Operational Reactor Safety
22.091/22.903

Lecture 18

Pilgrim Nuclear Power Station

Background Information

Mark Santiago – Pilgrim Training

Welcome to Pilgrim Station and the Chiltonville Training Center

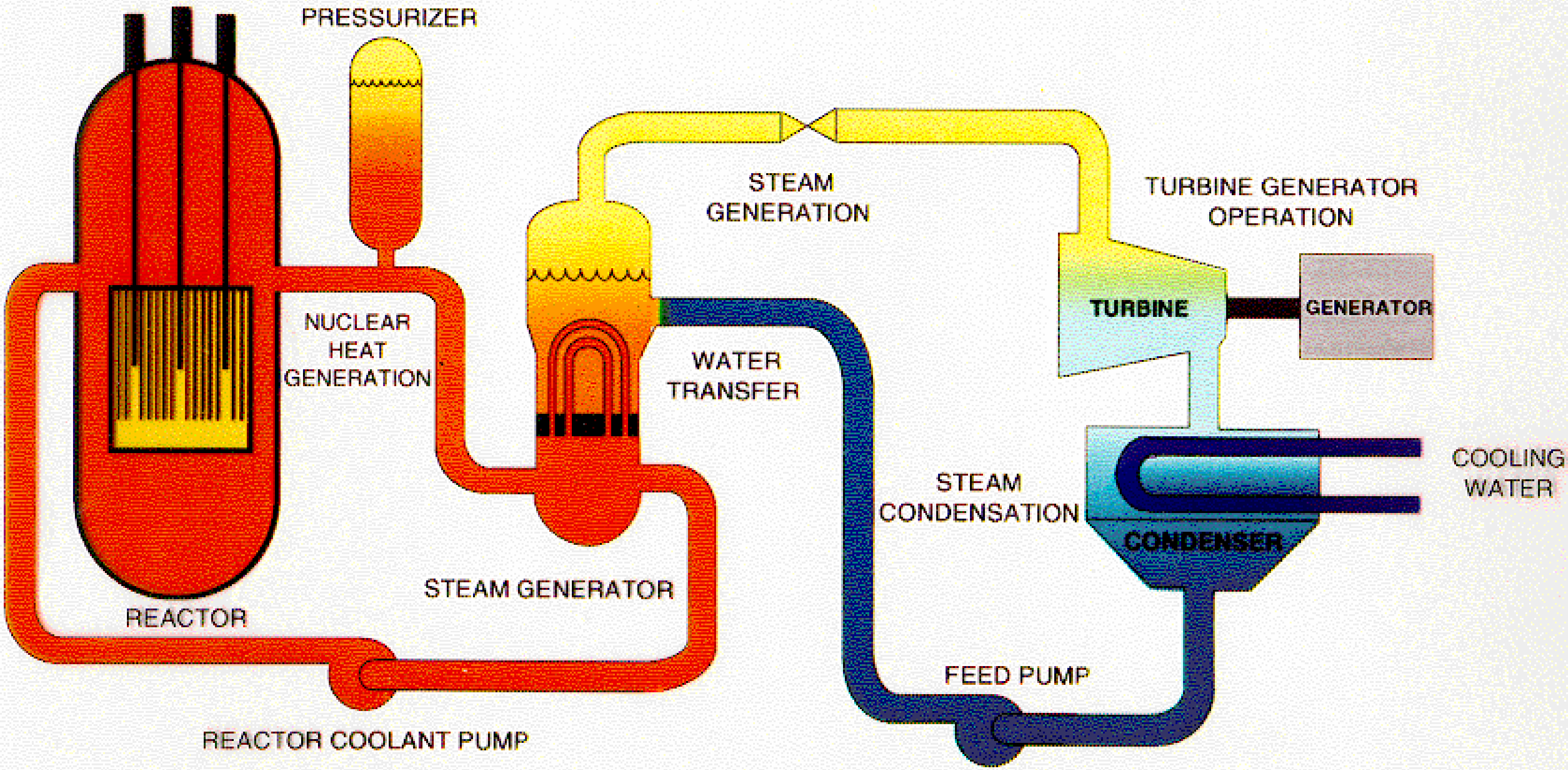
Pilgrim Station Facts

- 700 MWe Boiling Water Reactor
- 2028 MWth
- Nominal Operating pressure 1030 psig
- 145 control rods
- 580 fuel bundles
- Mark 1 Containment

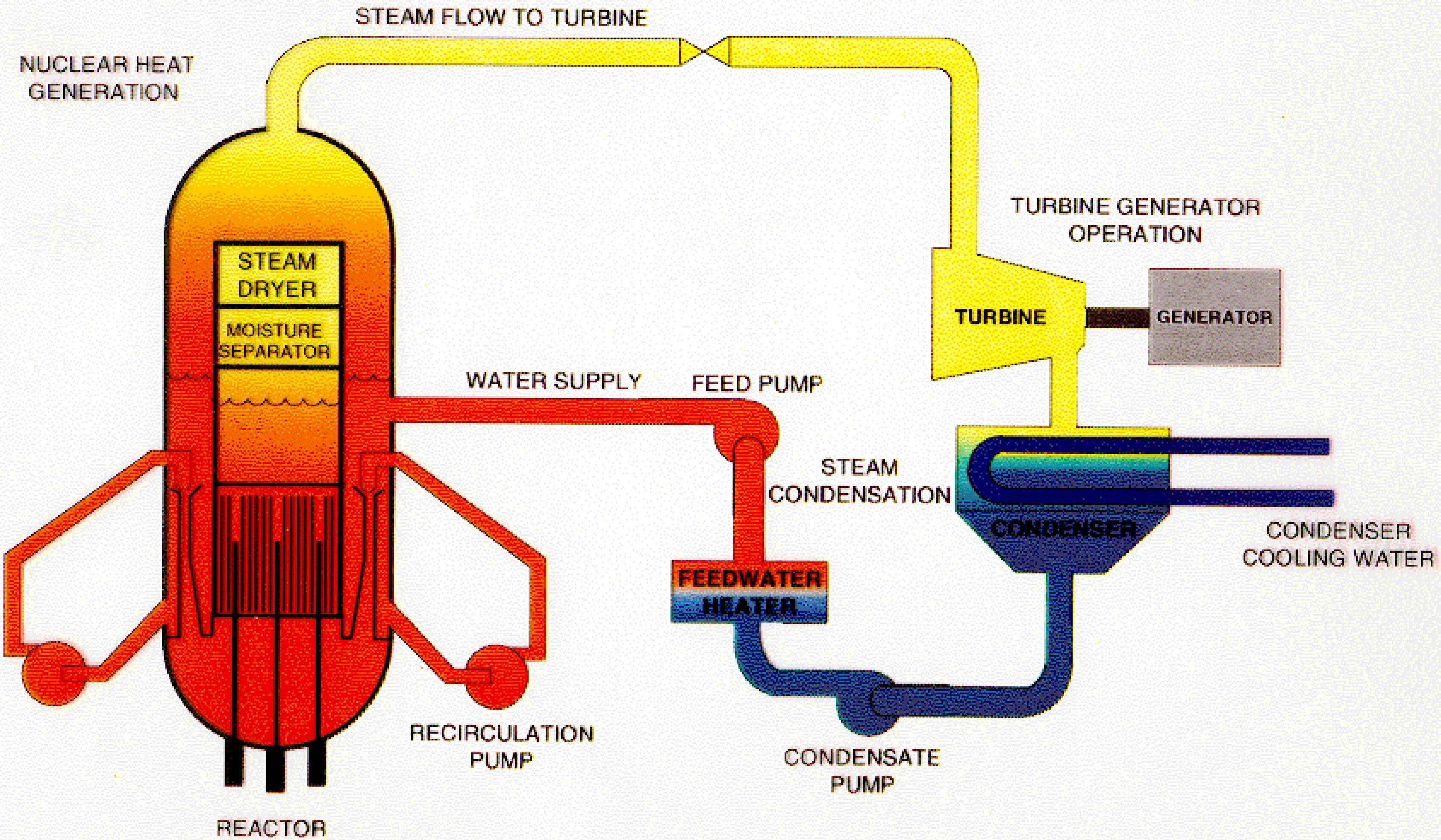
Pilgrim Station Facts

- Went commercial in 1972
- License will currently expire in Aug 2012
- Application for 20 Year license renewal submitted
  - Expect approval later in 2008.
- Operates on a 24 month refueling cycle
- Owned by Entergy
- Part of a twelve unit nuclear fleet
- Entergy has filed for a permit for new BWR in Mississippi.

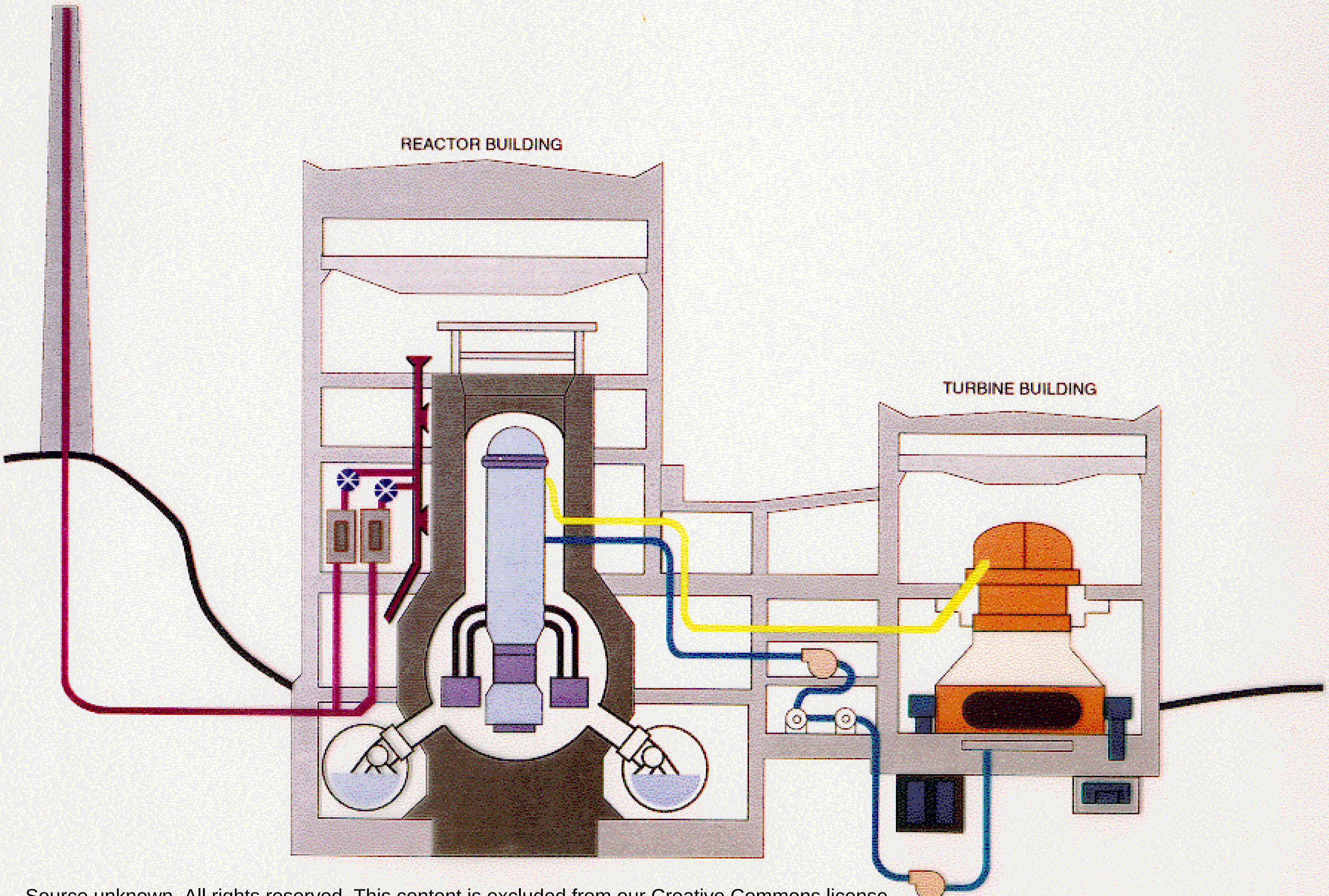
Pressurized Water Reactor

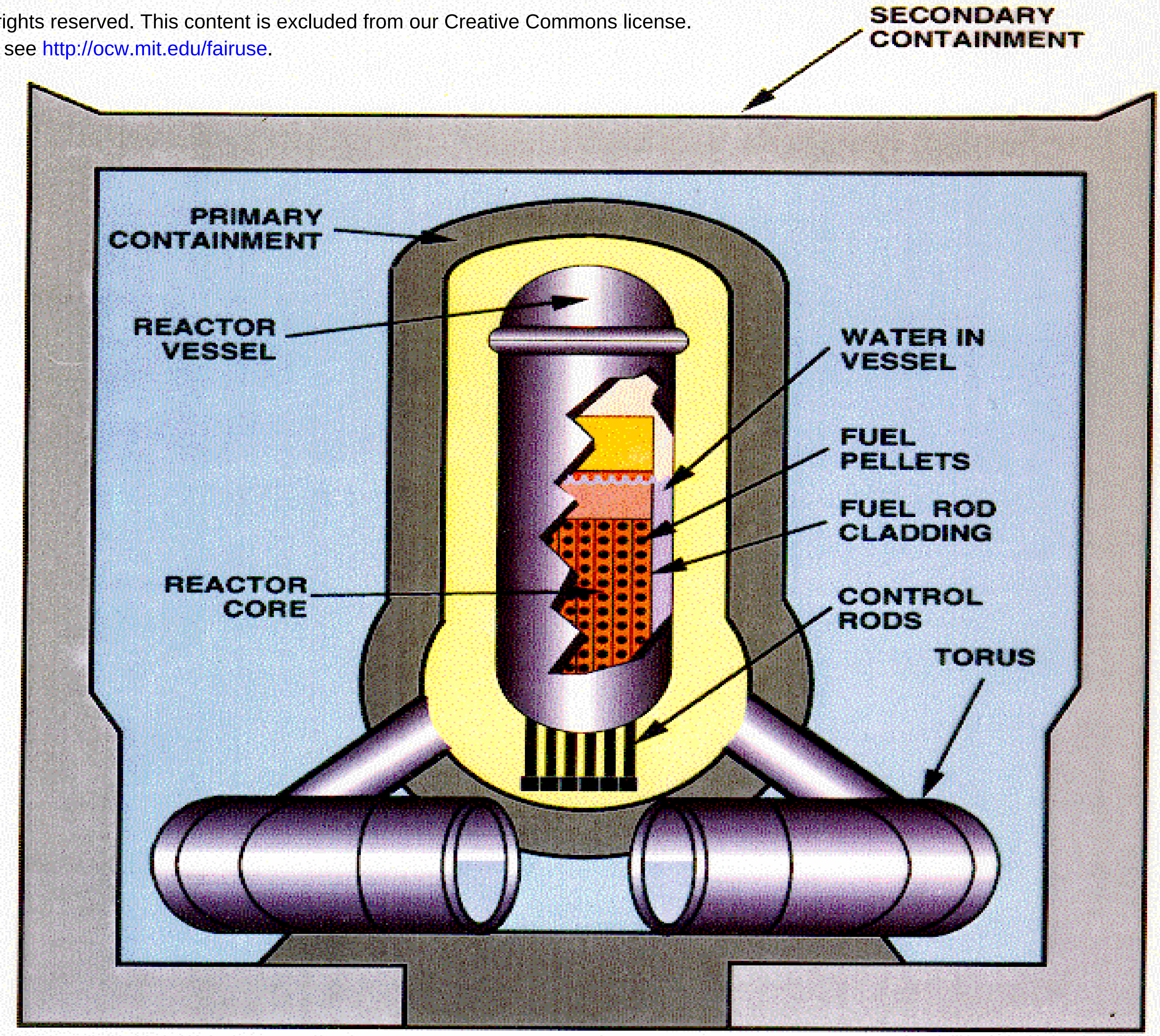


Boiling Water Reactor



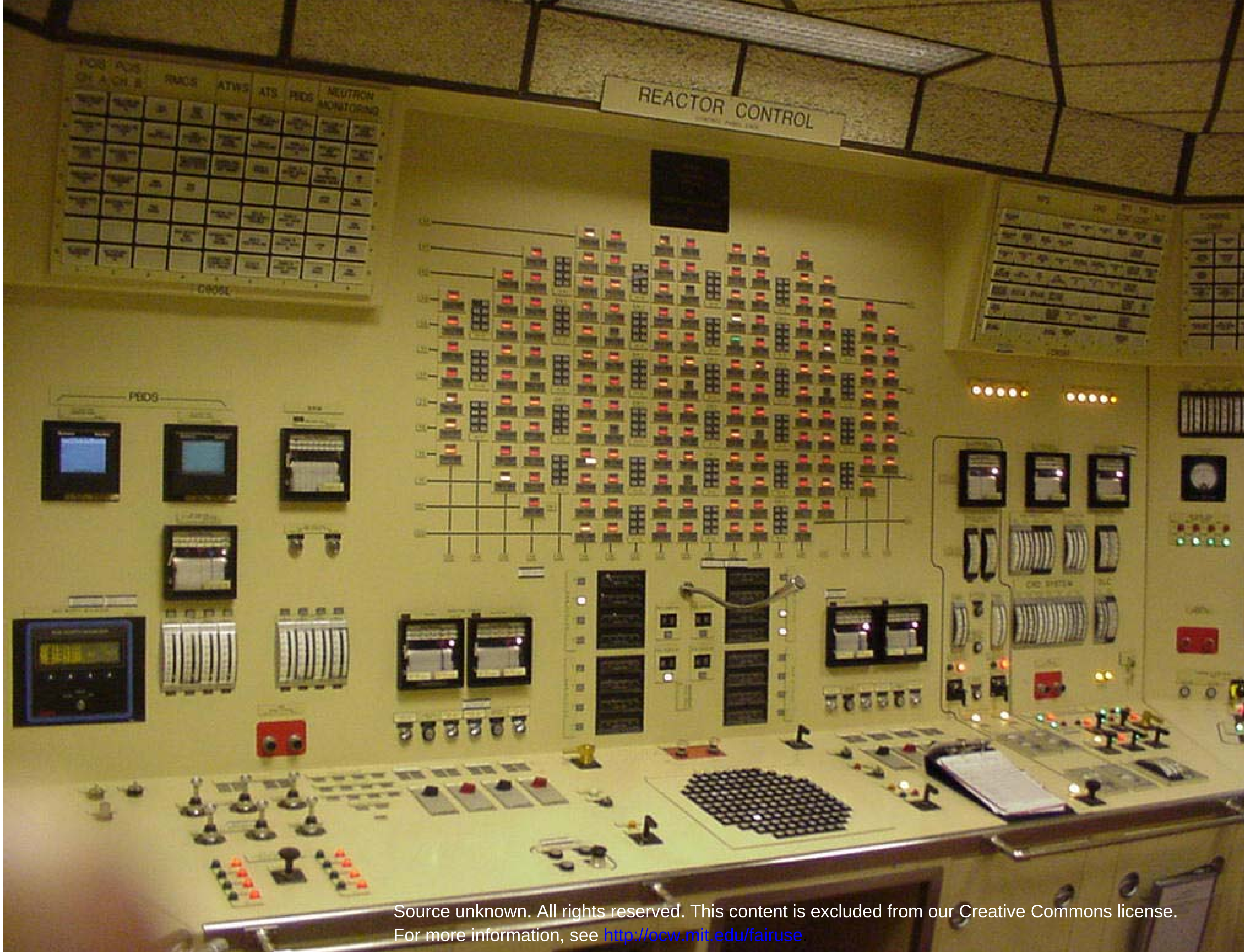
Reactor and Turbine Buildings



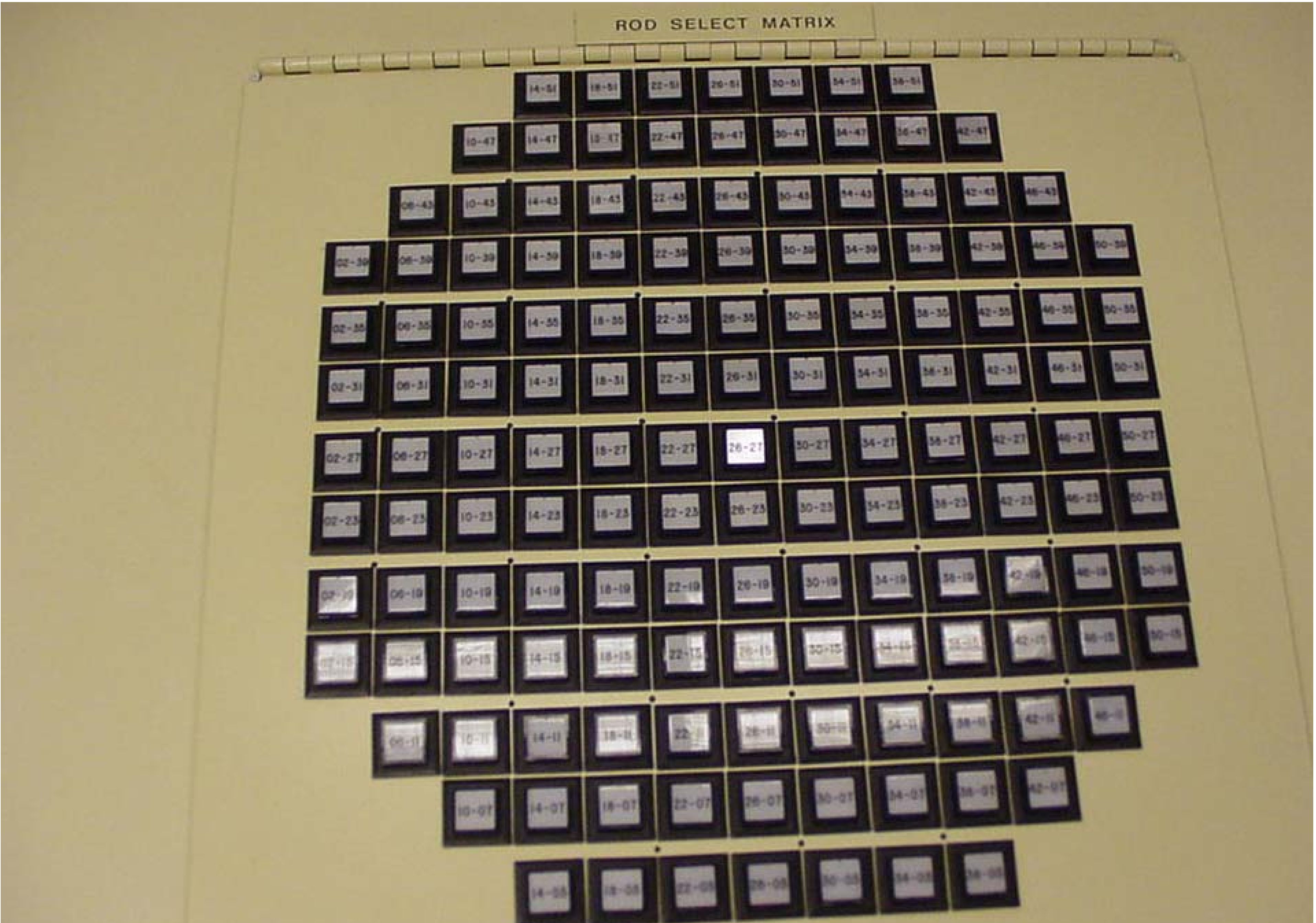

SECONDARY CONTAINMENT
PRIMARY CONTAINMENT
REACTOR VESSEL
WATER IN VESSEL
FUEL PELLETS
FUEL ROD CLADDING
REACTOR CORE
CONTROL RODS
TORUS

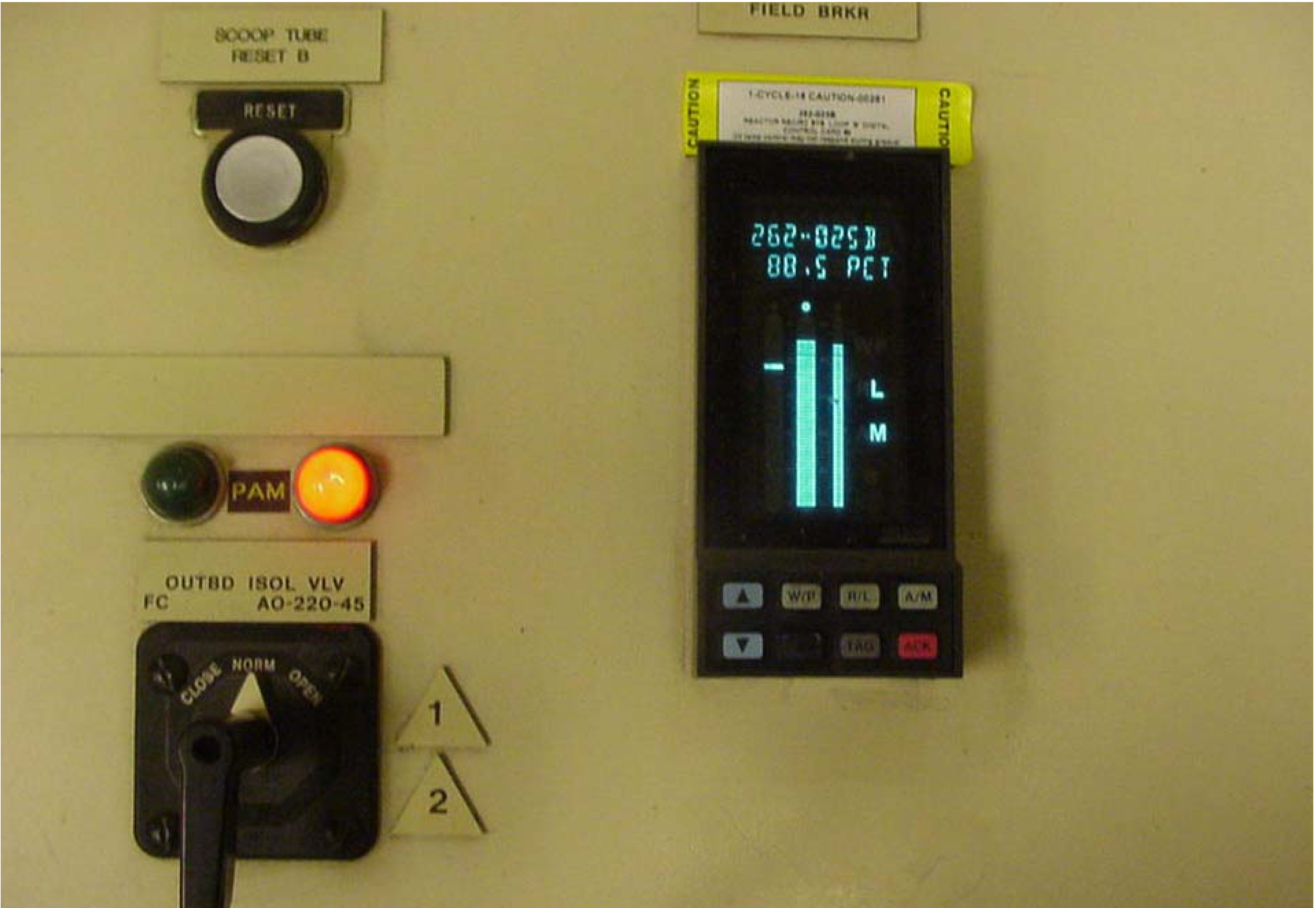
Reactivity Coefficients

- Void Coefficient
  (~ -1×10^{-3} Δk/k / % Voids)
- Moderator Temperature Coefficient
  (~ -1×10^{-4} Δk/k / ^{o}F)
- Doppler Coefficient
  (~ -1×10^{-5} Δk/k / ^{o}Fuel Temperature)

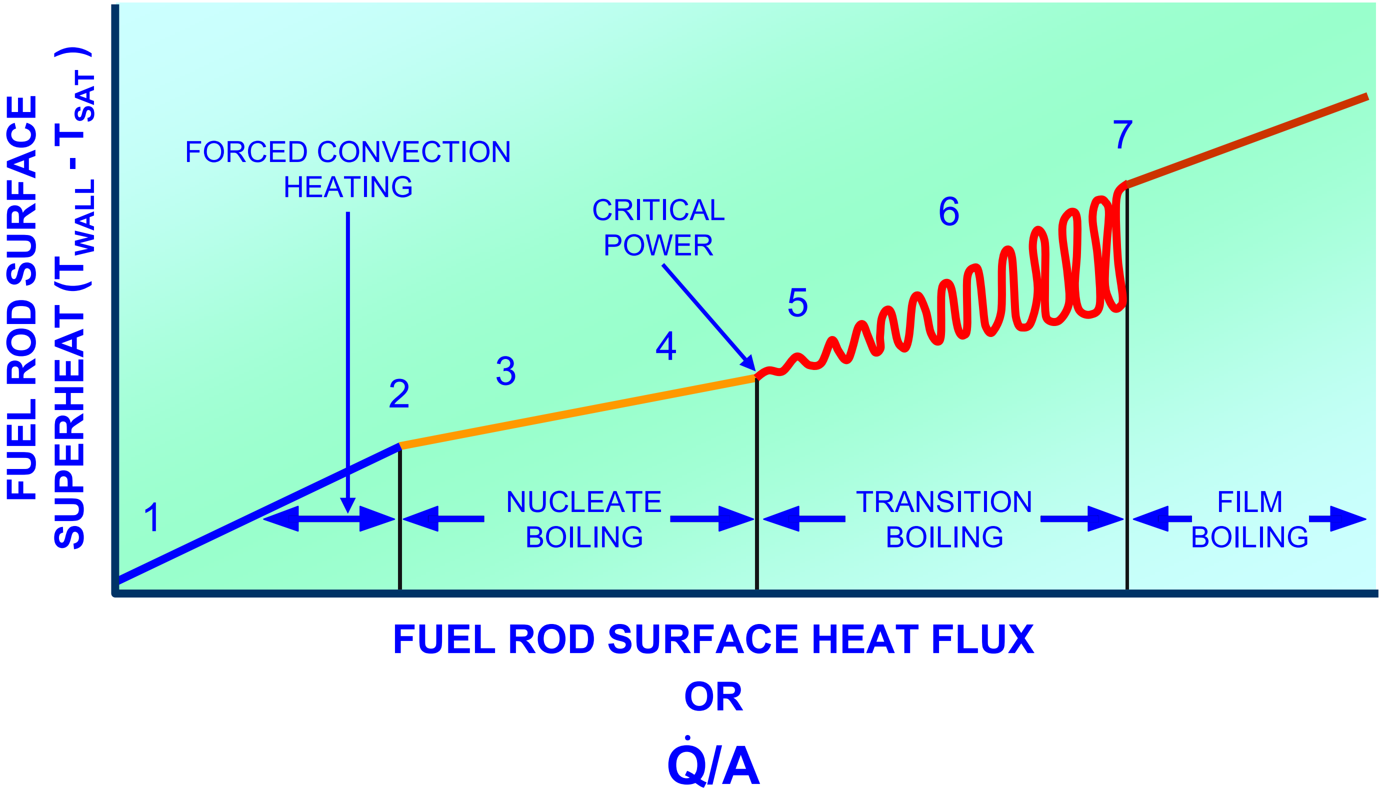

REACTOR CONTROL
PBDS
ATWS
ATS
NEUTRON MONITORING


51
47
43
39
35
31
12-45
IRM A
04-37
12-37


ROD SELECT MATRIX


SCOOP TUBE
RESET B
RESET
FIELD BRKR
CAUTION
262-025B
88.5 PCT
L
M
PAM
OUTBD ISOL VLV
FC
AO-220-45
CLOSE
NORM
OPEN
1
2
W/P
A/M

Critical Power



Critical Power Ratio (CPR)

$$CPR = \frac{CP}{AP} > 1.0$$

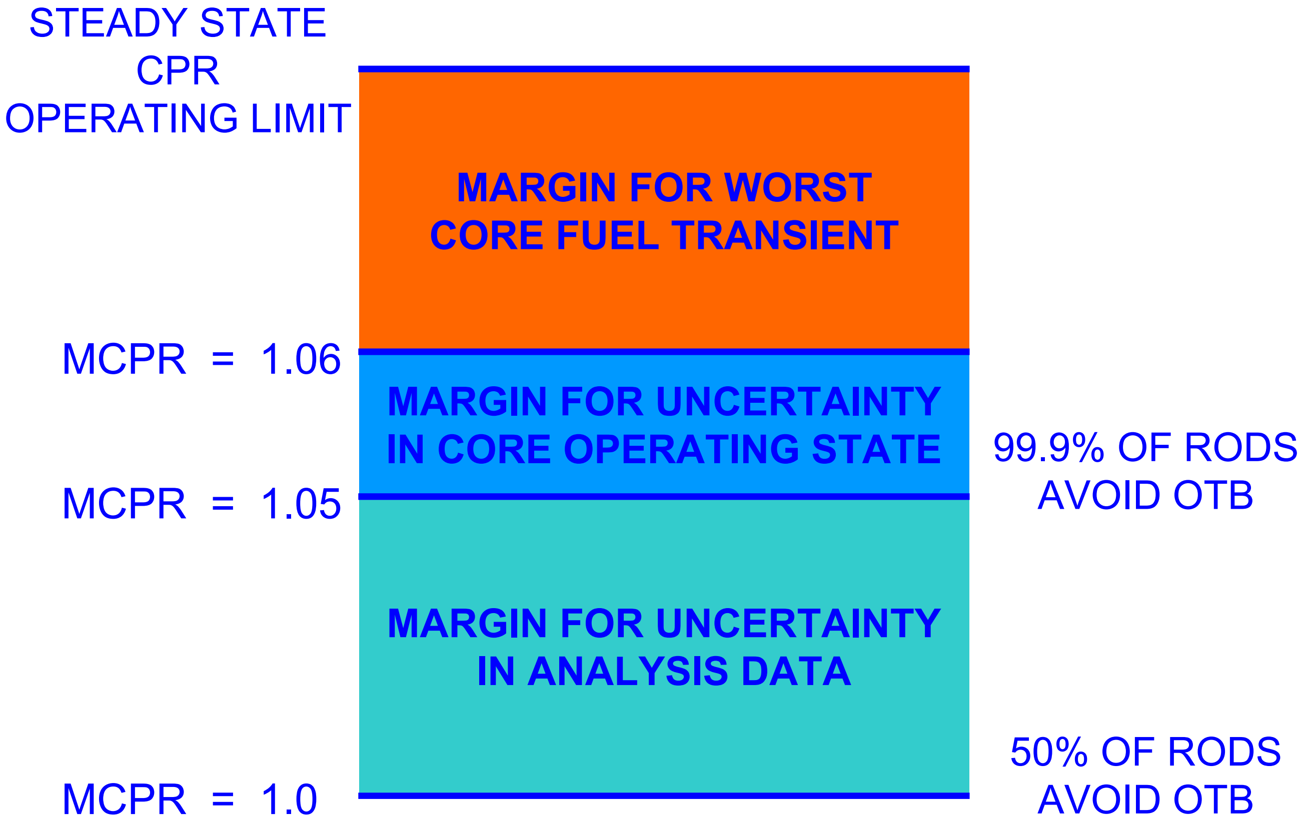
Where:

CPR = critical power ratio

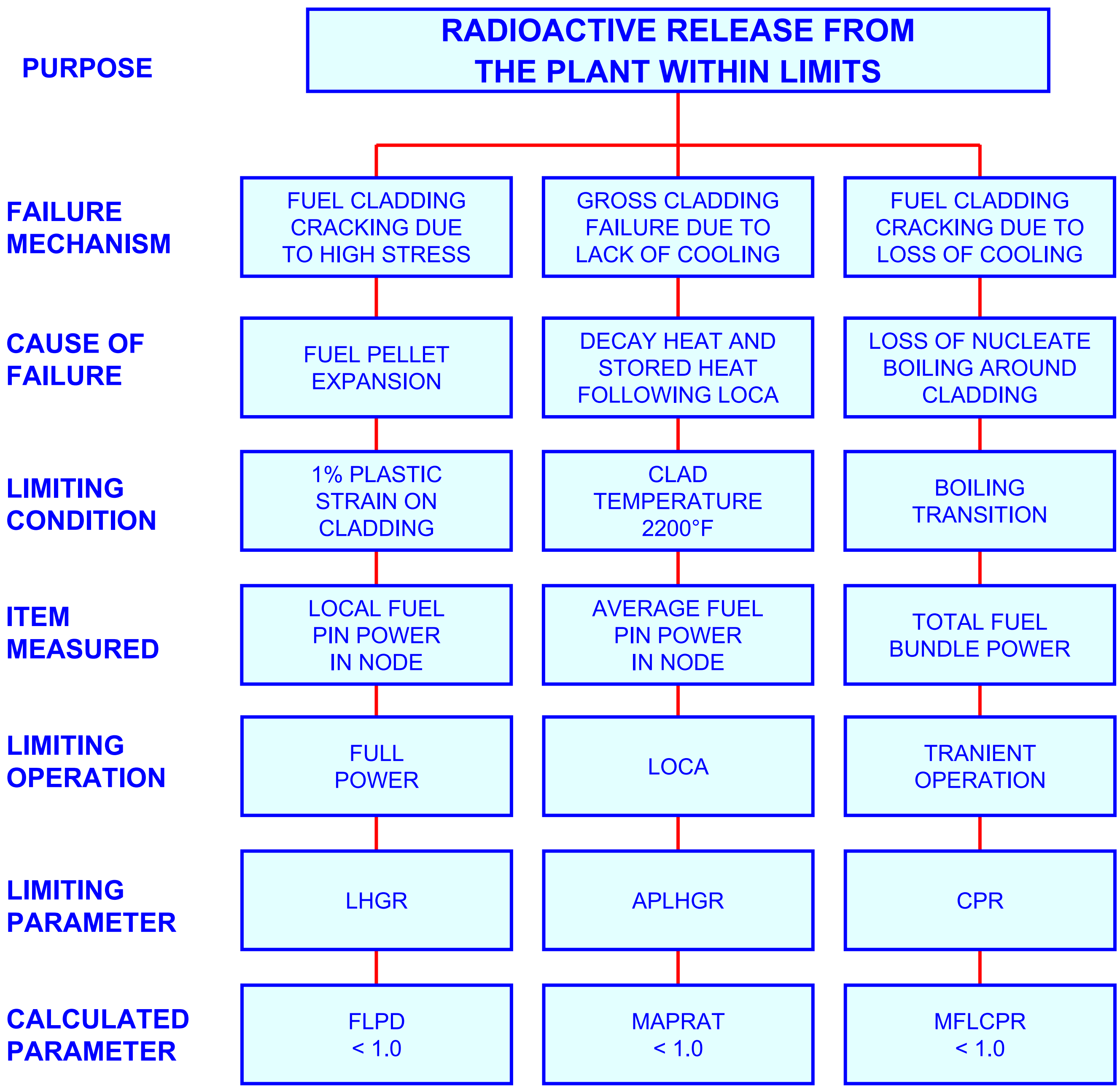
CP = bundle power at which OTB occurs

AP = actual bundle power

MCPR LIMITS



STEADY STATE
CPR
OPERATING LIMIT
MARGIN FOR WORST
CORE FUEL TRANSIENT
MCPR = 1.06
MARGIN FOR UNCERTAINTY
IN CORE OPERATING STATE
99.9% OF RODS
AVOID OTB
MCPR = 1.05
MARGIN FOR UNCERTAINTY
IN ANALYSIS DATA
50% OF RODS
AVOID OTB
MCPR = 1.0


PURPOSE
RADIOACTIVE RELEASE FROM THE PLANT WITHIN LIMITS
FAILURE MECHANISM
FUEL CLADDING CRACKING DUE TO HIGH STRESS
GROSS CLADDING FAILURE DUE TO LACK OF COOLING
FUEL CLADDING CRACKING DUE TO LOSS OF COOLING
CAUSE OF FAILURE
FUEL PELLET EXPANSION
DECAY HEAT AND STORED HEAT FOLLOWING LOCA
LOSS OF NUCLEATE BOILING AROUND CLADDING
LIMITING CONDITION
1% PLASTIC STRAIN ON CLADDING
CLAD TEMPERATURE 2200°F
BOILING TRANSITION
ITEM MEASURED
LOCAL FUEL PIN POWER IN NODE
AVERAGE FUEL PIN POWER IN NODE
TOTAL FUEL BUNDLE POWER
LIMITING OPERATION
FULL POWER
LOCA
TRANIENT OPERATION
LIMITING PARAMETER
LHGR
APLHGR
CPR
CALCULATED PARAMETER
FLPD < 1.0
MAPRAT < 1.0
MFLCPR < 1.0

PILGRIM CYCLE 17
3DM/P11
PERIODIC LOG
USER REQUEST
CALC RESULTS

SEQUENCE NO 2
6-MAR-2008 06:40 CALCULATED
6-MAR-2008 06:40 PRINTED
CASE ID FMLD1080306064009
RESTART FRFD1080306062454
LPRM SHAPE - FULL CORE

CORE PARAMETERS

POWER	MWT	2015.7
POWER	MWE	716.5
FLOW	MLB/HR	59.273
FPAPDR		0.656
SUBC	BTU/LB	27.58
PR	PSIa	1045.75
CORE	MWD/sT	23308.1
CYCLE	MWD/sT	3181.3
MCPR		1.712

Keff	1.0095
XE WORTH %	-2.15
XE/RATED	1.027
AVE VF	0.406
FLLLP	0.923

LOAD LINE SUMMARY

CORE POWER	99.4%
CORE FLOW	85.9%
LOAD LINE	110.5%

CORRECTION FACTORS: MFLCPR= 1.002 MFLPD= 0.996 MAPRAT= 0.996 ZBB= 3.19
OPTION: ARTS 2 LOOPS ON MANUAL FLOW MCPRLIM= 1.460 FCBB= N/A

MOST LIMITING LOCATIONS (NON-SYMMETRIC)

MFLCPR	LOC	MFLPD	LOC	MAPRAT	LOC	PCRAT	LOC
0.855	29-22	0.652	25-20-11	0.713	29-22-10	0.956	35-26-11
0.845	29-18	0.651	29-22-10	0.689	31-24-11	0.954	27-18-11
0.841	31-24	0.642	25-22-10	0.688	29-18-11	0.951	27-22-10
0.840	35-24	0.640	35-26-11	0.687	35-24-11	0.935	29-20-10
0.823	27-16	0.639	27-18-11	0.674	27-16-11	0.927	31-26-11
0.822	31-16	0.637	33-28-11	0.674	37-26-10	0.920	27-20-11
0.822	39-16	0.633	29-20-10	0.672	31-16-11	0.905	33-24-11
0.819	37-14	0.630	31-24-11	0.669	37-22-10	0.898	19-26-11
0.818	37-26	0.630	29-18-11	0.657	37-14-10	0.892	31-22-11
0.813	37-22	0.629	35-24-11	0.655	35-12-10	0.889	31-18-11

```
SEQ. A1        C=MFLCPR D=MFLPD M=MAPRAT P=PCRAT *=MULTIPLE
51

47
  L
43                          10

39
  L
35                  08              08

31
  L
27          10              08              10
                                      P
23
  L
19                  08    D     *       08

15
  L
11                          10

07
  L
03
      L       L       L       L       L       L
   02   06   10   14   18   22   26   30   34   38   42   46   50
```

CORE AVE AXIAL NOTCH	REL PW	LOC
00	0.140	24
02	0.249	23
04	0.764	22
06	0.886	21
08	0.986	20
10	1.013	19
12	1.029	18
14	1.061	17
16	1.026	16
18	1.070	15
20	1.254	14
22	1.313	13
24	1.388	12
26	1.434	11
28	1.419	10
30	1.379	09
32	1.332	08
34	1.243	07 <-
36	1.157	06
38	1.078	05
40	0.950	04
42	0.783	03
44	0.632	02
46	0.415	01

CORE AVERAGE RADIAL POWER DISTRIBUTION

RING #	1	2	3	4	5	6	7
REL PW	0.899	1.395	1.346	1.269	1.226	1.120	0.563

Abnormal Operational Transients

- Abnormal Operating Transients include the events following a single equipment malfunction or a single operator error that is reasonably expected during the course of planned operations.
- Power failures, pump trips, and rod withdrawal errors are typical of the single malfunctions or errors initiating the events in this category.

Reactor Limits

- To avoid the unacceptable safety results for abnormal operational transients, reactor operating limits are specified. Operating limits are specified to maintain adequate margin to the onset of boiling transition and failure due to cladding strain (CPR & LHGR). To ensure that adequate margin is maintained and an unacceptable result is avoided, a design requirement based on a statistical analysis was selected. This requirement would ensure that during an abnormal operational transient, 99.9% of the fuel rods would be expected to avoid boiling transition.

Abnormal Operational Transients Event Categories

- **Events Resulting in a Nuclear System Pressure Increase**
  - Ex: Turbine trip (turbine stop valve closure)
- **Events Resulting in a Reactor Vessel Water Temperature Decrease**
  - Ex: Inadvertent Pump Start
- **Events Resulting in a Positive Reactivity Insertion**
  - Ex: Continuous, inadvertent rod withdrawal

Abnormal Operational Transients Event Categories Cont.

- **Events Resulting in a Reactor Vessel Coolant Inventory Decrease**
  - Ex: Loss of feedwater flow
- **Events Resulting in a Core Coolant Flow Decrease**
  - Ex: Recirculation pump seizure
- **Events Resulting in a Core Coolant Flow Increase**
  - Ex: Recirculation Flow Control Failure - Increasing Flow

DESIGN BASIS ACCIDENTS

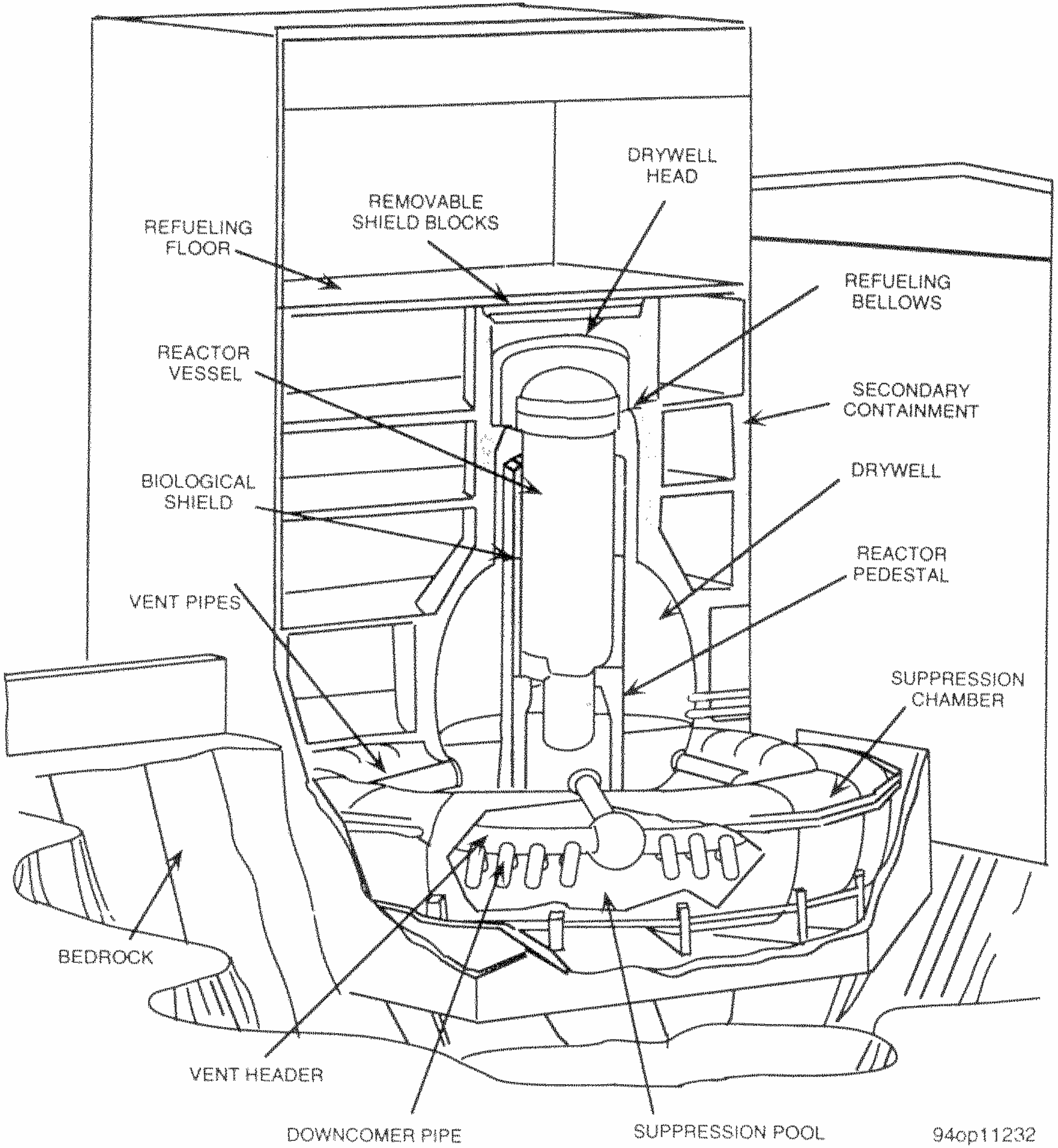
- A design basis accident is a hypothesized accident; the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive materials barriers, and the restriction of radioactive material release from the barriers.

Unacceptable Results

- radioactive material release which results in dose consequences that exceeds the guideline values of 10CFR100
- failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited
- nuclear system stress in excess of those allowed for the accident classification by applicable codes
- containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required
- overexposure to radiation of station personnel in the control room

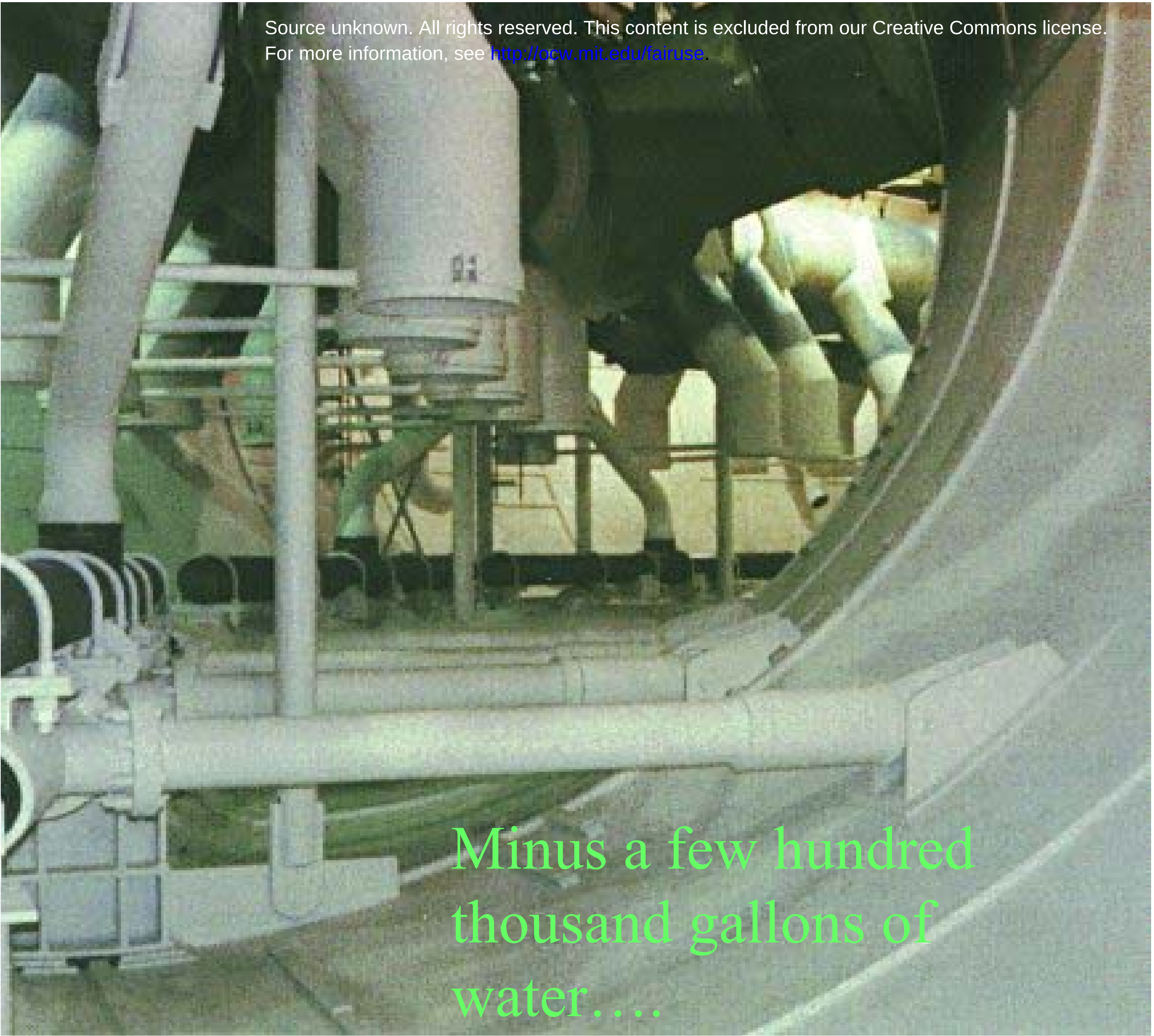
DESIGN BASIS ACCIDENTS

- **Control Rod Drop Accident**
- **Loss of Coolant Accident**
- **Main Steam Line Break Accident**



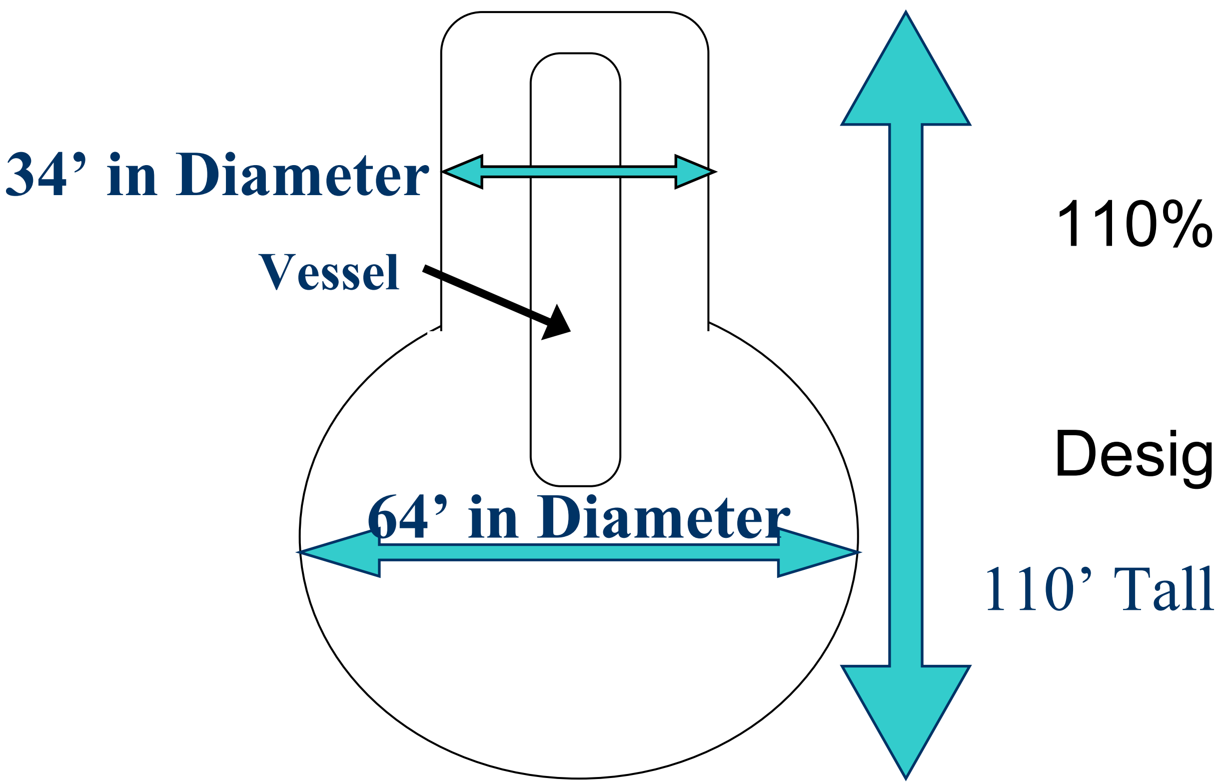
PRIMARY AND SECONDARY CONTAINMENT SYSTEMS
FIGURE 1 REV. 1

TORUS



Drywell

- Steel ASME Code pressure vessel
- Shaped like an inverted light bulb



Design Pressure

+ 56 psig

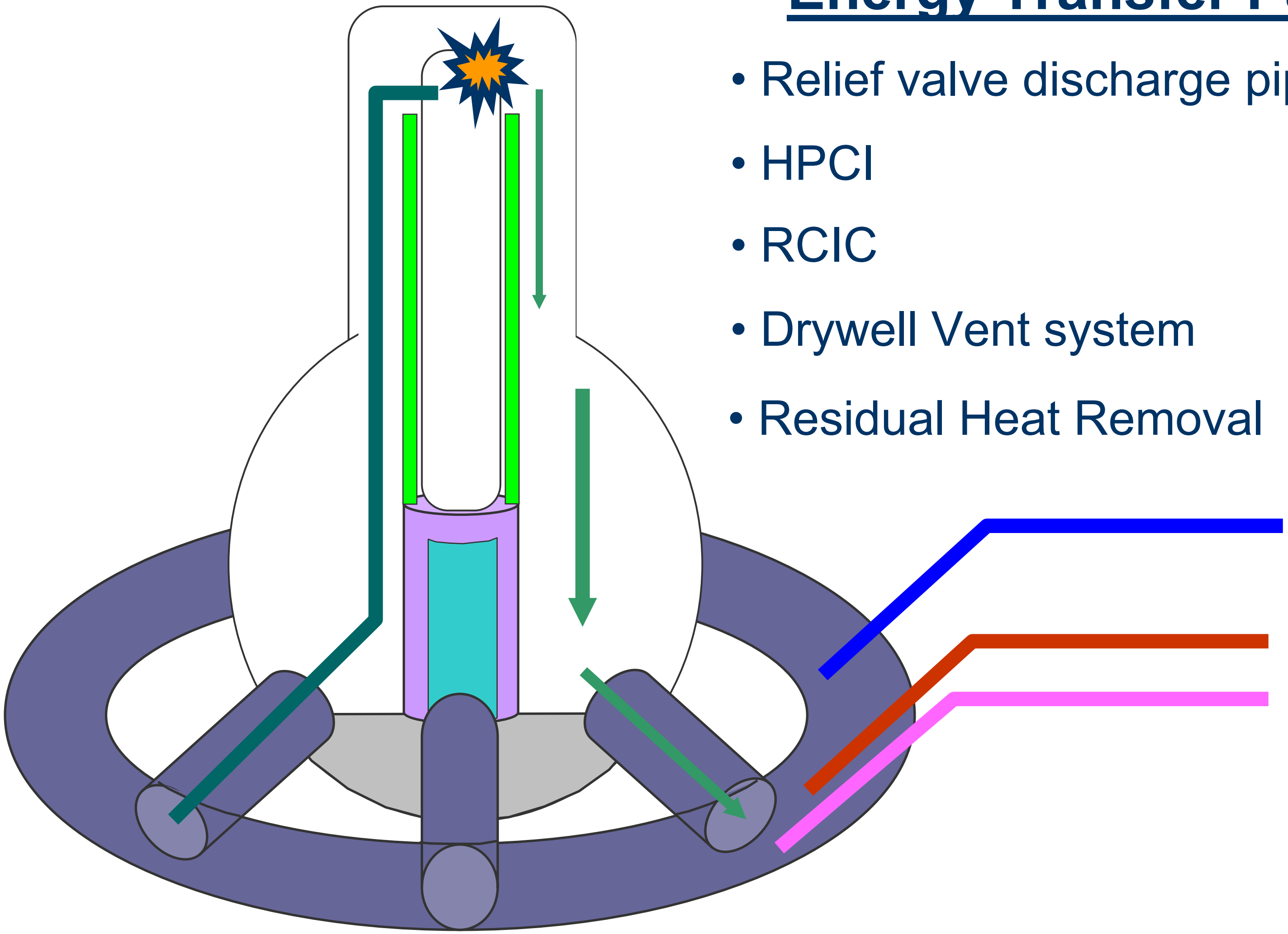
- 2 psig

110% overpressure

+ 62 psig

Design Temp (Tsat)
281°F.

Pressure Suppression Chamber and Pool



Energy Transfer Paths

- Relief valve discharge piping
- HPCI
- RCIC
- Drywell Vent system
- Residual Heat Removal

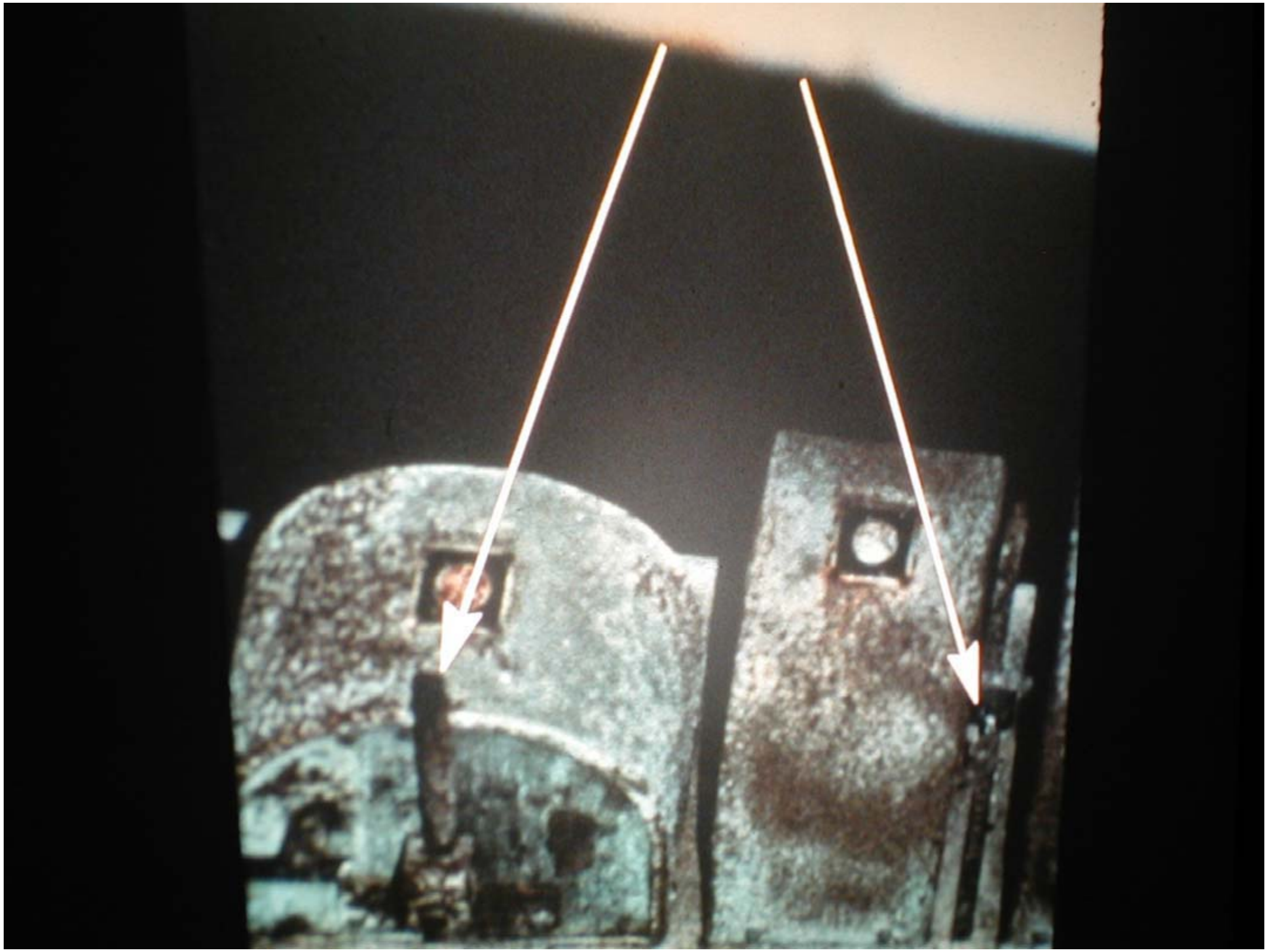
Hydrogen Event At TMI

- The first warning of the presence of hydrogen in the system was quite violent, but thanks to the heavily over engineered containment structure, it was almost anticlimactic save for its implications. A poorly shielded relay sparked, detonating the hydrogen in the containment. Containment building pressure zoomed to a frightening 28 pounds per square inch, and stayed there for nearly eight seconds as the hydrogen burned. The force shook the control room floor noticeably, and was thought to be equivalent to the explosion of several modern 1,000 pound bombs.

Hydrogen Combustion
The Burn

- Deflagrations are combustion waves which heat the gas by thermal conduction
- Travel Subsonically and cause low pressure loads on the containment

Hydrogen Event At TMI
Burnt crane controls



Hydrogen Combustion
The Boom

- Detonation heats the unburned gas by compression from shock waves.
- The waves travel supersonically and produce high pressure loads on the containment.

Containment Damage

- Hydrogen can create excessive drywell pressure
- Containment design pressure =
- 56 psi
- Estimated failure pressure =
- ~ 200 psi
- Estimated pressure with 30% metal-water reaction with a burn
- >> 200psi

Hydrogen Event At TMI
'Nuf said'

Drywell head

22.091 Nuclear Reactor Safety
Spring 2008