTORIZED
☒ UNCLASSIFIED

PROP

INPUT FORM

NUMBER OF COPIES RECEIVED: 1

DESCRIPTION

Ltr. re. their 7-12-76 ltr...and our 6-8-76 ltr
Responses concerning a reassessment of the Reactor Vessel Support System.......

(1 Signed Cy. Received)
(1 Pge)

ENCLOSURE

ACKNOWLEDGED

DO NOT REMOVE

PLANT NAME: Oconee # 1,2 &3

DISTRIBUTION FOR REACTOR VESSEL SUPPORT INFO FOR OPERATING REACTORS PER MR. TRAMMELL 7-12-76

SAB 8-20-76

SAFETY	FOR ACTION/INFORMATION
ASSIGNED AD:	Goller
BRANCH CHIEF:	Schwencer
PROJECT MANAGER:	Zech
LIC. ASST:	Sheppard

INTERNAL DISTRIBUTION

- REG FILE
- NRC PDR
- KNIGHT
- KOSZTOCZY
- CHECK
- C. TRAMMELL
- SHAO
- F. BARANOWSKY
- P. NORIAN
- R. BOSNAK
- V. NOONAN

EXTERNAL DISTRIBUTION

LPDR: Walhalla, S.C.

CONTROL NUMBER: 8412

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

August 13, 1976

Mr. Benard C. Rusche
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer
Operating Reactors Branch Number 1

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

Dear Mr. Rusche:

My letter of July 12, 1976, in response to your transmittal of June 8, 1976 concerning a reassessment of the Oconee Nuclear Station reactor vessel support system, stated that further information would be provided by August 13, 1976.

Accordingly, Duke Power Company has discussed this matter with other utilities that have Babcock and Wilcox (B&W) Nuclear Steam Supply Systems of the same design (177 fuel assembly, skirt supported vessels) and a utility users group has been formed with B&W. Plans and schedules concerning the resolution of this issue are currently being finalized. Subsequently, it is anticipated that a joint utility/B&W/NRC meeting will be scheduled in the near future to discuss the plans and schedule for this resolution.

Very truly yours,

William O. Parker Jr.
William O. Parker, Jr. by WAH

DCH:vr

8412

Docket Nos. 50-269
50-270
and 50-287

AUG 1 2 1976

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

RE: OCONEE NUCLEAR STATION, UNITS NOS. 1, 2, AND 3

Provided herein as Enclosure 1 is a description of events which occurred at Millstone Unit No. 2 during July 1976 relating to plant operation and equipment failures during a degraded grid voltage condition.

On July 27, 1976, all utilities with operating reactor facilities received telephone notification from the NRC of the events at the Millstone Unit No. 2 facility. At that time members of your staff were asked to investigate the vulnerability of your facility to similar degraded voltage conditions and to provide a response by telephone within 24 hours.

As a result of our initial investigation and evaluation of the potential generic implications of the events at Millstone and our preliminary discussions with several licensees, we consider it necessary to require all operating reactor licensees to conduct a thorough evaluation of the problem and to submit formal reports. Therefore, we request that you conduct an investigation of the issue as it affects your facility using the Request for Information detailed in Enclosure 2 as a guide, and provide the analyses and results within 30 days of your receipt of this letter.

The signed original and 39 copies of your response will be necessary.

OFFICE➤						
SURNAME➤						
DATE➤						

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Description of Events
   Millstone Unit No. 2
2. Request for Information

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Mr. Troy B. Conner
Conner & Knotts
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

DISTRIBUTION
NRC PDR(3)
Local PDR
Docket (3)
ORB#1 Reading
KRGoller
TJCarter
OELD
OI&E(3)
GZech
TVerdery
ASchwencer
SMSheppard
DEisenhut
TBAbernathy
JRBuchanan
ACRS(16)
VStello
DRoss

OFFICE	ORB#1	ORB#3	ORB#1			
x27433 SURNAME	GZech:tsb	TVerdery	ASchwencer			
DATE	8/12/76	8/12/76	8/13/76			

ENCLOSURE NO. 1

DESCRIPTION OF EVENTS

MILLSTONE UNIT NO. 2

On July 20, 1976, Northeast Nuclear Energy Company (NNECO) reported that, following a trip of Millstone Unit No. 2 on July 5, 1976, several motors powered from 480 volt (v) motor control centers failed to start as required. The failure of the 480 v motors to start was traced to blown control power fuses on the individual motor controllers. These controllers receive control power through 480 v/120 v transformers within the controller.

NNECO's investigation disclosed that, as a result of the plant trip, the grid voltage dropped from 352 kv to 333 kv. This voltage drop, in conjunction with additional voltage drops associated with the transformers involved, reduced the control power and voltage within individual 480 v controllers to a voltage which was insufficient to actuate the main line controller contactors. As a result, when the motors were signalled to start, the control power fuses were blown. Subsequent testing by NNECO showed that the contactors required at least 410 v to function properly.

NNECO concluded that under similar low voltage conditions, the operability of 480 v Engineered Safety Feature equipment could not be assured.

NNECO's immediate corrective action was to raise the setpoint of the Engineered Safeguards Actuation System (ESAS) "loss of power" undervoltage relays to assure that the plant would be separated from the grid and emergency power system (dual) operation would be initiated before the control voltage fell below that required for contactor operation. A trip of the undervoltage relays causes the emergency buses to be de-energized and a load shed signal to strip the emergency buses, the diesel generators to start and power the emergency buses, and required safety related loads to sequence start on the buses.

On July 21, 1976, NNECO reported that the earlier corrective action taken was no longer considered appropriate because during starting of a circulating water pump, the voltage dropped below the new ESAS undervoltage relay setting. This de-energized the emergency buses, caused load shedding to occur, started the diesel generators and began sequencing loads onto the emergency buses in accordance with the design. However, during sequencing of the loads onto the buses, the voltage again dropped below the undervoltage relay setting which caused the load shed signal to strip the buses. The result was energized emergency buses with no loads supplied.

REQUEST FOR INFORMATION

1. Evaluate the design of your facility's Class IE electrical distribution system to determine if the operability of safety related equipment, including associated control circuitry or instrumentation, can be adversely affected by short term or long term degradation in the grid system voltage within the range where the offsite power is counted on to supply important equipment. Your response should address all but not be limited to the following:

   a. Describe the plant conditions under which the plant auxiliary systems (safety related and non-safety related) will be supplied by offsite power. Include an estimate of the fraction of normal plant operating time in which this is the case.

   b. The voltage used to describe the grid distribution system is usually a "nominal" value. Define the normal operating range of your grid system voltage and the corresponding voltage values at the safety related buses.

   c. The transformers utilized in power systems for providing the required voltage at the various system distribution levels are normally provided with taps to allow voltage adjustment. Provide the results of an analysis of your design to determine if the voltage profiles at the safety related buses are satisfactory for the full load and no load conditions on the system and the range of grid voltage.

   d. Assuming the facility auxiliary loads are being carried by the station generator, provide the voltage profiles at the safety buses for grid voltage at the normal maximum value, the normal minimum value, and at the degraded conditions (high or low voltage, current, etc.) which would require generator trip.

   e. Identify the sensor location and provide the trip setpoint for your facility's Loss of Offsite Power (undervoltage trip) instrumentation. Include the basis for your trip setpoint selection.

   f. Assuming operation on offsite power and degradation of the grid system voltage, provide the voltage values at the safety related buses corresponding to the maximum value of grid voltage and the degraded grid voltage corresponding to the undervoltage trip setpoint.

   g. Utilizing the safety related bus voltage values identified in (f), evaluate the capability of all safety related loads, including related control circuitry and instrumentation, to perform their safety functions. Include a definition of the voltage range over which the safety related components, and non-safety components, can operate continuously in the performance of their design function.

h. Describe the bus voltage monitoring and abnormal voltage alarms available in the control room.

2. The functional safety requirement of the undervoltage trip is to detect the loss of offsite (preferred) power system voltage and initiate the necessary actions required to transfer safety related buses to the onsite power system. Describe the load shedding feature of your design (required prior to transfering to the onsite [diesel generator] systems) and the capability of the onsite systems to perform their function if the load shedding feature is maintained after the diesel generators are connected to their respective safety buses. Describe the bases (if any) for retention or reinstatement of the load shedding function after the diesel generators are connected to their respective buses.

3. Define the facility operating limits (real and reactive power, voltage, frequency and other) established by the grid stability analyses cited in the FSAR. Describe the operating procedures or other provisions presently in effect for assuring that your facility is being operated within these limits.

4. Provide a description of any proposed actions or modifications to your facility based on the results of the analyses performed in response to items 1-3 above.

Docket Nos. 50-269
50-270
and 50-287

AUG 1 1 1976

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

RE: Oconee Nuclear Station Units Nos. 1, 2, and 3

A number of reported instances of reactor vessel overpressurization in Pressurized Water Reactor (PWR) facilities have occurred in which the Technical Specifications implementing 10 CFR Part 50 Appendix G limitations have been exceeded. The majority of cases have occurred during cold shutdown in which the primary system has been in water solid conditions. These overpressurization events have been initiated by a variety of causes, including the following:

(1) Isolation of RHR system/letdown system while charging to a water solid primary system,

(2) Thermal expansion following the starting of a primary coolant pump due to stored thermal energy in steam generators,

(3) Inadvertent actuation of safety injection accumulators, and

(4) Initiation of operation of a reactor coolant pump or a high pressure safety injection pump.

In essentially all of the events reported, a single personnel error, equipment malfunction or procedural deficiency has been sufficient to cause the event.

We believe that appropriate steps should be taken promptly by all PWR licensees to minimize the likelihood of additional occurrences of reactor vessel overpressurization. To that end we recently completed a series of meetings with several PWR licensees and NSSS suppliers in which we discussed the reported overpressurization events and assessed the measures that are currently being employed to either avoid or reduce the probability of similar occurrences,or to control the pressure transient to less than Appendix G limits. Examples of those measures identified by the various licensees are as follows:

7

OFFICE➤						
SURNAME➤						
DATE➤						

(1) Complete avoidance of water solid conditions by either maintaining a pressurizer steam bubble or by providing a low pressure nitrogen blanket in the pressurizer when a steam bubble cannot be maintained,

(2) Disabling High Pressure Injection and Safety Injection pumps by disconnecting electrical power supplies when at low primary system temperatures,

(3) Installation of dual setpoint pressurizer power relief valve(s) to provide protection against exceeding Appendix G limits while at low primary system temperatures,

(4) Minimization of time at water solid conditions and upgrading plant procedures to include appropriate warnings and cautions when such operations are necessary, and

(5) Installation of relief valves in charging pump discharge lines with a setpoint to provide protection against exceeding Appendix G limits.

It was noted in our discussions with the PWR licensees that, for the majority of those plants involved, not all potential overpressurization events would be prevented by the measures they had identified and that some of the remaining measures may have undesirable effects on reactor safety.

Based on the information gathered to date, we have concluded that all PWR licensees should evaluate their system designs to determine susceptibility to overpressurization events. Specifically, you should provide the following:

(1) An analysis of the Reactor Coolant System (RCS) response to pressure transients that can occur during startup and shutdown. Any design modifications determined to be necessary to preclude exceeding Appendix G limits are to be incorporated in this analysis. The analysis should include a plot of pressure as a function of time until termination of the event. The analysis should assume the most limiting initial conditions (e.g., one RHR train operating or available for letdown, other components in normal operation when the system is water solid such as pressurizer heaters and charging pumps, and one or more reactor coolant pumps in operation) with the worst single failure or operator error as the initiating event. Justification should be provided for the choice of limiting conditions and worst single failure or operator error assumed in the analysis.

OFFICE➤						
SURNAME➤						
DATE➤						

(2) A description of those design modifications determined to be necessary, including equipment performance specifications and system operational sequences. The design basis used in the choice of equipment should be included, and

(3) A schedule for the prompt implementation of the proposed design modifications.

The basic criteria to be applied in determining the adequacy of overpressurization protection are that no single equipment failure or single operator error will result in Appendix G limitations being exceeded.

For those situations in which the necessary design changes identified cannot be implemented within the next few months, you should identify short-term measures to reduce the likelihood that overpressurization events will occur in the interim period until the permanent design changes can be made. Short term measures should be identified separately for immediate implementation within the terms and conditions of your license. Short term measures might include some combination of, but would not be limited to, the following suggestions:

(1) Procedural changes to minimize the time in which the primary system is in a water solid condition,

(2) Upgrading existing plant procedures and administrative controls to assure that appropriate warnings and cautions are included to alert the operator whenever the potential for primary system overpressurization exists,

(3) Provide alarms and/or indications to alert the operator whenever primary system pressure increases toward Appendix G limits,

(4) Introducing temporary plant modifications for pressure relief, and

(5) Assignment of additional personnel to monitor plant operations when water solid.

OFFICE➤						
SURNAME➤						
DATE➤						

AUG 1 1 1976

Modifications to preclude or minimize the probability of reactor vessel overpressurization events are plant dependent and the examples given may or may not be adaptable to your specific system design. Consideration must also be given to the potential effects of both the short term and long term measures you consider to assure that other aspects of nuclear safety are not compromised.

To verify compliance with Appendix G pressure-temperature limits during startup and shutdown, you should assure that the appropriate instrumentation is installed to provide a continuous permanent record over the full range of both pressure and temperature. This instrumentation should be in service during long periods of cold shutdown as well as during startup and shutdown operations. Reliance upon the plant computer to reconstruct a pressure transient is not considered sufficient because of the likelihood of computer downtime especially during plant shutdown conditions.

We request that you notify us within 20 days after receipt of this letter that you will provide all the information requested within 60 days or explain why you cannot meet this schedule and provide the schedule that you will meet.

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

FOR DISTRIBUTION SEE PAGE 5

Mr. Troy B. Conner
Conner & Knotts
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

OFFICE➤	ORB#1	ORB#1				
x27433:tsb SURNAME➤	GZech	ASchwencer				
DATE➤	8/11/76	8/ 11 /76				

AUG 1 1 1976

cc: Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

DISTRIBUTION
NRC PDR
Local PDR
Docket(3)
ORB#1 Reading
KRGoller
TJCarter
OELD
OI&E(3)
ASchwencer
GZech
SMSheppard
DEisenhut
TBAbernathy
JRBuchanan
ACRS(16)
VStello
DRoss

OFFICE ➤						
SURNAME ➤						
DATE ➤						