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ELECTRICITY SUPPLY BOARD
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Your ref:

Our ref: EN/SFTY. DOCKET NUMBER PROPOSED RULE PR 50 45 FR 65474

30th December 1980.

The Secretary,
U.S. Nuclear Regulatory Commission,
WASHINGTON D.C. 20555.

Attention of: M.S. Medeiros. JR.,
Office of Standards Development.

Proposed Rulemaking on:
"CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REGULATION".

Dear Sir,

My attention has just been drawn to the advance notice of this proposed rulemaking, which appeared on page 65474 of the Federal Register of October 2 1980.

I would like to submit the following general comments on this topic.

(1) I feel that Safety Analysis Reports & Regulations should not be restricted to the rather arbitrary "Design Basis Accidents", but should encompass the complete range of degraded-core and core-melt scenarios.

(2) I feel that the off-site consequences of core-melt can be considerably mitigated by additional design features (e.g. filtered venting, hydrogen combustion control, etc) which, for the most part, lend themselves to retro-fitting in present reactor plants.

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(3) I feel that the probability of core-melt in new reactors can be drastically reduced by adopting the design philosophy and features described in the enclosed pre-print.

(4) I feel that the probability of core-melt in existing reactors can be significantly reduced by the fool-proof monitoring of reactor water-level, and the provision of highly reliable, diverse, and redundant, reactor feed (injection) systems. (Minimum redundancy of 4 X 50% or 3 X 100% for sensors, controls, pumps, pipework and electrical supplies).

I would like to submit comments on each of the 18 particular items which you have enumerated, but this will take a few days and I note that the official closing date for comments is 31st December. I will therefore limit this letter to the above general comments and enclose a pre-print of the paper "Towards a More Forgiving Reactor" which I presented to the IAEA Reactor Safety Symposium in Stockholm in October.

Since presenting the paper I have been pleased to learn that at least one European manufacturer has been thinking along similar lines. They feel that present-day reactors are far too demanding and, while they may prove sufficiently safe in advanced countries with high standards of education technical expertise and organising ability, they are totally unsuited for the less-developed and less-stable areas of the world. They have therefore produced a new design which is remarkably similar to that which I have suggested for a "forgiving" light water reactor.

So far as new plant is concerned, the enclosed pre-print contains most of the comments I would wish to make on the proposed rulemaking and describes the design goals which I feel to be both desirable and feasible for all new reactor plants.

Unfortunately, it gives little guidance on how to upgrade the safety of existing reactor designs, so I will prepare answers to your 18 point questionaire in the hope that, though received after the dead-line, it may still prove helpful at your rule-making activities.

Yours sincerely,

Christopher O'Farrelly

C.O'FARRELLY,
Nuclear Safety Engineer,
Project Department.

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES

Stockholm, 20—24 October 1980

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TOWARDS A MORE FORGIVING REACTOR

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ABSTRACT

In the wake of Three Mile Island the development trend may lead towards further engineered-safety-systems, accompanied by more complex and sophisticated information-processing and control technology, thus tending towards the computer-controlled power station. An alternative approach is to go back to basic reactor concepts, to maximise inherent safety features, and to study the feasibility of modifying present designs so as to create simpler and more "forgiving" reactors.

Assuming that the concept of "buying time" is a valuable safety goal, the paper investigates the "forgivingness" of present-day reactors.

A "forgiving" reactor is one which has large time margins and can thus tolerate operator negligence or auxiliary plant failures of moderate duration. By adding enough engineered safety features, the calculated risk of most reactors can be made equally low, inspite of having quite different inherent characteristics. A "forgiving" reactor, on the contrary, would maximise inherent safety characteristics and depend heavily on passive systems.

For the purpose of assessing the inherent response of different reactors, it is assumed that transients occur without "scram" (or operator intervention).

Residual Heat Production is one of the chief characteristics of nuclear reactors and denies them the "walk-away" capacity of conventional power stations.

An ideal reactor would be capable of handling the decay heat by passive means for an unlimited time, so that the reactor could be left permanently unattended without fear of fuel damage or fission product release. The combination of 0.3% power for passive cooling and 1000 full-power-seconds for the inherent heat sink, is suggested as the basis of a possible Residual Heat Design Goal and the feasibility of this goal is tentatively explored.

HTGR cooling is examined and certain changes proposed where these would lead to more passive methods of heat dissipation and greater inherent heat sink capacity.

LWR cooling is discussed in a wide ranging perspective with a view to increasing the passive heat dissipation. A large inherent water heat sink is proposed and, to combat any loss of coolant, a method of blow-down + inherent refeed, to be triggered by a low water level signal.

Although the safety record of present-day reactors is excellent, the paper concludes that they are rather finely tuned, that improvements are feasible, and that the creation of a truly "forgiving" reactor should be a stimulating challenge for nuclear designers.

DISCLAIM

The long term design goals discussed in this paper represent the personal views of the author and are not related to any design work in his organisation.

UNITS

Unless stated otherwise in the text all quantities are expressed in Primary (i.e. non pre-fixed) S.I. Units.

MATHEMATICAL NOTATION

5E6 = 5×10^6 = 5 million; 5E-3 = 5×10^{-3} = 0.005.

1. INTRODUCTION:

The accidents at Windscale, Brown's Ferry and Three Mile Island have been characterised - not by the instant disaster of popular mythology - but by a long drawn-out agony, with operators struggling by the hour to reassert their control over a damaged or down-graded reactor.

Each of these accidents has highlighted a unique feature of nuclear reactors, the residual heat production, which endows the radioactive fission-products with an inherent mechanism for self-dispersion, and necessitates continuous heat-removal from the shutdown reactor. Should the heat-removal systems fail, then core-melt can occur at any time, even months after the reactor has been successfully shut-down.

Three Mile Island has shown up obvious deficiencies in education, training, regulation and organisation. It has also revealed some deficiencies of reactor design, especially in the areas of control, instrumentation and man-machine interfacing.

The resulting development trend may lead towards further engineered-safety-systems, accompanied by more complex and sophisticated information-processing and control technology, thus tending towards the computer-controlled power station.

An alternative approach is to go back to basic reactor concepts, to maximise inherent safety features, and to study the feasibility of modifying present designs so as to create simpler and more "forgiving" reactors.

1.1 Gaining Time

Given time, human beings are usually resourceful and imaginative in emergency situations and are often remarkably successful in solving problems, improvising equipment, and generally preventing an impending disaster (e.g. Windscale, Brown's Ferry, Apollo 13 and - with certain reservations - Three Mile Island). Without adequate time, human beings are generally useless in an emergency.

The principle of "buying time" is therefore an important safety concept for reactor operators.

A "forgiving" reactor is one which has large time margins prior to fuel or plant damage, and can thus tolerate operator negligence or auxiliary plant failures of moderate duration. This margin should be not less than one shift-length (8 hours) but should preferably exceed one day.

1.2 Reactor Design Philosophy

While present-day reactors are designed 'de jure' on a deterministic basis, they are, 'de facto', designed on a probabilistic basis, since the worst accident scenarios are not catered for. By the use of diverse systems, redundancy, and physical seperation, it is hoped to reduce the probability of these worst scenarios to an acceptably low level.

By adding enough engineered safety features, the calculated risk of most reactors can be made acceptably, and equally low, in spite of having quite different inherent characteristics. A "forgiving" reactor, on the contrary, would maximise inherent safety characteristics and depend heavily on passive systems.

The principal areas to be examined are:

(1) Inherent Transient Behaviour, including response to Loss-of-Coolant accidents (which are one of the chief safety issues for most power reactors).

(2) Residual Heat Dissipation.

2. INHERENT TRANSIENT BEHAVIOUR:

Normally reactors are protected against serious transients by a highly reliable protection system which ensures that the reactor is "scrammed" in time to prevent any adverse consequences. However, for the purpose of assessing the inherent response of different reactors, we will assume that transients occur without "scram" (or operator intervention).

The inherent core response to such transients depends on the specific power per unit mass of fuel (and its closely associated moderator) and on the following reactivity feedback coefficients.

(1) <u>Fuel Temperature Coefficient (αF)</u>

The increase in resonance capture of neutrons with U-238 temperature, known as the Doppler effect, is indeed a dispensation of nature. It provides a fundamental negative feed-back between fuel temperature and reactivity, and so leads to stable power reactors.

(2) <u>Moderator Temperature Coefficient (αM)</u>

In heterogeneous reactors, the moderator temperature may be less than that of the fuel, or following the fuel temperature with a long thermal time-lag. Variat[illegible] in moderator temperature cause changes in the mean velocity of ther[illegible]trons and, while this doesn't alter the fission rate for "1/V" [illegible] like U-235, it can have a significant effect for isotopes lik[illegible]39 which exhibit resonances in the thermal energy region. As a result, the moderator temperature coefficient can be either positive or negative depending on irradiation history, isotopic composition, and lattice constants.

If αM is positive, but numerically less than αF, the overall temperature coefficient ($\alpha = \alpha F + \alpha M$) will be negative and the core can still exhibit the complete stability required by an ideal "forgiving" reactor. If αM is positive and numerically greater than αF, the overall temperature coefficient will be positive and the reactor will exhibit long-term instability, necessitating either operator intervention or automated control. (The negative fuel coefficient will ensure short-term stability).

(3) <u>Coolant Density Coefficient</u>

Provided that the reactor is under-moderated, any decrease in coolant density (= increased voidage) will cause a reduction in reactivity. Since loss-of-coolant is a well-known accident scenario, a positive density coefficient (= negative void coefficient) is a sine qua non for any "forgiving" reactor. This requirement is generally met by all commercial water reactors (in the case of gas reactors the effect of this coefficient is negligeable).

In the special case of a 2-phase coolant (e.g. BWR) the collapse of coolant voids (due to a sudden pressure increase) would therefore initiate a reactivity excursion.

2.1 Basic Transient Responses

During each uncontrolled transient, the fuel temperature will tend to rise from its initial value (θo) but, thanks to the Doppler effect, there will be a negative reactivity feedback which will tend to stabilise the reactor. Theoretically the temperature should reach some equilibrium value but, due to thermal inertia, the actual peak temperature (θ max) is likely to overshoot this value by a significant amount.

If the actual peak temperature (θ max) is less than the critical temperature (θ cr) for fuel (or can) damage, then the inherent response to that particular transient is benign and no damage will result from scram failure.

For an ideal "forgiving" reactor, the temperature-rise-ratio

$$R = (\theta max - \theta o)/(\theta cr - \theta o) \quad 1 \text{ for all transients.}$$

While the potential initiating events and event-trees are numerous, the effects on the reactor core can be grouped into three basic categories:

(1) Reactivity excursions.

(2) Loss of coolant-flow.

(3) Loss of coolant.

2.1.1 Reactivity Excursions

(1) Maximum ("Slow") Reactivity Addition, e.g. all control rods and/or control poisons fully withdrawn from the cold core sufficiently slowly for the fuel temperature to rise in an equilibrium manner, i.e. the reactor is just critical at all times.

(2) Max. Single Item ("Instantaneous") Reactivity Addition, e.g. a single control element is ejected from the reactor core at full power, or, in the case of BWR, a sudden void collapse.

(3) Max Reactivity Addition Rate e.g. control rods and/or control poisons withdrawn, at maximum possible rate from the cold shutdown condition. If there is significant overshoot and the final temperature is very much greater than the equilibrium value calculated for item (1), it would suggest that the withdrawal rate is too great.

2.1.2 Loss-of-Coolant-Flow

This transient assumes that there is a total, instantaneous, loss of coolant flow at full power. The resultant fuel temperature rise will depend on the parameter (Po*l/Mc) where l = mean neutron lifetime and Mc = heat-capacity per ^{o}C of that portion of the core which is closely coupled to the fuel.

2.1.3 Loss-of-Coolant

This transient assumes that there is a total loss of coolant, at full power, due to a rapid depressurisation of the primary circuit. The transient temperature rise ratio (R) can only apply to reactors which, like the graphite reactors, have a permanent heat sink at close proximity to the fuel and so may be able to withstand a sudden loss of coolant without inevitable fuel damage.

It obviously can have no meaning in the case of reactors, such as LWR, which have practically no inherent heat sink other than the coolant, so that fuel temperatures would rise to self-destructive levels in small fractions of an hour. Such reactors must depend on engineered safety systems to prevent core melt.

2.2 Discussion

While it is envisaged that large LWRs should be capable of riding out a list of "anticipated transients" without scram, (Ref. 1) it is not clear that they could be designed to ride out all the basic transients of the previous sections in a passive manner.

The "SECURE" district heating reactor (Ref. 2) has an inherent shut-down system. A further clever feature ensures that boronated pool water is drawn into the core whenever a power/flow mismatch leads to abnormally high core outlet temperatures. Loss-of-coolant flow should be no problem, and loss-of-coolant impossible.

Turning to the HTGR, the core heat-up scenario for a 500MW (th) Pebble Bed Reactor (Ref. 3) features a total loss of all electric power without scram. It may therefore be possible to design certain types of HTGR to achieve R   1 for all the basic transients.

3. RESIDUAL HEAT REMOVAL:

Residual Heat Production is one of the chief characterisitcs of nuclear reactors and denies them the "walk-away" capacity of conventional power stations.

Following shut-down, and in the absence of all cooling systems, the reactor core must eventually melt but, with adequate time, there is every hope that some means of cooling could be improvised or, at the very least, an effective evacuation programme could be implemented.

Decay power and energy are shown in Fig. 1 and it appears that over 600 full-power-seconds (FP.s) of thermal energy must be removed from the fuel within the first day of shutdown. A "forgiving" reactor must achieve this without damaging the fuel or plant, by entirely passive means.

Fig. 2 shows the four traditional barriers to fission product release and indicates the effect of an unrestricted core heat-up. This scenario will be used to compare the inherent characteristics of reactors regarding residual heat removal. The later the fission products are released from the fuel, and the later they are released to the environment, the better.

Fig. 4 shows the estimated time history of some published heat-up scenarios, and gives an overall picture of the "forgiveness-capacity" of these reactors regarding residual heat removal.

An ideal reactor would be capable of handling the decay heat by passive means for an unlimited time, so that the reactor could be left permanently unattended without fear of fuel damage or fission product release.

Thermal convection and radiant heat transfer are passive means of cooling, and an inherent heat sink would be essential for such an ideal reactor.

For every value of passive cooling power which can be achieved, a corresponding value of inherent heat sink is required (Fig. 1). The combination of 0.3% power for passive cooling and 1000 FP.s for the inherent heat sink appear reasonable and will be used in this paper as a possible Residual Heat Design Goal. The feasibility of this goal will be tentatively explored in the following sections.

4. CONTAINMENT COOLING:

An equilibrium heat removal rate of 0.3% means that approximately 10MW must be removed from the containment of a 3000MW (th) reactor. The following methods of passive cooling suggest themselves.

4.1 Gravity-Feed Water-Cooling

This entails a lake or river at a higher level than the reactor and a permanent water supply of approximately 50 kg/s to the containment emergency heat exchanger(s). This water need not be wasted, on leaving the heat exchanger(s), and could be directed to a reservoir for various station requirements.

4.2 Natural Convection Water-Cooling

This entails a lake or ultimate heat sink, at a higher level than the reactor, and adequately sized insulated piping between the containment emergency heat exchangers and the lake, so as to maintain sufficient natural convection for 10MW of decay power.

4.3 Air Cooling of Containment Shell

In some ways this is the simplest and most fundamental method of dissipating the 10MW of decay heat.

Fig. 3-a shows an unshielded steel container of fairly typical dimensions, designed for an internal absolute pressure of 0.5MPa (75psia). Assuming the containment filled with saturated steam at this pressure, the steel shell should reach a temperature of almost 150°C. Under these circumstances radiation, alone, should easily dissipate 13MW of heat.

Unfortunately, from the heat dissipation point of view, it is now the normal practice to surround a steel containment with a massive concrete shield building designed to protect the reactor from various external effects such as gas-cloud explosions and certain types of air crash. So far as heat dissipation is concerned, the reactor has then been covered by something equivalent to a giant tea-cosy.

On the other hand it is only necessary to provide air ducts at top and bottom of the shield building to transform the interspace into a kind of natural draught chimney, as shown in Fig. 3-b.

The capacity for heat dissipation by natural convection depends on the steel surface area (S) the chimney height (Zc) and the chimney annular width (Yc). The heat dissipation is very sensitive to the relative height ratio (Zc/Yc) and, for a given surface and chimney height, there is an optimum chimney width for which the heat dissipation is maximum (See Fig. 3-d).

If the width is less than the optimum the outlet temperature approaches the surface temperature (Ts) but the mass flow is reduced resulting in a smaller power transfer.

If the width is greater than the optimum, the outlet temperature and power are both reduced. Fig. 3-b is based on a fairly typical width of 2 metres, and indicates a power transfer of approximately 4MW.

By means of light metal panels supported from the shield building, it should be possible to form an annular duct of optimum width around the steel containment as indicated in Fig. 3-c. Under these circumstances the power transfer should exceed 13MW, i.e. adequate to remove 0.3% of full reactor power.

5. HTGR COOLING:

The ceramic fuel particles are able to withstand very much higher temperatures than metal clad fuel elements. Each fuel particle is surrounded by its own mini containment of pyrolitic graphite and, according to Ref. 4, has to be heated to temperatures of 2500°C for some time before releasing its fission products.

In addition the graphite core retains its structural shape at temperatures exceeding 3500°C and so forms an inherent sink for residual heat absorption.

Fig. 4 shows the temperature and release-time history of two HTGR designs during unrestricted core heat-up.

5.1 Heat-Up Scenario (1)

This is the heat-up scenario for a 3000MW(th) Prismatic-Fuel HTGR as given in Ref 5.

A mean core temperature of 1260°C may be considered the upper limit at which forced circulation could be restored without damaging the steam generators and metallic parts of the primary circuit outside the core. This point of "no-return" is reached within a few hours.

At approximately the same time the mean gas temperature and pressure have risen to the level at which the safety valves open. It is assumed that, due to a high temperature, the valves stick open after a while and so depressurise the reactor.

Having absorbed about 400 FP.s of decay energy, the central core temperature reaches 2500°C in less than one day and begins to release fission products to the primary coolant and hence, via the open safety valve, to the containment. As large areas of the core heat up to 2500°C, fission products are continuously released over a period of days.

Provided the liner-cooling system operates correctly, it will be able to dissipate the residual power and stabilise the heat-up accident, without any damage to the PCRV concrete.

In the absence of PCRV liner cooling, the concrete temperatue rises and spalling occurs. This releases water and CO_2, both of which may react with the hot graphite to form CO and H_2, which then collect in the containment. After some days, the containment might be overpressurised by these gases, but it is more likely to fail as a result of a rapid combustion (or detonation) of the gases when they reach flammable concentration.

5.2 Heat-Up Scenario (2)

The scenario for a 500MW(th) pebble-bed HTGR, described in Ref 3, differs from the previous scenario in the fact that the maximum central core temperature peaks at approximately 2100°C, so that the bulk of fission products remain inside the fuel.

Hopefully these characteristics can be extended to the case of a 3000MW (th) power station plant.

5.3 Possible Methods of Improvement

The following suggestions would tend to make the prismatic HTGR closer to the ideal "forgiving" reactor, and are illustrated in Fig. 6 items 1 - 4.

5.3.1 Heat Sink (Item 1)

According to Figs. 2 and 4, central core temperatures reach 2500°C at a time when the energy absorbed by the core is only 450 FP.s.

Table I shows that the mean core heat capacity per °C for this HTGR design is very much smaller than it was in previous graphite reactors. Increasing this heat capacity (i.e. reducing W/m3 graphite) would enable the core to absorb 1000 FP.s without releasing fission products.

5.3.2 Liner Cooling (Item 2)

Present PCRV liner-cooling systems are designed to handle 0.3 - 0.6% of full power and, so long as they are functioning, there is no possibility of concrete spalling during a core heat-up accident, and hence no danger of overpressurising the containment.

Liner cooling depends on active systems with pumps maintaining outlet water temperatures of approx. 40°C but, for a core heat-up accident, it would be preferable to have an entirely passive cooling system. Concrete can withstand several hundred °C, so it should be possible, in emergency, to tolerate a water outlet temperature of 150°C and this should provide sufficient thermal head to permit 0.3% power transfer by natural circulation to the steel containment. The steel is externally cooled by natural draught (5).

5.3.3 PCRV Liner Insulation (Item 3)

According to the analysis of Ref. 5, the metal cover plates holding the mineral-wool insulation to the PCRV liner may be damaged during unrestricted core heat-up. Collapse of these cover plates would allow insulation to fall off, and so expose the liner to the full radiant heat flux from the core. While total removal of the insulation would be beneficial, its partial removal in one area may lead to local overheating, coolant dry-out and eventual collapse of the entire liner cooling system.

This mechanism for neutralizing the benefits of liner cooling is due to the discrete and discontinuous nature of the insulation. Insulation by means of refractory concrete would seem to avoid this potential for sudden, catastrophic, local overheating of the PCRV liner cooling system.

5.3.4 Natural Convection (Item 4)

Due to the height difference between reactor and steam generators, the earliest graphite gas-cooled reactors had excellent natural circulation, but this has been greatly diminished with the advent of prestressed concrete reactor vessels.

It would be most desirable to have at least 5% natural circulation at normal operating pressure, so that residual heat could be removed by the steam generators in the normal way without any need for operating gas circulators at shutdown.

If the reactor is depressurised by a factor of 10 the heat removal capacity by natural convection would still be 0.5% and (provided the core heat sink was 1000 FP.s) capable of stabilizing any unrestricted heat up accident. The potential depressurisation ratio should not be so great as to reduce the convective cooling capability below 0.3%.

5.3.5 Steam Generators

While there are obvious advantages with once-through steam generators, there is a lot to be said in favour of the larger time constants and simpler feed control-requirements of the drum type generators.

It is also desirable that all sections of the heat transfer tubes are designed to withstand the normal core outlet gas temperatures for a reasonable length of time.

5.3.6 Containment Overpressurisation

Since containment failure is due to the products of concrete disintegration, it might be possible to construct the PCRV from a different type of concrete with lower water content and no limestone.

The combustion (or detonation) of H_2 and CO is the most likely route to containment failure, and could be eliminated by inerting the containment atmosphere.

6. LWR COOLING:

There is little heat-sink capacity in present-day light-water power reactors and little possibility of passive heat removal from the primary circuit.

Fig. 4 - (3) shows the effects of unrestricted core heat-up of a PWR, based on the values in Table II. The safety valve having lifted, the core would uncover and fuel heat-up commence at t = 15E3 (≃ 4h). As the fuel heats-up, the cladding reacts with steam, releasing gaseous and volatile fission products which are shown passing to the containment via the safety/relief valve. Core uncovery and fuel melt-down should be complete by t = 20E3 (≃5½h) and a further period of an hour, or so, is required to boil-off the lower plenum water and melt through the pressure vessel.

Steam, hydrogen and CO_2 are given off as the basemet is attacked, and these may overpressurise the containment either directly, or as a result of rapid combustion (or detonation). According to WASH 1400 this might occur at a time ranging from 0.5 - 1 day, but according to Ref. 6 it shouldn't occur for 2½ days, (Containment breach 3 and 3a).

A fresh look at the inherent features of the LWR would be required if there is to be any hope of realizing the residual-heat design goal of section 3; i.e. equilibrium passive heat removal rate = 0.3% full power and inherent reactor heat sink capacity = 1000 full-power-seconds.

The diagrams, Fig. 5 illustrate an evolution of ideas concerning residual heat removal from the cores of light water reactors.

6.1 Simple Pool Reactor

Fig. 5-a shows a simple type of reactor where the core is immersed in a large containment vessel of water. On shutdown, the core is cooled by natural convection currents.

In the absence of adequate external cooling, the water temperature and associated vapour pressure will continue to rise until the safety/relief valve lifts.

As more heat is generated steam will escape through the safety/relief valve and the water level will decrease in the reactor vessel until eventually it begins to uncover the core. Further evaporation will progressively uncover the fuel, which will then depend on steam cooling. Steam cooling causes steam superheating which, as everyone now knows, is a clear sign that the core has been uncovered. Steam cooling may sometimes be adequate to prevent fuel melt but, in the case of Three Mile Island, it could not prevent clad temperatures rising, hydrogen generation, clad destruction and massive fission product release to the primary circuit.

The energy absorption up to boiling point is:

$$E - Eo = MC \times (Tsat - To)$$

To make good use of the water mass, the temp. difference (Tsat - To) should be as large as possible. Increasing Tsat means increasing the pressure rating and cost of the vessel. Since primary circuit and pool form a single entity, lowering To means lowering the useful output temperature. The simple pool system is therefore only suited to research reactors where low output temperatures are acceptable.

6.2 Thermally-Segregated Primary Circuit

The heat sink capacity of a pool type LWR can be greatly increased by employing a thermally-segregated primary circuit as shown in Fig. 5-b. By pressurizing the containment vessel with a cover gas, the pool temperature (To) can be kept to a low value (_ 35°C) while the primary core outlet temperature can approach the containment design boiling point.

In normal operation the coolant in the primary circuit is thermally insulated from the pool and prevented from mixing with the pool water by a small differential pressure. On shut-down the differential pressure is eliminated and the cold pool water enters the bottom of the reactor and flows upwards through the core by natural convection.

Such a system forms the basis of the "SECURE" District Heating Reactor described in another paper at this meeting. In this ASEA design, the pool consists of highly boronated water at approx. 35°C, while the core outlet temperature is approx 120°C. So long as the primary circulating pump is operating, a differential pressure maintains physical segregation between the primary coolant and the pool water. On pump trip, the boronated pool water enters the core and automatically shuts down the reactor.

The implied heat sink capacity of this design exceeds the 1000 full-power second criteria (see Fig. 6-(4) and there doesn't seem to be any reason why the remaining 0.6MW of residual heat shouldn't be dissipated by a completely passive containment cooling system.

While the Thermal Segregation of the previous section permits the pool to be kept at a lower temperature than the coolant, the core outlet temperature must be lower than the pool saturation temperature. This is determined by the safety/relief valve setting which must be less than the containment vessel design pressure.

A core outlet temperature of 120°C can be obtained with a cover gas pressure of 0.7MPa but, while 120°C is adequate for a district heating plant, it is far too low for electric power stations which need temperatures of at least 300°C (p_{sat} = 8.6 MPa).

With a primary pressure of 8.6 MPa, thermal rating of 3000 MW, and pool temperature of 35°C a volume of 2500m3 of water would be required to provide a heat sink of 1000 full-power-seconds. This is more than 10 times the volume of the present primary circuits and demonstrates that such large volumes of water cannot be maintained at primary circuit pressure.

It doesn't seem possible therefore to extend the Thermally-Segregated pool design to large size nuclear power stations.

6.3 Pressure-Segregated Primary Circuit

Fig. 5-c shows a high pressure reactor primary circuit immersed in a large volume of water at a containment pressure close to ambient. Assume that the reactor is successfully tripped, but all feedwater supplies are lost to both primary circuit (make-up water) and secondary circuits (steam generator feeds), leaving the reactor without any active residual heat removal system.

The primary pressure would rise until the safety/relief valves lifted. Steam would then be discharged and the reactor water level would start to fall. According to Fig. 4 it would only be a couple of hours until the top of the core was uncovered and core damage was imminent. Surrounded by a massive heat sink (but insulated from it) the fuel would find itself in a dire but paradoxical situation and could well parody the words of the Ancient Mariner:

> "Water, water everywhere, but none the fuel felt
> Water, water everywhere, yet all the rods did melt"

In view of the problems to be overcome by well-designed active systems, it is hardly surprising if the problem of refilling a pressurised (but partially empty) reactor from a low pressure pool, by totally passive means, is a daunting prospect.

The first step must obviously be to depressurize the reactor until it has the same pressure as the containment atmosphere. This can be achieved by opening a set of depressurization valves (of a non-re-closing type) and allowing the reactor to blow-off steam into the pool until its pressure equalizes that in the containment. At that stage pool water should flow by gravity through the non-return valves to the bottom of the core, refilling the reactor, and establishing a natural convection path for residual heat removal. The reactor would then form part of the water pool in a manner similar to that shown in Figs. 5-a and 5-b.

Depressurization (or blow-off) has the inherent disadvantage of deliberately increasing the coolant loss, causing an immediate increase in the downward velocity of the water surface.

Steam injecto[illegible] d seem to have a definite role to play in passive cooling systems (see R[illegible] and might well be used to overcome the inherent snags of depressuriza[illegible], as shown in Fig. 5-c. Instead of allowing the blow-off steam from the depressurization valve (DV) to be condensed directly in the pool, the steam would be fed to a series of injectors which would draw cold water from the pool and inject it into the bottom of the reactor. Since the mass ratio of feed/steam should always exceed unity, the effect of opening the depressurization valves (DV) should be an immediate reduction in water surface downward trend.

The nett effect of depressurization via steam-injectors is to utilize the thermal energy of the remaining reactor coolant to rapidly refill the (still partially pressurized) reactor from the cold low-pressure pool.

When the pressure is sufficiently low, the non-return valves open, allowing a gravity feed to refill the reactor (if necessary) and establishing a natural convection pathway for continued core cooling. As the pool heats up, the vapour pressure rises in the free space and steam condenses on the steel walls of the containment thus rejecting heat to the outside. By the time the pool has absorbed 1000 FP.s of decay heat, steam condensation on the externally cooled containment shell should dissipate 0.3% of full power.

There would be multiple depressurization valves, non-return valves and injectors, providing redundance and system reliability. Opening of a depressurization valve would be initiated by a lower water-level signal from a sensor (or group of sensors) in the reactor vessel head (PWR), or below normal water level (BWR).

The experience at Three Mile Island has clearly shown that reactor water level should be the key parameter to initiate all emergency feed (injection) systems.

Reactor pressure, and drywell pressure, are less fundamental parameters and though they can be useful as anticipatory signals, they can never replace the key role of reactor water level.

To ensure the passive nature of the system, each depressurization valve, control signal and associated water level sensor(s) could form an autonomous self contained system, independent of central control systems (manual or computerised) and external supplies (electric power, compressed air etc.).

Drywell

Fig. 5-d shows a similar arrangement, but restricting the water to a (cylindrical) tank, and leaving a drywell around the reactor.

In the event of a TMI type LOCA, caused by a leaking safety or relief valve, the reactor would be depressurized and refilled to produce a natural convection path, without flooding the drywell.

In the event of a leak or break in the drywell area, the reactor would be depressurized, refilled and a natural convection pathway established as before. However leakage would continue into the drywell until, eventually, the water level would equalize with that in the (cylindrical) tank as shown in Fig. 5-e.

Pressurized ECCS injection would still be required in the case of the largest design-basis LOCA and the advantages of better injection points, such as the lower plenum, (Fig. 7, feature 3) are described in Ref. 9.

In the passive ECCS Injection System described by Kleimola (Ref. 8) large quantities of chilled water (approx. 10 core volumes) are forced into the reactor by steam injectors fed from the steam generators (PWR), or from special hot water tanks (BWR).

Fig. 7 indicates the possible appearance of a PWR incorporating a passive heat sink of 1000 FP.s (1) with steel containment externally air-cooled by natural draught (5). Other features shown are: Depressurization-valve plus steam injector (2); Lower-plenum injection points (3); and Heat Transfer by steam condensation to steel shell (4); The same features can be applied in the case of a BWR which, being designed for boiling, is a simpler system so far as LOCAS are concerned.

One of the lessons of TMI is the fact that PWRs must be designed in such a manner that, having lost the effect of the pressuriser heaters for any reason, and reactor pressure having therefore dropped to the saturation value, they can still safely dissipate their residual heat in a boiling mode. (it is assumed that PWR pressuriser heaters are irreversibly switched-off, once the water level has dropped below its operating range).

7. CONCLUSIONS:

Given time, human beings can cope with many types of emergencies so that an increase in failure-delay-times is an important safety goal.

Present-day reactor designs have been examined and, though their overall safety record is excellent, it appears that, in some respects, they are rather finely-tuned and not quite as "forgiving" as one would wish. e.g. on shutdown, they must depend on engineered residual heat removal systems for extremely long periods of time while, in certain accidents scenarios, the point-of-no-return for irreversible plant or fuel damage is reached in a matter of hours - in some cases less than an hour.

If this situation was due to some inherent law of Nature, it would have to be accepted in order to enjoy the benefits of Fission Power; but since this is not the case, improvements may well be feasible, and the creation of a truly "forgiving" reactor should be a stimulating challenge for nuclear designers.

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ABBREVIATIONS

DV Depressurisation valve
RV Relief valve
SG Steam generator
SV Safety valve

SYMBOLS

A	m2	Cross section of chimney
c	J/kg°C	Specific heat
Cf	-	Skin friction coefficient
E	J	Energy
Eo	J	Initial energy of pool
l	s	Mean neutron lifetime
M	kg	Mass
Mc	J/°C	Heat capacity per °C
P	W	Thermal power
Po	W	Full (rated) power of reactor
Pr	-	Prandtl number
p	Pa	Pressure
p_c	Pa	Containment pressure
p_p	Pa	Primary circuit pressure
q	W/m2	Heat transfer rate per unit area
R	-	Temperature-rise-ratio (See 2.1)
Re	-	Reynold's number
S	m2	Surface area of containment
S2 = 2*S	m2	Total heat transfer surface of chimney
St	-	Stanton number
t	s	Time from reactor shutdown
T	°C	Temperature
To	°C	Initial temperature of pool
T1	°C	Inlet temperature
T2	°C	Outlet temperature
TL1	°C	Liner coolant inlet temperature
TL2	°C	Liner coolant outlet temperature
Tp	°C	Mean reactor coolant temperature
Ts	°C	Surface temperature of containment
Tsat	°C	Saturation temperature
W	m	Width between containment and shield building
Yc	m	Width of chimney
Zc	m	Height of chimney
θ	°C	Fuel temperature
θo	°C	Initial fuel temperature
θmax	°C	Maximum fuel temperature
θcr	°C	Critical temperature for fuel damage
ρ	kg/m3	Density

Table I - Graphite Mass & Heat Capacity (Full-power-second/°C)

Parameter	Symbol	MAGNOX		AGR	HTGR
		BRADWELL	WYLFA	HUNT. -B	FULTON
Power(th)	Po	531. E6	1875. E6	1500. E6	3000. E6
Core	M				0.32E6
	Mc/Po				0.19
Reflector	M				0.59E6
	Mc/Po				0.35
Total	Σ M	1.9	3.7 E6	1.0 E6	0.91E6
	Σ Mc/Po	5.36	2.96	1.13	0.54
Effective *	Mc/Po				0.33

=1800kg/m3 C=1800J/kg°C(@700) * Deduced from heat-up graphs

Table II Energy Requirements for LWR Boil-off & Heat-up

Energy in full-power-seconds	Symbol	PWR (3.5 GW)	SECURE (0.2 GW)
To boil off secondary	E_1	100	-
To raise temp to SV set point	E_2	15	1616
To boil-off water above core	E_3	50	4440
To boil-off water in core etc	E_4	50	1480
Tot. Energy until -FP.sec	E_1+E_2	115	1616
SV lifts -time (s)		8.5 E3	3.4E5
Tot. Energy until -FP.sec	$E_1+E_2+E_3$	165	6056
top of fuel is dry -time(s)		1.45E4	2.4E6
Tot. Energy until -FP.sec	ΣE	215	7536
fuel is totally dry-time(s)		2.0 E4	3.5E6

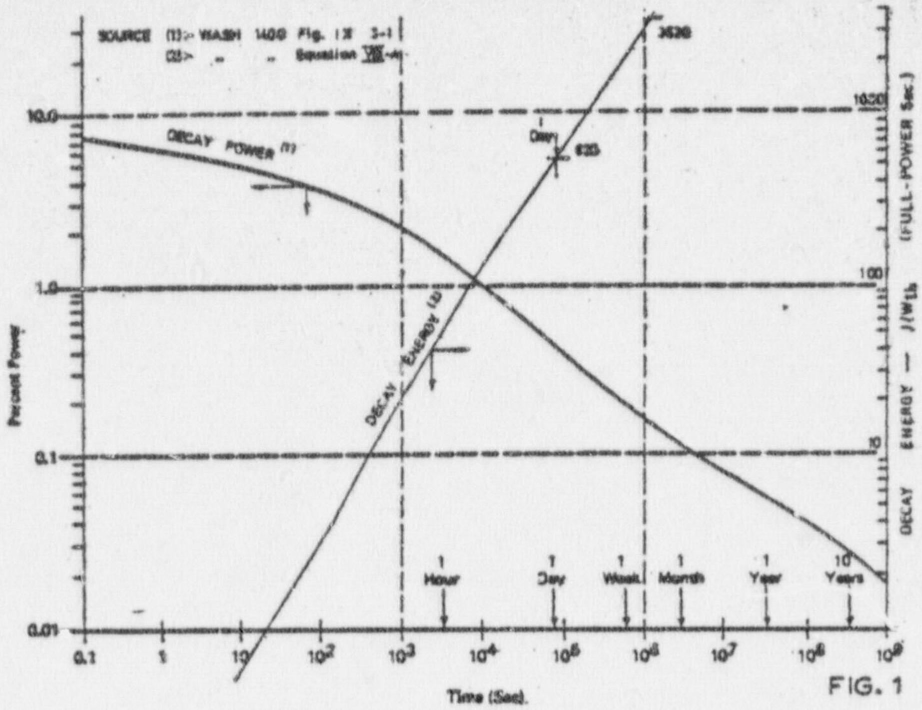


FIG. 1

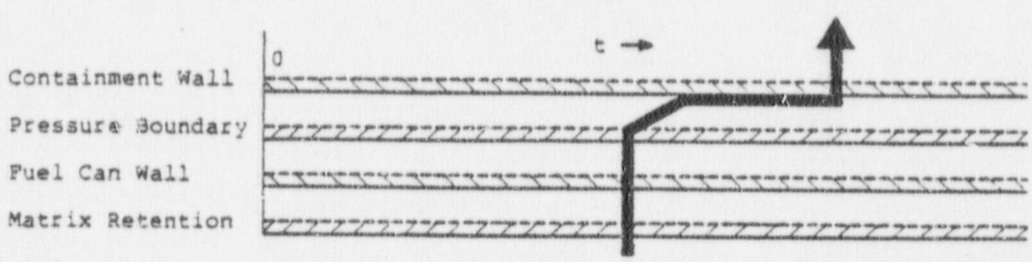


Fig. 2 - BARRIER BREACH DIAGRAM FOR UNRESTRICTED CORE HEAT-UP

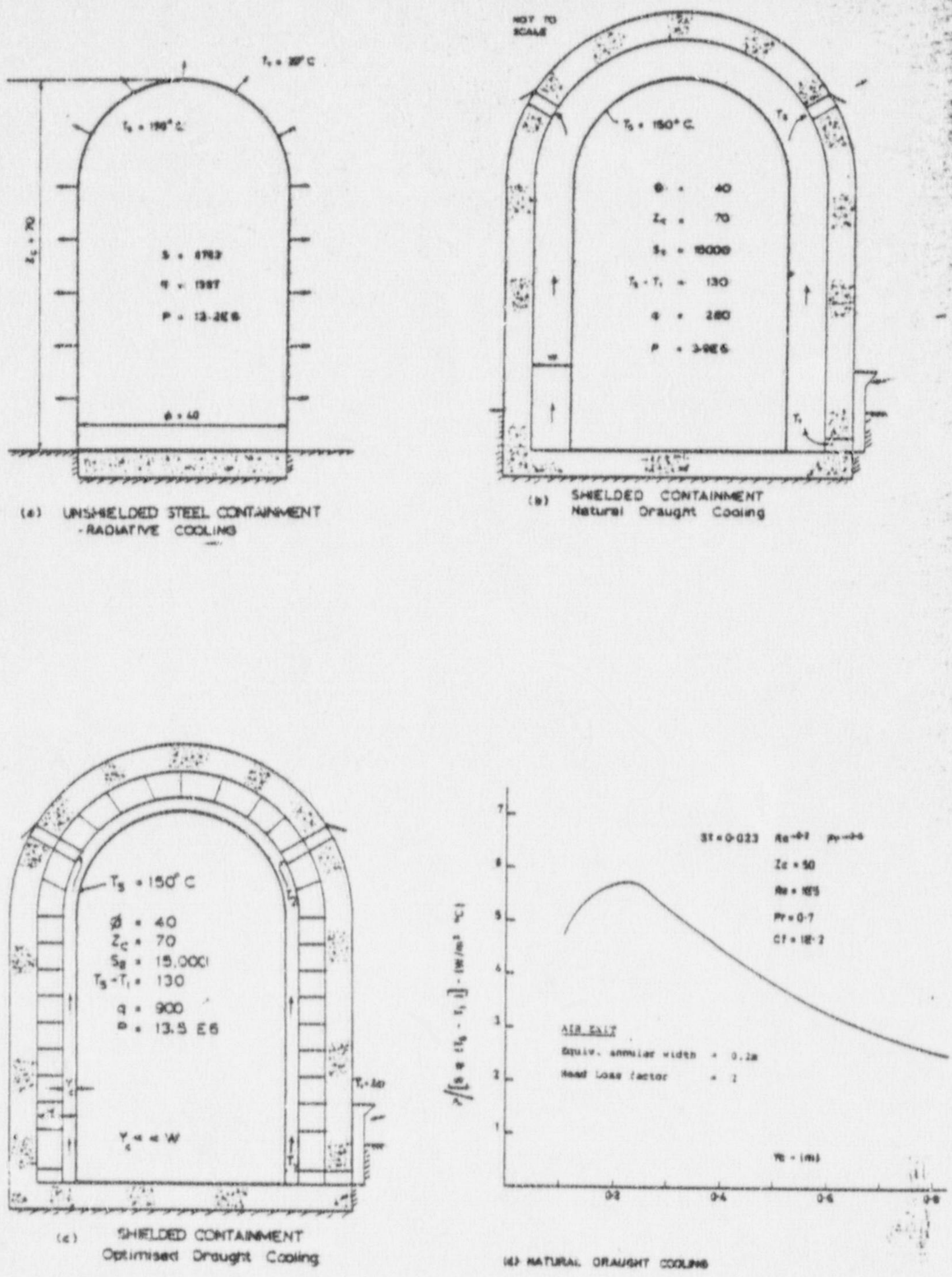


Fig 3 CONTAINMENT COOLING

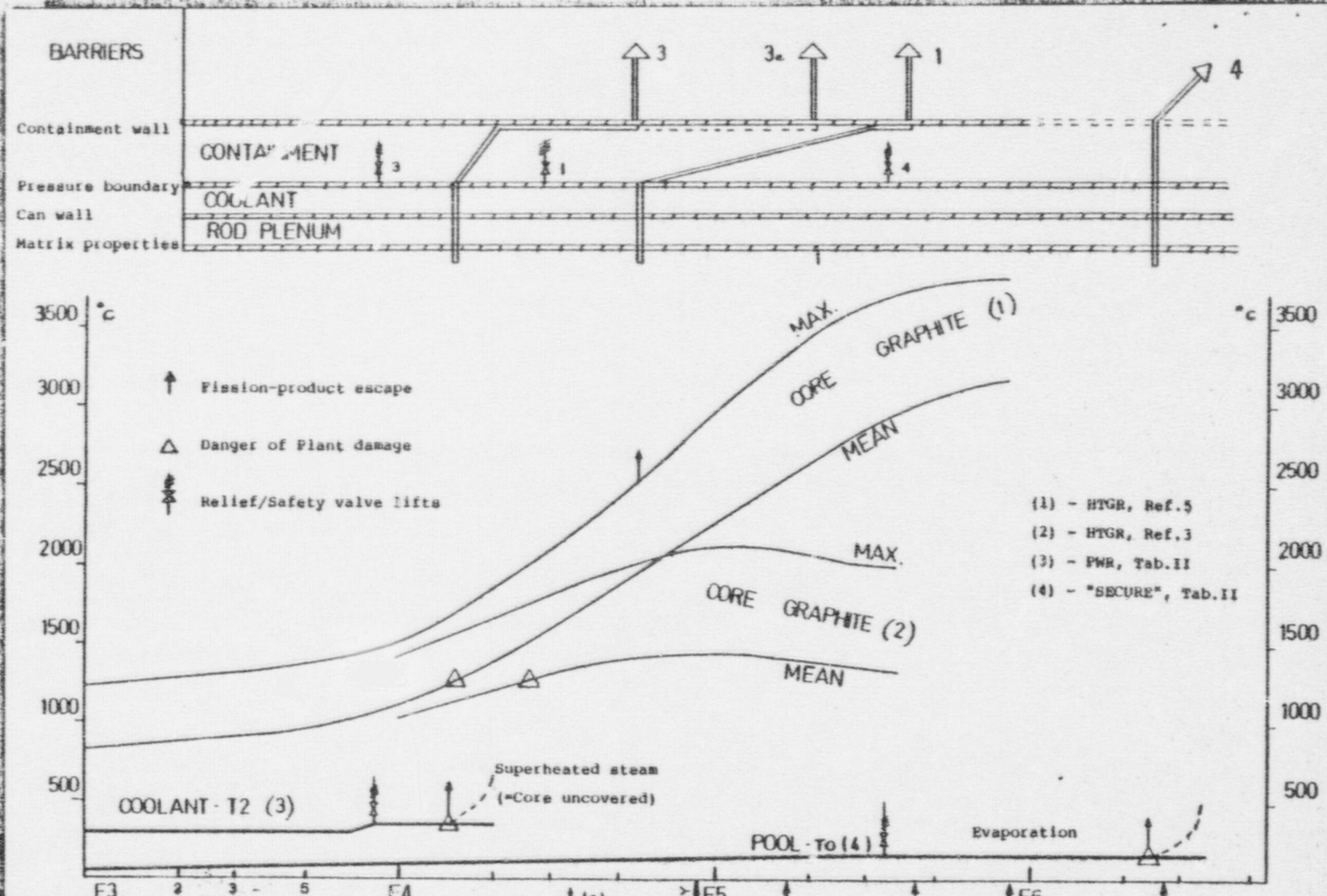
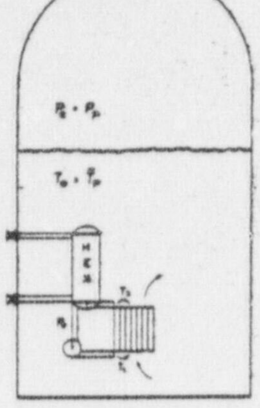
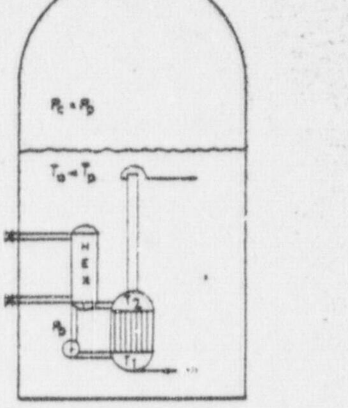


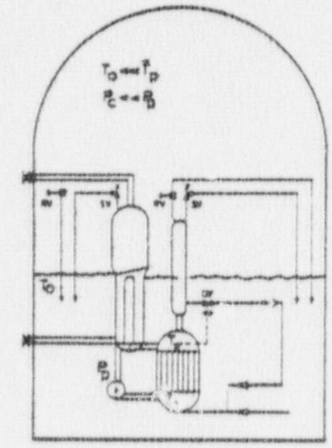
Fig. 4 – UNRESTRICTED CORE HEAT-UP



(a) SIMPLE POOL CIRCUIT



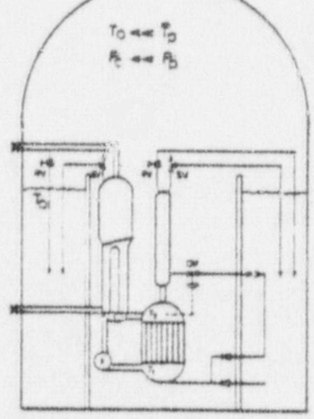
(b) THERMALLY SEGREGATED PRIMARY CIRCUIT



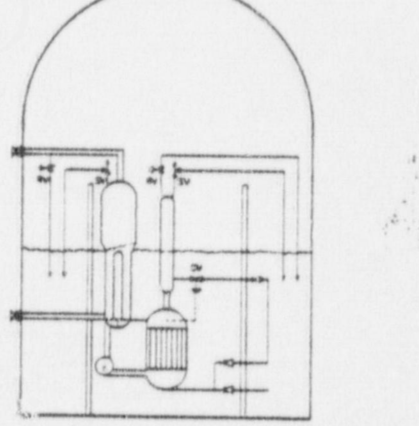
(c) PRESSURE - SEGREGATED PRIMARY CIRCUIT

Fig 5

LWR COOLING



(d) PRESSURE SEGREGATED PRIMARY IN DRYWELL
(Normal Operation)



(e) PRESSURE SEGREGATED PRIMARY IN DRYWELL
(Flooded Condition after Loca)

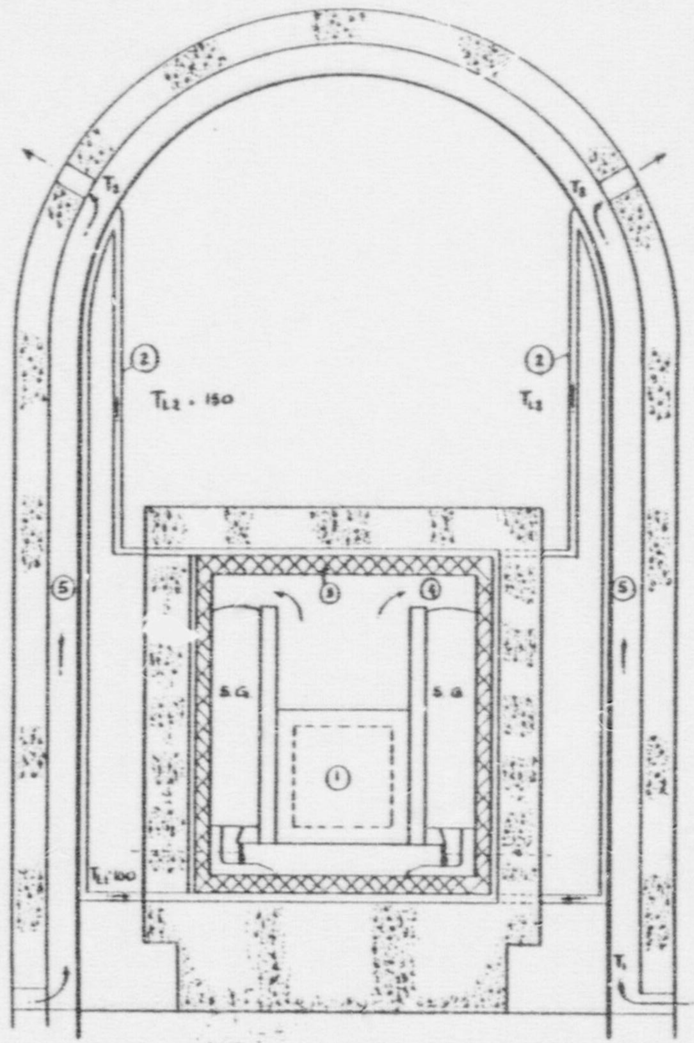
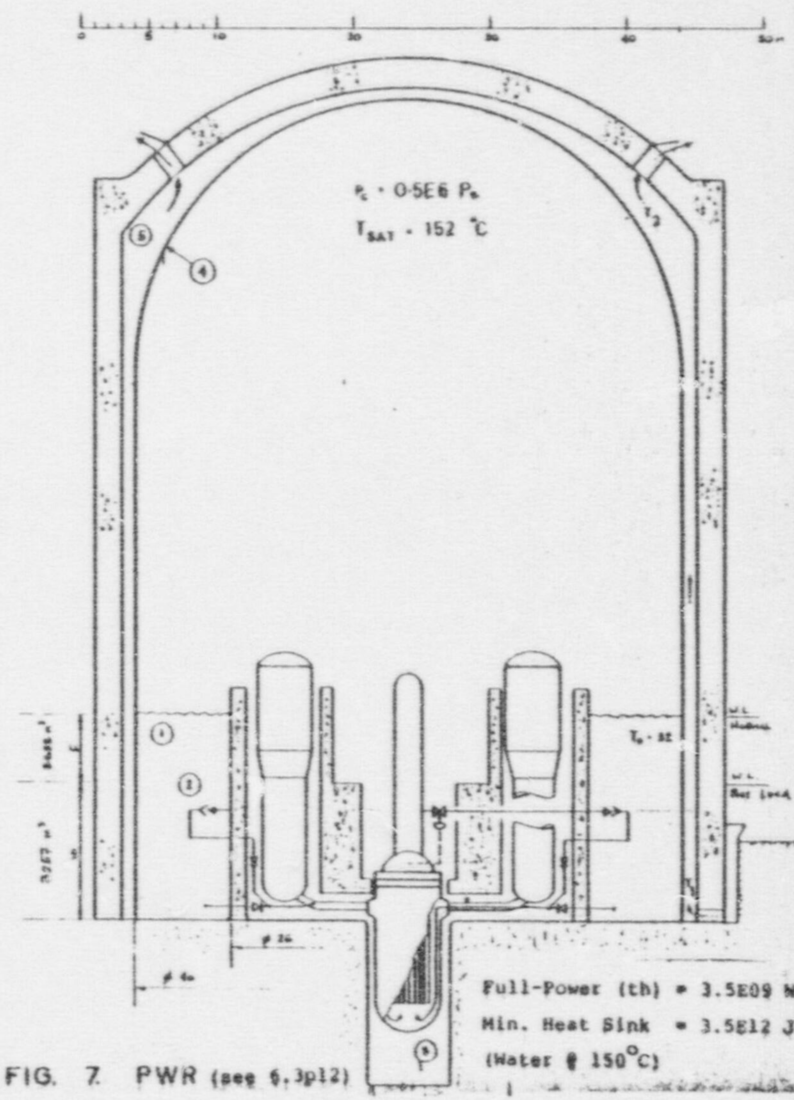


FIG. 6 DIAGRAM OF HTGR (see 5.3 p7)



Full-Power (th) = 3.5E09 W
Min. Heat Sink = 3.5E12 J
(Water @ 150°C)

FIG. 7 PWR (see 6.3p12)

BIOGRAPHICAL NOTE.

The author, who is a member of the European Community Reactor Safety Working Group 1, has been involved with Nuclear Power since the construction of the early British Magnox reactors. For many years however, he has specialized in monitoring nuclear safety problems for an electricity company which hasn't yet committed itself to any one design.

The author is thus able to review the reactor safety scene with detachment, and his comments, criticisms and suggestions reflect the questioning view-point of potential first-time purchasers, considering the nuclear-power option in the aftermath of Three Mile Island.

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