Issue 116: Accident Management

DESCRIPTION

Historical Background

This issue was identified in a NRR/DST memorandum[1] and addressed the potential risk reduction that might result from training operators and having procedures developed to assist the operators in managing accidents beyond the design basis.

Presently, there is little training or procedures required for operator use in dealing with accidents that go beyond inadequate core cooling with severe core damage or core-melt. Should a core-melt accident occur, operators without procedures and untrained in core-melt accident progression, phenomena, and consequences might:

(1) fail to take full advantage of opportunities to delay or minimize containment failure and the release of fission products (e.g., they might vent the containment when it is inappropriate to do so, or fail to vent when it is, or they might fail to take full advantage of the ways to accomplish containment heat removal or combustible gas control); or (2) give erroneous advice and guidance on the progression of the accident to offsite emergency response personnel, leading to inappropriate emergency response actions, such as unnecessary evacuation, evacuations from locations at which a shelter/relocate strategy would be superior, etc.

Safety Significance

EOP guidelines emphasize symptom management to the exclusion of root cause analysis. Under emergency conditions, shutting down a reactor and sustaining core cooling from the control room, based upon symptom management, is the first course of action. However, if operators cannot sustain core cooling with available equipment, or if they have succeeded temporarily with equipment whose failure is imminent, or if their success in sustaining core cooling gives the operating crew some flexibility, it is important to assess what failed, why it failed, what the failure causes mean to the future success of core cooling, and to consider makeshift repairs if necessary.

Possible Solution

A solution to this issue could provide for operator training and the development of procedures to guide operators in handling accidents beyond the design basis. Training would deal with the progression and diagnosis of severe accidents and the related phenomena and consequences. Included in this training would be the functions performed by the safety-related systems and the change in accident progression and consequences which results from the degradation or inoperation of each safety-related system alone or in combination. Procedures would guide operators in maintaining as much control as possible over the progress of an accident in order to reduce to a minimum the release of fission products and to assure that emergency response personnel are provided the best available information on the progression of the accident and the magnitude and timing of the expected releases.

PRIORITY DETERMINATION

Frequency Estimate

The evaluation of this issue is based upon the calculated change in risk for the two RSSMAP[2] reference

[1] Memorandum for W. Minners from F. Rowsome, "A New Generic Safety Issue: Accident Management," April 16, 1985. [8505080417]
[2] NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1981, (Vol. 2) May 1981, (Vol. 3) June 1982, (Vol. 4) November 1981

plants: Oconee 3 and Grand Gulf 1. It should be noted that the solution to this issue does not assume credit for a reduction in accident initiation frequency when some reduction may result; rather, credit is assumed for a reduction in the frequency of accidents in those release categories which have high source terms and the shift

of this reduction to accident release categories that have lower source term values. Hence, the overall accident frequency is not reduced, but the public dose is. However, proper training and procedures may result in some decrease in accident initiation frequency.

Consequence Estimate

The total whole body man-rem dose was obtained by using the CRAC Code[3] results for each WASH-1400[4] release category. The results were obtained assuming a uniform population density of 340 persons per square mile (which is the average for U.S. domestic sites in the year 2000) within the area between a radius of 1/2 mile and 50 miles from the plant. Typical (midwest plain) meteorology, no evacuation, and no ingestion were also

assumed. The core inventory at the time of the accident was assumed to be represented by a 3412 MWt (1120 MWe) PWR. Based on the above assumptions, the dose calculations obtained for each release category is shown in Table 116-1.

Category	**Core-Melt**	**Non-Core-Melt**
Table 116-1		
Public Whole-Body Dose Factors (Man-rem)		
PWR 1A*	5.4×10^6	
PWR 1B	4.4×10^6	
PWR 2	4.8×10^6	
PWR 3	5.4×10^6	
PWR 4	2.7×10^6	
PWR 5	1.0×10^6	
PWR 6	1.5×10^5	
PWR 7	2.3×10^3	
PWR 8		7.5×10^4
PWR 9		1.2×10^2
BWR 1	5.4×10^6	
BWR 2	7.1×10^6	

[3] NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.
[4] WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Atomic Energy Commission, October 1975.

BWR 3	5.1 x 10^6	
BWR 4	6.1 x 10^5	
BWR 5		2.0 x 10^1

* Assumed to be PWR 1

The actions taken by the operators, adequately trained and with procedures to guide them in handling accidents beyond the design basis, will result in moving a portion of the releases in the higher release categories into lower release categories. For PWRs, a portion of the releases in Categories 2 and 3 was assumed to result in releases in Categories 4 and 5, respectively. For BWRs, a portion of the releases in Categories 2 and 3 was assumed to result in a Category 4 release.

PRA results suggest that core-melt frequencies with plausible recovery/repair are about 10 times lower than they are predicted to be without credit for recovery actions, including actions outside the control room and makeshift repairs. The calculations for consequence reduction were bounded by using a 90% reduction as an upper bound and a 10% reduction as a lower bound. A median of 50% was selected for reference purposes. The effect of the reduction values on the accident occurrence frequency for the PWR release categories is shown in Table 116-2 and for the BWR release categories in Table 116-3. The effect of the reduction in the higher release categories is shown as an increase (positive value) in the lower release categories.

Reduction in Accident Frequency	**PWR Release Category**			
Table 116-2 Change in PWR Accident Release Category Frequency				
	(Event/RY)			
Reduction in Accident Frequency	**PWR Release Category**			
90%	-9.0 x 10^{-6}	-2.6 x 10^{-5}	+9.0 x 10^{-6}	+2.6 x 10^{-5}
50%	-5.0 x 10^{-6}	-1.3 x 10^{-5}	+5.0 x 10^{-6}	+1.3 x 10^{-5}
10%	-1.0 x 10^{-6}	-2.9 x 10^{-6}	+1.0 x 10^{-6}	+2.9 x 10^{-6}

Reduction in Accident Frequency	**BWR Release Category**		
Table 116-3 Change in BWR Accident Release Category Frequency			
	(Event/RY)		
	2	3	4
90%	3.1 x 10^{-5}	-1.3 x 10^{-6}	+3.2 x 10^{-5}
50%	-1.7 x 10^{-5}	-7.0 x 10^{-7}	+1.8 x 10^{-5}

10%	-3.5×10^{-6}	-1.0×10^{-7}	$+3.3 \times 10^{-6}$

The change in occurrence frequency for each release category was multiplied by the dose consequences shown in Table 116-1. The results were then summed algebraically for each reduction frequency to determine the annual change in public dose. Multiplying the annual change in public dose by the average remaining reactor life (28 years for PWRs and 27 years for BWRs) provided the average risk reduction expected for each reactor type. The results are shown in Table 116-4.

Reduction in Accident Frequency	**(Event/RY)**			
Table 116-4 Resulting Change in Public Dose				
	PWR		BWR	
	Man-rem/RY	Man-rem/R	Man-rem/RY	Man-rem/R
90%	-134	-3750	-207	-5580
50%	-67	-1880	-113	-3040
10%	-15	-420	-23	-630

Cost Estimate

Industry Cost: The cost to the industry includes several diverse items. The initial cost would be for technical efforts to develop the action plans for dealing with accidents beyond the design basis. These action plans could be developed by the Owners' Groups for the various NSSS vendor designs. Based upon these findings, the generic EOPs and training objectives can be developed, from which, plant-specific emergency procedures and training programs can be developed. Training must be provided to the licensed operating crews at each plant. For operator trainees, the training programs must be enlarged to include additional training for dealing with accidents beyond the design basis. Furthermore, refresher training courses on dealing with beyond design basis accidents must be included in the operator refresher training programs.

The initial cost is estimated to be $2M/reactor based on an estimate of $1M/reactor to develop the upgraded emergency operating procedures, as reported in a draft INEL report on the "On-Site Assessments of the Effectiveness and Impacts of Upgraded Emergency Operating Procedures." Due to the added complexity, it is estimated that the costs for the development of procedures and training to handle accidents beyond the design basis will be twice that for the upgraded EOPs.

Annual recurring refresher training costs were estimated to be $48,700 in 1989 dollars for six crews of five persons each. Based on an average reactor life of 28 years and a 5% discount rate, the refresher training total life cost is $842,000/reactor.

It has been assumed that the instrumentation added after the TMI-2 accident will be adequate to meet the operators' needs. Hence, no hardware changes or additions are included in the cost estimate, nor are there any replacement power costs. Thus, the total industry cost is estimated to be $2.8M/reactor or $375M for 134 plants.

NRC Cost: Based upon NUREG/CR-4568,[5] the NRC costs are estimated to be $220,000 for technical

[5] NUREG/CR-4568, "A Handbook for Quick Cost Estimates," U.S. Nuclear Regulatory Commission, April 1986.

resolution and $14,000/reactor for individual plant reviews. No appreciable annual costs are anticipated. For 134 plants, the total NRC cost is estimated to be $2.1M.

Total Cost: The total industry and NRC cost to implement the possible solution to this issue is approximately

$377M.

Value/Impact Assessment

Separate value/impact scores were calculated for the three cases of accident frequency reduction analyzed above.

(1) **90% Reduction**:	$S = \frac{580{,}000 \text{man} - \text{rem}}{1{,}358 \text{man} - \text{rem}/\$M} = 1{,}538 \text{man} - \text{rem}/\$M$
(2) **50% Reduction**	$S = \frac{258{,}000 \text{ man} - \text{rem}}{\$377M} = 684 \text{ man} - \text{rem}/\$M$
(3) **10% Reduction**:	$S = \frac{(65{,}500 \text{ man} - \text{rem})}{\$377M} = 137 \text{ man} - \text{rem}/\$M$

CONCLUSION

Comparing the value/impact score to the total man-rem reduction, this issue

is in the high priority category at the 10% accident frequency reduction level. However, at the time of this analysis, the issue was being addressed[6] in the Accident Management element of the NRC plan for the closure of severe accident issues, as described in SECY-88-147[7] and Generic Letter 88-20, Supplement 2,[8] and was not pursued separately.

[6] Memorandum for C. Ader from W. Minners, "GI 116, Accident Management," May 9, 1990. [9704090138]
[7] SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," U.S. Nuclear Regulatory Commission, May 25, 1988. [8806030338]
[8] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.