# SAFETY SIGNIFICANCE OF FIRE INDUCED SHORTS ON THE 1E12*MOVF009 and 1B21*MOVF019 VALVES

By

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## SAFETY SIGNIFICANCE EVALUATION

- Engineering Analysis Was Requested To Evaluate The Safety Significance Of Leaving The Valves Energized

- Of Specific Concern Were
  - 1E12*MOVF009 RHR SDC Suction Valve
  - 1B21*MOVF019 MSL Drain Valve

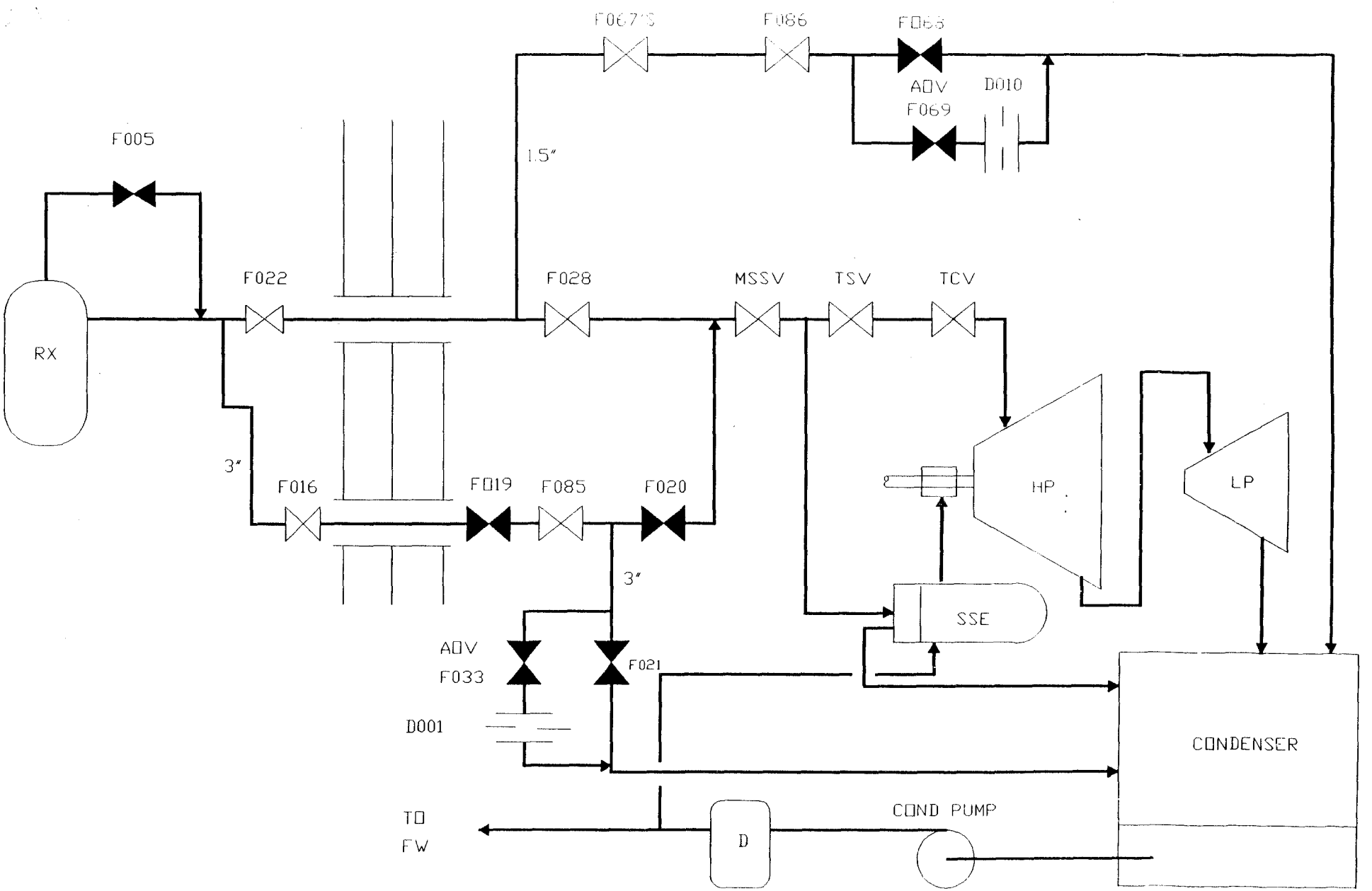
- Mechanistic Evaluations
  - Stress Analysis On RHR Piping (F009)
  - Off-site Dose Calculation For Discharge For Steam Piping (F019)

- Mechanistic Analyses Used Standard Design Basis Analytical Techniques

- Probabilistic Safety Assessments
  - Interfacing LOCA With Potential Core Damage (RHR System)
  - Steam Bypass To Condenser - No Core Damage (Main Steam)

- Probabilistic Analyses Based On Standard PRA Techniques
  - NUREG-1150's Interfacing System LOCA Methodology
  - Kuosheng's PRA Fire Risk Analysis

# MAIN STEAM, DRAIN LINES AND STEAM SEAL EVAPORATOR



## MECHANISTIC ANALYSIS OF 1B21*MOVF019 AND MOV*F021 VALVES FAILED OPEN

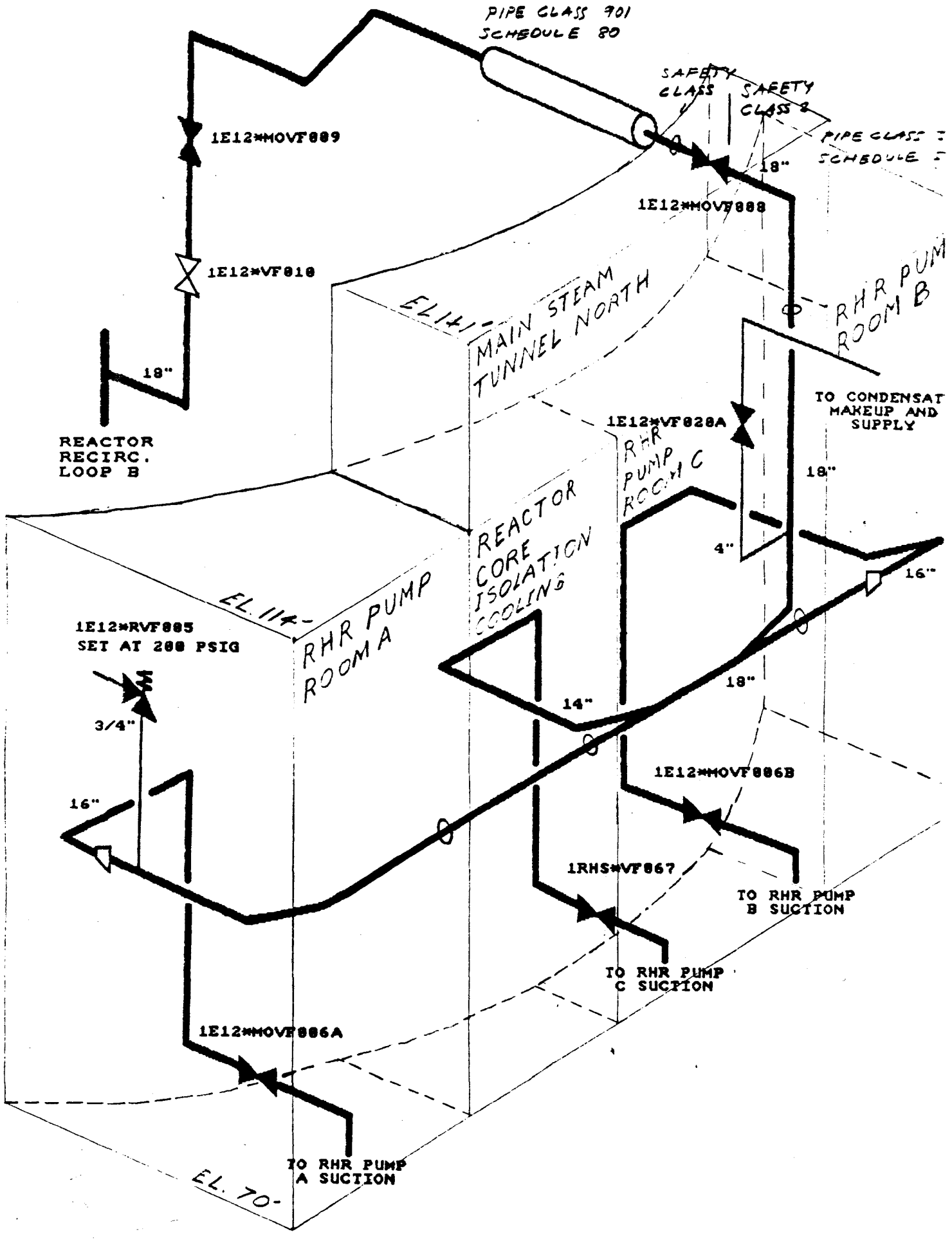
- Not An Important Contributor To Core Damage
  - Location And Effect Of Value On Other Reactor Systems
  - Low Flow (~ 124 GPM) To Condenser

- Primary Concerns Was Continued Blowdown And Offsite Release
  - Offsite Dose Calculation Was Based On Expected Activity In Reactor Water (Design Basis Value Based On NUREG-16)
  - Using Expected Activity Offsite Dose After 72 Hours Of Release Is Less Than 10CFR20 Limits (7.5 mrem For 72 Hour Release Versus 1500 mrem/Year 10CFR20 Limits)

- Offsite Calculations Are Conservative
  - During the 72 Hours For Going To Cold Shutdown The Reactor Pressure Decreases, Thereby Impacting Postulated Release Flow. The Flow (Release Rates) Was Held Constant For These Calculations.
  - Expected Activity Is Approximately 100 Times Greater Than Actual Activity. Therefore, Assumed Activity Is Very Conservative.
  - Operator Action To Shut F085 Should Occur Via AOP-31 And AOP-03.

- Therefore Consequences Of This Release Are Insignificant And Well Within The Design Basis Offsite Release Limits

FIGURE: RHR SHUTDOWN COOLING SUCTION LINE



## RHR PIPING STRESS ANALYSIS

- From Reactor Vessel To MOVF008 Design Pressure Is 1070 Psig And Temperature 552°F (ASME Class I)

- From MOVF008 To RHR Valves F006A, F006B, And F067 Design Pressure Is 200 Psig And Temperature 344°F (ASME Class II)

- Stress Analysis Based On 1070 Psig And 552°F For Class II Piping

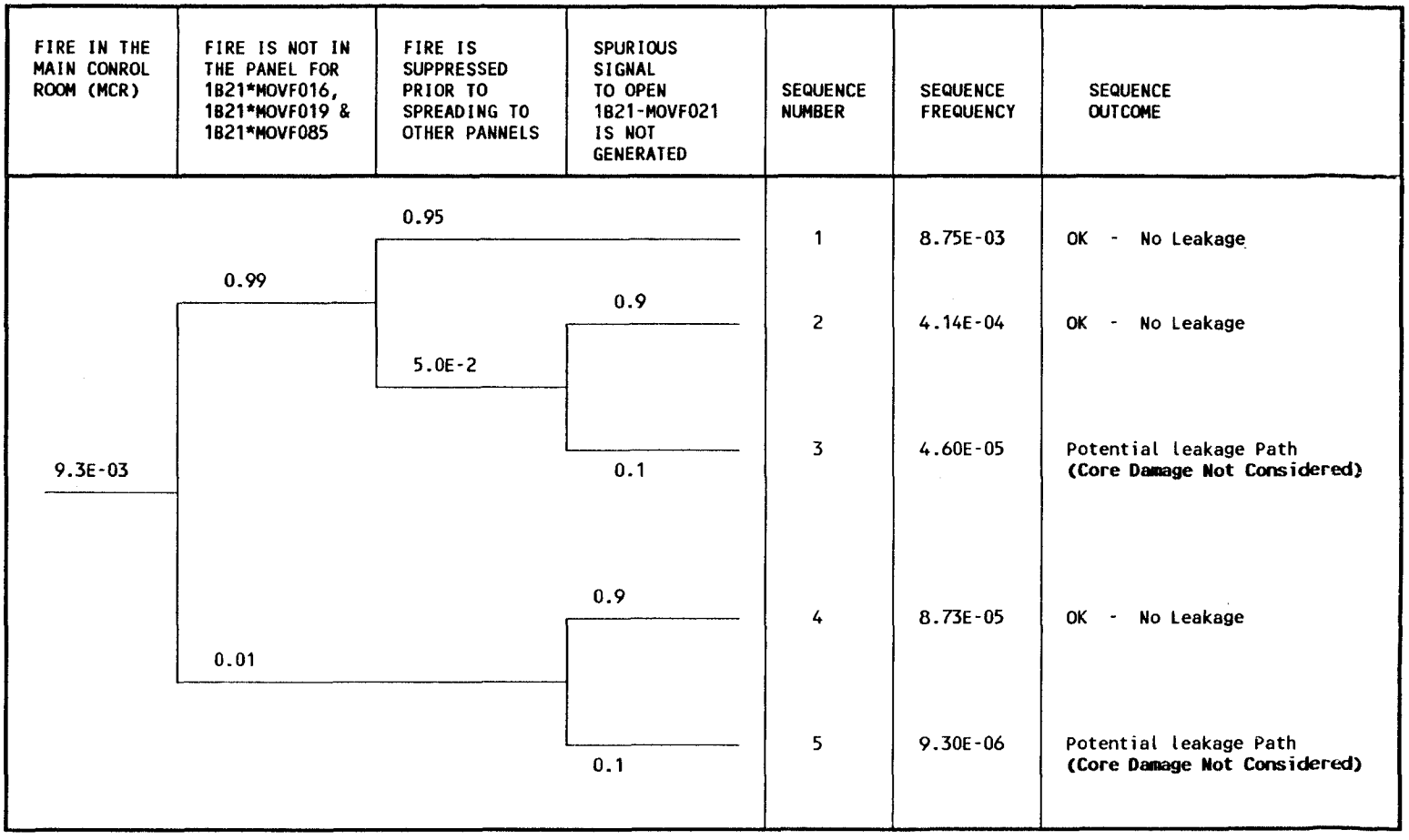
- All Stress Levels Meet Stress Allowables Per ASME-III And The Piping Material Is Within Yield Strength

- Therefore, For The Prescribed Pressurization Event The Piping Would Not Rupture

## DATA USED FOR PSA

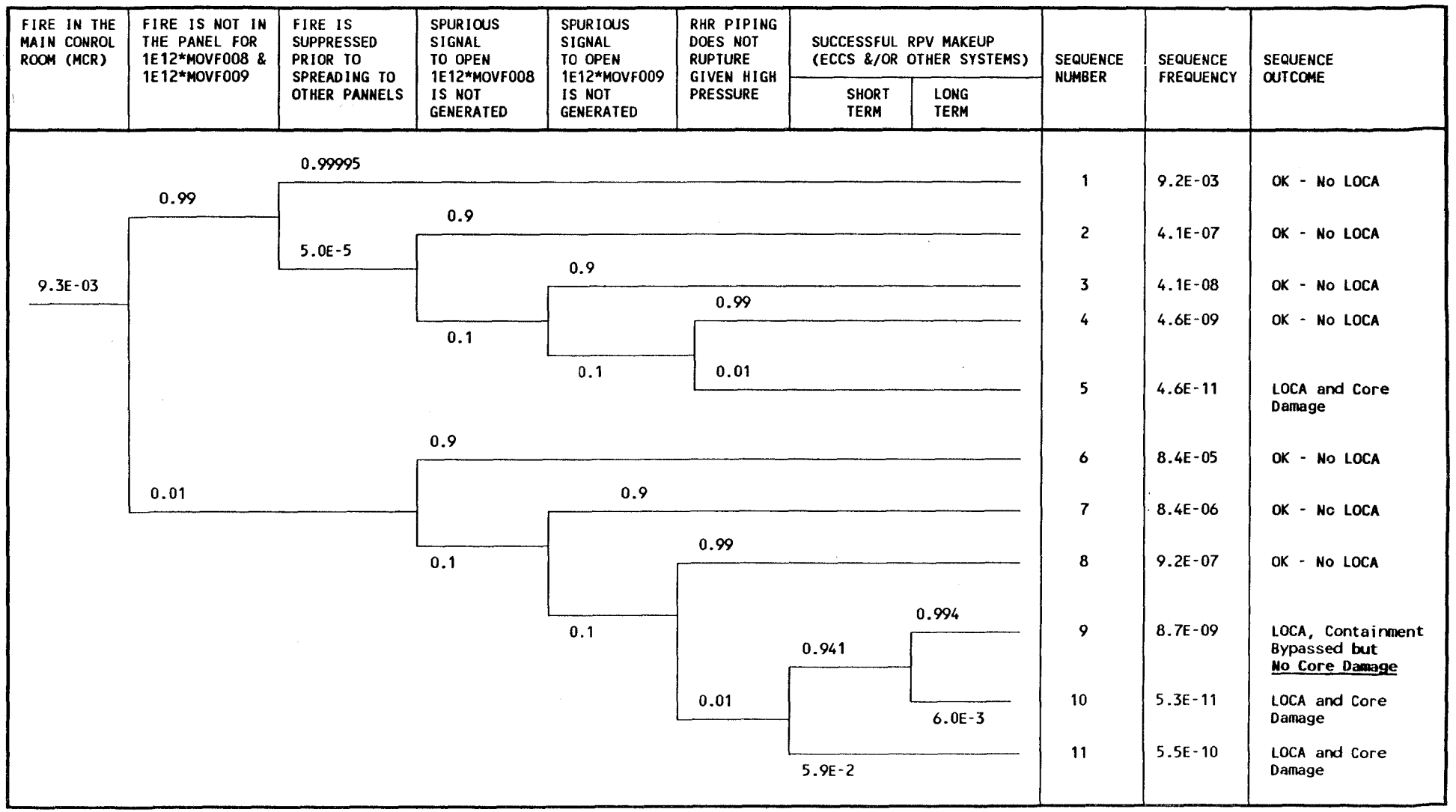
| | | | |
|---|---|---|---|
| 1. | Fire In RSS Panel | $3x10^{-4}$/yr | (Kuosheng PRA) |
| | Fire In MCR | $9.3x10^{-3}$/yr | (Kuosheng PRA) |
| 2. | F008 Opens Due To Hot Short Caused By Fire | 0.1/Event | (Sandia) |
| 3. | F009 Opens Due To Hot Short | 0.1/Event | (Sandia) |
| 4. | Pipe Rupture Due To Overpressurization | $1X10^{-2}$/Event | (NRC - NUREG-4550, AEOD/C502) |
| 5. | HPCS Fail To Maintain RPV Water Level (Short Term) | $5x10^{-2}$/Event | (RBS PRA) |
| 6. | Standard PRA Methodology Used | | |

**FIGURE 1.0 EVENT TREE FOR A CONTAINMENT BYPASS VIA CONDENSER DUE TO A FIRE IN THE MAIN CONTROL ROOM**




**TOTAL FREQUENCY OF POENTIAL RELEASE PATH = $5.5 \times 10^{-5}$ per Reactor-Year**

**FIGURE 2.0 EVENT TREE FOR A INTERFACING SYSTEM LOCA DUE TO A FIRE IN THE MAIN CONTROL ROOM**

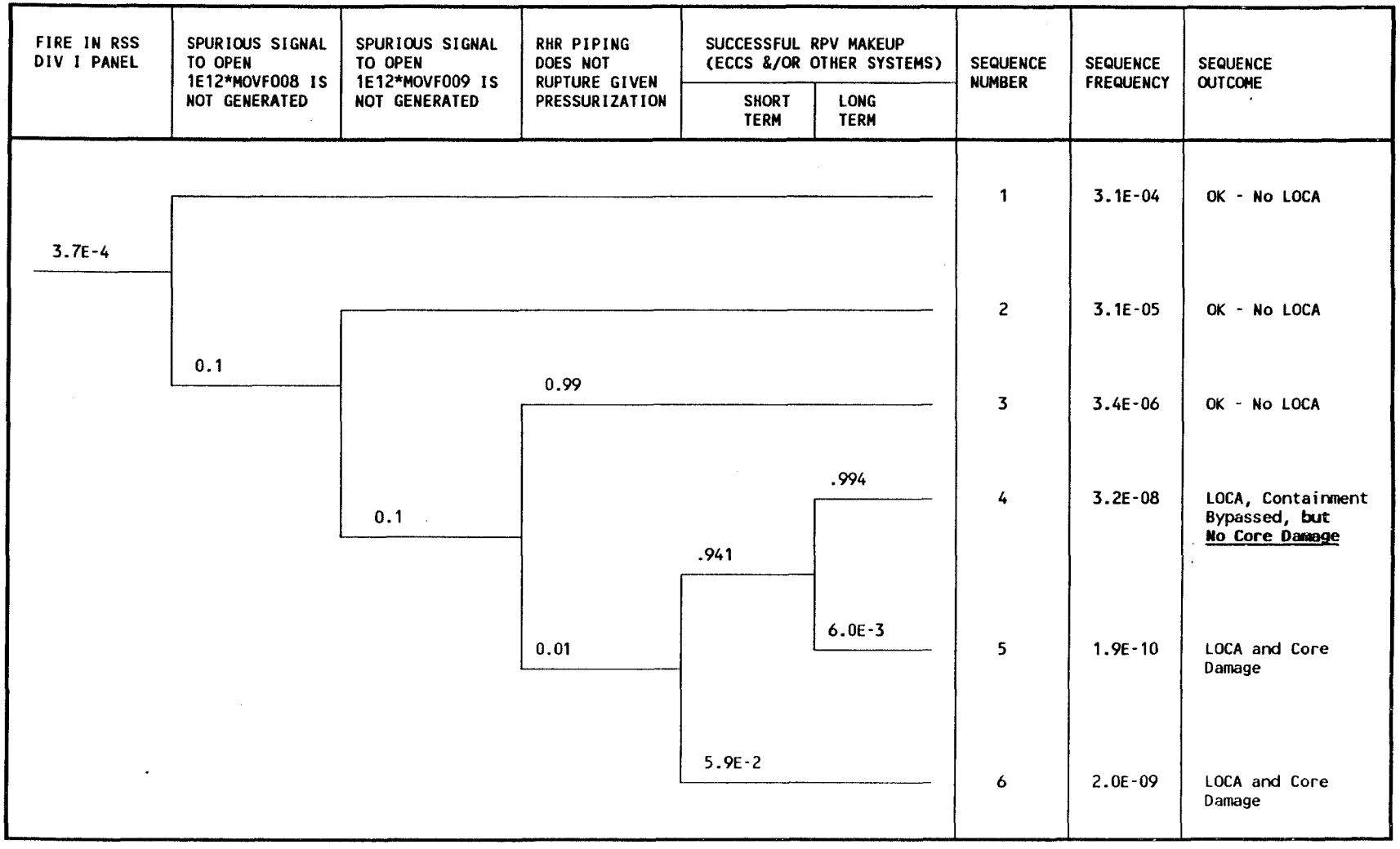



PROBABILITY OF PIPE RUPTURE = $1.0 \times 10^{-2}$

FOR ECCS - SHORT TERM ASSUMED THAT LPCS IS FAILED, ONLY HPCS CAN RESPOND TO BREAK

TOTAL CORE DAMAGE FREQUENCY DUE TO FIRES IN MCR = $6.5 \times 10^{-10}$ Per Reactor-Year

**FIGURE 3.0 EVENT TREE FOR A INTERFACING SYSTEM LOCA DUE TO A FIRE IN THE DIVISION I REMOTE SAFE SHUTDOWN (RSS) PANEL**




PROBABILITY OF PIPE RUPTURE = $1.0 \times 10^{-2}$

FOR ECCS - SHORT TERM ASSUMED THAT LPCS IS FAILED, ONLY HPCS CAN RESPOND TO BREAK

TOTAL CORE DAMAGE FREQUENCY DUE TO FIRES IN DIVISION I RSS PANEL = $2.2 \times 10^{-9}$ Per Reactor-Year

## SUMMARY OF RESULTS

### Interfacing LOCA (RHR System)

| | CDF |
|---|---|
| o F009 Open, Probability Pipe Rupture At $10^{-2}$/year | $3.2 \times 10^{-9}$/Reactor Year |
| o F009 Open, Probability Of Pipe Rupture At 1.0 | $5.8 \times 10^{-8}$/Reactor Year |
| o NRC Safety Goal For Core Damage | $10^{-4}$/Reactor Year |
| o Grand Gulf (NUREG-4550) | $4.0 \times 10^{-6}$/Reactor Year |
| o NRC Safety Goal For Large Release | $10^{-6}$/Reactor Year |

o Therefore Safety Implications Of This Situation On Core Damage Are Insignificant

### Steam Bypass (Main Steam)

o Probability Of Steam Bypass To The Condenser is $5.5 \times 10^{-5}$/Year

o If Occurred, Consequences Are Low

No Significant Offsite Dose Rates (Well Below 10CFR20 Limits)

- A 72 Hour Continuous Release Is Less Than 10CFR20 Limits And Well Within The Design Basis Of RBS 10CFR100 Limits

- No Core Damage

## CONCLUSIONS

- Probability Of Core Damage For Fire Induced Interfacing LOCA Is Less Than $10^{-8}$/year

- Probability Of Fire Induced Steam Bypass To The Condenser Is Low ($\sim 5.5 \times 10^{-5}$)/year)

  If Steam Bypass Occurred Offsite, Consequences Are Low (Less Than Offsite Operating Limit) And No Core Damage Is Expected

- Therefore, The Safety Consequences Of Having 1E12*MOVF009 And 1B21*MOVF019 Energized During Normal Operation Are Very Low

# Plant Response To Postulated Events

Plant Status

- Reactor at pressure

Postulated Event 1

- Main control room (MCR) fire, or Division I remote shutdown (RSS1) fire
- Fire Suppression systems and fire brigade assumed ineffective

Plant Response

- E12*F008 and E12*F009 open due to multiple hot shorts
- Potential overpressurization and damage to shutdown cooling suction piping

Response to potential overpressurization in residual heat removal (RHR) or reactor core isolation cooling (RCIC) cubicle:

- MCR annunication of pre-trip and trip condition by leak detection system area temperature and differential temperature monitors
- Back panel indication of area temperatures and differential temperatures
- Nuclear steam supply shutoff system (NSSSS) isolation of RHR/RCIC valves, MCR isolation annunciation, backpanel isolation indication, valve position indication
- Digital rad monitoring system (DRMS) alarm and indication of elevated activity due to postulated coolant leakage
- Alternate/supplemental temperature, radiation, and equipment status provided by local surveys
- Explicit procedural guidance in EOP-1, RPV Control, and EOP-3, Secondary Containment Control, to isolate the postulated discharge
- Symptom based EOP's may subsequently lead to reactor scram, or ultimately to emergency reactor depressurization
- Reactor scram, full NSSSS and balance of plant (BOP) isolation, and Emergency Core Cooling (ECCS) initiated prior to MCR evacuation

Postulated Event 1 - continued

Response to potential overpressurization in main steam tunnel

- Similar to response to potential overpressurization in RHR/RCIC cubicle
- Postulated discharge isolated manually or electrically with jumpers
- Automatic NSSSS isolation of main steam isolation valves (MSIV's) due to high area temperature, high area differential temperature
- Automatic reactor scram due to MSIV closure

Emergency Action Level

- Site Area Emergency (SAE) implemented due to a fire compromising the function of a safety system
- The full resources of the emergency plan would be available to assist on shift personnel in controlling the postulated casualty

Plant Status

- Reactor at pressure

Postulated Event 2

- MCR fire
- Loss of offsite power (LOOP)
- Fire suppression system and fire brigade assumed ineffective
- Loss of Division II

Plant Response

- Reactor scram on LOOP
- B21*F019 opens due to a hot short, opening a steam drain path to the condenser

Response To Postulated Steam Drain

- Relatively minor effect of increased reactor pressure decay would be noted during the shutdown process
- Required isolations verified by local operator action in accordance with AOP-3, Automatic Isolations

Emergency Action Level

- SAE implemented due to a fire compromising the function of a safety system
- The full resources of the emergency plan would be available to assist on shift personnel in controlling the postulate casualty