Duke Energy

W. R. McCollum, Jr.
Vice President

Duke Energy
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672
(864) 885-3107 OFFICE
(864) 885-3564 FAX

November 5, 2001

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, 270, and 287
Accident Sequence Precursor (ASP) of Operational Condition - Postulated High Energy Line Breaks in Turbine Building Leading to Failure of Safety-Related 4 kV Switchgear

In a letter dated March 28, 2001, the Nuclear Regulatory Commission (NRC) provided a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational condition that was reported in License Event Report (LER) No. 269/1999-001-01. Duke Energy Generation Services (Duke) responded to the report in a letter dated July 19, 2001.

The ASP addresses postulated high energy line breaks in the Turbine Building leading to the failure of safety-related 4kV switchgear. Duke seeks to provide a licensing position relative to the scenario discussed in the subject ASP report.

As stated in our above response, the subject scenario is not an event, rather it is an approved design feature of Oconee. In a response to NRC questions associated with the Giambusso letter (Supplement 1, to MDS Report No. OS-73.2, dated June 22, 1973), Duke was asked to provide an analysis of the station's ability to mitigate a postulated feedwater line break in the Turbine Building in the area of the 4160 volt switchgear. Duke's response clearly indicated that this scenario is an existing vulnerability, and detailed the means to mitigate such an event. The applicable excerpt from this response is included as Attachment 1 to this letter. Duke's response was subsequently approved by the NRC in an SER dated July 7, 1973. In addition, NRC inspection reports and associated correspondence have recognized this scenario as an approved design feature of Oconee.

A001

In conclusion, this condition has clearly been established with the NRC as a scenario within Oconee's licensing and design basis. Mitigation of this scenario has been factored into plant operations, procedures, training, and modifications since inclusion in the ONS design and licensing basis in 1973. Therefore, this is not an event, but a condition considered to be part of the design and licensing basis of the facility.

If there are any questions or further information is needed, please contact Reene' Gambrell at (864) 885-3364.

Very truly yours,

W. R. McCollum, Jr., Vice President
Oconee Nuclear Site

cc: Mr. L. N. Olshan, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14 H25
Washington, D. C. 20555

Mr. H. N. Berkow, Project Directorate
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14 H25
Washington, D. C. 20555

Mr. L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. C. A. Casto, Division of Reactor Safety
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. V. M. McCree, Division of Reactor Projects
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Station

Mr. Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

Attachment 1

Excerpt From

MDS Report No. OS-73.2, Supplement 1 dated June 22, 1973

Attachment 1
Excerpt From
MDS Report No. OS-73.2, Supplement 1 dated June 22, 1973

Question

Provide an analysis of the station's ability, after design changes are completed, to mitigate a postulated feedwater line break in the turbine building in the area of the 4160 volt switchgear.

Answer

The consequences of the postulated double-ended break of main feedwater lines A at the emergency feedwater connection inside the turbine building has been analyzed under the premise that the design changes as described in Section 4 of the high energy pipe break study, "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, and 3," assure that a redundant emergency feedwater supply line is available to each steam generator for long-term core cooling. The extent of damage to other equipment is assumed to be as follows:

1. Feedwater valve FDW-33 is destroyed.

2. The pipe whip of feedwater line A severs emergency feedwater line connection to main feedwater line B and destroys feedwater valve FDW-42, thus eliminating the normal channels of main and emergency feedwater flow to either steam generator.

3. The 4160 volt switchgear 1TC, 1TD, and 1TE is lost due to direct water/steam impingement.

Without the additional emergency feedwater supply lines to each steam generator the immediate consequences of the accident are similar to those presented in Section 14.1.2.8.3, "Results of a Complete Loss of All Station Power Analysis," of the Final Satety Analysis Report. As further stated in that section, immediate operation of the emergency feedwater system is not of a critical nature, i.e., the reactor can sustain a complete loss of electric power without emergency cooling for about 23 minutes before the pressurizer is filled with reactor coolant and for an additional period of 83 minutes before boil-off of the coolant will start to uncover the core. However, with the addition of emergency feedwater to either steam generator prior to filling the pressurizer with reactor coolant, sufficient decay heat removal can be provided to assure core coverage and the reactor coolant

system can be maintained for an extensive period of time in a hot shutdown condition. Once power is restored to the high pressure injection pumps, the reactor coolant system can then be cooled in an orderly manner with an adequate supply of borated water for coolant makeup and boron control.

The sequence of events and resulting consequences for the postulated feedwater line break with the availability of emergency feedwater are as follows:

1. Termination of all feedwater results in a reduction in secondary system heat removal capability. Feedwater line check valves prevent a secondary system blowdown through the feedwater line break.

2. Loss of electric power results in gravity insertion of control rods. Even if power is available after the break, increased reactor coolant system temperature and pressure result in a high pressure reactor trip within 15 seconds after the loss of feedwater.

3. Following reactor trip, turbine trip occurs with the closure of the turbine stop valves.

4. The main steam safety valves actuate after the turbine stop valves close to prevent excessive temperatures and pressures in the reactor coolant system. The safety valves close after about 20 seconds of steam relief if steam flow through the turbine bypass valves is available to relieve excess steam and provide for decay heat removal.

5. Thermal equilibrium is re-established in the reactor coolant system, i.e., the heat removal rate provided by steam relief is equal to the core decay heat input.

6. Once the steam generator liquid inventories have been vaporized in about nine minutes, the RCS will begin to heat up with actuation of the pressurizer safety valves at 2515 psia within five minutes after the steam generators are dry.

7. Steam relief by the pressurizer safety valves will continue until emergency feedwater flow is established to either steam generator within 15 minutes after the break. Since the addition of emergency feedwater to either steam generator occurs within the 23 minute period described in FSAR Section 14.1.2.8.3, and is sufficient for decay heat removal, the pressurizer is prevented from filling with reactor coolant.

8. The operator can then re-establish thermal equilibrium and begin plant cooldown at this time by emergency feedwater control and steam relief to the condenser or the atmosphere.

9. Prior to plant cooldown, the operator must manually restore power to any one of three high pressure injection pumps. Power to pumps is not a part of the 4160 volt switchgear affected by the accident but comes directly from the 4160 volt main feeder buses (See figure 8-3 of the FSAR). These actions can be easily accomplished within a 30 minute time period.

10. The operator utilizes high pressure injection flow for makeup and boron control during plant cooldown.

The postulated feedwater line break results in a reactor trip followed by reactor coolant system heatup prior to the orderly control of the transient by the operator so that the core can always be maintained in a subcritical condition. Also, the reactor coolant system pressure does not exceed code design limits at any time during the transient. Therefore, the core integrity is maintained during this event and an orderly cooldown to cold shutdown condition is accomplished by the installation of the redundant emergency feedwater supply lines.