UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30323-0199

MAR 3 1 1994

MEMORANDUM FOR: Gus C. Lainas, Assistant Director for Region II Reactors
Division of Reactor Projects I/II Office of Nuclear Reactor Regulation

FROM: Jon R. Johnson, Acting Director Division of Reactor Projects

SUBJECT: TASK INTERFACE AGREEMENT (TIA 94-013) TECHNICAL ASSISTANCE REQUEST, SAFETY IMPLICATIONS RELATED TO A POTENTIAL STEAM LINE BREAK BLOWING DOWN BOTH STEAM GENERATORS AT OCONEE

The purpose of this TIA is to determine if the Oconee licensee is operating their steam system in a condition which involves an unreviewed safety question and to evaluate the safety significance of a potential uncontrolled blowdown of both steam generators. The Oconee Main Steam System design includes two six inch pipe branches from main steam headers "A" and "B" that join into an eight inch line which supplies the startup steam header. Each six inch line has a motor operated isolation valve (MS-24 & 33) which according to FSAR Section 10.3.2, prevents blowdown of both steam generators from a single leak in the system. These valves are normally open and require operator action to close. Additionally, they are not classified by the licensee as safety-related, do not have 1E power supplies and are load shed following a loss of offsite power. The piping downstream of these isolation valves is not safety-related or seismically qualified (Oconee class "G"). Only one unit at a time supplies the startup steam header, which supplies all three units with auxiliary steam.

MS-24 & 33 were included in the licensee's Generic Letter (GL) 89-10 program because they are used to mitigate design basis licensing events, e.g. steam line break, and steam generator tube rupture. As part of the GL 89-10 program, the licensee determined by calculation that these valves would not close under the maximum differential pressure that would be developed initially following the failure of the class G piping downstream of the valves (differential pressure could initially be around 1050 psig, whereas the MOVs had been calculated to be able to close under 400 psig differential pressure). This determination resulted in the licensee questioning the ability of these valves to perform their intended safety function. This concern was evaluated under evaluation OSC-5060 (Enclosure 1). This operability evaluation essentially states that the valves are operable provided efforts to close the valves continue even if the initial attempt fails due to excessive differential pressure across the valves. The rationale for this conclusion was based on the assertion that "the exact pressure at which the valves close is not critical, so long as they do close."

The licensee felt that the pressure at which the valves would close was not critical because the steam line break evaluated in FSAR Section 15.13 was much larger than the postulated break in question, and was therefore bounding. Additionally, the licensee felt that by the time operators would take action to close the valves (ten minutes into the event, per the licensee) the steam

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pressure in both steam generators would have blown down to the point where differential pressure across the valve would not prohibit valve closure.

The region does not agree that the exact pressure at which the valves close is not critical or that the steam line break addressed in Section 15 of the FSAR is bounding in all respects. On December 13, 1993, in response to questions from the resident inspectors, the licensee provided calculated values of steam generator pressure versus time, up to the point the valves would probably be closed (assumed to be 10 minutes after the initiating event according to the licensee). The data indicated that both steam generators would be blown dry within 10 minutes. A notice of deviation was written (93-31-01) as a result (NRC Inspection report Nos. 50-269,270,287/93-31 Enclosure 2).

The licensee's response to the deviation (Enclosure 3) agreed that a postulated break in certain portions of the steam supply lines to the auxiliary steam header could lead to a simultaneous blowdown of both steam generators. However, they concluded that the break in question was bounded by existing FSAR analyses and that no corrective action was necessary other than changing the wording in the FSAR.

The licensee's response does not address all the safety implications associated with the blowdown of both steam generators from the postulated break. The break postulated in Section 15.13 of the FSAR, while considerably larger in size, is postulated to blowdown only one steam generator; we therefore fail to understand why this is bounding. We would agree that the resulting cooldown of the RCS would be worst case from a double ended rupture of a main steam line. However, the blowdown of both steam generators presents significant challenges to decay heat removal once the initial cooldown is completed.

For both the steam line break evaluated in Section 15, and the break postulated here, the blowdown results in a rapid RCS cooldown and corresponding RCS volume shrinkage which will cause a loss of subcooling margin (probable formation of steam bubbles in the hot legs and reactor vessel) requiring the operators to trip all reactor coolant pumps. However, the plant response from this point would be significantly different for the two scenarios. For the steam line break evaluated in Section 15, there is always an intact steam generator with feedwater level already established, therefore once the RCS begins to heat back up, natural circulation is easily established. For a break that blows down both steam generators, the RCS conditions at the end of the blowdown would be no forced circulation or natural circulation flow for some period of time. At this point RCS temperature and pressure will rapidly rise due to the isolation of all sources of feedwater to the steam generators. The resulting mitigation strategy would, at best, require trickle feeding hot, dry steam generators with cool emergency feedwater, or High Pressure Injection forced cooling. Neither of these rather drastic mitigation strategies are typically required following a steam line break. These concerns have not been adequately addressed by the licensee.

The regional staff knows of no other operating units where a single failure vulnerability to blowdown all a unit's steam generators is considered an acceptable part of the licensee's design or licensing basis.

The regional staff feels this issue has safety significance. The steam line in question is not Quality Related or seismically qualified and it is therefore not unreasonable to postulate its failure. Oconee's design does not incorporate any automatic features for steam break isolation upstream of the turbine stop valves. Typically, safety-grade identification and isolation systems are required for mitigation of steam system break events. The valves necessary to isolate the steam break in question (MS-24 & 33) are normally open, require operator action to close, and are load shed following a loss of offsite power. Operator response alone is available to mitigate this event, which would involve either feeding hot, dry steam generators with cool emergency feedwater or high pressure injection forced cooling. In essence, the postulated event is credible and could lead to more adverse consequences than would occur at other plants.

We therefore request you and your staff review this issue to determine:

1. Does the postulated event involve an unreviewed safety question per 10 CFR 50.59 in that it presents the possibility for an accident or malfunction of a different type than any evaluated in the safety analysis report?

2. Is the postulated break bounded by the steam line break evaluated in Section 15.13?

3. What are the safety implications? Should the licensee be required to eliminate this vulnerability?

4. Is the single failure vulnerability in question reportable under 10 CFR 50.72 as an unanalyzed condition or a single event that alone can prevent the fulfillment of a safety function.

If you have any questions regarding this matter, please contact Mark Lesser at (404) 331-0342 or Lee Keller at (803) 882-6927.

Jon R. Johnson

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Enclosures:
1. Oconee Engineering Evaluation OSC-5060
2. NRC Inspection Report Nos. 50-269, 270, 287/93-31
3. Oconee's Response to Deviation 93-31-01

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Should you have any questions concerning this letter, please contact us.

Sincerely,

/S/

Charles A. Casto, Chief
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Docket No. 50-302
License No. DPR-72

Enclosure:
NRC Inspection Report

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