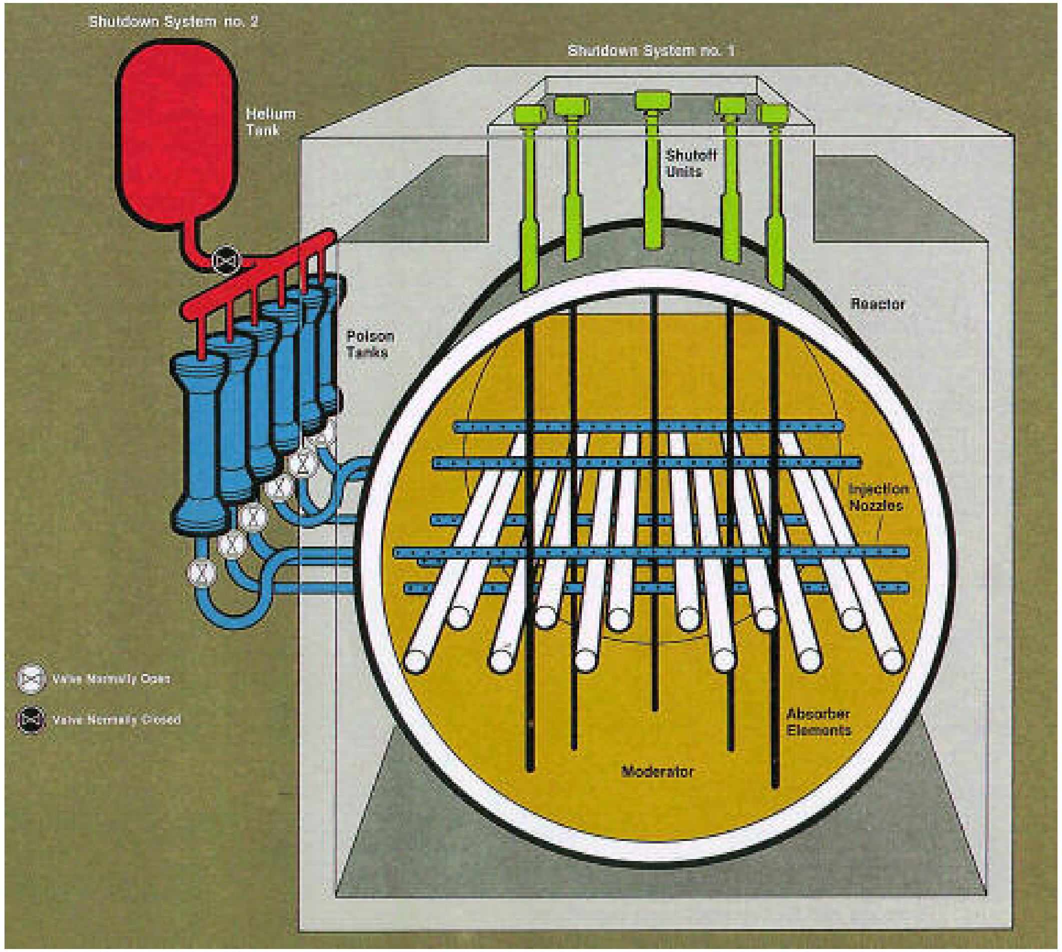
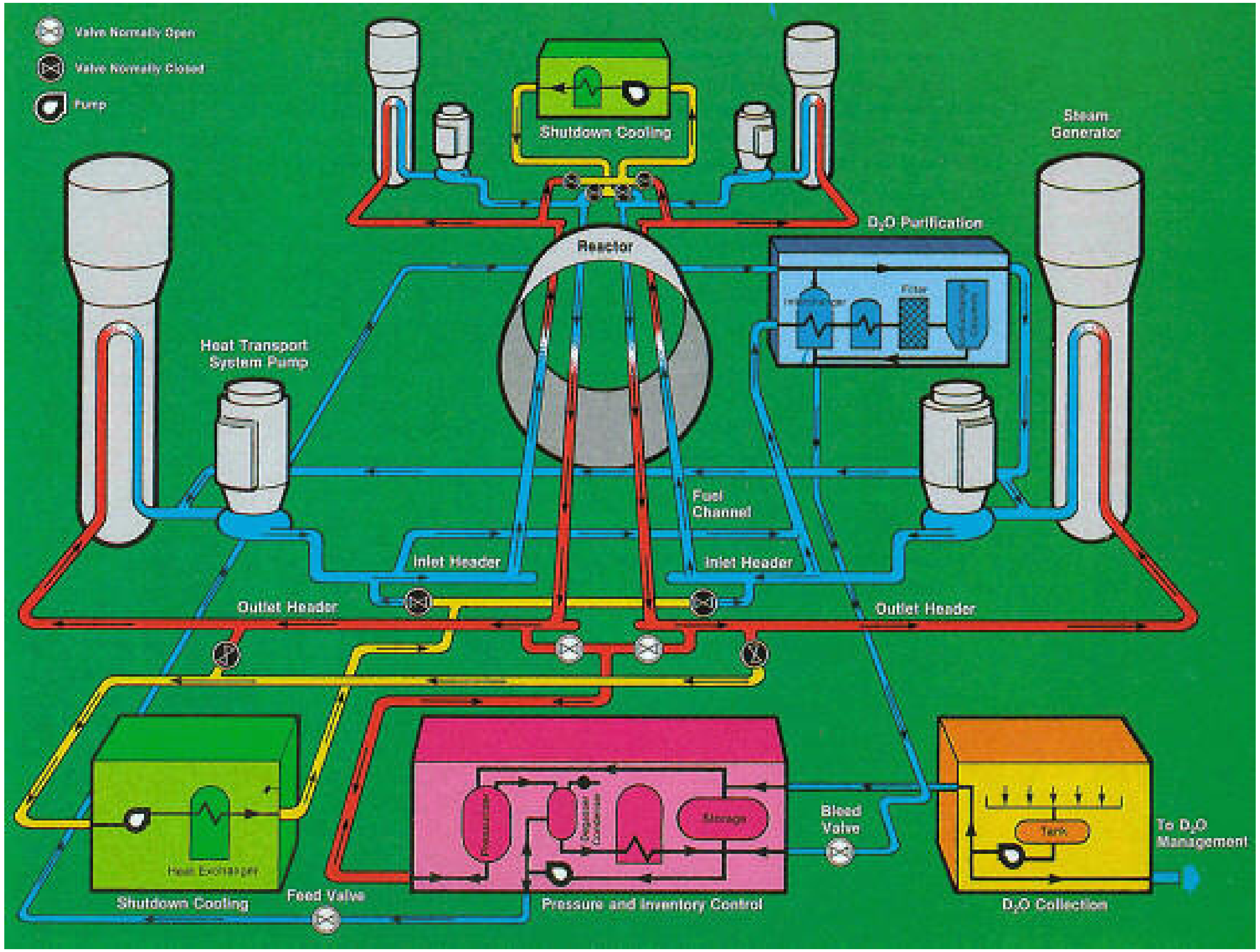
CANDU Shut down systems #1 and #2



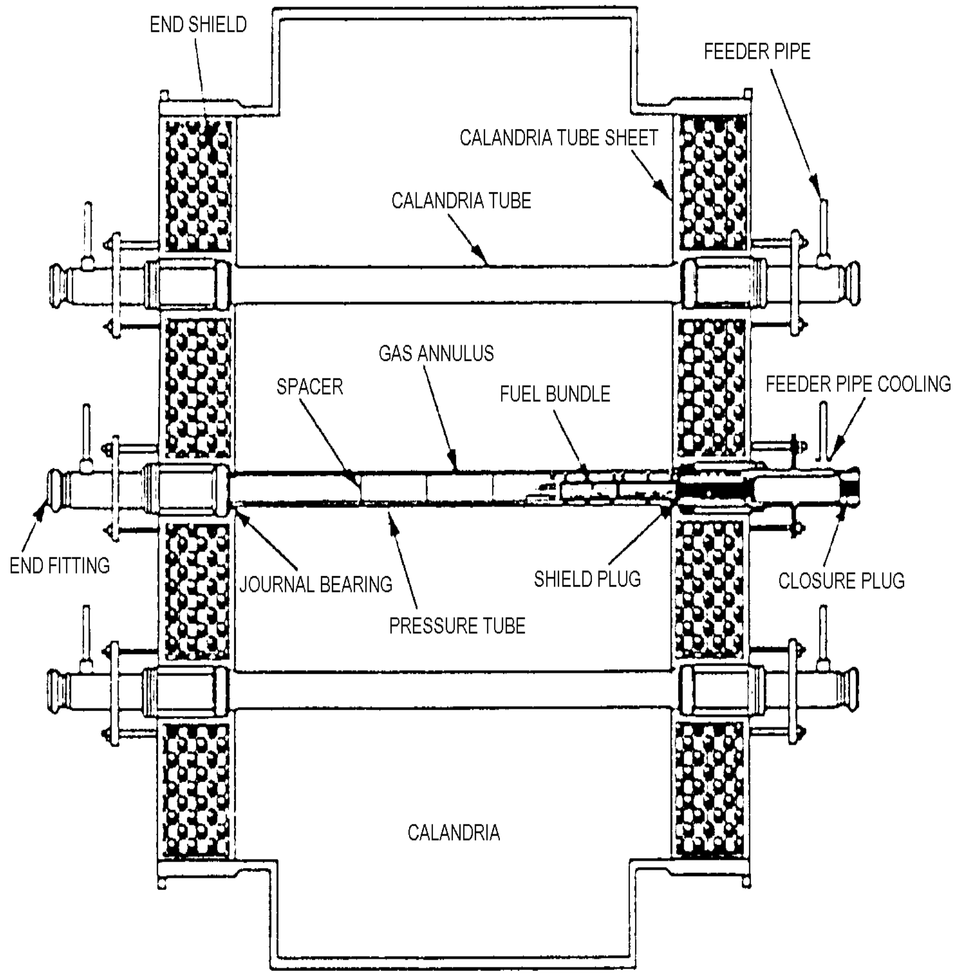
Decay Heat Removal



Reactor Core Structure (1)

- The moderator and coolant are separated by two tubes (calandria and pressure tubes) with a donut shaped space (annulus) between them. These tubes pass through a large stainless steel tank, full of moderator D_2O, called **calandria.** The **pressure tubes** contain the fuel bundles, and the coolant is pumped through these tubes.

- The **calandria tubes**, which form part of the **calandria**, prevent the moderator from contacting the high temperature coolant. The **annulus gas** in the space between the tubes insulates the cool moderator from the hot pressure tube and is used to detect leaks in either of the tubes.



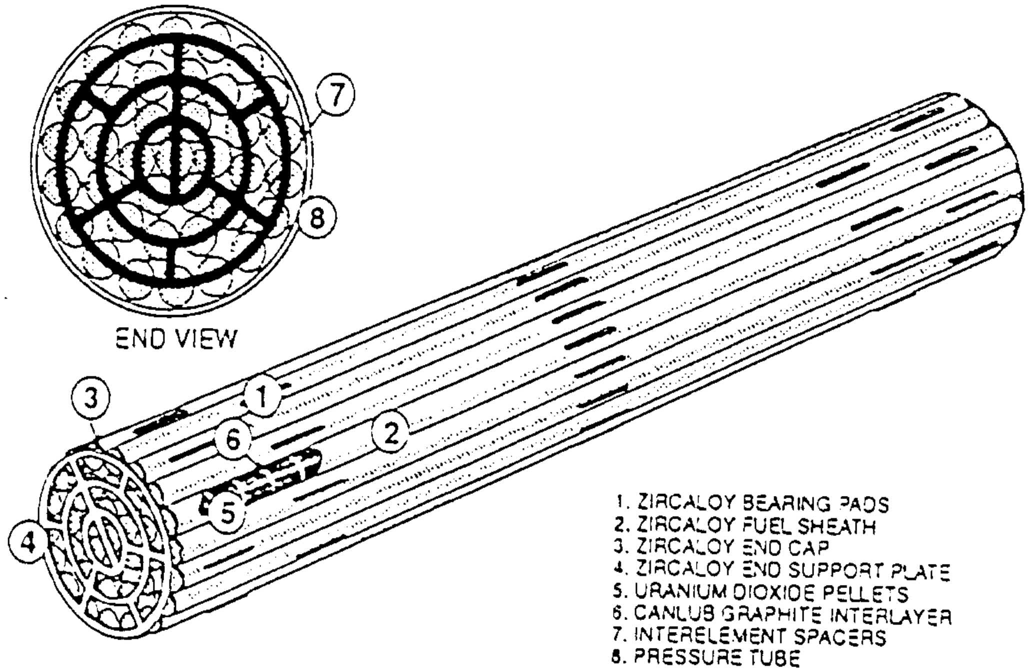
Reactor Core Structure (2)

- Cernavoda CANDU reactor has a steel lined concrete tank filled with light water that provides both thermal and radiation shielding.

- The moderator is separated from the coolant by adopting the pressure tube design. Pressure tubes running horizontally through the reactor contain the fuel. High pressure heavy water coolant passes through the pressure tube and over the fuel. This separation allows the moderator to be operated at a low temperature and pressure, avoiding the need for a large, heavy, expensive, high pressure vessel.

CANDU 6 Fuel Bundle

- Type: natural UO2
- Number of fuel bundles per channel:12
- Number of fuel pencils per fuel bundle: 37
- Fuel cladding: Zircalloy
- Fuel bundle weight: 25 Kg
- Fuel bundle length: 295 mm
- Enrichment: 0.714 U-235 (natural)



CANDU Single / dual failure concept

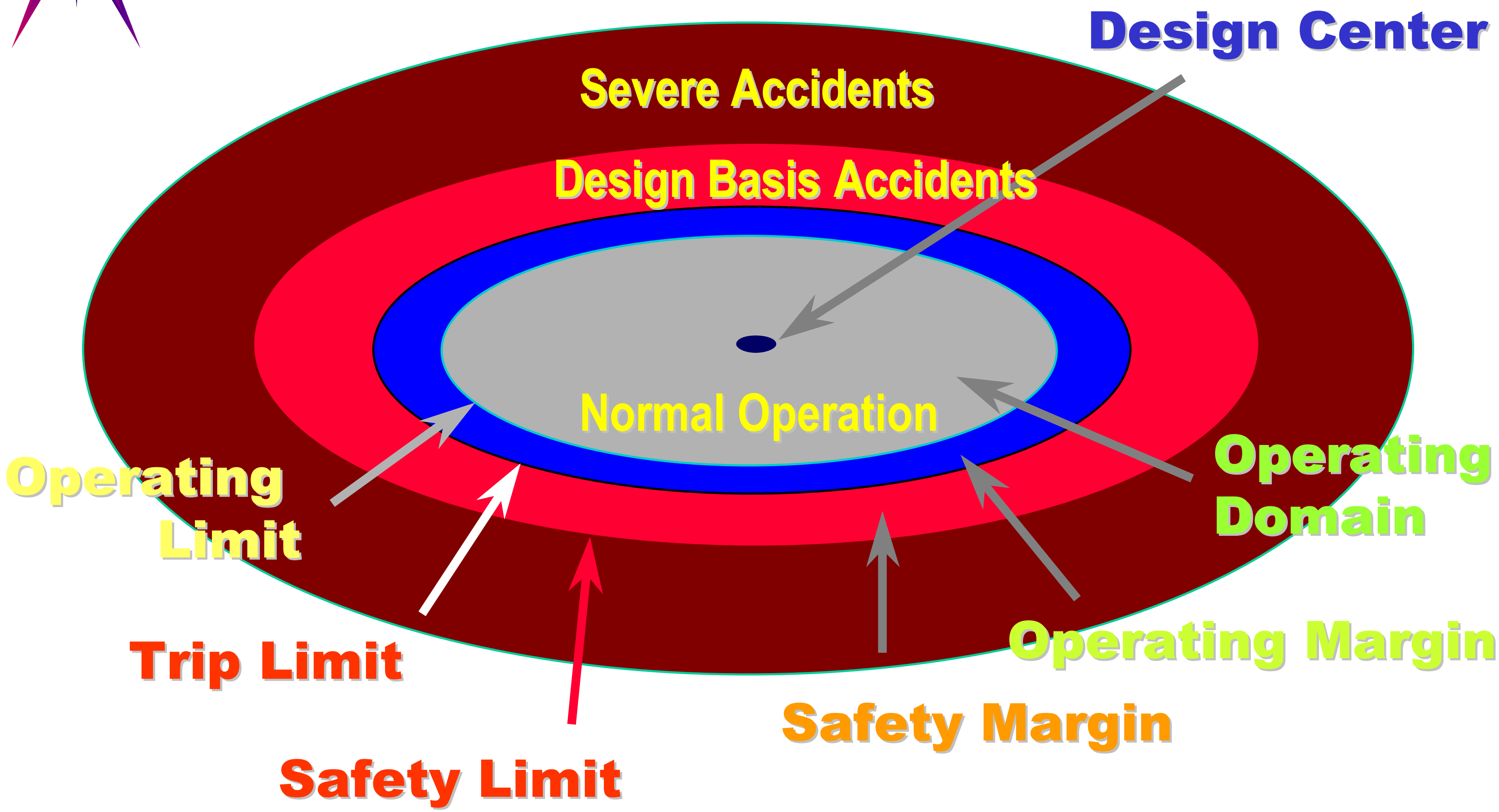
- **Single failure**: a failure of any process system which requires intervention of a Special Safety System
- Single failure criteria covers Anticipated Operational Occurrences (AOO's) and Design Basis Accidents (DBA).

- **Dual failure**: a simultaneously failure combination between a process system and a special safety system
- Double failure criteria covers Design Basis Accidents and Beyond Design Basis Accidents (BDBA).

According to plant design and operating procedures:

- main process system failure < 10^{-2} ev/year
- special safety system failure<10^{-3} ev/year

CANDU Defense in Depth concept



The Romanian legislation stipulates four licensing stages:

- **Site Licence** on the basis of an Initial Safety Analysis Report
- **Construction Licence** on the basis of a Preliminary Safety Analysis Report
- **Commissioning Licence** on the basis of the Final Safety Analysis Report - phase I
- **Operating Licence** on the basis of a Final Safety Analysis Report phase II, including two steps:

  - probationary operating licence
  - operating licence

- **Final Safety Analysis Report** (FSAR) is the main document on the base which Romanian Regulatory Body (CNCAN) issues the **operating and maintenance license .**

- The license is valid for two years. After this period a new review of FSAR (which include all modification and changes performed last two yeas) shall be submit to CNCAN, to issue a new license.

Structure of FSAR

- FSAR is structured according to Regulatory Guide 1.70:

  1. Introduction and general description of plant
  2. Site characteristics
  3. Buildings and structures
  4. Reactor
  5. Process Systems
  6. Safety Systems
  7. Unit control
  8. Electric Systems
  9. Auxiliary Systems

Structure of FSAR

10. Turbine/Electric generator and their auxiliaries
11. Radioactive waste management
12. Radiation protection
13. Conduct of operations
14. Commissioning program
15. **Safety Analysis**
16. Limiting condition for operation
17. Quality assurance during operation

In chapter 15 are presented the DBA considered for Cernavoda NPP Unit 1.

DBA considered are the following:

- large breaks in the primary circuit
- small breaks in the primary circuit
- pressure tube rupture
- channel flow blockage
- end fitting failure
- fuelling machine events while “on reactor”
- fuelling machine events while “off reactor”
- pipe breaks in Heat Transport System Auxiliary systems
- loss of class IV electric power
- heat transport pump seizure
- loss of reactivity control
- pressurization events -primary side

- depressurization events-secondary side
- feedwater line breaks
- steam main breaks
- steam generator tube rupture

Chapter 15.4 of FSAR ,Trip Coverage analyses the following situations:

- all Design Basis Accidents
- acceptance criteria for effectiveness of trip parameters
- coverage maps for each trip parameter
- from 0 to 100 % power range

The format used in describing each event is as follows:

1. Event evaluation
2. Event sequence and plant systems operations assumptions
3. Analysis methodology
4. Results
5. Summary and conclusions

Each initiating event is considered both as a single failure (i.e. Large LOCA, SBLOCA, MSLB) in case A of analyses but as dual failure, case B (Containment failures) and C (ECCS failures)

Computer codes used for Cernavoda 1 FSAR

In accident analysis for Safety Analysis Report for Cernavoda Unit 1 are involved 37 computer codes, in the following areas:

- **Reactor physics** (ALITRIG, INCON 2, PJET, RFSP)
- **Thermalhydraulics** (FIREBIRD 0.131, SOPHT, NUCIRC-MOD 1.603, ASSERT-IV, IBIF-THERMOSS-III)
- **Fuel behavior** (ELESIM-MOD 10, ELESTRESS, ELOCA-A, HOTSPOT-III, GENHTP)
- **Fuel channel** (CHAN –IIA, COBRA –IV, CONTACT-1, PDTFORM)
- **Moderator** (2DMOTH, PHOENICS, MODSBOIL, TUBRUPT)
- **Containment** (PRESCON2, GOTHIC, H2MIX, VENT)
- **Fission product release and dose calculation** (PEAR)

Domains covered

Deterministic SA is usually performed through the calculation of plant parameters with complex computer codes, solving a set of relevant mathematical equations describing a physical model of the plant.

Safety analysis presented by Cernavoda Safety Report are based on computer code sets which cover reactor physics, thermal-hydraulics, fuel, fuel channel, moderator, containment, atmospheric dispersion and dose calculations.

The codes are interconnected, the results of some codes being used as input by the other codes in order to fulfill the analysis purposes.

Acceptance criteria

They may:

- set numerical limits on the values of predicted parameters
- set requirements on plant states during and after an accident
- set performance requirements on systems
- set requirements on the need for, and ability to credit
- operator action.

Examples of general acceptance criteria for DBA

- Dose to the individuals and the public must remain bellow the defined values

- Fuel damage should be limited for each type of accident to ensure a coolable geometry

- The pressure in the coolant system should not cause a pressure - boundary failure in addition to the accident

- Systems required to mitigate the consequences of an accident should not be made ineffective because of conditions generated by accident (-containment should not be damaged in a LOCA to the extent that it can not perform its functions)

Examples of acceptance criteria for LOCA accident

- Doses to the most exposed individual is bellow the limit
- Channel geometry must remain coolable
  - the amount of fuel sheath oxidation must not embrittle the sheaths on rewet
  - the amount of sheath strain must be limited so that coolant can flow through the channel
- Channel integrity is maintained
- Pressure inside containment is bellow design pressure
- Pressure within containment compartments does not cause structural failures

Main steam line breaks - Examples of acceptance criteria

- -Doses to the most exposed individual is bellow the limit
- -The PHT integrity must be maintained (15 min /MCR ; 30 min/SCA)
- -Systematic fuel failures are prevented (no prolonged periods of dryout/stratified flow before trip)
- -Containment pressure must remain bellow the cracking value

The dose limits for accident condition

	SINGLE FAILURES		DUAL FAILURES	
	Efective	Thyroid	Efective	Thyroid
Individual [Sv]	$5 \cdot 10^{-3}$	$3 \cdot 10^{-2}$	0.25	2.5
Population [man-Sv]	10^2	10^2	10^4	10^4

Also, are defined an exclusion area of such size that an individual located at any Point on its boundary for 2 hours immediately following onset of the postulated fission product release would not receive a total radiation dose on the whole in excess of 0.25 Sv or a total radiation dose in excess of 1 Sv to the thyroid from iodine exposure.

A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 0.25 Sv or a total radiation dose in excess of 1 Sv to the thyroid from iodine exposure. Also the collective dose estimated in these conditions for 22.5 degree radial sector should not exceed 10^4 man-Sv.