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ACCESSION NBR:7910100415 DOC.DATE: 79/10/05 NOTARIZED: NO DOCKET #
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STELLO,V. Office of Inspection & Enforcement, Office of the Director

SUBJECT: Forwards info re potnetial unreviewed safety question on interaction between nonsafety grade sys & safety grade sys, in response to NRC 791005 ltr.No scenarios considered constitute hazard to public or basis to modify/revoke OL.

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KEITH W. AMISH
EXECUTIVE VICE PRESIDENT

TELEPHONE
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October 5, 1979

Mr. Victor Stello, Jr.
Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stello:

This letter is in response to an NRC Staff communication dated September 17, 1979 entitled "Potential Unreviewed Safety Question on Interaction Between Non-Safety Grade Systems and Safety Grade Systems." Rochester Gas and Electric has reviewed the applicability to Ginna of the Westinghouse concerns regarding the potential malfunction of non-safety related control systems due to adverse environment. Based on our review of the four scenarios judged by Westinghouse to have the most severe potential consequences, we have concluded that none of these scenarios constitute any hazard to the health and safety of the public. Three of the scenarios are bounded by previously documented and NRC-approved analyses. The fourth scenario, a small steam line break with rod withdrawal prior to trip, is not expected to be of any significance with respect to fuel damage or offsite radiation release. Attachment 1 lists the four noted areas of concern, and the evaluations made to judge the adequacy of Ginna systems to respond to these concerns.

RG&E is further planning to perform evaluations of any other possible interactions of non-safety related control systems and adverse environments. These evaluations will be performed in conjunction with Topics III-12, III-5.A, III-5.B, XV-2, and XV-6 of the Systematic Evaluation Program. Based on the evaluations presented in Attachment 1, RG&E considers that there is no basis that the RG&E NRC license to operate the R.E. Ginna Nuclear Station should be modified, suspended, or revoked.

Very truly yours,

Keith W. Amish

Keith W. Amish

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Subscribed and sworn to me
on this 5th day of October 1979

Stephen Kawba

NOTARY PUBLIC, State of N.Y., Monroe County
My Commission Expires March 30, 1980

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ATTACHMENT 1

SCENARIO 1: FEEDWATER LINE BREAK IN THE INTERMEDIATE BUILDING

Potential Control System Failure: A feedwater line break in the intermediate building creates an adverse environment. The atmospheric steam dump valve controller causes the valve to fail open, resulting in an uncontrolled steam release from the unbroken loop.

Evaluation: For the Ginna two-loop plant, this accident sequence results in a small feedwater line break in one loop, and a small steam line break in the other loop. The feedwater line break size in the area of concern (between the containment and the check valve) is restricted by the approved Ginna augmented inservice inspection program to a 2.9 inch diameter break. [Ref. 1]

The combination of a small feedwater line break and a small steam line break results in a smaller heat removal rate from the reactor coolant system (RCS) than the design basis cooldown event - a double-ended steam line break upstream of the steam line flow limiter.

Ginna has a safety-grade redundant auxiliary feedwater system (the standby AFW system) located separately from any adverse environment resultant from the postulated feedwater line break in the intermediate building. Operator action to align this source of auxiliary feedwater to the steam generator which did not have the postulated feedwater line break can be taken from the control room. Feedwater inventory is such that this action would not be required for in excess of 10 minutes (based on 10 minute operator action being acceptable for a double-ended feedwater line break, a much more severe loss of secondary inventory than this scenario). [Ref. 2, 3]

Once the standby auxiliary feedwater system is properly aligned, its capacity is such that it can more than compensate for the stuck open atmospheric dump valve.

When the steam generator which suffered the postulated feedwater line break is emptied, the accident mitigation systems will function exactly as required to mitigate the consequences of a small one-loop operation steam line break, an accident with only minor consequences, as reported in the Ginna FSAR and subsequently analyzed in Ginna transient analyses required during reload applications. [Ref. 4, 5] Thus, the postulated event is within the bounds of previous analysis.

SCENARIO 2: SMALL FEEDWATER LINE BREAK IN THE INTERMEDIATE BUILDING

Potential Control System Failure: A small feedwater line break can cause an adverse environment. This environment could cause the feedwater control system to malfunction, reducing flow to the unaffected steam generator, and thus resulting in a lesser secondary inventory in that steam generator at the time of reactor trip than originally assumed in the plant safety analysis (narrow range instrumentation at low-low level (15%), rather than at low level (30%)).

Evaluation: The reason this particular scenario can be of concern is that there might be a more rapid loss of secondary coolant heat sink than assumed in the plant safety analysis, (and thus a greater potential for fuel clad damage due to loss of heat sink). However, for the Ginna plant, this particular scenario is bounded by the original feedwater line break analysis (Ref. 3). In this original analysis, the assumed feedwater flow rate out the break was 2,500 lb/sec, with the unaffected steam generator at the low level setpoint at the time of reactor trip. To be able to cause feedwater controller failure together with a steam generator blowdown, the break in the newly-postulated scenario must occur in the intermediate building upstream of the first check valve, where the augmented Inservice Inspection program [Ref. 1] restricts the break size to a 2.9 inch diameter break. This results in a much slower loss of secondary inventory than the design basis full-diameter feedwater line break, which might occur inside containment. Even with the unaffected steam generator at "low-low level" at the time of reactor trip (a difference of about 9400 lb less than "low level"), the overall quantity of secondary coolant available for cooling of the RCS during this postulated scenario is significantly greater than that in the original feedwater line break analysis performed for Ginna. Thus the postulated event is within the bounds of previous analysis.

SCENARIO 3: FEEDWATER LINE BREAK INSIDE CONTAINMENT

Potential Control System Failure: The adverse environment could affect the pressurizer power operated relief valve (PORV) control system, causing the relief valves to fail open and remain open.

Evaluation: The feedwater line break with concurrent loss of auxiliary feedwater, which results in pressurizer relief (through the safety valves, rather than the PORVs), has previously been analyzed and found acceptable. [Ref. 2, 3] Since the pressurizer safety valve capacity is greater than that of the PORVs (576,000 lb/hr vs. 360,000 lb/hr), the previously analyzed accident is considered more severe both in terms of loss of reactor coolant inventory and containment mass and energy release.

The duration of reactor coolant blowdown in the analyzed feedwater line break accident was terminated in 30 minutes. This newly-postulated scenario would result in the continued blowdown of reactor coolant, becoming, essentially, a small LOCA. This accident could be mitigated either by the use of operator action to close the PORV block valves, thus terminating the LOCA, or by following established LOCA procedures. The ability of the Westinghouse PWR's to withstand this type of loss of coolant accident has been recently and thoroughly documented in WCAP-9600 [Ref. 6].

SCENARIO 4: INTERMEDIATE SIZE STEAM LINE BREAK INSIDE CONTAINMENT

Potential Control System Failure: The steam line break environment could cause an adverse environment inside containment, resulting in improper functioning of the excore neutron flux detectors. These detectors could thus send a spurious "low flux" signal to the automatic rod control system, causing the control rods to begin to improperly step out, prior to reactor trip. Core power level and neutron flux at the time of reactor trip would be greater under such conditions than assumed in the plant transient analysis.

Evaluation: The concern of this particular scenario is the potential for exceeding DNBR limits specified in the FSAR for a steam line break event (therefore increasing the potential for clad failure). The particular circumstances and assumptions of this accident sequence make it a relatively low probability event (Reference 7):

a) The steam line break must be between 0.1 and 0.25 ft^2

b) It must occur at 70-100% power

c) Conservative end of life core characteristics and instrument errors must exist

d) The steam line break environment must make the excore neutron flux detectors malfunction such that they send spurious "low flux" signal to the automatic rod control system, without causing a rod withdrawal block on negative flux rate. If the flux detectors continue to operate normally, or initiate a spurious high flux signal, causing reactor trip, this scenario would not be of concern.

Large and small steam line break analyses have recently been performed by Ginna. (Ref. 5) Even though these analyses used a very conservative F_Q^T of 2.80 (as opposed to the 2.32 Ginna is restricted to in its Plant Technical Specifications), it was shown that for a large steam line break, MDNBR was only 1.58, and for a small steam line break, DNBR was not even an acceptance criteria, since no return to criticality was expected. It thus appears that the Ginna plant has substantial margin prior to anticipating DNBR concerns.

Additionally, Westinghouse has performed a typical bounding analysis (Reference 7) of this scenario to calculate the extent of fuel damage. Based on the reduction of radial peaking factors with burnup, and using conservative end-of-life physics parameters, no fuel damage was calculated to occur.

Since there is no fuel damage anticipated as a result of these evaluations and analyses, and there is no loss of reactor coolant system integrity or containment integrity as a result of this scenario, and since the particular set of circumstances required to comprise this scenario made of a relatively low probability event, it is apparent that this scenario is of little consequence relative to overall plant safety. RG&E therefore feels that no additional commitments regarding this scenario need be made.

References

1. Amendment No. 7 to the Ginna license dated May 14, 1975.
2. Amendment No. 29 to the Ginna license dated August 24, 1979.
3. Letter dated May 24, 1974 from K. W. Amish, RG&E, to J. F. O'Leary, USAEC.
4. Application for Amendment to Operating License, September 12, 1975.
5. XN-NF-77-40, Plant Transient Analysis for the R. E. Ginna Unit No. 1 Nuclear Power Plant, submitted with Application for Amendment to Operating License, January 4, 1978.
6. WCAP 9600, Report on Small Break Accidents for Westinghouse NSSS System.
7. Letter dated September 27, 1979 from F. Noon, Westinghouse, to Bruce Snow, RG&E, "Environmental Interaction."