ROUTING AND TRANSMITTAL SLIP		ACTION
1 TO (Name, office symbol or location)	INITIALS	CIRCULATE
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2 T. Novak	INITIALS	FILE
W. Minners	DATE	INFORMATION
3 W. Vesely	INITIALS	NOTE AND RETURN
M. Taylor	DATE	PER CONVERSATION
4 G. Mazetis	INITIALS	SEE ME
	DATE	SIGNATURE

REMARKS

There has been considerable debate over the need for protection against ATWS events 'specially since the publication of WASH-1400. The attachment provides my views on the need for ATWS "fixes." Please provide your comments by 1/10/77.

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE
Ashok C. Thadani	1/3/77
	PHONE 27341

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 100-11.206

1/17/77
My Comments

Let's discuss when you get a chance

8104170452

A. Thadani

Why ATWS

1. BACKGROUND

The safety objective is that the likelihood of all accidents with significant consequences not included in the design basis envelope should not be greater than 10^{-6} per year. For the particular potential failure path of ATWS, the staff believes that a failure rate of the order of one-tenth of the overall safety objective is an appropriate goal. It should be noted that both the ACRS and the reactor vendors have agreed in the past that the 10^{-7} aiming point was appropriate for ATWS events. Use of this type of goal is helpful to the staff in determining whether additional design requirements are warranted such that the potential risk from accidents is very low. The staff does not believe that technology exists to rigorously demonstrate that the safety objective is achieved for ATWS events. Rather, an attempt is made to minimize multiple failures due to common causes and require that the equipment needed in the short term (few minutes) following an ATWS event be highly reliable and automatically actuated.

greements documented?

why not just the latter?

As discussed in detail in WASH-1270, the staff, based on the operating experience to date, including several instances of incipient or partial common mode failures of reactor shutdown systems and the evaluation of current reactor shutdown system designs, has concluded that this safety objective would not be met unless either shutdown system designs are improved to provide greater assurance of scram when needed, or measures are taken to make the consequences of ATWS acceptable. The first alternative, i.e., revision of designs to improve significantly the reliability of reactor shutdown systems by providing shutdown

Improve SD systems

Make consequences ok

systems which would be diverse to the current reactor protection system and control rod drives such that ATWS events need not be considered in the design envelope, was originally a requirement in WASH-1270 for all applications filed after October 1, 1976. Although this alternative is still acceptable, the staff has since concluded that the desired safety objective should can be approached for all classes of plant by means of the second alternative alone.

Same conclusion as in preceding page

?

The second alternative, i.e., including ATWS events within the safety design basis of plants and providing design changes to assure that the consequences of anticipated transients would be acceptable in the event of a postulated failure to scram, is the course currently being pursued by vendors and reviewed by the staff. With this safety goal as an aiming point, the staff has developed several design and evaluation guidelines which it believes are acceptable in evaluating the response of reactor systems to the design basis postulated ATWS events.

?, GE

2. REACTIVITY SHUTDOWN SYSTEM UNRELIABILITY

The reactivity shutdown system consists of the reactor protection system, that is the sensors, signal conditioning equipment, bistables, logic circuits, trip breakers and power supply, and the control rod system. The unreliability of the reactivity shutdown system has been estimated in WASH-1270 to be 10^{-4} per demand. This unreliability is a conservative estimate of the accumulated operational experience of Army and Naval power units in addition to U.S. and foreign central station power units. This estimate may not be a conservative estimate of the unreliability of any individual central station design, since the operating experience of individual designs is limited. Therefore, an evaluation of the unreliability of individual current designs cannot be determined by the direct application of the unreliability estimates developed in WASH-1270. Based on a review of current designs, the staff has concluded that the areas of reactor protection systems that are particularly susceptible to common mode failure must be corrected before an unreliability in the order of 10^{-4} per demand would apply to any individual system.

? The staff also recognizes that the unreliabilities of the reactor protection system and the control rod drive mechanisms individually are necessarily less than the overall unreliability of the ~~reactivity~~ ok shutdown system ~~as a while~~. However, while actual and potential common mode failures of control rod systems have occurred in operating plants, in our view this ~~in our view this~~ data ~~from operating plants~~ is insufficient to determine the distribution of the total unreliability among the portions of the reactivity shutdown system.

3. RISK ASSESSMENT

Since the publication of the Reactor Safety Study (WASH-1400) considerable controversy has arisen in terms of what fraction of risk can be attributed to ATWS and indeed if any design modifications are necessary for ATWS if the overall risk is not reduced substantially. We would make an effort to compare the ATWS risk studies performed by the RSS and the staff interpretation of what the risks might be.

3.1 Anticipated Transient Frequency

WASH-1270	WASH-1400 PWR	WASH-1400 BWR
1/RYR	3/RYR	10/RYR

3.2 Reactivity Shutdown System Unreliability Comparison

WASH-1270	WASH-1400 PWR		WASH-1400 BWR	
95% Confidence	Upper Bound	Median	Upper Bound	Median
1.0×10^{-4}	1.0×10^{-4}	3.6×10^{-5}	4.6×10^{-5}	1.3×10^{-5}

3.3 Accident Sequences

For the BWR, the plant used in the safety study for detailed examination was one to which a partial ATWS "fix" has already been applied. The BWR unit has a recirculation pump trip which would significantly reduce the peak pressure. The comparison shall be made in spite of this difference and an assessment of the impact on risk due to this fix will be made later in this report. The PWR unit analyzed has higher primary relieving capacity than some other PWR units, otherwise, the increased relieving capacity would result in lower calculated peak pressure. The calculated peak pressure is expected to be high enough that the isolation valves (RHR, CVCS etc.) may experience permanent deformation and consequently long-term shutdown capability may be lost. It is not clear whether the RSS had given consideration to this failure mode.

3.3.1 WASH-1400 Estimates

3.3.1.1 BWR Release categories

$T \equiv$ Anticipated Transients ~ 10/yr

$C \equiv$ Failure of both the RPS and either the recirc. pump trip or the SLCS 1.3×10^{-6}

$\alpha \equiv$ Containment failure mode Steam Explosion $\sim 10^{-2}$

$\gamma \equiv$ Containment Failure Mode Overpressure ~ 0.99

$(TC)\text{-}(\alpha) \approx 1.3 \times 10^{-7}$ (Category 1 release)

$(TC)\text{-}(\gamma) \approx 1.3 \times 10^{-5}$ (Category 3 release)

Risk from all Accidents

Category 1	1×10^{-6}
Category 3	2×10^{-5}

Therefore, by WASH-1400 estimates ATWS contributes greater than 50 percent to the Category 3 release consequences.

3.3.1.2 PWR Release Categories

$T \equiv$ Anticipated Transients of Significant Consequences ~ 3/yr

$K \equiv$ RPS Unavailability $\sim 3.6 \times 10^{-5}$?

$Q \equiv$ Primary System Relief/Safety Valves fail to Reclose $\sim 10^{-2}$

$M \equiv$?

$\alpha \equiv$ Containment failure, Mode Steam Explosion $\sim 10^{-2}$

$\beta \equiv$ Containment Leakage $\sim 10^{-4}$

$\epsilon \equiv$ Containment Rupture by Melt through ~ 1.0

$TKQ-\alpha = 3 \times 10^{-8}$ (Category 3 Release)

$TKQ -\alpha = 1 \times 10^{-8}$ (Category 3 Release)

$TKQ-\beta = 3 \times 10^{-10}$ (Category 5 Release)

$TKQ-\epsilon = 3 \times 10^{-6}$ (" 7 ")

$TKMQ-\epsilon = 1 \times 10^{-6}$ (" " ")

Risk from all Accidents

Category 3	4×10^{-6}
" 5	7×10^{-7}
" 7	4×10^{-5}

Therefore by WASH-1400 estimates ATWS in general is not a significant contributor except for category 7 where it contributes about 10 percent to the releases.

3.3.2 NRR Staff Estimates Using WASH-1270

3.3.2.1 BWR Systems

$T \equiv$ Anticipated Transient	$\sim 1/yr$	
$C' \equiv$ RPS Failure	$\sim 10^{-4}$	
$U \equiv$ SLCS Failure	$\sim 10^{-1}$	(WASH-1400)
$H \equiv$ HPCI Failure	$\sim 10^{-1}$	(" ")
$S \equiv$ S/V Fails to Reclose	$\sim 10^{-1}$	(" ")

$TC' -(\alpha) = 10^{-6}$

i.e. even if the standby liquid control system were to be manually initiated several minutes after the initiation of an ATWS event, the consequences would be very severe in that core covery would be difficult and in addition the suppression pool conditions could be such as to cause damaging vibrations and consequently affect long term core cooling capability.

TC' - $\gamma \approx 10^{-4}$

TC'U-α (WASH-1400 TC-) $\approx 10^{-7}$

TC'U-γ (" TC-γ) $\approx 10^{-5}$

TC'H-α $\approx 10^{-7}$

TC'H-γ $\approx 10^{-5}$

TC'S-γ $\approx 10^{-5}$

Assuming similar release categories as in Section 3.3.1.1

ATWS Risk from Release Category #1 $\sim 1.2 \times 10^{-6}$

" " " " " #3 $\sim 1.3 \times 10^{-4}$

Therefore by NRR estimates the probability of exceeding 10 CFR 100 limits for a BWR plant is approximately 10^{-4} per year and therefore protection against ATWS events is warranted.

3.3.2.2 PWR Systems

ATWS calculations for various types of PWR systems vary in the severity of consequences. For example, if no plant modifications are assumed, the calculated peak primary pressure would be in the range of 4000—5000 psia for different PWR systems using essentially a realistic model. We believe that such high pressures are unacceptable and could result in core melt. One such path that could lead to core melt is the loss of isolation valves operability. While the safety study assumes that more complex ATWS events (TKQ, TKMQ) lead to core melt, the staff believes that the consequences of ATWS (TK) are potentially serious and in view of the "realistic" nature of the calculations in contrast to the highly conservative calculations used for determining licensing acceptability, the staff judges that the TK sequences would exceed 10 CFR 100 limits Further, the PWR unit analyzed in the RSS, due to higher relievin capacity, would be expected to result in less severe consequences than some other PWR units. On the other hand, if either auxiliary feedwater system or turbine trip were disabled due to the CMF that caused scram failure, the RSS would have concluded this sequence to result in core melt.

TK - α	$\sim 10^{-6}$	(Category 3)
TK - ϵ	$\sim 10^{-4}$	(Category 7)

Therefore by NRR estimates if either turbine trip or auxiliary feedwater system is not available for plants with higher relievin capacity (e.g. unit analyzed in the RSS study) and for other PWR units, the probability of exceeding the 10 CFR 100 limits is approximately 10^{-4} per year.

3.4 WASH-1400 Vs. Staff Risk Comparisons

		WASH-1400	STAFF
Total Probability*	BWR	2×10^{-5}	Discussion B
Exceeding of 10 CFR 100	PWR	$\sim 10^{-5}$	10^{-4}
	BWR	$\sim 10^{-5}$	$\sim 10^{-4}$
ATWS Probability of Exceeding 10 CFR 100	PWR	$\sim 2 \times 10^{-6}$	$\sim 10^{-4}$
	BWR	~50%	Single Most Do
ATWS Contribution	PWR	~20%	Single Most Do

*Estimate based on discussion with RSS Personnel. Although t RSS considered core melt probabilities, not all release categories would probably result in exceeding 10 CFR 100 doses. However, the RSS estimates that the probability of exceeding 10 CFR 100 limits is about 10^{-5} per reactor year.

In some limited areas quantitative probabilistic safety goals or criteria exist in NRC requirements. In WASH-1270 and elsewhere the AEC Regulatory staff suggested as a long-range safety goal a frequency of exceeding 10 CFR 100 guidelines in the 10^{-6}/ry range.

garbled

This is derived in WASH-1270 from a nationwide goal therein proposal of 10^{-3}/year or lower frequency of accidents involving significant overexposure of members of the public, and a projected ~~AD 2000~~ number of power reactors of the order of 1000 by the year 2000. Today, with ~50 power reactors operating, the proposed nationwide goal would lead to a present-day safety goal of 2×10^{-5}/ry for exceeding 10 CFR 100 guidelines.

But since present-day reactors are expected to be operating in AD 2000, the long-range goal will presumably apply to them also in the future.

Logic?

Since not all potentially serious postulated accidents are ATWS events, a goal for ATWS frequency should be lower than the goal for severe accident frequency. WASH-1270 suggests a factor of 10 for this margin.

A goal of this ~~sort~~ nature is compatible with present licensing practice in some areas. An example of such an application of probabilistic reasoning in safety evaluation is the evaluation currently performed of airplane crashes. If the calculated frequency of plane crashes onto a proposed facility that could interfere with safe shutdown is lower than a threshold value, no plant protection is required. A predicted frequency higher than the threshold value results in inclusion of airplane crashes of some severity in the safety design basis of the facility.

The WASH-1400 calculations show that the likelihood of exceeding 10 CFR 100 is of the order of 10^{-5} per reactor year and any requirement to protect to a level of 10^{-7} per year could be construed as unnecessary or not needed. Our regulatory policies have continuously evolved since design and construction of Peach Bottom Units 2 and 3 and Surry Unit 1, the base plants of WASH-1400. Several improvements in the ECCS systems, for example, such as automatic transfer from injection to recirculation mode and incorporation of two discharge lines instead of one from the refueling water storage tank would reduce the overall risk and we consider it inappropriate to base regulatory decisions only on the calculated risks for those two plants. Rather more work is needed before risk assessment methodology can be used routinely in licensing decisions. It is also the staff belief that ATWS 'fix' would reduce risk from other contributors.

Point?

For example, the two significant contributors to the core melt probability in WASH-1400 for the BWR plant are the transient events followed by the failure of either scram system or the decay heat removal system. It is believed that any ATWS fix that includes high pressure feed system would also reduce the core melt probability due to decay heat removal system failure.

In summary, the staff believes that the ATWS events are ~~the most~~ important ~~dominant~~ contributors to the probability of exceeding 10 CFR 100 limits. Accordingly, the staff would require that protection against ATWS events be provided to protect the health and safety of the public.

bcc: S. Hanauer
R. Heineman
W. Minners
R.Easterling
T. Novak
G. Mazetis
A. Thadani

Distribution:
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NRC PDR

MAY 11 1977

Combustion Engineering, Inc.
ATTN: Mr. A.E. Scherer
Licensing Manager (9460-401)
Nuclear Power Systems Division
1000 Prospect Hill Road
Windsor, Connecticut 06095

Gentlemen:

CENPD-158, REVISION 1, "ANALYSIS OF ANTICIPATED TRANSIENTS WITHOUT SCRAM IN COMBUSTION ENGINEERING NSSS'S"

The Nuclear Regulatory Commission (NRC) staff has completed its initial review of Combustion Engineering's (C-E) topical report, CENPD-158, Revision 1.

In our review of CENPD-158, Revision 1, we have noted several parametric analyses which have been useful in our evaluation of the sensitivity of ATWS analyses to variations of input parameters and primary relief area. However, additional information concerning the sensitivity of the analyses to parametric variations is necessary before we can complete our review of C-E's analyses. The enclosure identifies the specific information required.

If you have any questions, please contact us.

Sincerely,

Original signed by
D. F. Ross

Derwood F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety
Office of Nuclear Reactor Regulation

Enclosure:
Information Request

cc: Mr. C.B. Brinkman
C-E, Bethesda, MD

DUPE 8104170054

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DATE ➤	05/11/77	05/11/77	05/11/77	05/11/77		

ENCLOSURE

ATWS

COMBUSTION SET OF CALCULATIONS

The following tables describe the set of calculations required to perform simplified study to determine the effects of sensitivity of key parameters on the consequences. The table was developed using a factorial experiment method to glean maximum amount of information while minimizing the calculational effort. In order to understand the sensitivity of these parameters at pressures of interest (~3200 psi) the reference case should assume all systems to be operable, including pressurizer relief valves, ten percent relief and safety valve accumulation for water relief, and primary flow area of .075 ft^2. These studies should be conducted for the 2560 MWt and 3800 MWt plants addressed in CENPD-158, Revision 1.

Following completion of the sensitivity studies and assuming distributions or bounds on these parameters, probability distribution associated with parameter values would be convoluted using Monte Carlo technique to determine the frequency of ATWS events that might exceed criteria.

Run	Power A	Tav B	MTC C	Doppler D	Gap E	SGI F	T/T G	RCS H	AF I
1	0	0	0	0	0	0	0	0	0
2	+	+	+	+	+	+	+	0	0
3	+	+	+	-	-	-	-	0	0
4	+	-	-	+	+	-	-	0	0
5	+	-	-	-	-	+	+	0	0
6	-	+	-	+	-	+	-	0	0
7	-	+	-	-	+	-	+	0	0
8	-	-	+	+	-	-	+	0	0
9	-	-	+	-	+	+	-	0	0
10	0	0	0	0	0	0	0	+	0
11	0	0	0	0	0	0	0	0	+

INITIAL CONDITIONS

Combustion Engineering

Levels	Power	T_{AV} °F	PCM MTC	ΔK/K/°F Doppler	Mils Gap	lbm Total SGI	Sec T/T(1)	Ft^3 RCS	Sec AFW(2)
+	104%	0+4	- 4	0+25%	0+60%	0+5%	0+20	0+5%	0+20
-	96%	0-4	-15	0-25%	0-60%	0-5%	0-10	0	0
0	100%	0	- 6	0	0	0	0	0	0

CE - Reference Case Pressurizer Total Relief & Safety Valve Flow Area Should Be 0.075 ft^2

0 - Reference Value

(1) - Turbine Trip Time

(2) - Auxiliary Feedwater Actuation Time