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IPTS STUDY FOR H.B.ROBINSON* (HBR-HYPO)

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PRESENTED AT U.S.-JAPAN SPECIALIZED TOPIC WORKSHOP (STW) ON PRESSURIZED-THERMAL-SHOCK

HOLIDAY INN CROWNE PLAZA HOTEL
ROCKVILLE, MARYLAND
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SUMMARY

A primary purpose of the U.S. Nuclear Regulatory Commission (NRC) Integrated Pressurized-Thermal-Shock (IPTS) Program, completed in 1985, was to develop an integrated probabilistic approach for evaluating pressurized water reactor (PWR) pressure vessel integrity; and the scope included the application of the methodology to three "high risk" PWR plants. The three plants selected were Oconee Unit 1, Calvert Cliffs Unit 1, and HBRobinson Unit 2 (HBR-2); and the plant studies were conducted in that order. As a result of this sequence and the developmental nature of the program, the HBR-2 study was the more complete and state-of-the-art.[1] However, by the time the HBR-2 study was conducted, a reevaluation of vessel chemistry and reference nil-ductility transition temperature (RT_{NDT}) had indicated relatively low concentrations of copper and nickel and low values of initial RT_{NDT} (RT_{NDT_0}), resulting in very low probabilities of failure. Thus, for illustrative purposes, copper, nickel, and RT_{NDT_0} were increased so that RT_{NDT} (2σ) = 270°F for the critical weld at 32 EFPY. This value of RT_{NDT} corresponds, of course, to the NRC PTS-Rule screening criteria (10 CFR 5.61). This hypothetical "plant" was referred to as HBR–HYPO, and it was identical to HBR-2 in every respect except for the concentrations of cooper and nickel and the value of RT_{NDT_0} for the welds.

HBR-2 is a three-loop, 2300 MW(th), pressurized water reactor plant. The high-pressure, emergency-core-coolant injection system has a rather low maximum head (1500 psi), although the relatively low flow-rate charging pumps can achieve 2500 psi (safety-valve setting) under conditions of no substantial leakage. On the steam side (secondary system), steamline flow restrictors limit the flow to 120% of normal flow in the event of a large steamline break between the steam generators and the main steamline isolation valves, thus limiting the severity of thermal shock to the pressure vessel that can result from blowdown cooling of the secondary.

Eight categories of event initiators were considered in the postulation of the PTS transients. They included four direct initiators (loss-of-coolant, steamline breaks, steam generator secondary-side overfeed, and steam generator tube rupture), and four indirect initiators (reactor trip, electrical-system failures, instrument-air system failures, and component and service-water system failures). Subcategories included size and location of breaks, plant states (zero and full power), and number of loops affected. External events (containment flooding, fires, etc.) and seismic events were not included.

Event trees with thousands of end points (PTS transients) were generated for each initiator and, after estimating the frequency of each end point, the transients were categorized for the thermal/hydraulic and fracture-mechanics analyses. Those with frequencies less than 10^{-7}/reactor year were relegated to residual groups, which contributed little to the

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overall frequency of failure. This left about 200 transients that were further categorized for the detailed thermal/hydraulic analyses. The RELAP-5 code,[2] modified to include adequate secondary-system detail, was used for the thermal/hydraulic analysis, and the OCA-P code[3] was used for the probabilistic fracture-mechanics analysis.

A detailed thermal/hydraulic analysis was performed for only 13 transients, while others were performed using simplified means, including interpolation of the detailed results. Fluid mixing was considered, for the case of stagnation flow, to obtain minimum downcomer coolant temperatures, and a one-dimensional thermal analysis of the vessel wall was performed using this minimum temperature.

Five categories of PTS transients were included in the RELAP-5 detailed analyses. They included steamline breaks, overfeed to the steam generator, small-break LOCAs, steam-generator tube ruptures, and loss of heat sink. Subcategories, involving hot zero power, full power, different line size breaks, etc., resulted in the total of 13 cases.

All of the vessel flaws considered in the analysis were surface flaws oriented in either an axial or circumferential direction and were assumed to have the same surface density in plate regions as in the weld regions. To account for space-wise variations in chemistry, RT_{NDT_0}, and fast neutron fluence, the vessel beltline region was divided into several subregions and the conditional probability of failure calculated for each. Axially oriented flaws in axial welds were assumed to have a surface length equal to the height of a shell course, while the surface length of all other flaws was assumed to be infinite. The flaw depth distribution function and the flaw density were those suggested in the Marshall report. The uncertainty in the flaw density was assumed to be quite large, and this had a significant impact on the calculated valued of the mean frequency of failure.

The possible benefits of warm prestressing were not included because of uncertainties regarding the time at which $\dot{K}_I = 0$ and the effective increase in K_{Ic} resulting from a specific load/temperature history.

Results of the fracture-mechanics analysis indicated that axial welds were the major contributor to the conditional probability of failure [P(F|E)], and most of the critical flaw depths were in the range of 0.25 to 0.65 in. Also, reducing the duration of the transient from the assumed value of 2 h to 1 h significantly reduced P(F|E). For the six most dominant transients, the decrease ranged from a factor of 2 to 30.

For the nominal conditions assumed for the study and 32 EFPY, P(F|E) for the most severe transient was 7×10^{-4}, and for the first and second most dominant transients it was 3×10^{-7} and 9×10^{-7}, respectively. For HBR-2, the values for the same three transients were $<10^{-10}$.

The total frequency of vessel failure [$\Phi(F)$] was obtained by summing the products of transient frequency and P(F|E) for all of the transients. The largest single products corresponded to the dominant transients, that is, the transients that contributed the most to $\Phi(F)$.

"Best-estimate" values of $\Phi(F)$ were obtained by using best-estimate values of all parameters in the study and simulating seven parameters in the probabilistic fracture-mechanics analysis. At 32 EFPY, $\Phi(F) = 1.4 \times 10^{-8}$/reactor year. For the six most dominant transients, the individual values ranged from 4×10^{-10} to 4×10^{-9}. Eighty-eight percent of $\Phi(F)$ was associated with reactor trips followed by stuck open steamline relief valves and dump valves.

A mean value of $\Phi(F)$ was obtained by performing an uncertainty analysis, which in principle considered uncertainties in all parameters. The mean value of $\Phi(F)$ so derived was 8×10^{-6}, which is slightly above the NRC limiting value of 5×10^{-6} (R.G. 1.154).

The single largest uncertainty in the calculation of Φ(F) was the number of flaws per vessel. This uncertainty contributed a factor of 45 to the mean value of Φ(F). Thus, improving the data base for flaw density could be very beneficial. Of course, another potential area of large uncertainty is the frequency of a transient. Great care must be exercised in selecting frequencies of initiating events and event-tree branch probabilities.

A comparison of the HBR–HYPO results with those for Oconee and Calvert Cliffs indicates that small differences in plant design and operating procedures can make a big difference in Φ(F), and thus generic IPTS studies are not adequate. Furthermore, it appears that the NRC screening criteria may not be appropriate for all U.S. PWRs.

REFERENCES

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2. C. D. Fletcher et al., *Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the H. B. Robinson Unit 2 Pressurized Water Reactor*, EG&G Idaho, Inc., Idaho Falls, Idaho, NUREG/CR-3977, April 1985.

3. R. D. Cheverton and D. G. Ball, *OCA-P, A Deterministic and Probabilistic Fracture-Mechanics Code for Application to Pressure Vessels*, NUREG/CR-3618 (ORNL-5991), Oak Ridge National Laboratory, Oak Ridge, TN, May 1984.

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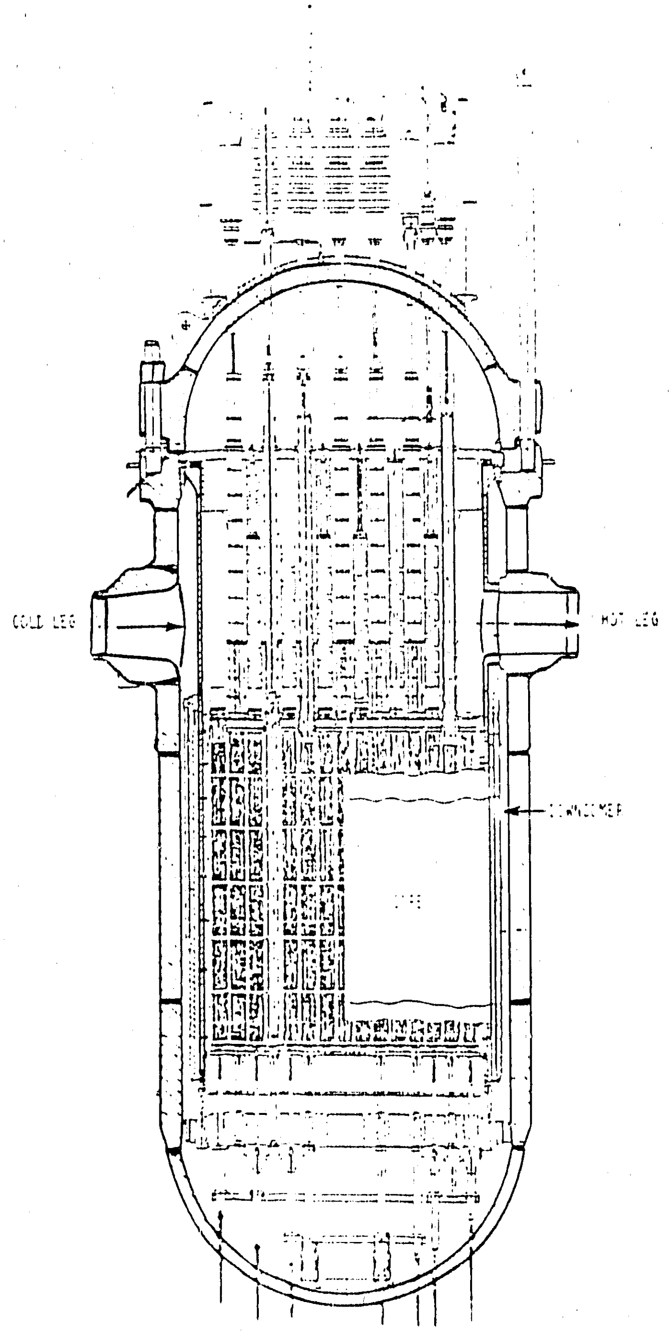
THREE PWR PLANTS, REPRESENTING THREE U.S. NSSS VENDORS AND EXPECTED TO HAVE RELATIVELY HIGH CALCULATED FREQUENCY OF VESSEL FAILURE, WERE SELECTED FOR INCLUSION IN THE IPTS STUDY

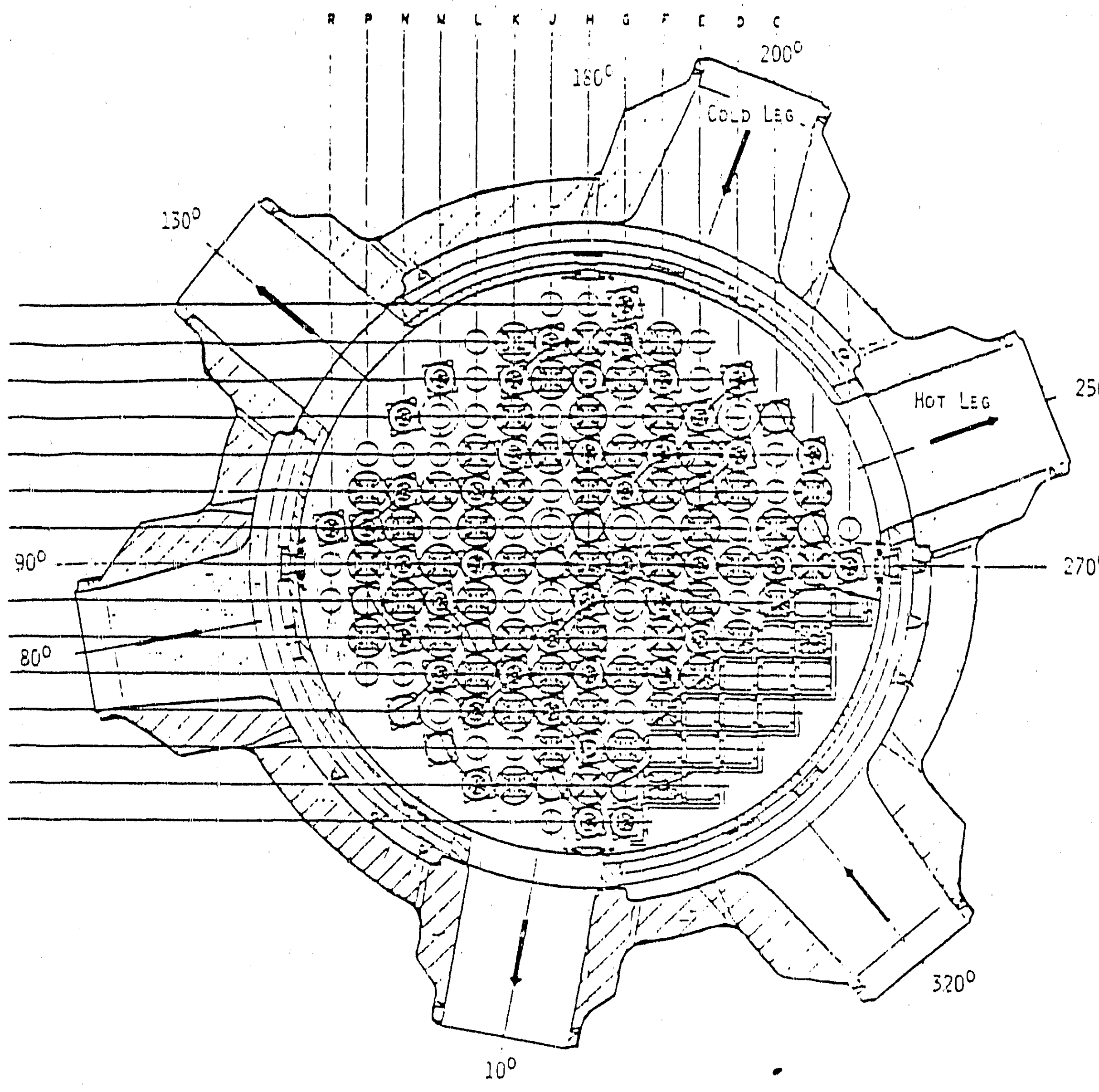
PLANT	NSSS VENDOR	VESSEL FABRICATION
• OCONEE UNIT 1	B&W	B&W
• CALVERT CLIFFS UNIT 1	CE	CE
• H.B.ROBINSON UNIT 2	W	CE

HBR-2 VESSEL "TYPICAL" OF 2300 MW (th) PLATE-TYPE PWR VESSEL

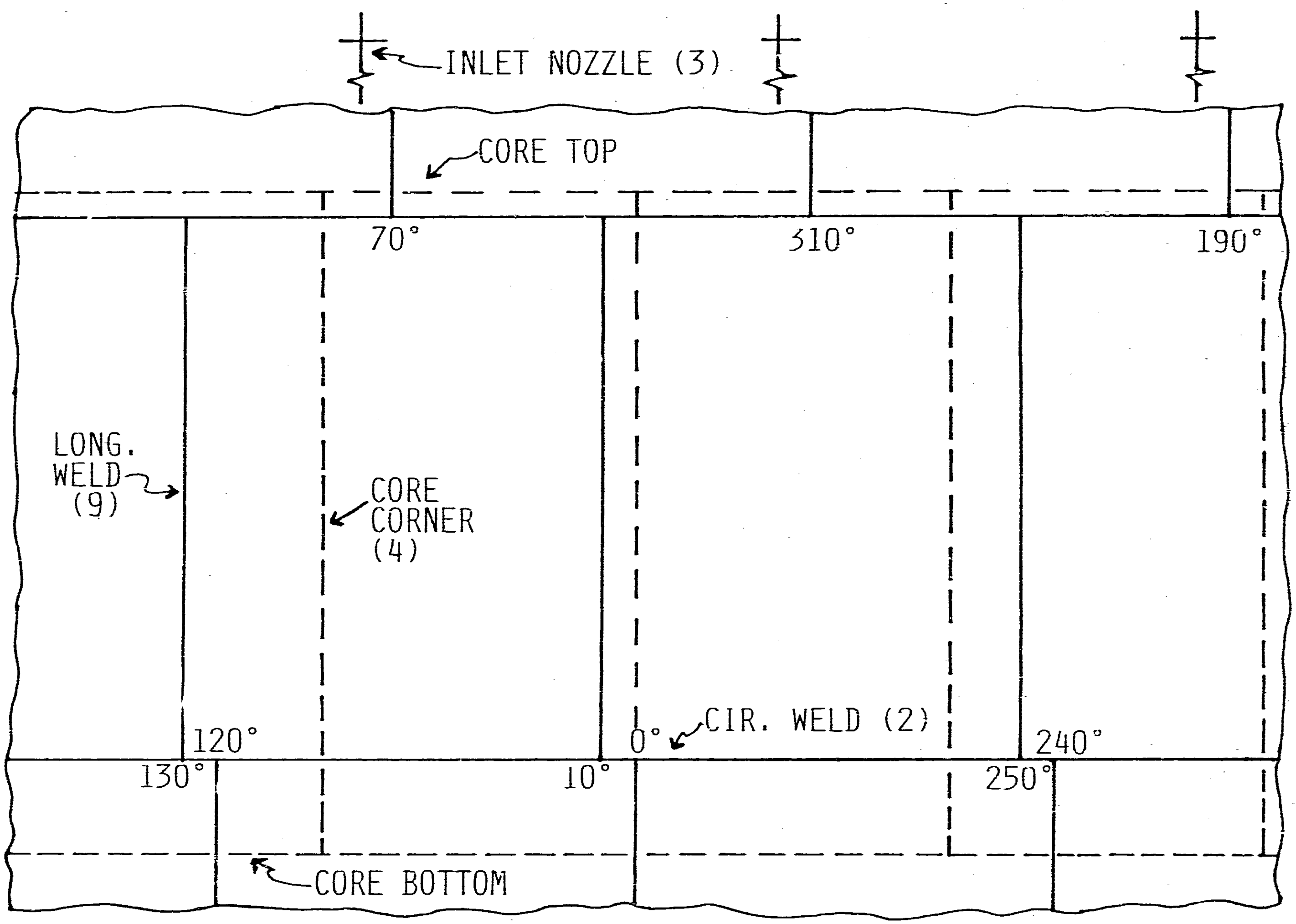


- OD = 174 in. (4.42 m)
- WALL THICKNESS = 9.4 in. (240 mm)
- PLATE CONSTRUCTION (A302B)
- CIRCUMFERENTIAL AND AXIAL WELDS IN BELTLINE REGION

PLAN VIEW OF HBR REACTOR AND VESSEL



DEVELOPED VIEW OF HBR PRESSURE VESSEL OUTER SURFACE



HBR-2 NOT ACTUALLY HIGH-RISK PLANT; THUS, HYPOTHETICAL HBR VESSEL (HBR–HYPO) CREATED FOR ILLUSTRATION OF IPTS METHODOLOGY

- ACTUAL CHEMISTRY OF BELTLINE WELDS DETERMINED USING HEAD-WELD BOAT SAMPLE

- HBR-2 PTS TRANSIENTS AND TRANSIENT FREQUENCIES USED IN ANALYSIS

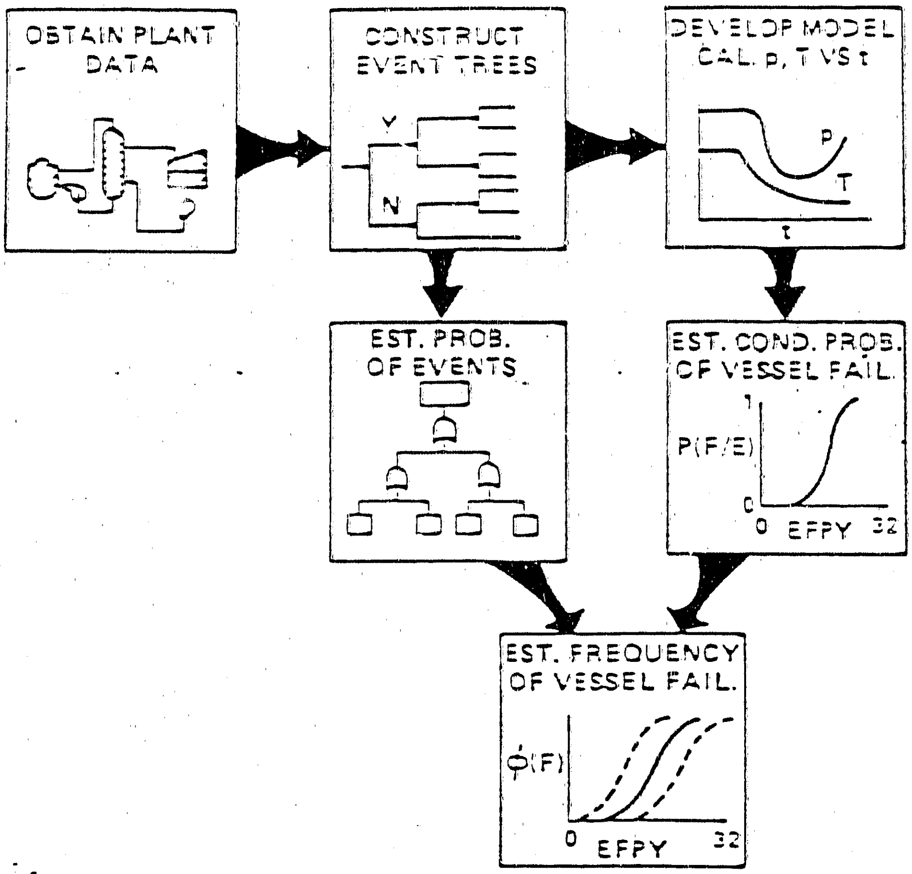
CHEMISTRY AND $RTNDT_O$ SELECTED FOR HBR–HYPO SUCH THAT RTNDT (2σ, DOMINANT REGION) = 270°F (132°C) AT 32 EFPY, USING PTS TREND CURVE

Region	Cu (wt%)		Ni (wt%)		$RTNDT_O$ (°F)	
	HBR	HYPO	HBR	HYPO	HBR	HYPO
Axial welds	0.22	0.22	0.04	0.80	–56	0
Cir. welds	0.17	0.22	1.0	0.80	–56	0
	0.19	0.22	0.8	0.80	–56	0
Plate	0.12	0.12	0.1	0.80	46	0

CHEMISTRY AND $RTNDT_O$ SELECTED FOR HBR–HYPO SUCH THAT RTNDT (2σ, DOMINANT REGION) = 270°F (132°C) AT 32 EFPY, USING PTS TREND CURVE (cont'd)

Region	Fluence ($10^{19} n/cm^2$)	Region Volume (ft^3)	RTNDT (2σ) (°F)
	Axial welds		
1A	1.24	0.14	223
1B	0.82	0.14	206
1C	0.41	0.14	181
2A	3.15	1.06	270
2B	1.03	1.06	215
2C	2.07	1.06	248
3A	1.95	0.28	245
3B	1.27	0.28	224
3C	1.27	0.28	224
	Cir. welds		
4A	1.64	3.5	236
4B	1.95	3.5	245
	Plate		
O2	1.95	71	155
O3	4.16	328	177
O4	1.64	35	150

IPTS APPROACH CONSISTS OF SIX BASIC STEPS



HBR-DISTINGUISHING PLANT-DESIGN FEATURES

- THREE LOOPS
- SAFETY INJECTION SYSTEM
  - HPI (3 PUMPS; 1500 psi; 375 gpm each)
  - LPI (2 PUMPS; 175 psi; 3000 gpm each)

    353,000 gal (90°F) + sump
  - ACCUMULATORS (3; 650 psi; 6000 gal each)
- CHARGING PUMPS (3, 2500 psi, 77 gpm each)
- STEAM FLOW RESTRICTORS (LIMIT STEAM FLOW TO 120% FOR MSLB)

EIGHT CATEGORIES OF <u>EVENT INITIATORS</u> WERE CONSIDERED

- DIRECT INITIATORS
  - LOCA
    - * THREE SIZES
    - * TWO PLANT STATES
      - ❑ ZERO POWER
      - ❑ FULL POWER
  - STEAMLINE BREAKS
    - * TWO SIZES
    - * TWO LOCATIONS
      - ❑ UPSTREAM OF MSIVs
      - ❑ DOWNSTREAM OF MSIVs
    - * TWO PLANT STATES

EIGHT CATEGORIES OF <u>EVENT INITIATORS</u> WERE CONSIDERED (cont'd)

- DIRECT INITIATORS (cont'd)
  - OVERFEED (SG SECONDARY)
    - * THREE TYPES
      - ❑ SINGLE LOOP
      - ❑ MULTILOOP
      - ❑ DELAYED
    - * TWO PLANT STATES
  - STEAM GENERATOR TUBE RUPTURE (SPECIAL LOCA CASE)
    - * TWO TYPES
      - ❑ SINGLE TUBE
      - ❑ MULTITUBE
    - * TWO PLANT STATES

EIGHT CATEGORIES OF EVENT INITIATORS WERE CONSIDERED (cont'd)

- INDIRECT INITIATORS (INITIATION FOLLOWED BY FAILURE OF SYSTEM COMPONENTS)
  - REACTOR TRIP
  - ELECTRICAL SYSTEM FAILURES
  - INSTRUMENT-AIR SYSTEM FAILURE
  - COMPONENT AND SERVICE-WATER SYSTEM FAILURE

SOME CATEGORIES OF EVENTS WERE NOT CONSIDERED IN IPTS STUDY

- EXTERNAL EVENTS (CONTAINMENT FLOODING, FIRES, ETC.)
- OPERATOR ACTIONS NOT ASSOCIATED WITH APPROPRIATE PROCEDURES
- SEISMIC EVENTS

EVENT-TREE BRANCH PROBABILITIES OBTAINED FROM SEVERAL SOURCES

- INITIATING EVENT AND EQUIPMENT FAILURES
  - NREP GENERIC DATABASE
  - NUCLEAR PLANT OPERATING EXPERIENCE
- OPERATOR FAILURES
  - INFLUENCE DIAGRAMS
  - HANDBOOK OF HUMAN RELIABILITY
  - TIME RELIABILITY CURVES

TRANSIENTS CATEGORIZED FOR THERMAL/HYDRAULIC AND FRACTURE-MECHANICS CALCULATIONS

- THOSE WITH $\Phi(E) \leq 10^{-7}$ ASSIGNED TO "RESIDUAL" GROUPS

- ~200 TRANSIENTS WITH $\Phi(E) > 10^{-7}$

- DURATIONS OF ALL TRANSIENTS = 2 h

THERMAL/HYDRAULIC CALCULATIONS PERFORMED TO OBTAIN T, p, h VS TIME

- RELAP5 MODIFIED TO INCLUDE DETAILS OF SECONDARY SYSTEM

- COMPLETE MODEL USED FOR 12–14 TRANSIENTS (VERY EXPENSIVE)

- SIMPLIFIED MODEL AND INTERPOLATION USED FOR OTHERS (MUCH LESS EXPENSIVE)

- MIXING CONSIDERED

  - PURDUE REMIX CODE AND 1/2-SCALE PTS FACILITY

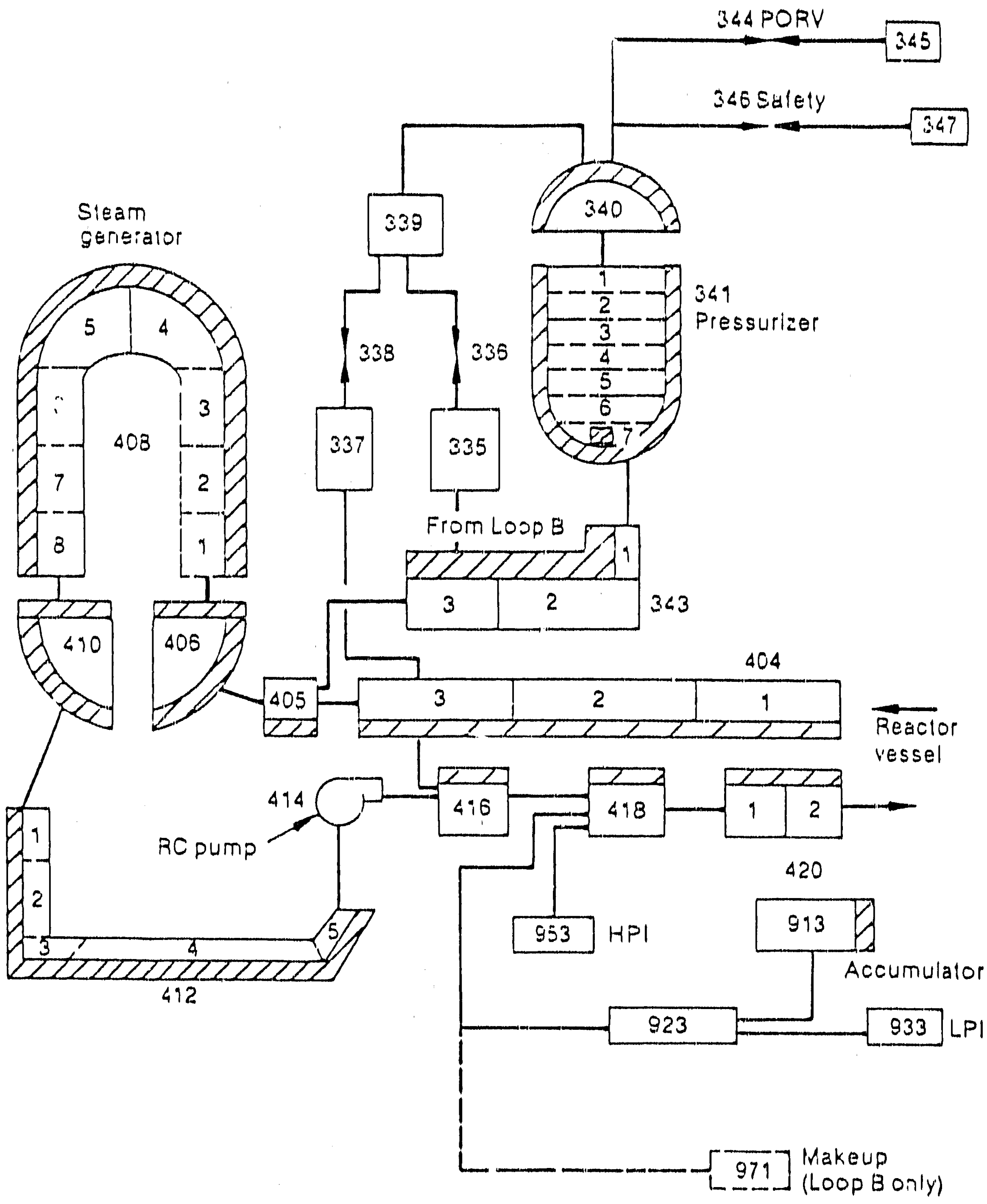
    CREARE 1/5-SCALE FACILITY

FIVE CATEGORIES OF PTS TRANSIENTS WERE INCLUDED IN THE RELAP5 DETAILED ANALYSIS

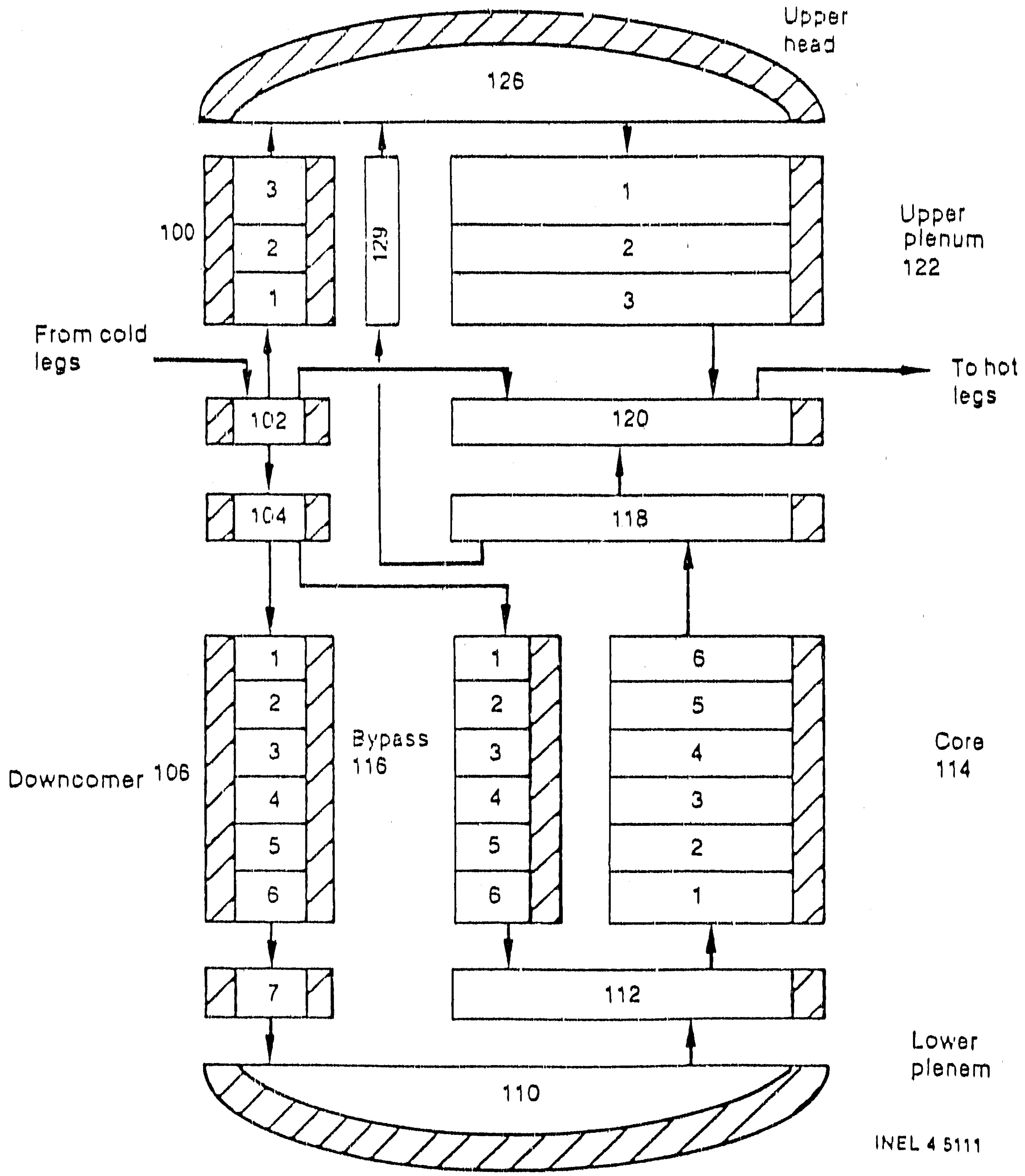
Scenario Number	Initial Plant Condition	Initiating Event
	Steam-line breaks	
1	Hot 0% power	1.0-ft^2 break in main steam line
2	Hot 0% power	Double-ended main steam-line break
3	Hot 0% power	Stuck-open STM PORV
4	Full power	Three SDVs fail open
	Runaway feedwater	
5	Full power	Overfeed with auxiliary feedwater
	Small-break LOCAs	
6	Full power	2-1/2-in. hot leg break
7	Full power	PZR PORV-size break
8	Hot 0% power	2-1/2-in. hot leg break
9	Full power	2-in. hot leg break
10	Hot 0% power	PZR PORV-size break
	SG tube ruptures	
11	Hot 0% power	SG tube rupture
12	Full power	SG tube rupture
	Loss of heat sink	
13	Full power	Loss of heat sink with primary system feed-and-bleed recovery

The acronyms used in this table (in order of their appearance) are: STM PORV = steam power-operated relief valve, SDV = steam dump valve, PZR PORV = pressurizer power-operated relief valve, and SG = steam generator.

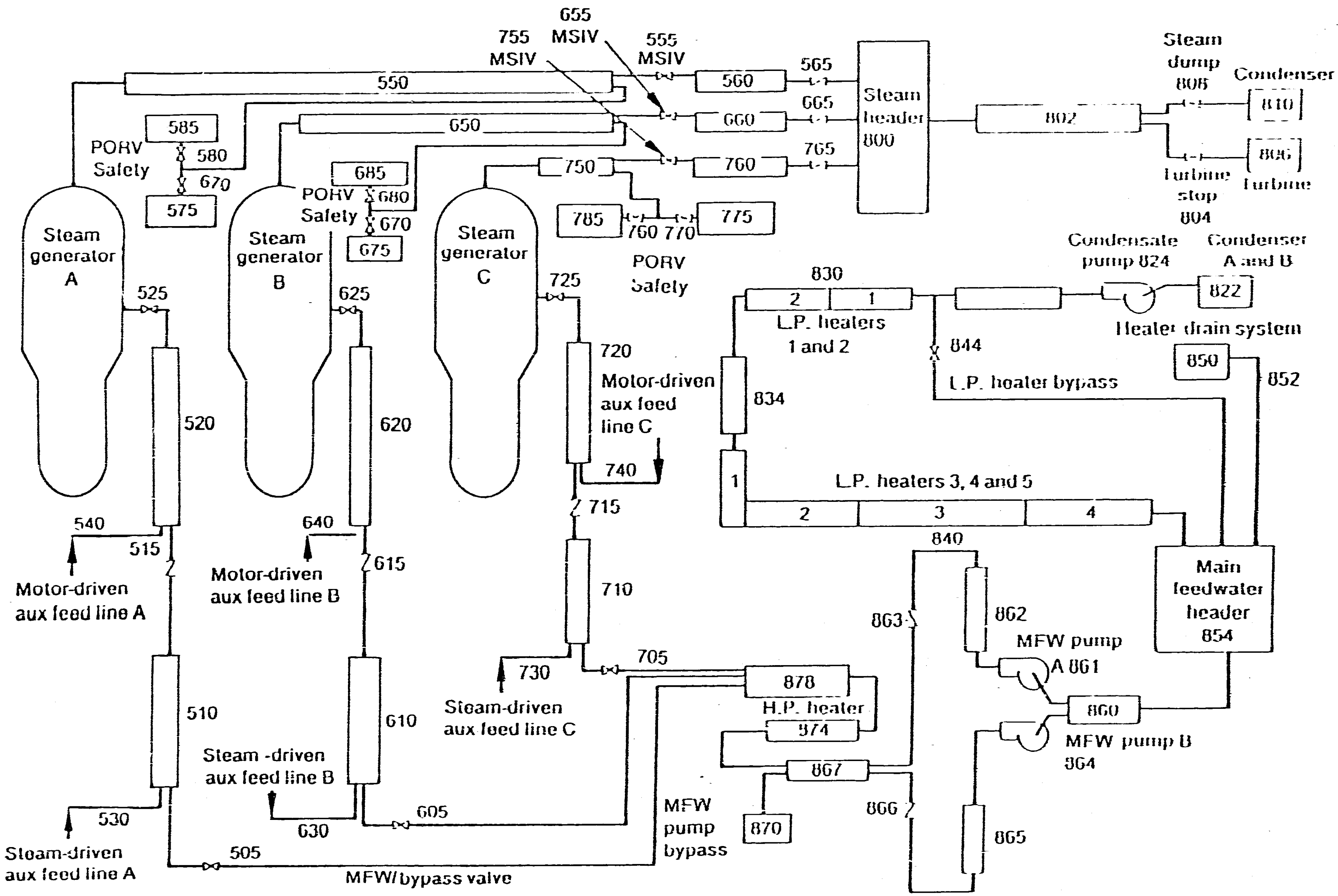
RELAP5 NODALIZATION OF PRIMARY COOLANT LOOPS (LOOP C SHOWN) FOR HBR-HYPO



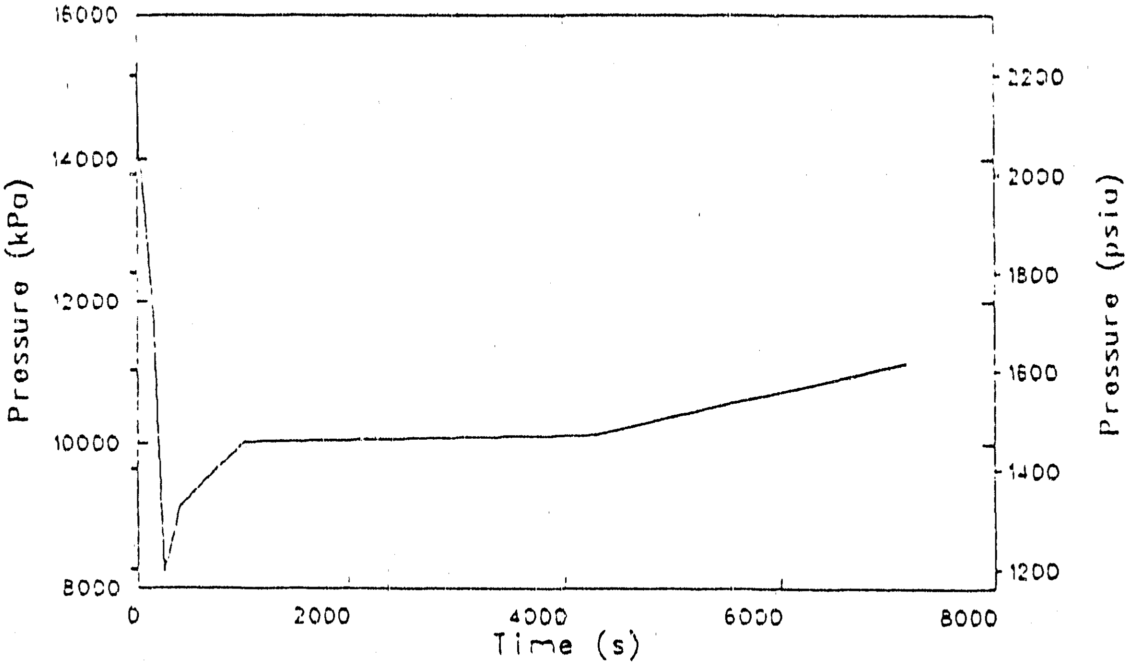
RELAP5 NODALIZATION OF REACTOR VESSEL FOR HBR-HYPO

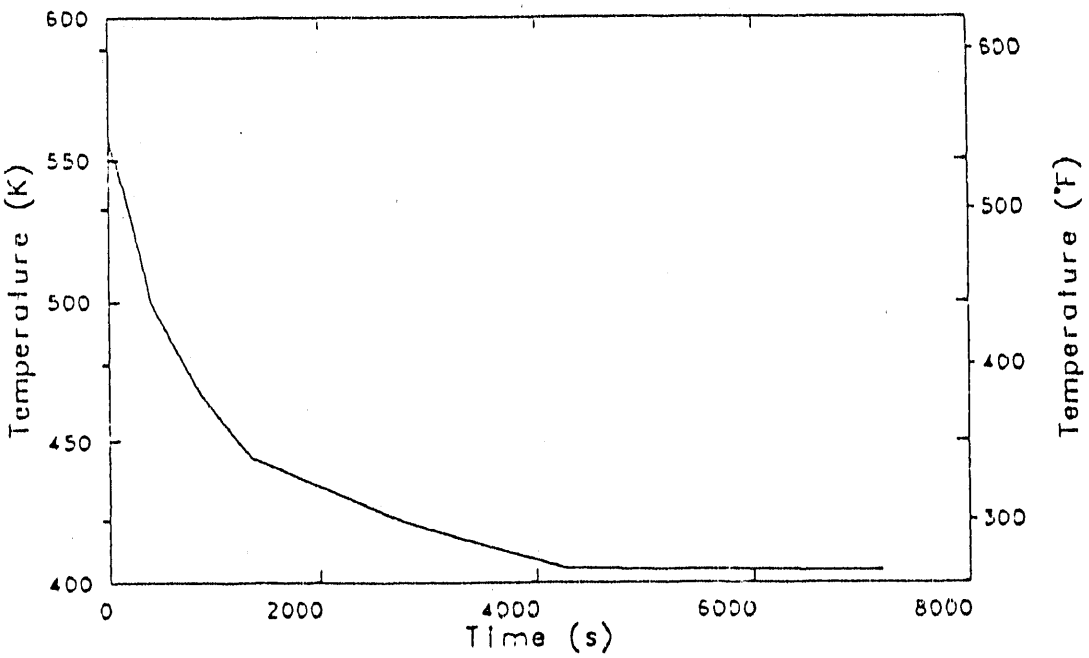


RELAP5 NODALIZATION OF FEEDWATER TRAIN AND STEAM LINES FOR HBR–HYPO

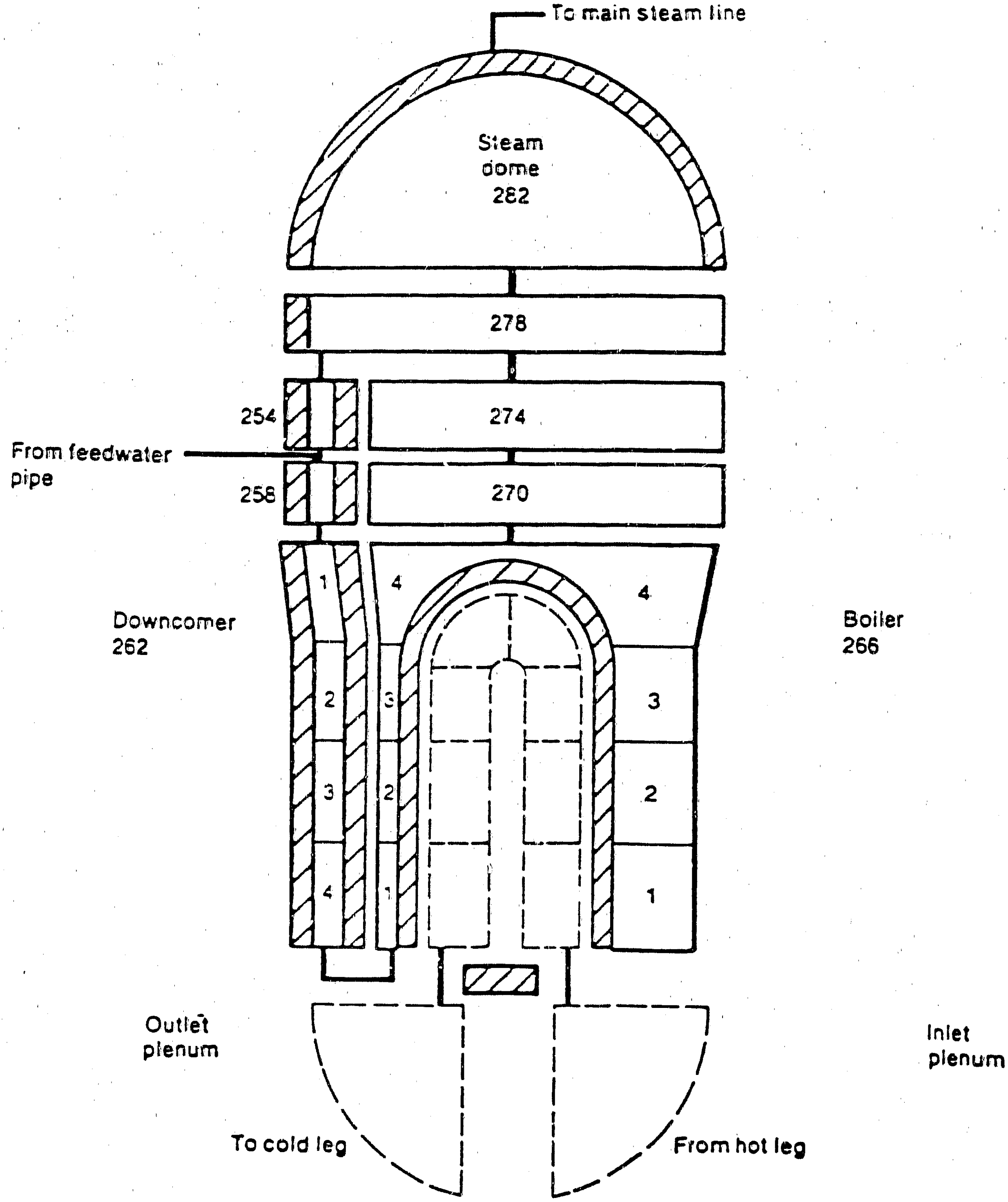


HBR-HYPO MOST DOMINANT TRANSIENT: REACTOR TRIP WITH 3 STM PORVs OPEN



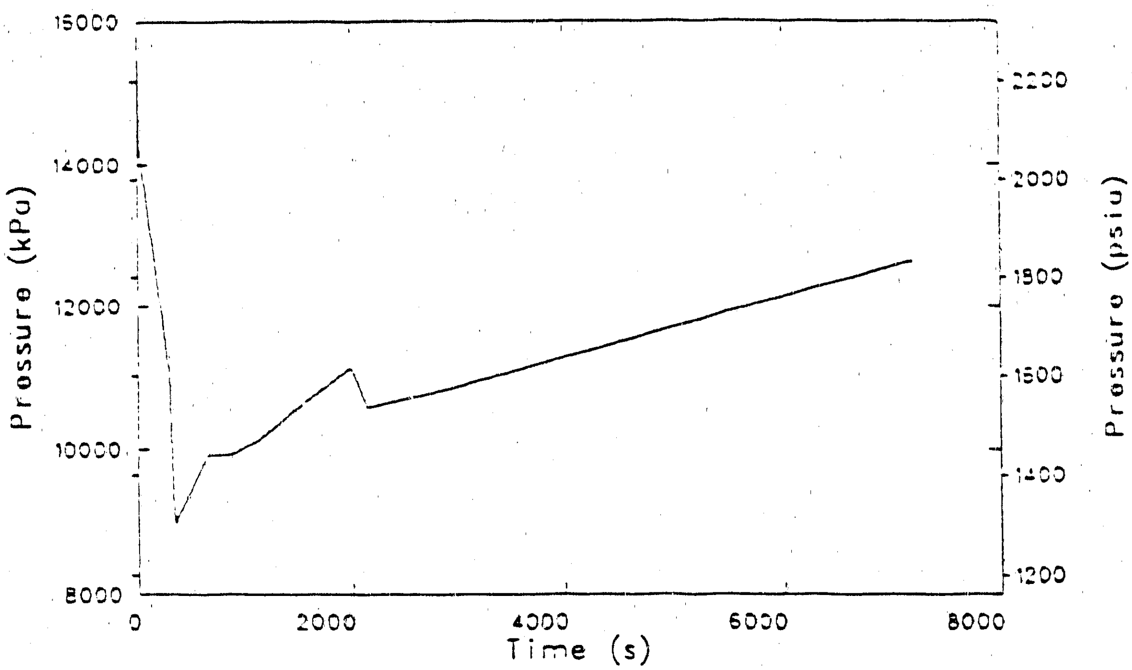


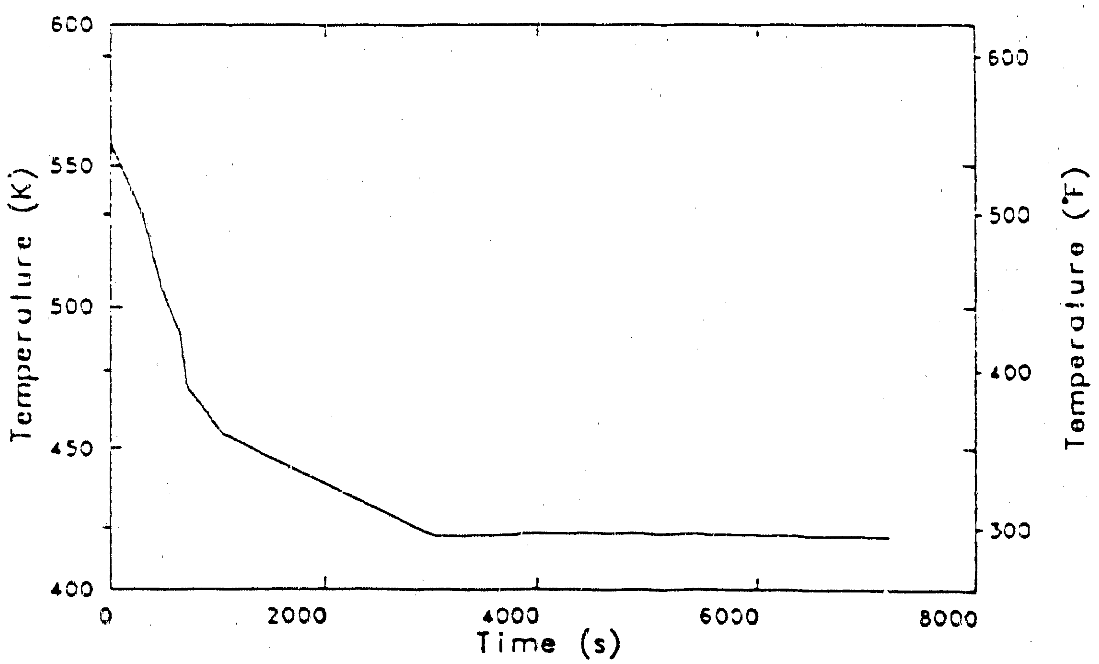
RELAP5 NODALIZATION OF STEAM GENERATOR (SG A SHOWN) FOR HBR–HYPO



Loop	Component Numbers
A	2xx
B	3xx
C	4xx

HBR-HYPO SECOND MOST DOMINANT TRANSIENT: REACTOR TRIP WITH 2 STM PORVs OPEN





SCOPE OF PROBABILISTIC FRACTURE-MECHANICS ANALYSIS

- P(F | E)
- SENSITIVITY OF P(F | E) TO SIMULATED PARAMETERS
- EFFECT OF WARM PRESTRESSING (WPS)
- EFFECT OF REMEDIAL MEASURES
  - REDUCTION IN FLUENCE RATE
  - IN-SERVICE INSPECTION
  - LIMIT ON REPRESSURIZATION
  - ANNEALING

PROBABILISTIC FRACTURE-MECHANICS ANALYSIS PERFORMED WITH OCA-P

- BASED ON MONTE CARLO METHODS
  - MANY VESSELS SIMULATED
  - DETERMINISTIC FM ANALYSIS FOR EACH
  - $P(F|E) = \dfrac{\text{NUMBER OF FAILURES}}{\text{NUMBER OR VESSELS}}$
- BASIC INPUT FROM SYSTEMS ANALYSIS:

  T_C , p, h = f(t)
- PERFORMS THERMAL, STRESS, AND FM ANALYSIS

SEVEN FM PARAMETERS SIMULATED IN IPTS STUDY

Parameter	Standard Deviation[a] (σ)	Truncation
Fluence (Φ)	0.3 $\mu(\Phi)$	$\Phi = 0$
Copper	0.025%	0.4%
Nickel	0.0	—
$RTNDT_o$	17°F[b]	*b*
$\Delta RTNDT$[c]	24°F[b,c]	*b*
$\Delta RTNDT$[d]	0.14 $\mu(\Delta RTNDT)$[d]	$\pm 3\sigma$
K_{Ic}	0.15 $\mu(K_{Ic})$	$\pm 3\sigma$
K_{Ia}	0.10 $\mu(K_{Ia})$	$\pm 3\sigma$
Flaw Depth		2.2 in.[e]

[a]Normal distribution used for each parameter.

[b]$\sigma_{(RTDNT)} = \left[\sigma^2_{(RTNDT_o)} + \sigma^2_{(\Delta RTNDT)}\right]^{1/2}$, truncated at $\pm 3\sigma$.

[c]Accounts for uncertainty in correlation.

[d]Accounts for uncertainty in Cu, Ni, and F_0 when $RTNDT_s$ is used as independent variable.

[e]For initial flaws only.

PROBABILISTIC FRACTURE-MECHANICS MODEL

- OCA-P (PROBABILISTIC FM CODE)
- LEFM
- 1-D THERMAL AND STRESS ANALYSIS
- CLADDING A DISCRETE REGION
- $T, p, h_f = f(t)$

PROBABILISTIC FRACTURE-MECHANICS MODEL (CONT'D)

- FLAW-DEPTH DENSITY FUNCTION (MARSHALL)
- FLAW SURFACE DENSITY = 0.036 flaws/ft^2
- $\overline{K}_{Ic} = 1.43\ K_{Ic}$ (ASME XI)
- $K_{Ic}\ (-2\sigma) = K_{Ic}$ (ASME XI)
- $K_{Ia} = 1.25\ K_{Ia}$ (ASME XI)
- $K_{Ia}\ (-2\sigma) = K_{Ia}$ (ASME XI)
- K_{Ia} (max) = 200 ksi$\sqrt{\text{in.}}$

PROBABILISTIC FRACTURE-MECHANICS MODEL (CONT'D)

- $\overline{\Delta RTNDT} = f(\Phi, Cu, Ni)$; (PTS TREND CURVE)

- $\Phi = \Phi_0 e^{-0.24a\ in.^{-1}}$

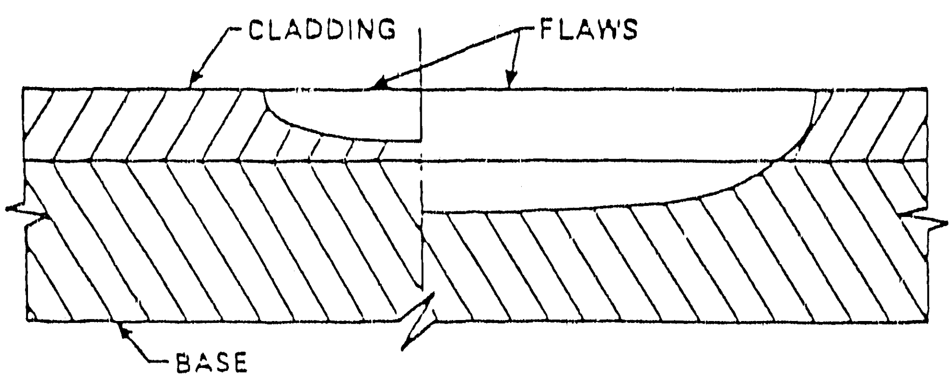
- FAILURE CRITERION: $K_I > K_{Ia}$ TO POINT OF PLASTIC INSTABILITY

FLAWS CONSIDERED WERE EITHER AXIALLY OR CIRCUMFERENTIALLY ORIENTED SURFACE FLAWS NORMAL TO SURFACE

- RESULT OF CLADDING PROCESS, STRESS–CORROSION CRACKING, ETC.

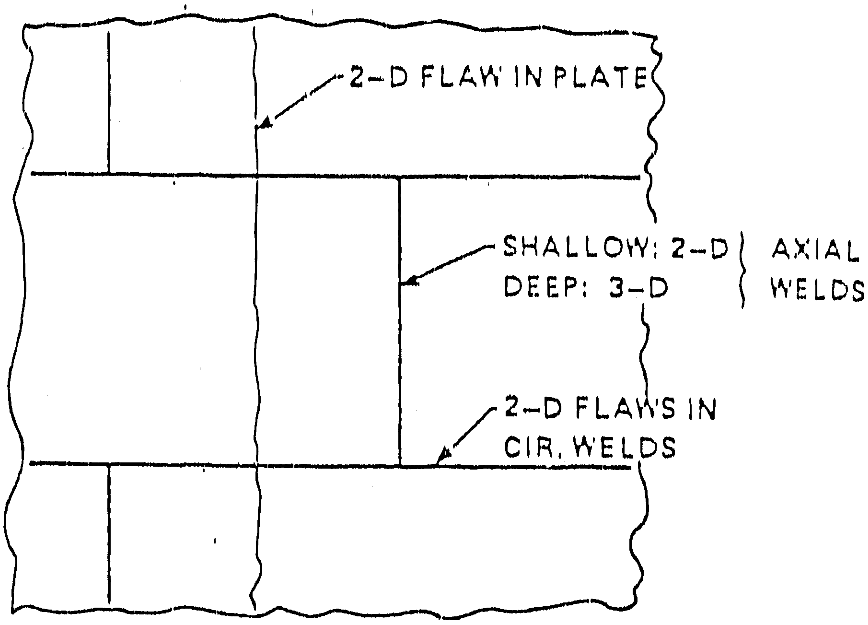
- VERY LITTLE NDE DATA

- LARGE UNCERTAINTY IN FLAW DENSITY



ORNL-DWG 84-6260 ETD

TWO FLAW GEOMETRIES (2–D, 3–D) AND THREE FLAW REGIONS (PLATE, AXIAL AND CIRCUMFERENTIAL WELDS) CONSIDERED

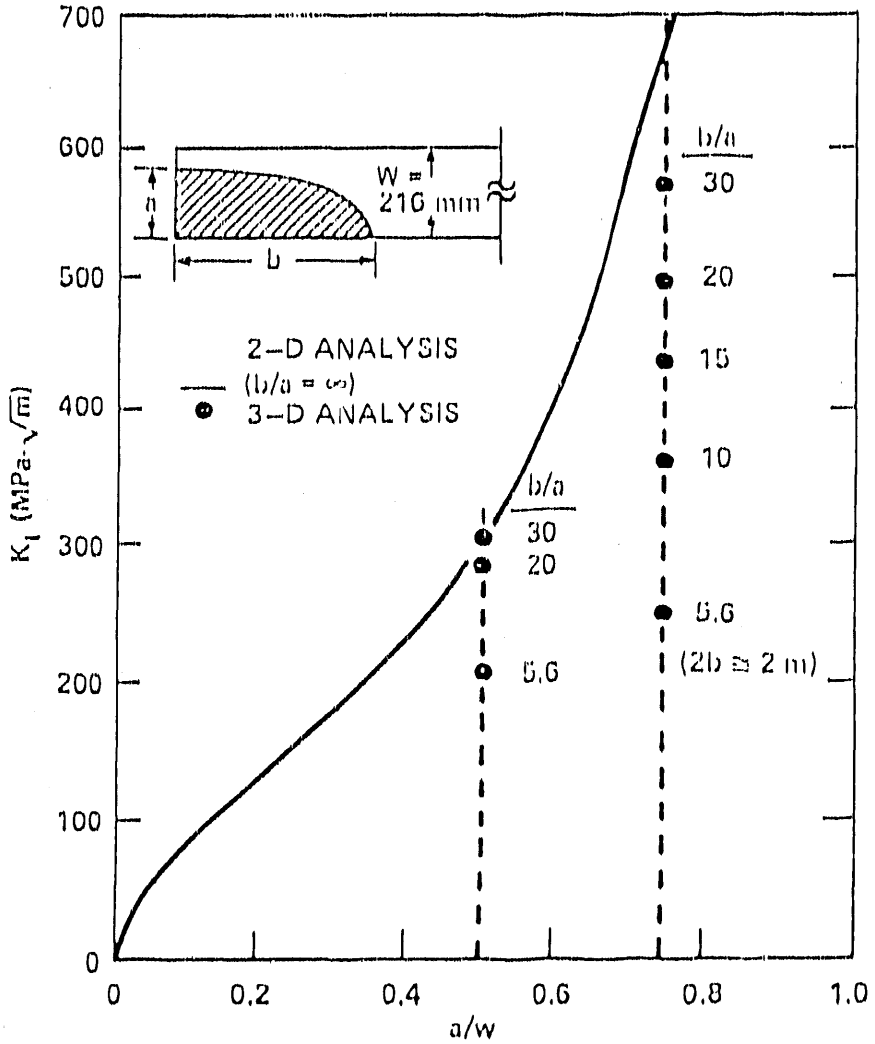


DEVELOPED VIEW OF BELTLINE REGION

MARTIN MARIETTA

ORNL DWG 83 4617A ETD

DEEP AXIAL FLAWS WITH SURFACE LENGTH EQUAL TO HEIGHT OF SHELL COURSE NOT INFINITELY LONG



AXIAL WELDS ARE DOMINANT CONTRIBUTOR TO P(F|E) FOR OCONEE-1, CC-1, HBR–HYPO

- Cu IN WELDS RELATIVELY HIGH
- K_I (AXIAL) > K_I (CIRCUMFERENTIAL)
- FLAW SURFACE DENSITY ASSUMED EQUAL FOR ALL REGIONS

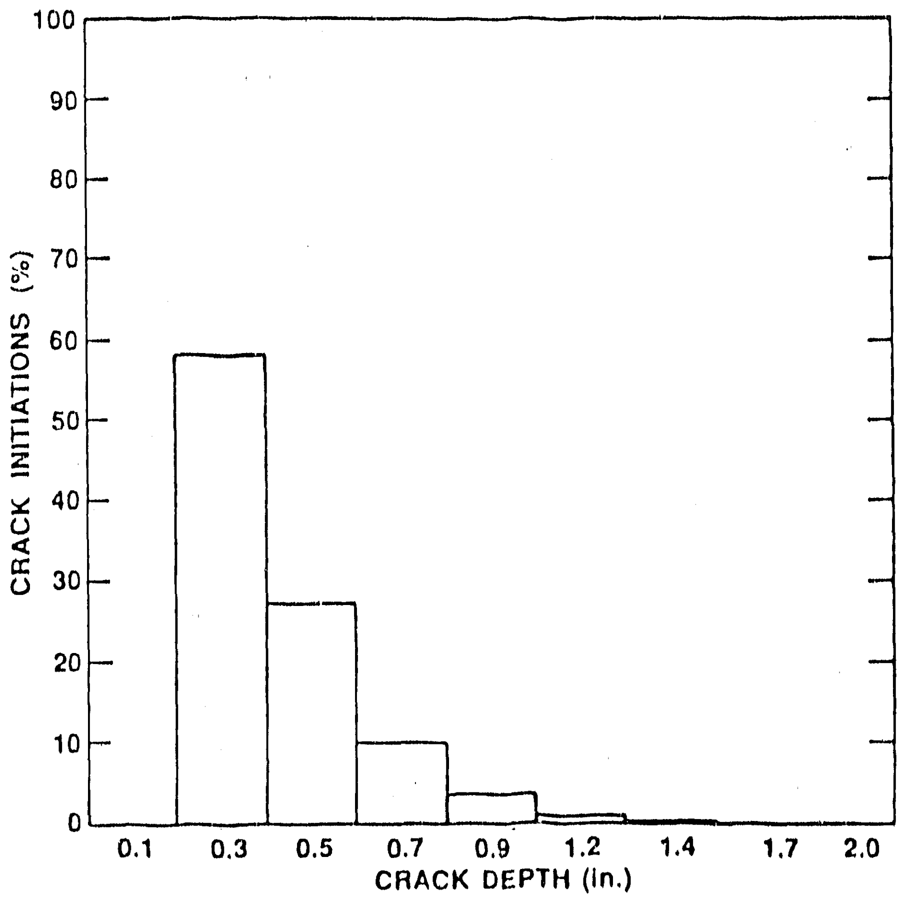
P(F|E) AT 32 EFPY FOR DOMINANT TRANSIENTS: $<10^{-10}$ TO 2×10^{-3}

TRANSIENT	P(F|E) AT 32 EFPY			
	PLANT			
	OCONEE-1	CC-1	HBR-2	HBR–HYPO
MOST SEVERE	5×10^{-3}	4×10^{-3}	$<10^{-10}$	7×10^{-4}
1st DOMINANT	2×10^{-3}	3×10^{-4}	$<10^{-10}$	3×10^{-7}
2nd DOMINANT	6×10^{-4}	1×10^{-5}	$<10^{-10}$	9×10^{-7}

MOST INITIAL INITIATION EVENTS WITH VERY SHALLOW FLAWS

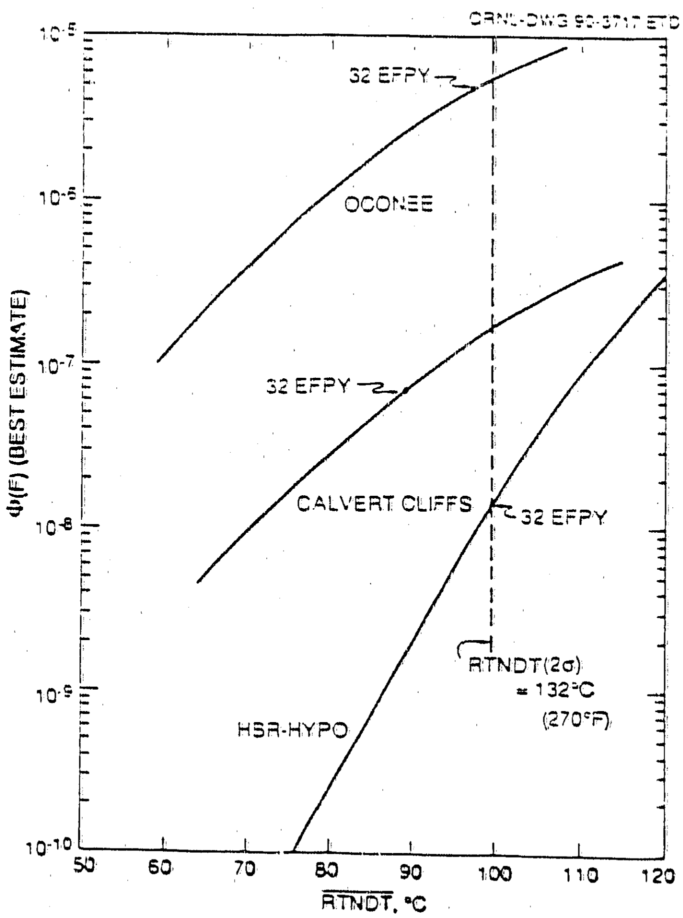
- SHALLOW FLAWS MORE LIKELY TO EXIST

- THERMAL, STRESS, AND FLUENCE GRADIENTS TEND TO FAVOR INITIATION OF SHALLOW FLAWS



"BEST ESTIMATE" $\Phi(F) \leq 5 \times 10^{-6}$ at 32 EFPY; SIGNIFICANTLY DIFFERENT FOR THREE PLANTS EVEN THOUGH RTNDT (2σ) ABOUT THE SAME

PLANT	RT_{NDT} (2σ) AT 32 EFPY (°F)	Φ(F) AT 32 EFPY (F/RY)
OCONEE-1	265	5×10^{-6}
CALVERT CLIFFS-1	252	7×10^{-8}
H. B. ROBINSON-2	135	$<10^{-11}$
HBR–HYPO	270	1.4×10^{-8}



DOMINANT TRANSIENTS DIFFERENT FOR THREE PLANTS

- EVENT FREQUENCIES AND BRANCH PROBABILITIES SENSITIVE TO PLANT DESIGN AND OPERATIONAL DETAIL
- VESSEL FLUENCES AND MATERIAL CHEMISTRY DIFFERENT
- CONCLUSION: GENERIC EVALUATION NOT ADEQUATE

DOMINANT TRANSIENTS[a]

Oconee[b]	Calvert Cliffs	HBY-HYPO[b]
Reactor trip[c] (42%)[d]	Small-break LOCA[e] (91%)	Reactor trip (88%)
Large SLB (14%)	Small SLB[e] (9%)	3 STM PORVs (36%)
Small-break LOCA (12%)	Residuals ($\Phi(E) < 10^{-4}$) (<1%)	2 STM PORVs (24%)
Excess MFW (9%)		≥3S DVs (15%)
Loss of MFW (9%)		Small-break LOCA (3%)
Inadvertent SI (5%)		Large SLB (2%)
SGTR (2%)		Small SLB (1%)
Small SLB (2%)		Residuals ($\Phi(E) < 10^{-7}$) (5%)
Residuals (5%)		

[a]Based on "best estimate" values of $\Phi(F)$ RTNDT (2σ) = 132°C (270°F).
[b]All transients, other than some residuals, from full power.
[c]Spurious trips followed by (1) excessive steam flow (TBVs and/or PORVs stuck open) and/or (2) excessive feed water flow.
[d]Percentage of $\Phi(F)$.
[e]Hot zero power.

UNCERTAINTY ANALYSIS PERFORMED TO ACCOUNT FOR PARAMETERS NOT SIMULATED IN "BEST ESTIMATE" ANALYSIS

- PTS TRANSIENT PROBABILITY
  - INITIATING-EVENT FREQUENCY
  - BRANCH PROBABILITY
- THERMAL/HYDRAULICS
  - TEMPERATURE (COOLANT IN DOWNCOMER)
  - PRESSURE (PRIMARY SYSTEM)
  - RESPONSE SURFACE USED TO GENERATE IMPACT
- FRACTURE MECHANICS
  - FLAW DENSITY
  - $RTNDT_o$ (mean value)
  - $\Delta RTNDT$ (mean value)
  - K_{Ic} (mean value)
  - RESPONSE SURFACE USED TO GENERATE IMPACT

RESULTS OF UNCERTAINTY ANALYSIS

- FLAW DENSITY SINGLE LARGEST UNCERTAINTY
- $\Phi(F)$ (mean) >> $\Phi(F)$ ("best estimate")
- NRC SCREENING CRITERIA MAY NOT BE APPROPRIATE FOR ALL U.S. PWR PLANTS

PLANT	RTNDT (2σ) °C (°F)	$\Phi(F)$ "BEST ESTIMATE"	$\Phi(F)$ MEAN
Oconee-1	132 (270)	6×10^{-6}	$\sim 5 \times 10^{-5}$
Calvert Cliffs-1	132 (270)	1.7×10^{-7}	6×10^{-6}
H. B. Robinson-2	132 (270)	$<10^{-11}$	
HBR–HYPO	132 (270)	1.4×10^{-8}	8×10^{-6}

END

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