EVALUATION OF

STEAM LINE BREAK CONSEQUENCES

ASSOCIATED WITH

REMOVAL OF RUPTURE MATRIX SIGNALS

FROM

EMERGENCY FEEDWATER VALVES

Prepared For

Florida Power Corporation CR-3 Project

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SUMMARY

The present design of the CR-3 Steam Line Rupture Matrix System (SLRMS) isolates main steam, main feedwater, and emergency feedwater in the event of a main steam line or main feedwater line break or other transients which result in low steam header pressure signals. A substantial improvement in secondary cooling reliability can be achieved by elimination of the automatic isolation of the emergency feedwater system by the SLRMS. Such a change, however, may result in more severe steam line break consequences than presently analyzed in the CR-3 FSAR. This report evaluates the containment pressure and core reactivity response resulting from continued addition of emergency feedwater to the affected steam generator following a steam line break. The conclusion of the study is that the incremental risk of containment over-pressurization or return to criticality is negligible. Since the benefits in terms of secondary cooling reliability are expected to be significant, implementation of the change is recommended.

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I. Background and Scope

One of the principal objectives of the CR-3 Nuclear Safety Task Force was to identify design or operational features which could lead to loss of all secondary cooling, thus requiring operator action and increasing the potential for water relief through the pressurizer relief or safety valves. A number of simplified event tree analyses were performed by the Task Force. Several of these event tree analyses, including those for steam line break, loss of main feedwater, excessive main feedwater, loss of AC power, and loss of ICS power, include sequences which lead to initiation of the steam line rupture matrix. This, in turn, causes isolation of both main and auxiliary feedwater, and without subsequent operator action, prevents heat removal on the secondary system and leads to the HPI heat removal mode. Because of the comparitively low probabilities assigned to operator action, the event trees thus show an undesirably high likelihood of ending up in this less desirable cooling mode.

In light of these conclusions and similar conclusions reached by the Task Force in reviewing the interaction of the Steam Line Rupture Matrix System and the EFW system, the Task Force recommended, subject to NRC staff approval, that the rupture matrix signals be deleted from the emergency feedwater valves.

These conclusions and the associated recommendation are fully consistent with recommendation 4.A of NUREG-0667, "Transient Response of B&W Designed Reactors" dated May, 1980, which states:

> "4. Steam Line Break Detection and Mitigation System
>
> A. Eliminate Adverse Interaction with AFW System"

NUREG-0667, page 7-26, Item "f" further supports this recommendation as follows:

> "f. Main steam and feedwater line break design bases
>
> Main steam and feedwater line breaks have been taken as design basis challenges for the AFWS in some but not all operating PWRs. AFW must be isolated from the affected steam generator and yet AFW must be supplied to the surviving steam generator(s) despite a single active failure.

Such accidents pose very little risk. They are rare and they do not directly threaten core cooling. We see virtually no risk reduction potential in extending these requirements to all PWRs, and the requirements might safely be relaxed where the provisions for automatic isolation of the "affected" steam generator or the valving necessary to satisfy the single failure criterion is found to degrade AFWS functional reliability for the very much more common loss of feedwater events."

While there are compelling reasons for making the change (removal of the rupture matrix signals from the EFW valves), the Crystal River-3 Final Safety Analysis Report includes analysis of the double-ended steam line break which assumes that EFW is isolated and thus does not contribute to the mass and energy release to the containment or core reactivity effects. The purpose of this report is to evaluate the containment pressure and core reactivity response resulting from continued addition of emergency feedwater to the affected steam generator following the design basis steam line break analyzed in the CR-3 FSAR.

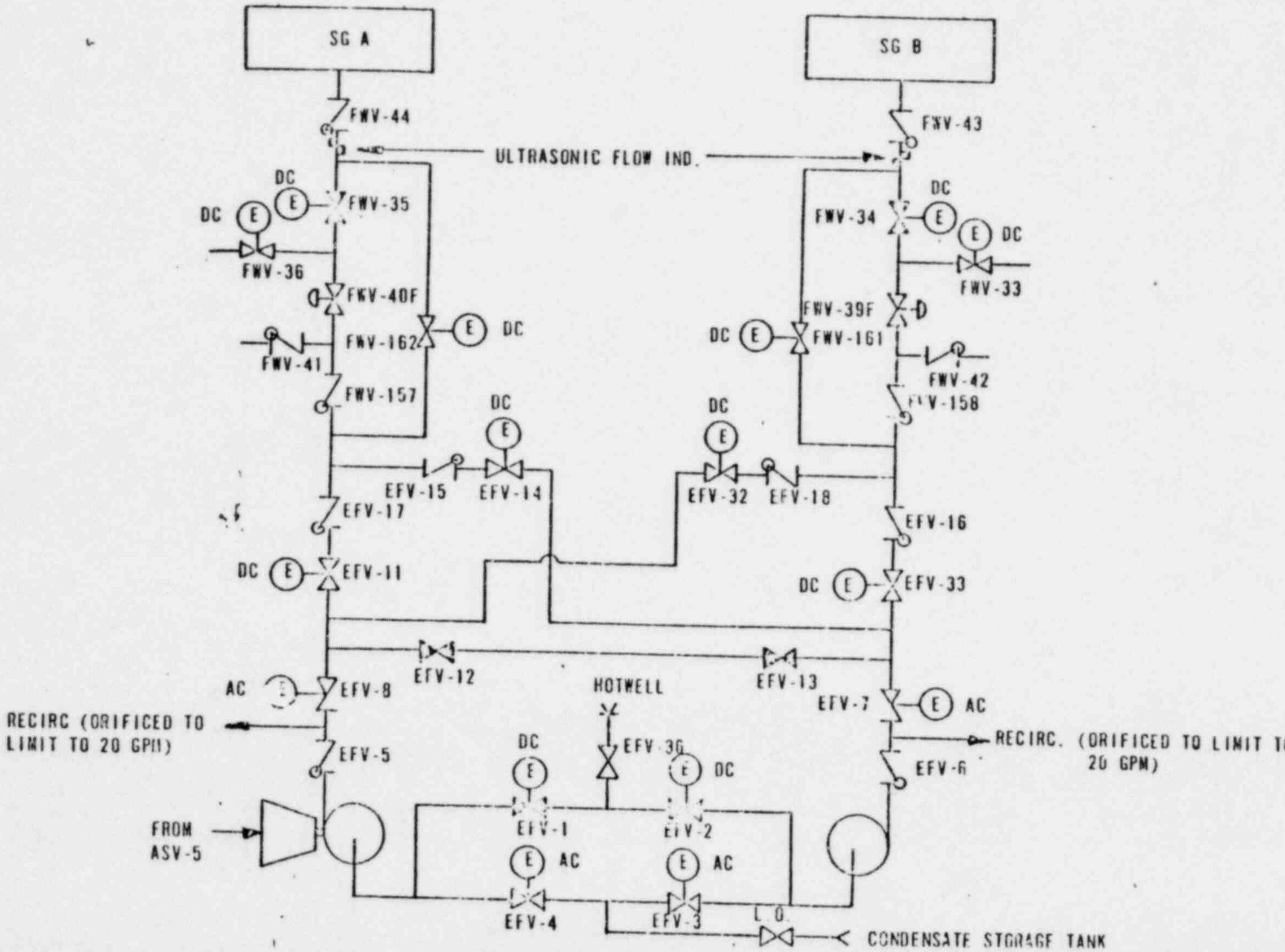
II. Change Description and Associated Analytical Parameters

The attached Figure 1 depicts the present CR-3 EFW system. Valves FWV-161 and 162 are preset at 22% open to pass a minimum of 500 gpm. The Steam Line Rupture Matrix System isolates valves FWV-161 and 162, FWV 33 through 36 as well as main feedwater and main steam if low pressure is sensed in both steam generators. The proposed short-term change would eliminate the rupture matrix signals to FWV-161 and 162, thus allowing EFW to be delivered to both steam generators, even if the rupture matrix actuates on both steam generators. (Note: The CR-3 Nuclear Safety Task Force also recommended further investigations in the long-term of possible changes to the Rupture Matrix System to provide automatic isolation of the EFW to the effected steam generator without degrading the reliability of the EFW system for other events).

Gilbert Associates Incorporated has evaluated the proposed change and provided the necessary input for the containment analysis (reference T. C. Reitz's letter to E. C. Simpson, FCA-1110 dated 4/25/80). The case which provides the maximum EFW flow to the affected steam generator is both EFW pumps operating and the intact steam generator at 1050 psig (the setting of the lowest bank of steam safeties) and the affected generator at 0 psig. For this case, the flow to the affected generator is limited by the preset valve (FWV-161 or 162) to 880 gpm. GAI also advised in the referenced letter that containment design pressure is 55 psig and that the building spray flow rate and delay time are 3,000 gpm and 68.2 seconds, respectively. These values were used in the analysis described in the next section.

FIG. 1

CRYSTAL RIVER UNIT 3-AFWS



III. Containment Overpressurization Analysis

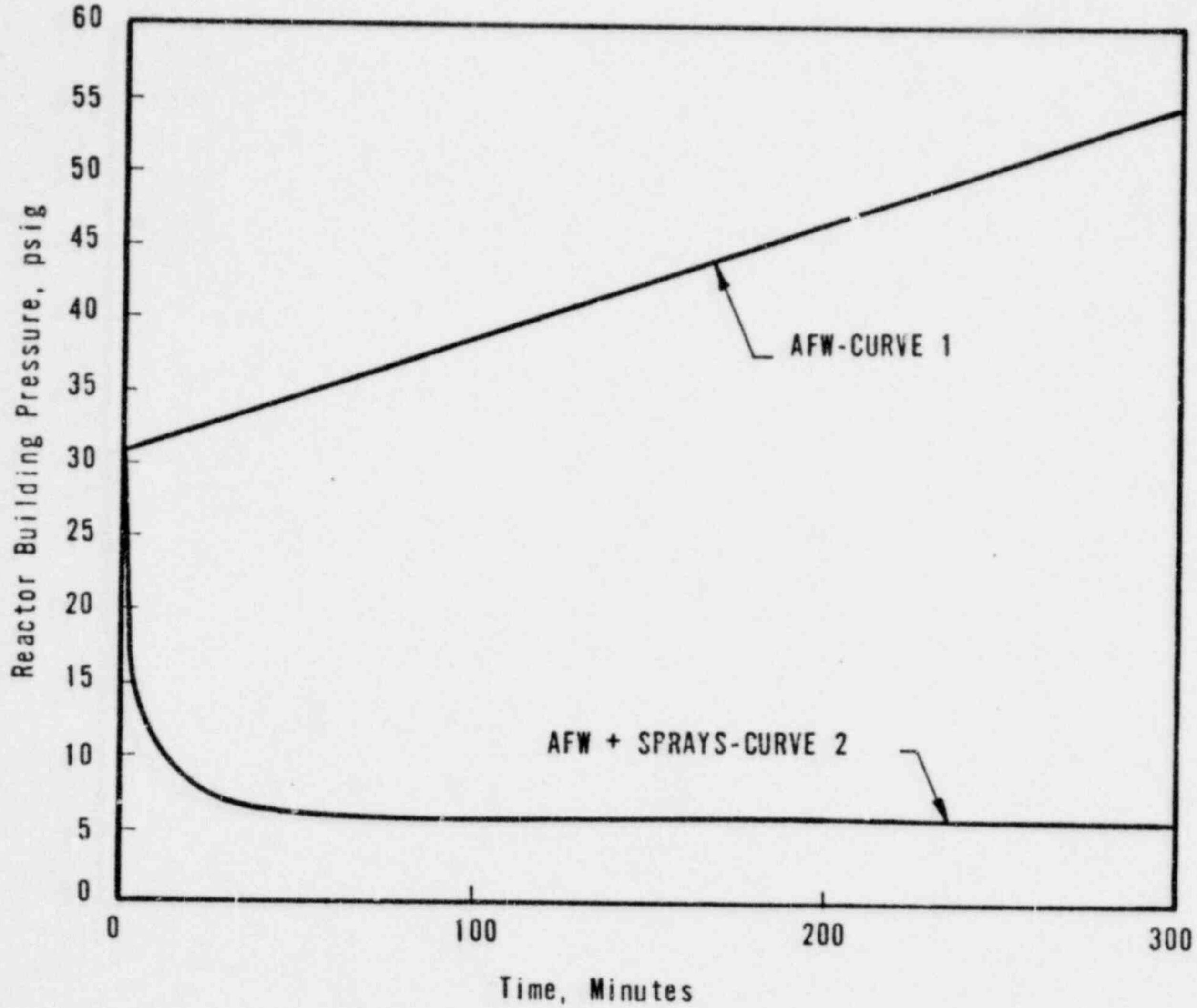
An evaluation of the reactor building pressure response to a double-ended steam line break followed by unmitigated emergency feedwater flow has been made. The effect of the mass and energy releases on the reactor building pressure has been evaluated using the methodology described in Section 14.2.2.1.5 of the Crystal River III FSAR. It was assumed that the containment is an adiabatic closed system. Steam generator blowdown is assumed to be instantaneously released to the containment as saturated steam. Emergency feedwater is added to the blowdown as a function of time (880 gpm @ 1170 BTU/lbm). Static calculations were then performed at specific times in order to determine the duration of unmitigated EFW flow that would be required to exceed the reactor building design pressure of 55 psig. It has been determined that approximately 5 hours of continuous EFW flow released to the containment as saturated steam would be required to raise the containment pressure to it's design limit.

The reactor building spray system is normally used for heat removal and pressure suppression in an elevated containment environment. Using reactor building sprays of 3000 gpm provides significant pressure suppression, such that the energy removal rate of the sprays is sufficient to compensate for extended EFW flows on the order of many hours.

A curve of reactor building pressure versus time after EFW initiation, with and without sprays is shown in Figure 2. Sufficient time exists to allow for operator action to evaluate the steam generator situation and terminate EFW to the affected steam generator.

Containment evaluations have been performed for several B&W plants considering unmitigated secondary system releases either from main or emergency systems. It can be concluded from these studies that main feedwater isolation following SLB is essential to prevent overpressurization in relatively short periods of time. Releases typical of emergency feedwater flowrates can be handled by building sprays for extended periods of time.

FIGURE 2. CRYSTAL RIVER III - REACTOR BUILDING PRESSURE VS. TIME OF AFW FLOW



IV. Assessment of Potential Reactivity Consequences

In addition to the potential for containment overpressurization addressed in Section III, the continual addition of EFW to the affected steam generator following a double-ended steam line rupture contributes to the overcooling and potential for recriticality and must be evaluated as a separate consequence. However, as described below, the probability of a steam line break which results in a return to criticality is small enough to be regarded as incredible. Further, the present CR-3 FSAR analysis of a return to power following an SLB is expected to be similar to the case of continued EFW flow to the affected steam generator.

The limiting case for assessing return to power is the double-ended rupture of the main steam line with the most reactive control rod stuck out of the core. Further, EOL core conditions (conservative maximum negative moderator coefficient) and other assumed conservatisms are employed. Additional overcooling caused by preventing isolation of the affected steam generator could lead to a return to power.

The probability of a main steam line break in the size range of interest has been estimated to be $\sim 1 \times 10^{-4}$ per reactor year in the Rasmussen Report (WASH 1400). B&W has previously estimated the probability of any MSLB (including small breaks) to be 1.8×10^{-4} per reactor year. For purposes of this evaluation, a conservative probability of a double-ended rupture of the main steam line was selected as 1×10^{-4} per reactor year.

Even if such an event were to occur, and the overcooling effect were to be increased beyond that analyzed, no return to criticality would result if all control rods were to drop. Therefore, an evaluation was made to determine the probability of any control rod to not trip on demand. The NRC Gray Book reports that, as of June, 1979, there had been 253 reactor trips at B&W operating plants (excluding TMI-2). In no case was there a failure of any control rod to fully insert. Using an upper 50% confidence level estimate for the failure of any particular rod to insert and assuming a Poisson distribution for such failures, the probability of at least one rod sticking in any scram demand is calculated to be 2.75×10^{-3} per trip demand. (Note that corresponding probability of the most reactive rod not inserting is $\sim 5 \times 10^{-5}$ per trip demand).

Based on the above, a conservative combined probability of any stuck rod concurrent with a double-ended main steam line break is shown to be less than 2.75×10^{-7} per reactor year.

It is concluded that, even if events were to occur which could greatly increase the overcooling effect associated with main steam line break, the probability of such an event leading to recriticality is acceptably small.

Section 14.2.2.1.3 of the CR-3 FSAR evaluates the return to criticality associated with continuous flow of main feedwater to the affected steam generator, and concludes that the consequences are acceptable. While no analysis was performed for the case of continued EFW flow, it is likely that such an analysis would show results very similar to the FSAR analysis results. (The FSAR analysis is terminated by core flood tank discharge and a comparable termination of continued overcooling by the EFW would be expected.)

In summary, the probability of an SLB which results in recriticality is negligibly small, and, though not specifically analyzed, the continued EFW flow case would be expected to show results similar to the present FSAR analysis results.

V. Conclusions and Recommendations

Based on the foregoing evaluations and analysis, the following conclusions were reached:

1) If the containment spray system actuates, no operator action is required to prevent exceeding the containment design pressure following a design basis steam line break. Even if the spray system and passive heat sinks are conservatively neglected, approximately 5 hours are available for operator action to isolate emergency feedwater to the affected steam generator before the containment design pressure would be exceeded.

2) The probability of a steam line break is low ($\sim 10^{-4}$ per reactor year). The probability of an SLB and a stuck control rod (a necessary prerequisite for a return to criticality) is conservatively estimated to be less than 2.75×10^{-7} per reactor year. Furthermore, though not specifically analyzed, the continued EFW flow case would be expected to show results similar to the FSAR analysis results for continued MFW flow.

3) Based on 1) and 2) above, we conclude that the incremental risk of containment overpressurization or return to criticality following a double-ended SLB and no automatic isolation of EFW to the affected steam generator is negligible.

4) The proposed change (elimination of the rupture matrix signals from the EFW valves) is consistent with the NRC recommendations (NUREG-0667) and similar analysis performed for other projects.

5) In light of the expected significant improvements in secondary cooling reliability which can be realized and the corresponding minimal risk of containment overpressurization or core damage which would result, the proposed change should be implemented.