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RISK EVALUATION OF THE ALTERNATE-3A MODIFICATION TO THE ATWS PREVENTION/MITIGATION SYSTEM IN A BWR-4, MARK-II POWER PLANT

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Introduction

The purpose of this paper is to present a risk evaluation of the ATWS Alternate 3A modification (ATWS-3A) proposed by NRC staff in NUREG-0460[1] to the ATWS prevention/mitigation system in a BWR nuclear power plant. The evaluation is done relative to three risk indices: the frequency of core damage, the expected early fatalities, and the expected latent fatalities.

The evaluation of the ATWS-3A modification is based on the Probabilistic Risk Assessment study of the Limerick Generating Station (LGS-PRA)[2] as it was revised by BNL[3]. Two analyses were performed; one assumed the present design of the ATWS prevention/mitigation system of the Limerick plant, and the other assumed that the features of the system attributable to the ATWS-3A modification were removed. The risk indices are evaluated and compared in both cases. In the LGS-PRA[2] and in the BNL revision[3] a detailed analysis of the ATWS prevention/mitigation function was performed using event trees to analyze the possible combinations of subsystem failures and their effects. This constitutes a realistic, less conservative approach to the evaluation of the ATWS contribution to the core damage frequency and to the risk, compared to assuming that failure to scram results in core damage with probability of unity.

A summary of the basic characteristics and results of the BNL revision to the LGS-PRA is given in Section 2. A short description of the ATWS-3A prevention/mitigation system of the LGS is presented in Section 3. The modeling of the ATWS sequences in the LGS-PRA is described in Section 4. The effects of removing the ATWS-3A modification are discussed in Section 5. Finally, the results and conclusions are given in Section 6.

2. Summary of the BNL revision of the LGS-PRA

The LGS-PRA was initially submitted to NRC in March 1981. Subsequently, in September 1981 and in December 1981, Revisions 1 and 2, respectively, were submitted to NRC. BNL, under an NRC contract, began the review of the PRA in January 1982 and the findings of this review were issued in NUREG/CR-3028 in February 1983.

The LGS-PRA employed an "extended" WASH-1400 methodology which included a more detailed accident sequence delineation and quantification as well as a more detailed analysis of core melt phenomenology. In addition, the LGS-PRA considered the design differences between the Limerick plant and the WASH-1400 BWR plant. The accident sequence analysis considered the following eight groups of accident initiators:

(i) Turbine Trip
(ii) Loss of Main Feedwater and Power Conversion System (or MSIV closure).
(iii) Loss of Offsite Power (LOOP)
(iv) Inadvertent Opening of the Relief Valve (IORV)
(v) Manual and Orderly Shutdown
(vi) Large LOCAs
(vii) Medium LOCAs
(viii) Small LOCAs

A failure to scram following any of the first four initiators will result in ATWS sequences which were analyzed separately.

The accident sequences were divided into four classes (plant damage states) as follows:

(i)	Class I	Core damage sequences are characterized by the loss of coolant makeup and core damage before containment failure; the core damage occurs relatively fast and the containment is at low pressure at core melt.
(ii)	Class II	Sequences are characterized by loss of long-term containment heat removal function which results in containment failure followed (by definition) by core melt.
(iii)	Class III	Sequences involve an ATWS and a relatively fast core melt before containment failure; the containment is intact but at high pressure at core melt.
(iv)	Class IV	Sequences involve an ATWS and a relatively fast core melt in an already failed containment; the containment fails fast from overpressure.

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The BNL modifications to the LGS-PRA involved both the systems analysis and the core melt and containment analysis. These modifications resulted in different accident sequence (and hence class) frequencies and in different consequences for each particular class of accident. The changes in the class frequencies stem from different estimates of the frequencies of the accident initiators and from differences in the system analysis that include intersystem dependences and individual system modeling. The changes in the estimated consequences for each class resulted from differences in the modeling assumptions and computer codes used. The results of the BNL revision are given in Reference 3 and summarized here in Table 1.

3. The ATWS-3A modification

In the LGS-PRA it is stated that the Limerick Generating Station will have an ATWS prevention/ mitigation system at least as good as the ATWS Alternate 3A modification identified by the NRC staff in NUREG-0460. This system includes the following major features [2]:

(i) Redundant and diverse safety grade level sensors in the scram discharge volume,

(ii) Alternate rod insertion (ARI) circuitry and solenoid valves,

(iii) Recirculation pump trip (RPT),

(iv) Automatic standby liquid control (SLC) system to inject boron solution,

(v) Feedwater runback,

(vi) A lower MSIV isolation setpoint from "level 2" to "level 1".

Item (i) above is one of the modifications resulting from the Brown's Ferry incident of June 1980 and is implemented to assure a high level of reliability for the mechanical rod insertion subsystem. This modification is not considered as part of the ATWS-3A modification. The ARI (item ii) is a system composed of redundant valves on the scram valve pilot air header in order to provide additional diversity and redundancy in the electrical portion of the scram system. This system is part of the ATWS-3A modification. Items (iii) and (v) are standard features of the BWR designs for ATWS prevention and mitigation and they are not considered as part of the ATWS-3A modification. The standby liquid control system (Item iv) is designed to

Table 1: Frequency of each accident sequence class and for each accident initiator. Expected acute and latent fatalities for each accident class.

Initiator	CLASS I	CLASS II	CLASS III	CLASS IV	TOTAL FREQUENCY OF CORE DAMAGE (per accident initiation)
1. Turbine Trip	1.2×10^{-5}	1.4×10^{-6}	1.5×10^{-6}	2.3×10^{-7}	1.5×10^{-5}
2. MSIV Closure	3.8×10^{-5}	1.5×10^{-6}	6.2×10^{-7}	5.0×10^{-8}	4.0×10^{-5}
3. LOOP	3.9×10^{-5}	6.3×10^{-7}	6.1×10^{-7}	9.0×10^{-9}	4.0×10^{-5}
4. IORV	5.0×10^{-6}	4.5×10^{-7}	6.0×10^{-7}	2.5×10^{-8}	6.1×10^{-6}
5. Manual Shutdown	4.0×10^{-7}	1.6×10^{-8}	----	----	4.2×10^{-7}
6. Large LOCA	----	6.4×10^{-8}	4.2×10^{-9}	4.0×10^{-9}	8.2×1^{-8}
7. Medium LOCA	4.7×10^{-8}	1.9×10^{-9}	1.9×10^{-9}	7.1×10^{-8}	7.1×10^{-8}
8. Small LOCA	2.9×10^{-8}	2.0×10^{-9}	2.4×10^{-8}	9.7×10^{-10}	5.6×10^{-8}
TOTAL FREQUENCY (Per Class)	9.5×10^{-5}	4.1×10^{-6}	3.4×10^{-6}	3.0×10^{-7}	1.0×10^{-4}
Expected Acute Fatalities	2.1×10^{-1}	4.3×10^{-1}	2.1×10^{-1}	8.1×10^{1}	--------
Expected Latent Fatalities	1.7×10^{3}	1.4×10^{3}	1.7×10^{3}	1.3×10^{4}	--------

inject a boron solution into the core in the event of failure to insert control rods, which would render the core subcritical. The system consists mainly of three positive displacement pumps, three explosive-actuated valves, the poison storage tank, and associated piping, valves, controls, and instrumentation. The additional features of the design over the standard BWR designs are the addition of a third pump and the automatic initiation. These features are considered part of the ATWS-3A modification. Item (vi) affects the relative frequency of those transients that cause the MSIVs to close and thus render the feedwater and main condenser unavailable for coolant injection and decay heat removal, respectively. Since transients involving loss of feedwater and main condenser are more severe than turbine trip transients (for which these systems are available), this MSIV setpoint change, which allows the MSIVs to remain open until the reactor coolant level reaches a lower point in the vessel, results in a decrease of the risk. This change, however, was not considered to be part of the ATWS-3A modification.

In summary, the ATWS-3A modification in LGS was considered to be the addition of the ARI system and the addition of the third pump and the automatic initiation of the SLC system. These modifications were assumed to be part of the current Limerick design and were included in the LGS-PRA.

4. Modeling the ATWS Sequences in the LGS-PRA

The various systems included in the design of the Limerick station to defend against ATWS events can be divided into two groups: those designed to prevent an ATWS from occurring (prevention), and those designed to mitigate an ATWS in the event that prevention fails (mitigation). Two types of event trees were developed in the LGS-PRA[2] that delineate the system combinations that lead to failure to prevent an ATWS and failure to mitigate an ATWS, respectively. The prevention and mitigation event trees for the turbine trip initiator are given in Figures 1 and 2, respectively. These event trees are briefly described below and the major differences of the corresponding trees for the other initiators are discussed at the end of this section.

The ATWS prevention tree includes the following systems (see Figure 1):

(i) the mechanical subsystem (C_M) of the reactor protection system;
(ii) the electrical subsystem (C_E) of the reactor protection system;
(iii) the recirculation pump trip (R); and
(iv) the Alternate Rod Insertion system (K);

The various combinations of system successes and failures in the prevention tree are divided into three groups: those that imply successful rod insertion, those that include failure to insert the control rods and failure of the recirculation trip, and those that include failure of rod insertion and successful recirculation pump trip. The sequences that include failure of the recirculation pump trip, i.e., failure of reducing the effective power level of the core from 100% to approximately 30%, are assumed to lead directly to a class III or class IV accident sequence depending on the availability of the condenser. The sequences that include failure of rod insertion but successful recirculation pump trip can be mitigated by bringing the reactor to a cold shutdown condition if a number of systems are available for mitigation.

The ATWS mitigation tree includes the following systems (see Figure 2):

(i) <u>Standby Liquid Control System</u> (C) Two cases are distinguished for this system; all three pumps have failed (no injection at all) or at least one pump is injecting. This distinction is necessary because it affects the success of the coolant injection function.

(ii) <u>Opening of the Relief Valves</u> (M) Failure of the relief valves to open was assumed to lead to a large LOCA and a Class IV accident sequence.

(iii) <u>Reclosing of the Relief Valves</u> (P) Failure of at least two relief valves to reclose affects the operability of the Reactor Core Isolation Cooling System (RCIC) and the requirements for the containment heat removal system.

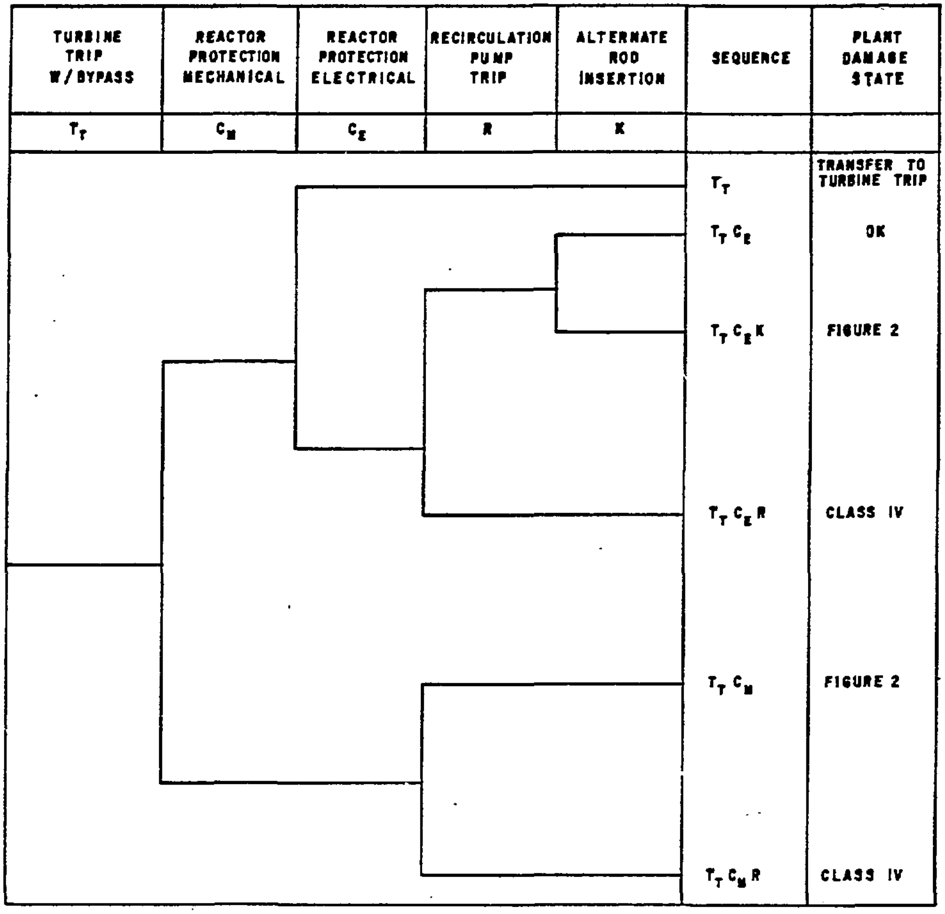
(iv) <u>Failure of Coolant Injection</u> (U) Coolant injection can be accomplished through the Feedwater system (FW), the High Pressure Core Injection System (HPCI) and the RCIC. If there are stuck open relief valves on failure of one of the SCC pumps, feedwater is assumed not to be adequate. In addition the FW is not available (owing to eventual MSIV closures) in the case of only one SLC pump injecting. Similar remarks apply for the RCIC System. The event tree construction reflects these dependences.

(v) <u>Inadvertent Actuation of the Automatic Depressurization System</u> (D) During the ATWS event it is very likely that an automatic initiation signal will be generated for the ADS. The operator must inhibit ADS initiation to prevent boron dilution.

(vi) <u>Inadvertent Operation of High Pressure Injection System</u> (U_H) To avoid poison removal/dilution from the reactor vessel the high pressure injection systems (FW, HPCI or RCIC) must be the reactor vessel water level controlled to maintain below Level 8.

(vii) <u>Containment Heat Removal</u> (W) This function can be performed either via the normal heat removal path through the condenser or through the RHR heat exchangers. Depending on the amount of

Figure 1: - Turbine Trip ATWS Prevention Event Tree

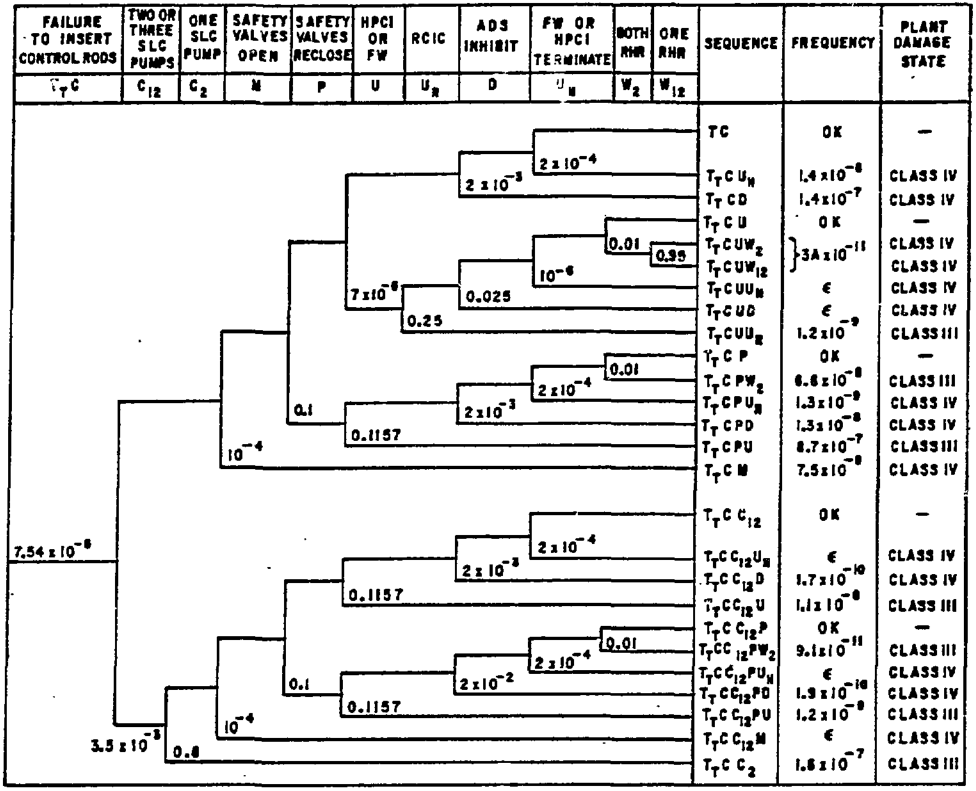


heat released to the suppression pool (SORVs fail to close, one SLC pump injecting) one or two RHR heat exchanges are adequate for containment heat removal. The event tree reflects these dependences.

Sequences that involve loss of coolant injection result in Class III accidents while sequences that involve poison removal/dilution result in Class IV accidents. Sequences that involve loss of containment heat removal result in Class III accidents owing to a dependence of the high pressure injection system on the containment pressure and suppression pool temperature. The high pressure injection system turbines (both HPCI and RCIC) have a high turbine exhaust pressure trip provision. High containment pressure (resulting from failure of containment heat removal) will result in a turbine trip. In addition the lube oil for these turbines is cooled by the suppression pool; failure of containment heat removal results in a rise of the pool temperature beyond the design temperature of the lube oil. This results in failure of the turbines. For these reasons, accident sequences involving failure of the containment heat removal function result in Class III accidents.

Four prevention and four mitigation event trees were developed corresponding to the first four transient initiators (see Section 2). For the orderly manual shutdown, an ATWS event was considered extremely unlikely and in any event the reactor power would have been reduced to very low power. For the medium and small LOCA initiators, failure to scram leads to event sequences similar to the IORV ATWS, and the corresponding ATWS trees are applicable. A large LOCA combined with a failure to scram was assumed to lead directly to a Class IV accident. The turbine trip and Loss of Feedwater ATWS prevention event trees are identical (see Figure 1). The LOOP and IORV ATWS prevention event trees assume a success of recirculation pump trip with probability of unity. In the LOOP tree, this is due to the loss of the power source for the recirculation pump. For the IORV tree, recirculation pump trip is not necessary for a successful mitigation of an IORV ATWS (see REF.3 on ATWS success criteria).

Figure 2: Turbine Trip ATWS Mitigation Event Tree



ϵ = NEGLIGIBLE, $T_TC = T_TC_M + T_TC_EK$ (SEE FIGURE 1.)

The mitigation trees for the other three transient initiators are given in REF.3, p.5-70. The major differences are the following. Between the turbin trip and the MSIV/closure event trees, the only differences arise from the unavailability of feedwater for core injection immediately following the accident and the main condenser for containment heat removal later in the accident.

The LOOP ATWS event tree is identical in structure to the MSIV/closure tree, but the unavailabilities of the various systems are higher because their only power source is the emergency AC power (diesels). For the IORV transient the SLC is initiated by either a low water level, or a high reactor pressure or a manual singal. Actual injection of the SLC system begins after a defined time has transpired and the reactor core power is still not downscale. In addition the FW and the main condenser are not available for core injection and containment heat removal, respectively, and the RCIC is inadequate for core injection.

The results presented in Table I for classes III and IV were based on this modeling of the ATWS sequences.

5. Effect of removing the ATWS-3A Modification

As discussed in Section 3 above the two features of the Limerick ATWS prevention/mitigation system that can be characterized as forming the Alternate 3A modification are: (a) the Alternate Rod Insertion system; and (b) the third pump and the automatic initiation of the Standby Liquid Control System. The removal of these features affects both the prevention and mitigation probabilities of an ATWS event. The removal of the ARI system affects the ATWS prevention event trees (Figure 1). The unavailability of this system is set equal to unity in the revised trees. The changes in the SLC system affect the mitigation ATWS event trees. In the absence of an automatic initiation capability for the SLC, the unavailability of the system is dominated by the failure of the operator to manually initiate the system.

6. Summary of Results

The effects of removing the ATWS-3A modification are given in Table 2. As a result of how the various classes are defined, only class III and class IV plant states are affected. It can be noted that there is a 40% increase in the ex-

Table 2 Effects of Including ATWS-3A Modification in the Prevention/ Mitigation System of a BWR Mark-II Nuclear Power Plant

Accident Sequence Class	Expected Frequency of Core Damage Per Year of Reactor Operation		Expected Latent Fatalities Per Year of Reactor Operation		Expected Acute Fatalities Per Year of Reactor Operation	
	WITH ATWS-3A	WITHOUT ATWS-3A	WITH ATWS-3A	WITHOUT ATWS-3A	WITH ATWS-3A	WITHOUT ATWS-3A
I	9.5×10^{-5}	9.5×10^{-5}	1.6×10^{-1}	1.6×10^{-1}	2.0×10^{-5}	2.0×10^{-5}
II	4.1×10^{-6}	4.1×10^{-6}	5.7×10^{-3}	5.7×10^{-3}	1.8×10^{-6}	1.8×10^{-6}
III	3.4×10^{-6}	3.6×10^{-5}	5.8×10^{-3}	6.1×10^{-2}	7.1×10^{-7}	7.6×10^{-6}
IV	3.0×10^{-7}	8.0×10^{-7}	3.9×10^{-3}	1.0×10^{-2}	2.4×10^{-5}	6.5×10^{-5}
TOTAL	1.0×10^{-4}	1.4×10^{-4}	1.8×10^{-1}	2.4×10^{-1}	4.7×10^{-5}	9.5×10^{-5}

pected core damage frequency (per year of reactor operation). Similarly, increases of about 1.3 and 2.0 are observed for the total expected latent fatalities and the total expected acute fatalities respectively. A closer examination of the individual classes reveals that there is about a factor of 10 increase in all three risk indices for the class IV plant state and a factor of about 2.6 increase for the class IV plant state.

References

1. "Anticipated Transients Without Scram" NUREG-0460.
2. "Probabilistic Risk Assessment of the Limerick Generating Station," Philadelphia Electric Co., March 1981.
3. I. A. Papazoglou et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," NUREG/CR-3028, February 1983.