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NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

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ACRS Members
ACRS Technical Staff

RESPONSE OF COMBUSTION ENGINEERING PLANTS TO THREE MILE ISLAND TYPE TRANSIENTS

NRC Staff and Combustion Engineering met on May 1, 1979 in Bethesda, MD to discuss generically the effect that the transients which occurred at Three Mile Island would have on Combustion Engineering plants.

LOSS OF FEEDWATER

Historically, the Combustion Engineering operating plants response to loss of feedwater has been that within 15 to 30 seconds they get reactor trip on low steam generator water level. The turbine will trip on reactor trip and steam bypass will occur. The power operated relief valve (PORV) will not open under this condition.

At least 10 to 15 minutes are available to manually initiate auxiliary feedwater before steam generator heat removal capability is degraded.

STUCK OPEN PRESSURIZER RELIEF VALVE

Rapid depressurization would occur with a reactor trip on low pressure in about 5 seconds. ECCS would be initiated by low pressure in about 15 to 30 seconds. Flashing in the RCS would occur with an insurge into the pressurizer. Combustion Engineering anticipates that the operator would close the block valve to stop the transient.

CE PLANT GENERIC DESIGN FEATURES WHICH MITIGATE THREE MILE ISLAND TYPE TRANSIENTS

1. The anticipated response to moderate frequency events, such as loss of feedwater, does not open the PORV.

2. A large secondary water inventory allows sufficient time to manually initiate auxiliary feedwater (about 13 minutes for those plants that don't have automatic initiation and a little less for those with automatic initiation).

3. The pressurizer level is not used to automatically initiate safety systems. Low pressurizer pressure normally initiates ECCS.

4. The low HPSI shutoff head will not lift the pressurizer relief or safety valves.

5. The NSSS elevation layout requires only 20-25% of RCS inventory to cover the core. The top of the active core is about 14 feet lower than the bottom of the tube sheet in System 80 plants.

SMALL PIPE BREAKS

Combustion Engineering has analyzed 14 different pipe break sizes and has found that 0.05 ft^2 cold leg break is the worst (much worse than break at top of the pressurizer) 0.1 ft^2 breaks and larger do not require steam generator heat removal since sufficient heat will be removed through the break.

The advantages of "U" tube steam generators, where the hottest water in the primary interfaces with the secondary side of the steam generator at the lowest level first, over a straight-through type design were noted as follows:

a. On depressurization, a steam bubble will form in the reactor vessel and not in the top of the steam generator, since the top of the steam generator is cooler.

b. Heat transfer to the secondary occurs with any secondary side water level.

c. A moderate volume of non-condensibles does not cause a catastrophic loss of heat sink.

NATURAL CIRCULATION

Loss of the all AC power will allow something greater than 40 minutes before the steam generators boil dry, assuming no auxiliary feedwater. Natural circulation following loss of AC power will be about 2% of full flow. Flow can be verified by ΔT measurements across the reactor and steam generators. No direct flow measurements are available.

Combustion Engineering indicated that Michelson's concern, that during a small break, water slugs and non-condensibles when going from pool boiling to natural circulation might prevent establishing natural circulation, would not be a problem due to the high differential pressure driving force from the hot steam in the reactor to the relatively cold steam generator tubes which would be rapidly condensing steam and would push gases from the tubes.

CONTROL SYSTEMS

Operational history of CE plants control systems failures indicates that there have been only 6 out of 180 total (3%) scrams in 28 years of operation attributable to NSSS control system sensor and logic induced protective system challenges. These 6 scrams were due to main feedwater control challenges.

228 196

Combustion Engineering has been invited to discuss some of these same items at the 229th ACRS Meeting on May 10, 1979.

G. R. Quittschreiber

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Senior Staff Engineer

Attachments:
Handout Material

C-E OPERATING PLANT RESPONSE TO A STUCK PRESSURIZER RELIEF VALVE

- RAPID DEPRESSURIZATION OF RCS
- TM/LP TRIP ON LOW THERMAL MARGIN
- ECCS INITIATION AT APPROXIMATELY 1600 PSIA
- FLASHING IN RCS RESULTING IN PRESSURIZER INSURGE
- FILLING OF PRESSURIZER RESULTING IN TWO-PHASE RELIEF THROUGH PORV
- FAILURE OF DRAIN TANK RUPTURE DISK AND FLUID RELEASE TO THE CONTAINMENT
- OPERATOR CLOSES PRESSURIZER BLOCK VALVE TERMINATING DEPRESSURIZATION

2

SIMULTANEOUS HOT LEG/COLD LEG INJECTION COOLS CORE AND FLUSHES BORIC ACID FROM VESSEL

REFILL OF RCS DISPERSES BORIC ACID THROUGHOUT SYSTEM AND SG'S ARE ABLE TO COOL RCS TO SDC TEMPERATURE

BREAK SIZE, FT^2	RCS PRESSURE AT t = 8 HOURS, PSIA
10	20
5	20
2	20
1	20
0.5	20
0.2	75
0.1	125
0.05	150
0.02	225
0.01	410
0.005	840
0.002	1425
0.001	1620
0.0005	1680

228 199

OVERLAP OF ACCEPTABLE LTC MODES IN TERMS OF COLD LEG BREAK SIZE

SOURCES OF NON CONDENSIBLES

	VOL (FT^3)	
	STP	SAT @ 1300 PSI
DISSOLVED H_2 (50 SCC/KG H_2O)	454.	10.44
HE FILL GAS/ROD+	.019	4×10^{-4}
GAP FISSION GAS @ EOC/ROD+	.011	2.5×10^{-4}
DISSOLVED N_2 IN SIT	2636.	60.6
N_2 IN SIT	85,600.	1969.0
ZR-H_2O CLAD/% REACTION++	4,378	100.6

\+ 54,956 RODS

++ 57,000 LBM ZR

S.B. ASSUMPTIONS

REACTOR TRIP ON L.P.P.	(1728 PSIA)
SIAS ON L.P.P.	(1578 PSIA)

PUMP TRIP ON REACTOR TRIP

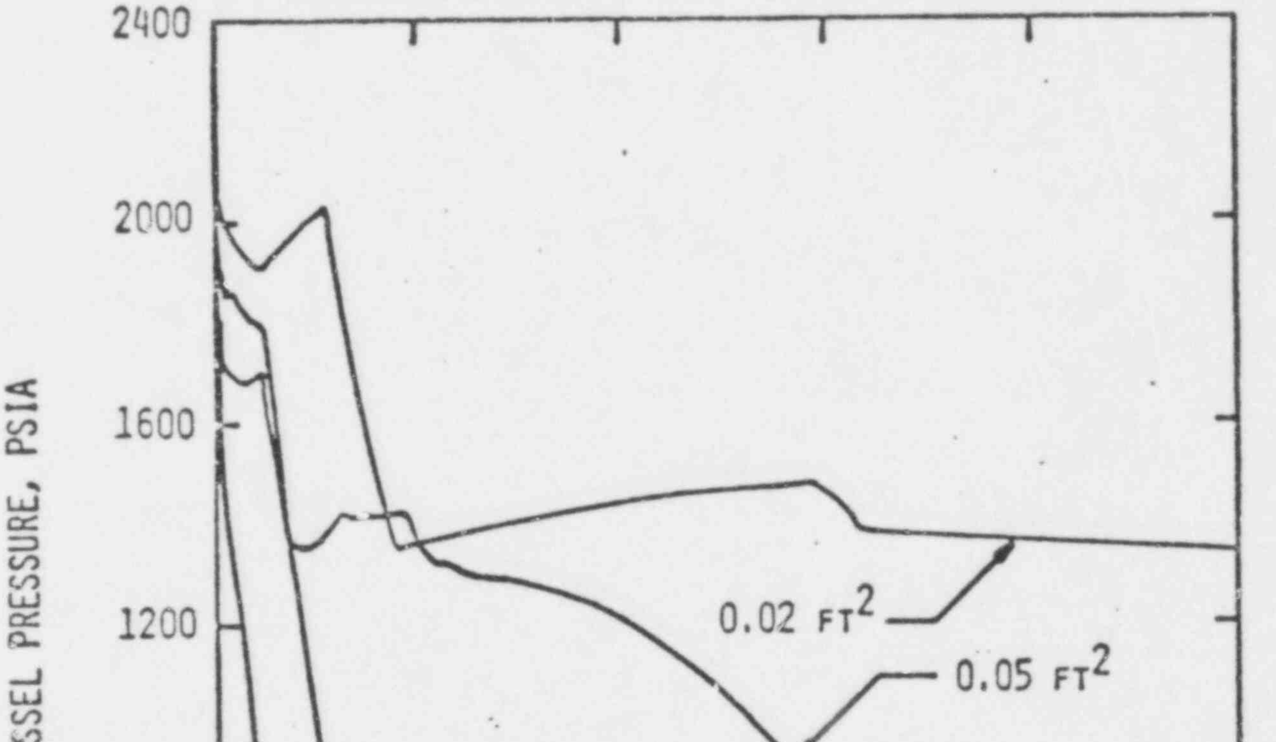
LOSS OF MAIN F.W. ON REACTOR TRIP

MAIN STEAM VALVES CLOSED ON REACTOR TRIP

NO OPERATOR ACTION FOR FIRST PHASE

SEC. SIDE AT SAT. CONDITIONS OF LOWEST SAFETY VALVE SETPOINT

228 201


2400
2000
1600
1200
800
400
0
INNER VESSEL PRESSURE, PSIA
0
600
1200
1800
2400
3000
TIME, SEC
0.02 FT2
0.05 FT2
0.2 FT2
0.5 FT2

228 202

C-E CONTROL SYSTEMS OVERVIEW

- AUTOMATIC TURBINE/REACTOR CONTROL BETWEEN 15% AND 100% POWER EXCEPT FOR MANUAL CONTROL OF BORON CONCENTRATION AND AXIAL POWER DISTIRBUTION

- CONTROL SYSTEMS LARGELY SEPARATE FROM AND INDEPENDENT OF PROTECTIVE SYSTEMS

- REDUNDANT 2/2 LOGIC EMPLOYED TO ENHANCE SAFETY AND AVAILABILITY BY PREVENTING SINGLE CONTROL SYSTEM FAILURES FROM CAUSING STRONG, RAPID CONTROL SYSTEM ACTIONS
  - STEAM BYPASS CONTROL SYSTEM

- CONTROL SYSTEMS DESIGNED TO ENHANCE SAFETY AND AVAILABILITY BY RESTORING ACCEPTABLE CONDITIONS WITHOUT CHALLENGING THE PROTECTIVE SYSTEM DURING A NUMBER OF DESIGN BASIS EVENTS

- EXTENSIVE DIAGNOSTIC/TEST FEATURES PROVIDED TO ENHANCE RELIABILITY

ECCS OPERATION

	DEMAND	FAIL TO OPERATE
DEMAND ECCS OPERATION		
PORV OPENING	1	0
POST-TRIP EXCESS FW	3	0
FW BYPASS VALVES STUCK OPEN	1	0
UNBLOCKED SIGNAL DURING COOLDOWN	1	0
INADVERTENT ECCS ACTUATION		
RPS/ESFAS TESTING	3	0

ECCS OPERATION

DEMAND ECCS OPERATION	-	6.
FAILURE TO OPERATE	-	0
INADVERTENT ACTUATION	-	3
FAILURE TO OPERATE	-	0

228 205

AUXILIARY FEEDWATER OPERATION

	TIMES USED	FAILED TO OPERATE
AS EMERGENCY FEEDWATER		
LOSS OF FEEDWATER	1	0
LOSS OF OFFSITE POWER	9	0
LOSS OF CONDENSER VACUUM	7	0
AS AUXILIARY FEEDWATER		
AUTOMATIC TRIPS	163	0
OTHER SHUTDOWNS	249	0

PORV ACTUATION

DEMAND VALVE OPENINGS

	OPEN	FAIL TO CLOSE
HIGH PRESSURE TRIP DURING SPURIOUS TURBINE RUNBACK AT POWER -	1	0
HIGH PRESSURE TRIP DURING LOSS OF LOAD - POWER ASCENSION TEST -	1	0

INADVERTENT ACTUATION

	OPEN	FAIL TO CLOSE
RPS MAINTENANCE - SPURIOUS OPEN SIGNAL - PRE POWER TESTING AT HOT SHUTDOWN -	1	1
PORV MAINTENANCE - SPURIOUS OPEN SIGNAL - COLD SHUTDOWN -	1	0

PORV OPERATION

DEMAND VALVE OPENINGS	-	2
FAILURE TO CLOSE	-	0
INADVERTENT ACTUATION	-	2
FAILURE TO CLOSE	-	1

LOSS OF OFF-SITE POWER

TIME = 0.0

REACTOR TRIPS
TURBINE TRIPS
RCPS COAST DOWN
MAIN CONDENSER IS LOST
MAIN FEEDWATER IS LOST

AUXILIARY FEEDWATER IS NOT INITIATED

PRESSURIZER ASSUMPTIONS

NO CHARGING/LETDOWN
NO HEATERS/SPRAY
NO WALL HEAT TRANSFER

STEAM GENERATOR PRESSURE IS MAINTAINED BY

1. SECONDARY CODE SAFETY VALVES, OR
2. SECONDARY DUMP VALVES