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Small reactors with simplified design

Proceedings of a Technical Committee meeting held in Mississauga, Ontario, Canada, 15–19 May 1995

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FOREWORD

There is a potential future need for small reactors for applications such as district heating, electricity production at remote locations and desalination. Nuclear energy can provide an environmentally benign alternative to meet these needs. For successful deployment, small reactors must satisfy the requirements of users, regulators and the general public.

The IAEA has been following the developments in the field of small reactors as a part of the sub-programme on advanced reactor technology. In accordance with the interests of Member States, a Technical Committee meeting (TCM) was organized in Mississauga, Ontario, Canada, 15-19 May 1995 to discuss the status of designs and design requirements related to small reactors for diverse applications. The papers presented at the TCM and a summary of the discussions are contained in this TECDOC which, it is hoped, will serve the Member States as a useful source of technical information on the development of small reactors with simplified design.

EDITORIAL NOTE

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SUMMARY

1. INTRODUCTION

Potential applications for small commercial reactors could be envisaged in the following areas:

- Supply of electrical power in remote locations (e.g. Arctic, Great Sahara Desert, Pacific Islands, etc.) for high-value special-purpose applications such as microwave communications and radar navigational aids.

- Supply of electric power and heat or desalinated water in remote areas.

- Utilization of heat in industrial or housing complexes near large population centres.

- Prolonged deep sea applications where the need for oxygen for power generation has to be minimized.

The initial demonstration of a small commercial reactor would likely involve a development project located near a community capable of providing the required technical nuclear infrastructure, such as a national nuclear research centre. In this case, the power produced could be used to support the needs of the site.

The presentations by the participants concentrated on design aspects and user requirements for reactor systems that have less dependence on human intervention. The main mechanism is reliance on passive safety features such as decay heat removal by natural circulation, utilization of stored potential energy and negative temperature coefficients of reactivities. Some papers considered various design options to enable simplified management to the extent where an autonomous mode of operation is possible for limited periods. Other contributions mainly presented existing designs in the small reactor range. One paper covered public acceptance, which may be a more stringent requirement in a small community and should be dealt with properly (Appendix II). In addition to the papers presented, one working session was devoted to discussion on general issues concerning measures to reduce the operational costs.

Small reactors with reduced power ratings can offer potentially significant cost reduction through design simplification and/or reduction of local infrastructure requirements. For small reactors having a long grace period for human intervention, it would be possible to minimize the number of highly skilled staff on-site by assigning some of the responsibilities to staff at a remote location. The staff at the central site would also continuously monitor the status of several reactors simultaneously.

Use of multiple, identical reactor units would have an advantage in achieving a high capacity utilization together with adequate flexibility and redundancy. It would also enable serial production, high-quality shop prefabrication, modularization and efficient application of remote monitoring.

2. DEFINITION OF TERMINOLOGY

When discussing the general considerations for the use of small reactors and identifying relevant issues for their successful deployment, it is important to have a common understanding of some of the key terms used. For the purpose of this report, the following definitions are adopted:

Small commercial reactor

A small commercial reactor is, for the purpose of this report, defined as a nuclear reactor with power levels up to 400 MW(th) (i.e. up to ~100 MWe). This class includes a very small reactor category (about 10 MW(th)) for which siting conditions may be relaxed as a result of low risk levels and long grace periods during postulated accident events.

Minimized staffing

Minimized staffing refers to the reduction of overall manpower by the sharing of skilled personnel among similar facilities. It is achieved by optimizing the division of responsibilities and functions between local staff, central support and the plant itself.

Remote monitoring

Remote monitoring is the provision of continuous surveillance of the condition of one or more plants from an off-site location that allows an appropriate response within the grace period permitted by the design.

3. GENERAL CONSIDERATIONS

3.1. Design

Nuclear reactors must compete with other energy sources except in a few special cases for which the performance characteristics of nuclear systems cannot be matched by alternate technologies (e.g. certain space power applications and submarine propulsion).

Not all cost components scale with power, and therefore the specific energy cost for smaller reactors is often high. Consequently, small reactors should not be designed merely as scaled down versions of larger units, but should take advantage of design simplification that can be achieved with smaller sizes. Realistically, small reactors can more readily incorporate cost saving design features and passive systems. They can also benefit from associated low redundancy requirements. Moreover, the small component and facility size of small reactors allows the use of low cost construction technologies (e.g. standard components, modular designs, shop construction, etc.).

High quality heat (i.e. heat at high temperature) is required for many applications such as generation of electricity with a high energy conversion efficiency. In the near term, such systems are likely to use water as the coolant. Accordingly, these systems would benefit directly from the worldwide experience with large PWR and BWR systems. In the longer term, higher temperature systems using organic fluids and liquid metals such as sodium and lead may be more advantageous in improving conversion efficiency.

3.2. Economics

Significant capital savings can be achieved through economies of standardization associated with the implementation of a large population of similar reactors. The contributing factors include the following:

- reduced cost of licensing through streamlined licensing procedures and standard product licences;
- reduced design and safety analysis costs;
- reduced equipment costs through bulk procurement;
- other capital cost savings resulting from single land procurement, common rights of way, shared distribution systems, etc. through co-location of several similar plants

At low power ratings additional savings may result from the elimination of some components or systems through design simplification (e.g. primary coolant pumps and emergency coolant injection system). The cost savings associated with the use of advanced construction and fabrication techniques for modern large reactors would also apply to small reactors.

Significant benefits can also be realized from reduced operation and maintenance costs through remote operation. Thus, small reactor systems should be designed to minimize the requirement for close and frequent supervision by skilled operators and maintenance personnel. Labour costs can be reduced by associating an operating and maintenance labour force with several small plants in an efficient manner.

A considerable reduction in maintenance requirements and related costs would advance the competitiveness of small reactors with other conventional power sources. Also, as reactors are more vulnerable during maintenance routines, reduced maintenance schedules have the added potential benefit of enhancing overall system safety.

To reduce the adverse impact of a system shutdown, the total load should be divided, where possible, among two or more small reactors. This approach would provide flexibility for the scheduled or unplanned maintenance of one reactor while essential services are provided by the other(s), thus reducing the need for backup power systems.

An advantage of operating a large population of closely located small reactors is that maintenance activities could be shared. A nominal two week annual maintenance period for each of a set of 25 reactors, for example, would make full use of a traveling maintenance crew and would thereby reduce maintenance costs.

As with all nuclear power systems, small reactors would have a substantial advantage relative to conventional power sources with respect to transportation and storage needs for fuel.

3.3. Safety

Small reactors have a potential for substantial simplification of the design, facilitating more economical operation and maintenance. A simple design allows for a corresponding simplification of the control procedures and significantly reduces the likelihood of operator errors.

Low stored energy is a characteristic in small reactors which permits use of passive safety systems for safety functions such as decay heat removal. This in turn eliminates the redundancy needs of active components. This is believed to improve plant safety, while simultaneously reducing plant costs (for components and building volume). The simplicity attained by these features will be conducive to increasing plant availability and lowering the size of the operating crew, as well as reducing the probability of serious accidents.

3.4. Licensing

The reactor licensing burden must be commensurate with the product size and with well defined conditions and procedures. This could be achieved by a process involving a Standard Product License for obtaining a site specific license.

A global programme towards establishing safety goals and developing standards for small reactors could not only form the basis for national standards but facilitate their harmonization and thus enhance the acceptance of small reactor concepts and their economy.

If minimization of staffing is contemplated for a small reactor, it must be addressed up front in both the design and in the licensing, since it may influence siting, design and infrastructure requirements. The actual implementation of such an operation may have to be introduced gradually.

Generically, in order to obtain a license for operation with reduced staffing, the licensee must demonstrate that such operation will not affect the safety of the reactor. The safety case is not compromised by a proper distribution of responsibilities and functions among the on-site staff, off-site staff and the reactor systems themselves.

Small reactors contain less stored energy and often operate at a lower pressure in the primary heat transport system than typical large power reactors. In addition, they have lower core power density and passive systems or features instead of active ones for heat transport and control. These characteristics are basically conducive to operation without fully licensed operators on-site. Some of the prerequisites for such operation are:

- There should be need for operator control during normal operation (only infrequent adjustments to compensate for burnup should be required).

- Use of remote plant monitoring and diagnostics, including assessment of abnormal operating conditions.

- Use of automatic safety functions with a local and remote shutdown capability.

- Provision of automatic and/or remote testing of safety systems to ensure their availability.

- Extended period between physical inspections or maintenance of reactor systems.

- No easy access to fresh or spent fuel and the reactor core.

- No significant off-site consequences for any credible accident.

- A long grace period (measured in days) for an intervention to prevent severe accidents.

- The primary containment vessel should not be breached by reactor pressure vessel melt-through following core damage.

It is assumed that adequate personnel will be continuously available on-site to fulfil the requirements of physical security, maintenance of conventional and auxiliary systems, managing emergency situations (both conventional and radiological), reporting and supervising of the nuclear system operation, without having direct access to the reactor but being able and authorized to shut it down. However, for truly remote sites, where there is no risk of harm to the public and no security concerns, even the conventional maintenance personnel may be remote. Service actions on the reactor systems would be performed by a specialized crew which would include licensed operators and would be dispatched from a central location when required.

While the described mode of operation is technically feasible, there are likely many regulatory hurdles to overcome, including the following:
- detailed safety assessment,
- identification and compliance with NPT requirements,
- definition of the composition of staff at central locations,
- the number of sites to be supported by one team.

The safety level of small reactors could easily exceed the goals stated for future power reactors. For example, several small reactor designs quote a core melt frequency $< 10^{-7}$/a. Also, consequences of severe accidents will be limited which may reduce public concerns related to nuclear technology which is a prerequisite with respect to ‘urban’ location.

3.5. Technology

Small commercial reactors are and would be designed for different operating conditions, and hence involve somewhat different technologies than for large power plants. A number of small reactor concepts exists, but proven technologies available on a commercial basis are not evident.

Specific technical features needed for some small reactor applications could include a high core operating temperature or a reduction or elimination of high pressure systems. Further, a focus on improved energy conversion efficiency for reactors of small size is required.

Inherent safety features as well as the use of passive systems, e.g. natural heat dissipation and natural circulation for core cooling, should as far as possible be emphasized in the design. Complexity, cost and the need for manual intervention would be lowered by the absence of large numbers of active systems.

Current industry capabilities indicate that a fully automatic plant is technically achievable. A large amount of remote sensing and information processing already occurs in large power plants; it is a small technology step to interpret the data and orchestrate predefined responses without operator action. Automation also lends itself to remote monitoring since the information required for automatic control is generally the same as or a subset of that needed for plant monitoring. This information allows a qualified operator to assess the operating status of the plant irrespective of his or her location and, with automatic responses, allows the plant to recover from upset conditions before the operator needs to be on site.

3.6. Public acceptance

Public acceptance is a critical issue for the introduction and widespread use of small commercial reactors. Enhanced safety characteristics of small reactors offer an opportunity to build public confidence, but it will take time, information and effort to overcome a legacy of generally negative associations and safety concerns with nuclear technology of almost any type. These are derived, in part, from historical accidents in nuclear reactors, military activities, and a perception of unresolved radioactive waste management issues. In particular, local public acceptance may be aided by a strong initial presence of the vendor living within the community and working at the application site to demonstrate that no undue hazard exists.

Local public acceptance may be linked to immediate economic benefits, such as employment opportunities. For a small reactor with minimized staffing, long term employment opportunities must be derived largely from the local use of the energy produced, rather than from its generation and distribution to an external market.

The safety features of small reactors should help in gaining public acceptance, possibly through a convincing demonstration of safety behaviour on prototypical systems. Besides, the general simplicity of small reactor designs should aid public understanding of the working principles involved, the likelihood of system failures and the degree of protection provided.

It is unlikely that the size of the plant alone will greatly influence public acceptance of a nuclear power plant. The safety argument must address both the actual technical safety features and issues as well as the detailed concerns perceived and expressed by the public; this must be done without undermining public perception of the reactor safety which may depend on other features or systems.

Small reactors should be capable of being easily implemented at many locations and in a time scale comparable to competing power systems. This would be possible only if the regulatory process is streamlined and public acceptance is readily achieved.

4. RELEVANT ISSUES

The use of small commercial reactors in remote or urban locations with minimized staffing raises a number of issues that would have to be addressed.

4.1. Security

Small reactors generally use enriched uranium based fuel inside a sealed core structure that is difficult to gain access to (e.g. a reactor vessel), usually inside a second 'guard' or containment vessel and enclosed by a biological shield structure or immersed in a protected pool. The design may also provide for refuelling at the factory (for very low power ratings), and this will avoid the need for a special spent fuel storage pool. Such a facility would reduce the risk of proliferation which would be reflected in cost savings.

Physical security can be provided through remote security monitoring using modern communications technology, low cost alarm systems, detection sensors, and video surveillance cameras. Local police and fire brigades can be made responsible for security monitoring and fire fighting at minimum cost, in the same manner as would apply for any small industrial facility. By minimizing security activities from the outset and assigning standard security obligations to available local organizations, savings in operating costs are possible.

4.2. Reliability, maintainability, transportation of radioactive materials, decommissioning

High reliability is very important in remote regions because skilled maintenance staff may have to travel long distances at significant cost, and a high capacity factor is needed for economic viability. Reliability can be achieved through design simplicity, e.g. by using natural circulation cooling, large operating margins and the use of multiple power reactor units of small output capacity.

The design should ensure that routine maintenance activities are only required on an infrequent basis, can be performed by local staff (with instruction from experts at a central support station, if necessary) and do not require high skill levels.

At very low power levels, e.g. < 10 MW(th), it may be feasible to centralize most nuclear infrastructure activities by designing the system to be transported to and from the application site as a sealed unit. At higher power levels, some fuel handling operations would be performed on site at infrequent intervals (e.g. every two to four years) and with full outside support. Fuel movement could be minimized by providing spent fuel storage within the reactor vessel itself or in an associated pool structure. However, there should be no long term fuel storage capability, e.g. more than 5 years, at the facility, and final decommissioning would most likely involve removal of all equipment and return of the site to its original state.

4.3. Human error and/or malevolence prevention

The potential for human error can be minimized by limiting, through design, the scheduled human actions needed for normal operation and for maintenance to a minimum, compatible with the safety goals and economic objectives. Such limits can be applied in terms of both the variety of required actions (through design simplicity) and their frequency (through larger operating margins).

The need for local human action can be reduced through the use of automated systems, for example, testing the availability of safety-related systems using automated and remotely operated equipment. The frequency of such testing depends on the reliability of safety-related equipment and the duration of power operation.

Security against malicious acts can be enhanced by limiting physical access to potentially sensitive equipment or material and to the rooms where they are housed, e.g. to the control rod drive mechanisms. In addition, the possible consequences of human error or malevolent acts can be mitigated by means of additional safety barriers, such as a guard vessel, pool cover, or remotely activated poison injection system.

4.4. Response time

Occurrences at a small reactor facility may be initiated by internal or external events and they may require either a local or an off-site response. The response requirements will depend on the siting of the reactor, e.g. whether it is in an urban area, in a remote area with nearby inhabitants, or in an uninhabited area.

In general, small reactors would be designed to withstand all credible external events at the site. For remote areas, where it may not be possible to get skilled staff to the site for several days, the plant must be designed accordingly, i.e. with no need for prompt intervention to ensure safety against credible external events. In urban areas, in addition to matching the design of the reactor to the response time, consideration must also be given to the possible effects of nearby human activities, e.g. large fires or explosions in nearby industrial facilities, vehicle or rail accidents in transportation corridors, and aircraft impact.

As previously noted, the reactor design must allow an appropriate allocation of operating responsibilities and functions among the local staff, the remote staff, and the reactor itself. Internal events must either be self-limiting, or allow enough time for the local or remote staff to respond. The plant should in general be designed so that no evacuation is required for any design basis accident, as well as for severe accidents. This is necessary since the public will not accept potential evacuation as being worth the relatively small incremental benefit provided by the plant relative to alternate technologies.

Nevertheless, an emergency plan will be required in most jurisdictions for sites with a nearby population. There must always be local staff whose responsibility is to initiate and manage the early stages of an emergency plan, until the local authorities take over the off-site aspects, and remote staff arrives to manage the on-site aspects. The local staff must be identified and trained in this function, even if they are never expected to exercise it. In an urban area, local staff will generally be responsible for some aspects of the small reactor, and there will be a good infrastructure which can provide police services to manage a limited evacuation. In a remote inhabited area, the infrastructure may be less extensive, so that the reactor owner will have to develop procedures and identify people responsible for executing them in an emergency.

4.5. Load characteristics, backup power source

The choice between a base-load or a load-following operating mode for a small commercial reactor should be based on economic considerations, taking into account application requirements, i.e. electricity, heat, cogeneration, or desalination, and operating conditions (linkage to a central grid network or local supply only), as well as the required backup energy source capacity.

For very small commercial reactors with remote monitoring, the load-following operating mode appears to be more reasonable and preferable as it does not result in large economic penalties and requires less operator intervention in the reactor control process. Load-following appears to be most useful for electric power and co-generation applications in remote isolated locations and should be provided when needed to meet user requirements.

In particular, load-following would support the efficient use of nuclear energy and reduce the need for alternate energy sources to meet temporary peak demand loads. Most small reactor core designs have an intrinsic capability for some degree of load-following operation as a result of their negative reactivity coefficients.

Base-load operation is especially reasonable for a dedicated heating plant because the heating demand load changes rather slowly and the capital cost of a backup or power peaking heat source is not large.

4.6. Use of pressurized systems

In general, low pressure systems will be simpler and have lower capital and operating costs than systems operating at higher pressure. Staffing of higher pressure systems may be subject to existing state regulations requiring the full time attendance of certified operators for all pressure vessels above a specified power and pressure level.

Passive protection against loss of coolant accidents is, in some designs, provided by a second pressure vessel surrounding the reactor pressure vessel. This guard vessel ensures that the core is never uncovered; to ensure that this second barrier is never breached, a reliable means of condensing steam should be provided.

Siting a very small pressurized reactor in an urban area may not be feasible without additional precautions because of the hazard of pressure vessel failure from internal or external events (e.g. gas explosion). Useful precedents to consider in this context are the low pressure (1.4 to 2.5 MPa) heating reactors NHR-5 (5 MW(th)) at the Institute of Nuclear Energy Technology, Tsinghua University, Beijing and NHR-200 (200 MW(th)). The siting criteria for NHR-200 involve a zone of no permanent residence to a radius of 250 m and a zone of restricted industrial development to a radius of 2 km.

To guarantee the integrity of both the primary and secondary pressure barriers, the design must allow for periodic in-service inspections to identify unacceptable flaws and cracks. Avoidance of catastrophic failure of pressurized components depends on a quality assurance programme which guarantees that the pressure vessel codes for design, testing and inspection are diligently applied. The code requirements for nuclear pressure vessels and other pressurized components will not depend significantly on reactor size. Periodic testing and inspection of pressure vessels will require specialized staff and special training, which would not be required for nonpressurized systems.

For electricity production and cogeneration, pressurized reactors have a large economic advantage because of their higher conversion efficiency. This economic advantage is balanced by the economic penalty of providing two independent pressure barriers which are essential to limit the consequences of primary coolant leaks.

5. CONCLUSIONS

Several general conclusions can be summarized concerning the use of small commercial reactors with simplified design and management aspects.

1) The subject of small commercial reactors continues to be an area of active interest with design concepts from several countries, despite the present realization of only a few prototype systems.

2) The commercial viability of small power reactors appears to require that a large number of very similar units be built.

3) It is unlikely that current, small, non-power reactors (SLOWPOKE, TRIGA, Pulstar, etc.) can be scaled up, or that existing medium and large power reactor designs (PWR, BWR, CANDU, HTR, AGR, LMR, etc.) can be scaled down to meet the economic and safety requirements for small, e.g. in the 10 to 50 MW(th) range, commercial reactors; an exception is district heating where the technological demands are modest.

4) The safety characteristics of a small commercial reactor must be such that public acceptance and licensing are assured independent of site location (i.e. there must be no need for emergency evacuation). The safety level achievable with current water cooled reactor systems would impose some restrictions on urban siting.

5) The design of a small reactor should be capable of addressing a wide range of applications, such as electricity generation, district heating, water desalination and cogeneration. Among these possibilities, the ability to generate some electricity is considered essential, and applications involving co-generation of electricity and heat are considered most likely.

6) The near term implementation of small commercial reactors would likely be based on water cooled designs and would primarily be aimed at remote, isolated locations where energy costs are high. In this context, integral reactor arrangements with natural coolant circulation and a closely fitted guard vessel enclosing the reactor pressure vessel are claimed to offer cost and safety advantages. However, a significant shift toward higher temperature systems and improved conversion efficiency may be necessary to achieve widespread future competitiveness with fossil fuelled energy sources. A rise in fossil fuel prices could, however, encourage pool-type heating reactors in urban locations.

7) Reactor operation with minimized staffing appears to be technically feasible, particularly for low-power systems, provided the responsibilities for safe operation are properly divided between local staff, central support and the machine itself, and that the design provides a sufficiently long grace period for required intervention.

8) Modern communications and control technologies are available to provide reliable means to monitor reactor operation remotely, thereby enabling operation with minimum on-site staff and significant cost savings without jeopardizing safety. Support by existing local infrastructure, such as police and fire fighting services, should be used to the maximum extent possible.

9) A global market survey is needed to identify the potential number of small commercial reactor units required, the optimum size(s) and underlying cost factors. This survey should principally address existing applications for stand-alone electric power generation, district heating and water desalination, and should identify the present day total unit energy cost involved and load variability. An international effort could solicit such information from Member States directly and co-ordinate the compilation and harmonization of existing data from other agencies.

10) Collaborative development and co-operation involving potential user groups and vendor consortia will be needed to establish an optimum technical design, an economically viable and sustainable market, and broad acceptance from the public, political leaders, investors, insurance underwriters and regulatory authorities.

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SMALL REACTOR OPERATING MODE

V.G. SNELL
Atomic Energy of Canada Ltd,
Mississauga, Ontario,
Canada

Abstract

There is a potential need for small reactors in the future for applications such as district heating, electricity production at remote sites, and desalination. Nuclear power can provide these at low cost and with insignificant pollution. The economies required by the small scale application, and/or the remote location, require a review of the size and location of the operating staff. Current concepts range all the way from reactors which are fully automatic, and need no local attention for days or weeks, to those with reduced local staff. In general the less dependent a reactor is on local human intervention, the greater its dependence on intrinsic safety features such as passive decay heat removal, low-stored energy and limited reactivity speed and depth in the control systems. A case study of the design and licensing of the SLOWPOKE Energy System heating reactor is presented.

1. CLIENT REQUIREMENTS

Small commercial reactors must satisfy three very different "clients". The most obvious client is the customer who uses the product–he demands safety, reliability, and acceptable costs. These demands become technical requirements on the design. The second client group is the general public, who must balance the real benefits of the facility with their perception of the risk. Often the function provided by a small reactor can also be provided by a competitive technology, such as natural gas for space heating or oil for desalination. The advantage provided by a nuclear source must be significantly greater than the competitive technology if the reactor is small: whereas people accept electrical power reactors because of their large economic benefit, they will not do so for small commercial applications unless they are unique, or have large advantages over the competition, or are transparently safe. The third client is the regulator, who demands licensability based on national experience with which he is familiar, and an assurance that he will not be faced with contamination of a populated area as a result of an accident. All three client requirements must be met.

2. DESIGN REQUIREMENTS RELATED TO OPERATING MODE [1],[2]

Operation of a reactor requires allocation of certain responsibilities, functions and duties. In principle these are the same as for a power reactor. However the design of a small reactor may allow a different allocation, while maintaining the essential requirements of public safety, plant reliability and low cost. For example, the design may reduce the need for prompt local response, and allow operating duties either to be centralized away from the reactor, or dispensed with entirely. The latter case requires confident demonstration of inherent safety. In general, the greater the inherently-safe characteristics of the design, the easier it is to reallocate the operating responsibilities, functions and duties.

One should start by assuming that **all** of the operating responsibilities, functions and duties of a power reactor apply to a small commercial reactor. One should then assess each one in turn, asking whether it can or should be allocated to the local staff, remote staff, or handled by the machine itself due to its inherent characteristics. In other words, one does not start from a reduced staff; one derives it from the machine characteristics, siting and usage.

We now list some safety-related characteristics for small reactors which normally are present to facilitate reduced staffing. Not all are relevant to any one application; not all must be satisfied to allow reduced staffing; and each must be considered in detail for any actual case.

1. The power, and hence the fission product inventory, are generally low.
2. There is highly restricted access to, and infrequent changes to, the core. The control devices for load-following are slow-moving and stability is aided by negative reactivity coefficients.
3. All the reactor safety systems are automatic or self-actuating. Effects of failures in the safety systems are mitigated by inherent properties or self-actuating processes. Automatic initiation of a safety system cannot be easily disabled by an operator. Safety devices are testable on power, without risk of a spurious shutdown. All critical components of a safety system are fail-safe or have independent back-up. Two independent and diverse shutdown systems are provided unless automatic shutdown can be guaranteed by inherent physical or chemical properties.
4. The mechanism for removal of decay heat has the same reliability and effectiveness as an automatic safety system for a long period after shutdown. Passive decay heat removal is the usual approach.
5. The stored energy in the coolant is low. A sudden loss of coolant is prevented, usually by double barriers. Following a small loss of coolant, there is no need for an external supply of water, nor for human intervention, for a long period of time.
6. The plant can withstand (without the need for prompt intervention) credible external events typical of the environment, such as severe natural phenomena, industrial accidents in nearby facilities, and the consequential effects of natural and man-made disasters on nearby facilities. Fires within the facility either are made impossible, or cannot affect the nuclear safety, or can be handled by fire-fighting staff in nearby settlements. There are generous time allowances built into the design for such staff to arrive and be effective.
7. The primary cooling system provides significant retention of fission products released from the fuel.
8. Storage of used fuel is on site, usually in the reactor pool or vessel, so that shipments of used fuel are infrequent or absent.
9. The confinement prevents release of radioactive gases which escape from the primary cooling system in both normal operation and an accident. In general accidents do not pressurize the confinement.

These characteristics enable the plant to operate with less need for routine operator intervention or presence. It is clear from this list that the safety of such small reactors relies much more on inherent design characteristics, and (to a lesser extent) on shutdown systems, than on engineered heat removal or prompt human intervention.

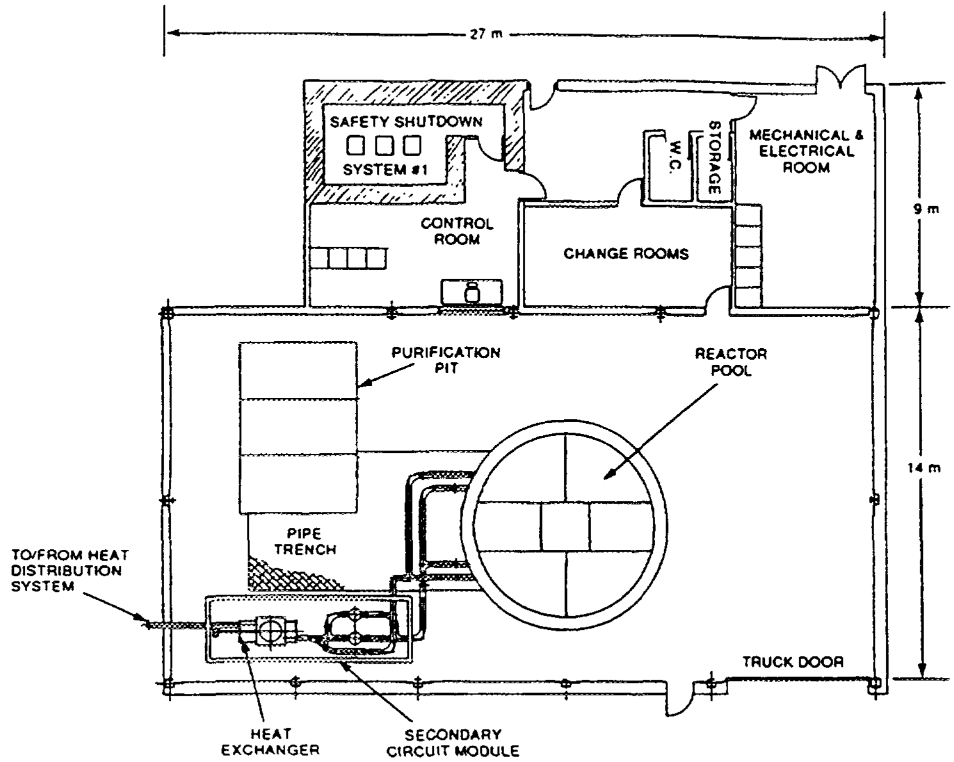
It is the emphasis on accident prevention through inherent characteristics which permits both urban siting and remotely-monitored operation. They are not new concepts by themselves– both have been accepted for a number of years in Canada for the 20 kW SLOWPOKE-2 research reactors, on the basis of an inherent design characteristic: because of the limited amount of reactivity available to the control system, and the negative temperature coefficients of reactivity, they do not need *any* engineered shutdown system, and can *therefore* be left unattended for periods of up to 24 hours. The SLOWPOKE Energy Systems 10 MW heating reactor retains many passive safety characteristics–for example, decay heat removal is passive–but because it operates at higher power, it requires an engineered shutdown mechanism and in fact is provided with two separate, independent, and diverse shutdown systems, to eliminate the potential of failure to shutdown for an abnormal event. This redundancy reflects a philosophy pioneered by CANDU nuclear generating stations.

3. SMALL HEATING REACTOR DESIGN EXAMPLE [2]

To illustrate how the small reactor safety principles discussed in Section 2 are incorporated in a design concept, the Canadian SES-10 heating reactor will be used as an example. The main protection against a major release of radioactivity from the 10 MW core is the fuel itself. By restricting the maximum fuel temperature in normal operation and in accidents, most of the fission products are retained within the uranium oxide pellets, and only a small fraction of the fission product gases escape to the narrow gap between the ceramic pellets and the metal sheath. If the sheath should fail, iodine would remain in the large volume of pool water and a small quantity of radioactive xenon and krypton could escape to the cover gas above the pool surface where it would be retained by the confinement barrier. The ultimate release to the environment from a single sheath failure would be well within regulatory limits for normal operation.

Other important safety features of the SES-10 concept (Figure 1) are listed below:

- A pool type reactor avoids the need for a nuclear pressure vessel and high pressure piping system.
- Operation below 100C and near atmospheric pressure avoids a large source of stored energy and loss of coolant by depressurization.
- Natural circulation of primary coolant in the pool avoids a loss-of-primary flow accident.
- Double containment of the pool in a steel vessel and concrete vault prevents loss of coolant by leakage. An air gap between the two containers permits both detection and control of leakage.
- Slow-moving control devices permit slow-response safety systems.
- The negative reactivity effects of coolant density and fuel temperature attenuate the power transient following control system faults.
- The large heat capacity of the pool delays the core temperature rise following loss-of-secondary coolant flow.
- The top surface of the pool is not open to the reactor building as in many pool-type reactors, but is enclosed with a steel cover plate and concrete shield. The vapour space between the pool surface and the cover plate is used to monitor and control gases released from the pool. The cover gas is confined by expandable gas bags to compensate for swell due to temperature changes.
- A mechanical shutdown system, and a second shutdown system dependent on phase change in devices installed in the fuel bundles are provided. Both systems are actuated by gravity. The first system is active and has dedicated instrumentation for detecting potentially hazardous conditions. The second system is passively triggered by the temperature of the material only.
- Long-term decay heat removal is by conduction to the ground through the concrete wall of the pool.



PLAN VIEW OF BUILDING

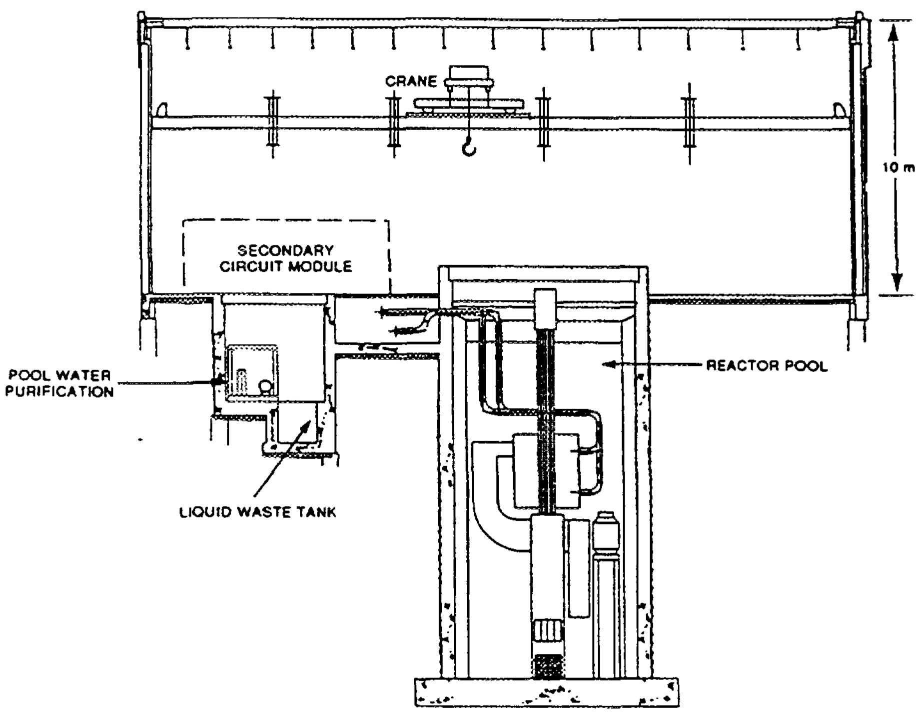


FIGURE SIDE ELEVATION OF BUILDING

4. SMALL HEATING REACTOR LICENSING EXAMPLE [3],[4],[5]

It was clear during the early design of SES-10 that licensing would be a challenge, because there were no well-established rules of licensing small reactors in any case, and SES-10 was proposed to operate in an urban environment using a unique operating concept. The operating model distinguished two types of operators: local and remote. Both had well-defined roles and responsibilities. Local operators were generally stationary engineers, charged with running the conventional heating plant and with monitoring the automatic operation of SES-10. They did not need to be in the control room all the time; they would go there periodically and be paged in case of an abnormal event. They would verify that the correct automatic action had occurred, and, if it did not, or in case of doubt, they could shut the reactor down (but not start it up). Because of the large pool, shutting the reactor down generally gives days before any further action need be taken. They, and the local fire and security staff, were also responsible for the conventional safety of the plant. The local operators would also communicate periodically, and in case of emergency, with the remote operators. They would receive specific training appropriate to these duties.

The remote operators were trained nuclear operators, periodically inspecting a remote monitoring station, which displayed all the key plant parameters, and from which these operators could diagnose an event and shut the reactor down. In case of emergency, they were summoned to the remote station by a pager initiated automatically by the monitoring computers, backed up by the local operator. Again, because the action time was days, there was no requirement for instantaneous response. The remote operators would also travel to site for skilled nuclear tasks such as refuelling or repairs to the control and safety systems. AECL and the Atomic Energy Control Board (AECB), the Canadian regulatory body, jointly agreed to consider the major issues for licensing SES-10 in advance of a formal application for a construction licence. This gave both parties the opportunity to consider the policy and concept issues posed by the design and mode of operation, so that a potential purchaser could be assured of low licensing risk.

During the initial review of the design, the AECB identified ten issues to be solved in the pre-licensing review. The major ones were as follows:

1. The worth of individual control rods would be determined by the following requirements:
   a. Calculate with confidence the consequences of inadvertent rapid withdrawal of the rod, regardless of the means, *or*
   b. Show that the probability of achieving prompt criticality or very severe fuel damage due to inadvertent withdrawal was less than 10^{-6} per year, *or*
   c. Subdivide the control rods so no one rod was worth more than 5 mk[1], *or*
   d. Use some other unspecified approach to convince the AECB the design is safe.

2. Use a risk target in design which reflects the lower benefit compared to power reactors.

3. Identify those events whose likelihood is so remote that they are excluded from the design basis. (These were generally accidents such as massive structural failure, which are precluded by use of appropriate codes and standards).

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[1] 1 mk ≈ 1/6 β

4. Use two poison storage tanks to give confidence that the reliability of liquid injection of 999 times in 1000 could be achieved.

5. Define the design specifications, and their basis, for the reactor building. The AECB expected that not all events leading to fuel damage could be ruled out of the design basis.

6. A comprehensive PRA could be required to demonstrate that risk targets were met.

7. Chemical interactions which produced sudden quantities of hydrogen, or precipitate the dissolved liquid from the second shutdown system, should be considered.

8. The design specifications for the operating mode should be considered early in the review.

The major issues from AECL's point of view were #1, #8, and #5. AECL took two years to systematically address these issues and respond to the AECB.

Issue #1 required a redesign of the reactor core, to dramatically reduce the worth of individual control rods. This was achieved by increasing the number of rods (and the core size), and by the use of burnable poison in the fuel assemblies, so that the control rods held the minimum reactivity required to start up the reactor and follow the load.

Issue #2 was considered by a joint industry-AECB ad-hoc technical working group, which proposed public risk criteria appropriate to small reactors [4],[5]. AECL identified the low-probability events for Issue #3, although little debate occurred on this. Issue #4 and part of #7 were made obsolete when the liquid poison shutdown system was replaced by the passive devices in the fuel elements. Passive self-starting hydrogen recombiners were used in the cover gas space. For issue #6, a commitment was made to perform a risk assessment, as is done on other AECL small reactors and on CANDUs.

Issue #8 was dealt with at length. A senior expert, formerly in Operations in Ontario Hydro, Canada's largest nuclear utility which now operates 20 CANDUs, joined the SES-10 team, and provided the operations framework for the design. A compilation of the operating responsibilities, functions, duties and training of operators in CANDU reactors was first undertaken, to ensure that none were overlooked. Next a set of basic operating principles was extracted from these; meeting these principles was a prerequisite for sound operation in any reactor facility. Finally these principles were used in allocating all the operating functions to the local operators and the remote operators. The roles and duties of each type of operator were carefully defined, and AECL ensured that the necessary technical infrastructure would be in place at each location to support them.

On issue #5, the design was changed so that the cover gas space was no longer vented either in normal operation nor in accidents. The confinement boundary therefore consisted of the reactor pool, the cover gas enclosure, and expandable gas bags which collected gas driven from the cover gas space due to the swell of the pool water either in normal startup, or in an accident. The redundant shutdown systems, and the ability to reject decay heat to the ground, ensured that design-basis accidents would not result in pool boiling (a relief path was provided for incredible accidents).

AECB accepted the revised core design in principle, subject as is usual in the regulatory arena to detailed review. Almost a year later they accepted the operating scenario as a satisfactory basis to proceed with further development of the design, subject again to future details on the safety analysis and the operating plans. On containment, AECB recognized that the absence of energetic dispersal of fission products was as prerequisite for the containment concept to be viable, but deferred an overall conclusion until further design details were reviewed.

5. CONCLUSIONS

Reduction of operating staff, or reallocating their duties elsewhere, derives from the design concept rather than being an *a priori* feature. Such reallocation is made easier by use of inherent safety characteristics which reduce the need for prompt local action to ensure safety. Since there is little precedent for such concepts, discussions with the regulatory agencies at an early stage are essential. The SES-10 example shows that the regulatory agency was willing to consider such innovation, as long as the designer was prepared to respond in a significant way to their concerns.

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SMALL NUCLEAR REACTOR SAFETY DESIGN REQUIREMENTS FOR AUTONOMOUS OPERATION

K.S. KOZIER
Whiteshell Laboratories,
Pinawa, Manitoba

S. KUPCA
National Defense Headquarters,
Ottawa, Ontario

Canada

Abstract

Small nuclear power reactors offer compelling safety advantages in terms of the limited consequences that can arise from major accident events and the enhanced ability to use reliable, passive means to eliminate their occurrence by design. Accordingly, for some small reactor designs featuring a high degree of safety autonomy, it may be possible to delineate a "safety envelope" for a given set of reactor circumstances within which safe reactor operation can be guaranteed without outside intervention for time periods of practical significance (i.e., days or weeks). The capability to operate a small reactor without the need for highly skilled technical staff permanently present, but with continuous remote monitoring, would aid the economic case for small reactors, simplify their use in remote regions and enhance safety by limiting the potential for accidents initiated by inappropriate operator action.

This paper considers some of the technical design options and issues associated with the use of small power reactors in an autonomous mode for limited periods. The focus is on systems that are suitable for a variety of applications, producing steam for electricity generation, district heating, water desalination and/or marine propulsion. Near-term prospects at low power levels favour the use of pressurized, light-water-cooled reactor designs, among which those having an integral core arrangement appear to offer cost and passive-safety advantages. Small integral pressurized water reactors have been studied in many countries, including the test operation of prototype systems.

1. INTRODUCTION

For more than fifty years, nuclear power reactor technology has evolved to meet the needs of its customers, providing concentrated, competitive energy sources for large-scale energy consumers, particularly electrical utility grids, as well as compact propulsion systems for military vessels, icebreakers and a few merchant ships. This evolution has progressed toward larger reactor units to benefit from economy of scale. The nuclear safety of present-day, large nuclear generating stations (about 1 GW_e or 3 GW_t) depends on active, engineered systems, supported by a complex infrastructure, highly trained personnel and rigid regulatory oversight.

Future growth in the application of nuclear reactor technology, besides addressing the need for electricity in today's developing economies, may come from more-widespread "secondary" energy markets involving smaller reactor units (typically less than 100 MW_t) for district heating, water desalination and small-scale electricity generation in remote areas. A large number of such applications may become attractive, provided nuclear energy generating costs for small-scale systems are competitive with alternate energy sources without compromising safety and reliability.

Significant cost savings can likely be obtained for small power reactors through design simplification and standardization, up-front or "type" reactor licensing, shortened construction schedules, and shop fabrication of major components. However, the "fixed" cost components for essential equipment, such as nuclear instrumentation, and minimum staffing requirements will represent a larger fraction of their generating cost.

Consequently, small reactor designs having features that allow operation with reduced crew size are favoured. Such operation requires a greater level of safety autonomy, that is, the reactor would have to achieve a safe, stable end state by passive means for all realistic upset scenarios without operator assistance and it would have to provide a sufficient response time for any eventual intervention using off-site resources.

To attract a sufficient level of global economic activity to warrant the development of a new small reactor power supply, a flexible design is required that addresses a wide range of applications for both heat and electricity. While unpressurized pool-type reactors are an attractive option for many applications requiring low-temperature hot water for residential district heating, the need to generate electricity with acceptable thermodynamic efficiency using proven, affordable technology, at present, requires higher temperature systems that produce steam.

Therefore, the near-term development of small power reactors ($< 100\ MW_t$) would likely be based on pressurized light water reactor (PWR) technology using conventional steam turbogenerators to produce electricity. This development would involve low technical risk since it would share component technology and global experience from the large power reactor programs.

Among the PWR concepts, the integral PWR (or IPWR), in which the steam generator is contained within the reactor pressure vessel (RPV), emerges as a strong candidate for a simplified, low-cost, compact nuclear power source in the low power range, yet incorporating enhanced passive safety features. Accordingly, this paper considers the safety requirements for the autonomous operation of small power reactors with particular reference to IPWRs.

2. GENERAL REQUIREMENTS FOR SMALL POWER REACTORS

Some important factors to consider in the selection of a small nuclear power reactor technology are:

(i) **Technical Feasibility and Appropriateness.** The technology selected must be well suited to the intended application and the required operating performance should lie conservatively within the envelope of proven performance in similar systems. To gain customer acceptance, reduce project risk and resolve generic licensing issues, it is essential that the technology be proven through the construction, licensing and test operation of prototype systems. The investment cost of such demonstration is likely affordable for a small reactor unit and would be expected to support hundreds of carbon-copy application units. The supplier of the technology must also have a proven track record for product support, as the facility lifetime will be 25-50 years.

(ii) **Safety.** The radiological hazard presented by a small nuclear power reactor is one or more orders of magnitude lower than that presented by a large power reactor by virtue of the former's low fission product inventory. Low power levels and power densities also place much less demanding performance response requirements on safety systems. Nevertheless, not all reactor hazards scale with power output, so that the design must still address the well known nuclear safety design principle of Defence In Depth, satisfy the Single Failure Criterion, and provide adequate redundancy, independence, diversity, and physical separation for achieving the safety functions. Also, the special safety systems must be designed to be fail-safe to the extent possible. A long response time prior to required operator intervention is essential.

In general, the collective risk to society from the operation of a set of $\boldsymbol{n}$ identical, small power reactors, each delivering the same lifetime energy output $\boldsymbol{e}$, but to different populations p_i, must be shown to be less than that from the operation of one large reactor of energy output $E = ne$ to a population $P = \sum_i p_i$ in similar circumstances.

(iii) **Licensability.** There is a strong need for an up-front "type" reactor licence from the nuclear regulator that separates the power supply technology from the specific site location and the intended application. The development of a global small reactor market would be fostered greatly by the establishment of streamlined reactor licensing requirements and uniform acceptance criteria based on international consensus, as revisiting the same issues in each local jurisdiction would not be cost effective.

Existing power reactor regulations in local jurisdictions have evolved in support of specific technologies as required, and may not be entirely applicable or appropriate for alternate technologies and small systems. For example, Canadian power reactor regulations were established for the distinctive CANDU pressurized heavy water reactor system and have prescriptive requirements for dual, independent safety shutdown systems that are completely separate from the regulating system. As such, these regulations are not directly applicable to existing LWR designs.

It is often stated that small reactors should achieve more stringent safety standards than larger units, such as a core melt frequency of $< 10^{-6}$ per year, since their benefit to society is less than that of large reactors.

(iv) **Cost Effectiveness.** Maximum use should be made of high-quality, mass-produced components for large power reactors, such as standard fuel assemblies that are widely available on a commercial basis, e.g., having Zircaloy-clad UO_2 rods with $< 5\%$ ^{235}U enrichment. The overall design should be as simple and as compact as possible to minimize radiation shielding mass and cost, as well as reactor containment and building size and cost.

On-site nuclear infrastructure costs must be minimized, including operating and maintenance staff. In particular, on-site activities and facilities associated with core refuelling, fuel handling and transport, and high-level radioactive waste management should be minimized and simplified; for example, through the use of long-life reactor cores (e.g., as in Rolls-Royce & Associates submarine reactors [1]), the accommodation of spent fuel within the RPV, or the transport of the entire reactor with its containment as a sealed unit to a qualified, central nuclear maintenance facility (e.g., for barge-mounted or marine propulsion systems).

(v) **Public, Investor and Political Acceptability.** Public perceptions of risk are such that the consequences of severe reactor accidents in large power reactors are not generally considered acceptable despite technical assurances that their probabilities of occurrence are very low. In contrast, with a large number of identical small reactors it may be possible to have acceptable consequences to otherwise severe accident events in individual units and lower total consequences despite a greater frequency of occurrence as a result of the large number of units. Improved acceptability may require convincing demonstration to average citizens that a self-evident improvement in safety practice relative to existing large reactors has been achieved.

Even when the safety risk of accidents is limited, the financial burden posed by events such as at Three Mile Island may carry unacceptable risk for investors. Financial exposure would be greatly reduced for utilities operating multiple small units.

Cross-border radioactive contamination associated with the Chernobyl reactor accident has left an indelible impression that nuclear safety is a transnational, global concern, despite its general implementation and regulation on a national basis. Consequently, local political acceptability of reactor technology would be encouraged if the technology were developed and implemented with multinational participation and broad application to a variety of environments (e.g., urban, remote, Arctic, tropical, desert, etc.).

(vi) **Reliability and Maintainability.** High reliability is essential in remote locations where skilled technical resources may be unavailable or very expensive and high availability is necessary to maximize the load factor for cost effectiveness. Accordingly, the design must be simple with a minimum number of auxiliary systems and critical components. Also, it should be tolerant of the failure of individual components, perhaps by the incorporation of factory-installed, redundant spares.

(vii) **Flexibility.** Potential applications for small nuclear reactor systems span a large range:

- electric power generation for small, isolated grid systems, or barge-mounted units,
- heat production for medium- and large-capacity water desalination systems,
- process steam generation for chemical plants and industrial applications, such as heavy oil recovery,
- residential district heating (e.g., low-temperature hot water district heating systems),
- combined systems (cogeneration) for electric power generation plus district heating or water desalination,
- propulsion of large, fast merchant ships, such as container ships and crude oil carriers, or icebreakers, and
- propulsion of small submarines and submersibles for oceanographic research, industrial support such as for off-shore oil drilling platforms, and cargo transport.

3. AUTONOMOUS REACTOR OPERATION

The technical feasibility of autonomous reactor operation has been demonstrated by the successful operation of Russian space satellites powered by small nuclear reactors for periods of up to one year. However, terrestrial power reactors operate in a cost-competitive, public environment and are subject to different regulations that have largely evolved in support of the commercial nuclear power industry with a focus on large generating stations.

From a technical safety perspective, the operation of small nuclear power reactors in a self-governing mode without the continuous attendance of highly skilled operators seems feasible for some reactor designs in well defined situations of limited duration.

Autonomous reactor operation does not mean that the reactor would not be supervised. The physical state of the reactor would be monitored continuously with transmission of essential data to a central station that would observe and record the behaviour of several units simultaneously. The monitoring station might be located at some distance from the actual plant and would be continuously staffed by trained personnel with ready access to the required expertise. Certain safety functions such as reactor shutdown would be possible to initiate remotely, but other actions like reactor restart would require the physical presence of a licensed operator at the reactor site.

A local representative would carry out any instructions received from the central monitoring station and could also serve as an initial contact point for local citizens, should a need arise. Simple, fail-safe means would be provided for the local representative to place the reactor in a safe shutdown state at any time. For example, the local representative would have access to a reactor trip button from outside the reactor room that disrupts electrical power to the control rods allowing them to drop into the core by gravity.

The time interval during which the reactor could be left in autonomous operation would depend on design details and operating needs. For example, periodic operator action would be needed to keep the primary cooling water chemistry from drifting beyond acceptable limits. If need be, a timer clock could be used to initiate reactor shutdown automatically if it is not reset by an operator at the required maintenance intervals.

At present, only a few very-low-power research reactors are licensed by nuclear regulatory authorities for unattended operation with remote monitoring and only for short durations. A good example is the unpressurized 20-kW_t SLOWPOKE-2 [2] research reactor which is licensed by the Atomic Energy Control Board (AECB) in Canada for operation at full power for up to 24 hours with no one in the reactor building and with the reactor room locked. The SLOWPOKE units are located in conventional buildings in high-density population, urban environments such as at the University of Toronto in downtown Toronto. Two TRIGA research reactors have similarly been licensed in other countries for unattended operation [3].

Extending the regime of autonomous reactor operation to higher power levels and to longer operating periods is a significant technical challenge, yet progress to this end seems achievable. For example, an important goal in the plans to develop the 10-MW_t SES-10 (SLOWPOKE Energy System) [4] dedicated heating reactor was to demonstrate the capability to operate for extended periods without an operator in the reactor room, but remotely monitored at all times.

Additionally, several, much-larger advanced light water reactor (LWR) designs, such as the 1000-MW_t Safe Integral Reactor (SIR) [5], claim that no operator action would be required for up to 72 hours following any design basis accident event. Also, the reactors on nuclear submarines routinely operate for patrol periods of about 70 days without outside assistance.

Of course, autonomous operation for extended periods would require that any essential functions normally provided by technical staff, such as instrument calibration or testing the availability of safety systems, would either have to be performed automatically by remote means or not be required for the duration of autonomous operation. Thus, cost savings from reducing staff requirements may be offset to some degree by the need for additional monitoring instrumentation.

Conceptually, the nuclear safety risk presented by a 10-MW_t pressurized water reactor may not be much different from that presented by a 10-MW_t unpressurized pool-type reactor, provided adequate cooling provisions are available to absorb the additional stored thermal energy in the former system when required. However, for pressurized systems, additional regulatory issues arise as a result of existing regulations for the autonomous operation of non-nuclear steam heating plants.

For example, the Power Engineer's Act [6] for the Province of Manitoba in Canada, although it does not apply to nuclear plants licensed by the AECB, allows for the operation of certain steam plants without constant supervision for periods not exceeding 72 hours with written authorization from the provincial Minister of Labour. However, such operation is permitted only for small systems up to a power level of 500 kW_t and a pressure of up to 1030 kPa, provided the boiler is installed in an unoccupied building, is equipped with a full set of safety controls and an approved visual readout system, and the plant and each safety device are tested by a power engineer of the class required.

Additional regulatory considerations and possible design limitations would arise from Non Proliferation Treaty (NPT) related safeguards requirements for significant quantities of fissionable materials and physical security requirements to restrict entry to controlled areas to authorized personnel and maintain adequate surveillance.

Implementation of autonomous operation of a specific reactor system would likely proceed gradually, following a prolonged period of supervised operation to demonstrate appropriate safety behaviour.

4. THE NUCLEAR SAFETY ENVELOPE

Reactor operation involves a continuum of operating states that can be grouped into several bands as shown in Figure 1. The lowest region corresponds to the safe shutdown state. It is followed by a transition zone in which the major plant physical variables, such as temperature and pressure, change during reactor start-up. The next state represents normal power operation. The passage from the shutdown state to normal power operation requires the presence of a qualified reactor operator, but the reverse transition can be performed locally or remotely by deliberate intervention or automatically through actuation of safety trip systems.

Above the normal power operation state is a band representing abnormal operation. The transition to this state could result from the drift of a physical variable, a component failure, or some internal or external initiating event. Operation in the abnormal state does not imply an immediate safety hazard, but it indicates a potential threat to one of the three fundamental reactor safety functions (control, cooling and containment) should further degradation occur. Consequently, alarm notification would occur locally and at the remote monitoring station. Failure to restore normal operation within a specified time period would initiate automatic reactor shutdown.

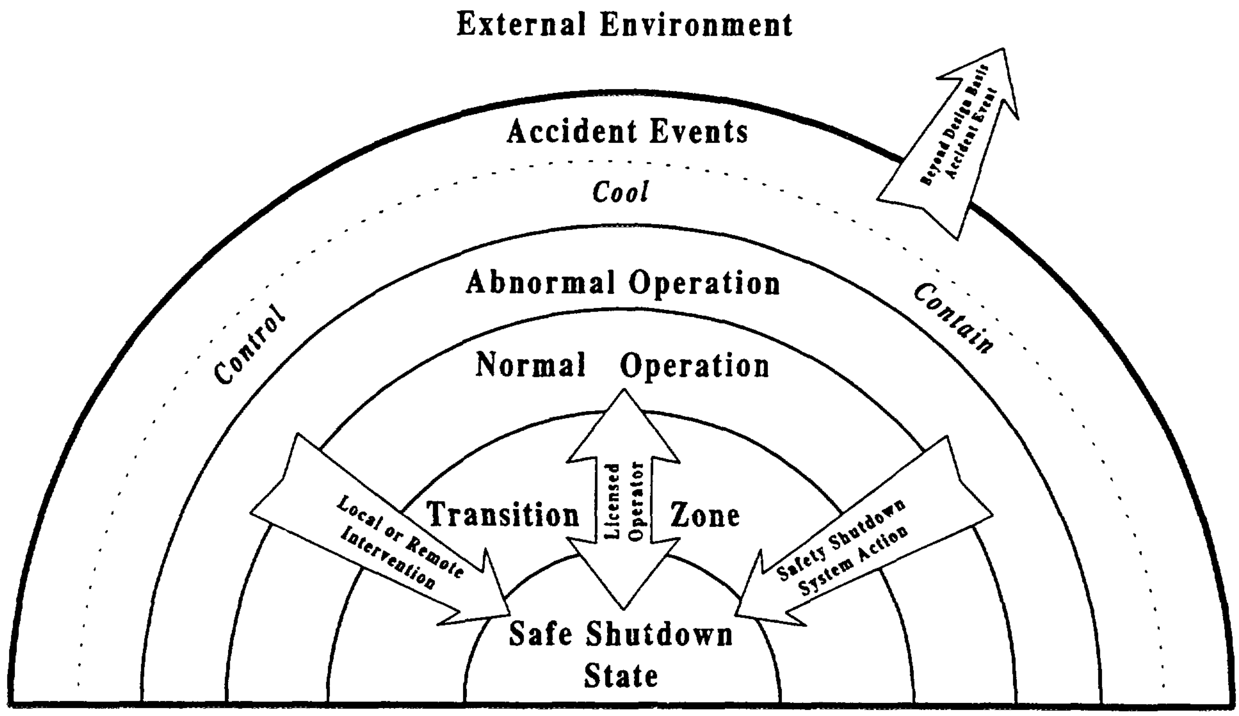

Figure 1. Nuclear Safety Envelope for Autonomous Reactor Operation

The next band corresponds to accidents that directly impact the three safety functions and are not arrested or mitigated by the built-in safety systems, such as reactor trip. Proper application of Defence In Depth may still prevent releases to the environment. However, a trained nuclear accident response team would be dispatched from the central monitoring station.

The upper, open-ended band represents low-probability severe reactor accidents with consequences beyond the facility boundary. It is not expected that any off-site evacuation of the public would be needed under any circumstances for small power reactors, however, minor releases of radioactivity within the specified regulatory limits may occur. The accident response team would secure the site, restore a safe, contained shutdown state, and perform any required cleanup.

Autonomous reactor operation requires that the reactor is in the normal operating state and that the transition to an accident which may lead to a potential release to the environment is sufficiently long for the response team to arrive and take appropriate action. In essence, the presence or absence of staff at the site should have no bearing on the progression of any foreseeable event for an extended period exceeding the authorized period of autonomous operation.

5. REACTOR SAFETY FUNCTIONS

A robust safety envelope requires appropriate attention to the design of the three fundamental safety functions, such that firm limits are imposed on the possible consequences of a wide range of accident events.

5.1 REACTOR CONTROL

Autonomous reactor operation requires intrinsic self-regulation of reactor power. This behaviour can be obtained by negative reactivity feedback such as that provided promptly by Doppler broadening of neutron absorption resonances in ^{238}U resulting from increased fuel temperature, and quickly by effects associated with coolant temperature increases and density decreases (i.e., negative void coefficient), and fuel thermal expansion. However, adequate safety consideration would also have to be given to possible reactivity insertion transients initiated by overcooling events.

In certain exceptional cases, such as for UZrH-type fuel used in TRIGA reactors, intrinsic protection can also be provided against large, rapid reactivity insertion transients, as might occur during control rod withdrawal at cold reactor start-up. In most cases, however, such protection can only be achieved by placing a conservative upper bound on the maximum excess reactivity that could be made available at any given instant. For example, for the SLOWPOKE-2 research reactor, the maximum credible excess reactivity is only about 3.9 mk [2], so that the worst-case reactivity insertion transient that could occur under any circumstance is well below prompt critical (about 8.0 mk for SLOWPOKE-2) and is limited to a value that can be demonstrated to produce acceptable transient behaviour.

For reactors that normally operate at higher power levels, burnable neutron poisons are an appropriate means to help provide the reactivity needed to compensate for long-term fuel depletion. Burnable poisons are an effective way to limit the amount of reactivity that needs to be supplied by movable control devices, provided the poisons are retained in the core with the fuel under all reactor conditions.

However, power reactors require significant amounts of reactivity (i.e., well above the amount needed to go prompt critical if added suddenly) that must be provided by movable control absorber devices (or removable poison dissolved in the primary coolant) under the direction of a licensed operator and following approved procedures during reactor start-up and the transition to equilibrium full-power operation. This positive reactivity is needed to compensate for losses associated with increased core temperature, reduced coolant density including bubble void formation, and equilibrium fission product poison loads, especially ^{135}Xe. Consequently, it is only possible to limit the amount of reactivity that could theoretically be inserted to small, intrinsically safe values when the reactor is already in the normal full-power operating mode with all movable control devices very near their maximum withdrawal positions (and when the dissolved poison concentration is close to zero).

During the specified autonomous operation interval, the control absorber devices could be kept at their maximum withdrawn positions (but available to drop into the core on safety system command). Operation in this manner would ensure that the maximum possible reactivity insertion rate would be limited by the maximum rate at which changes could be made to the physical state of the core, especially coolant temperature and density. Provided such changes are limited in magnitude and can only be introduced relatively slowly, it may be possible to demonstrate that the self-regulating characteristics of the reactor will ensure that any transient overpower is limited to acceptable values and that the stable end-state that is established does not exceed safety limits (e.g., RPV design pressure).

If, for example, the secondary system load demand is also constant during this interval, the primary coolant temperature and pressure would vary only gradually in response to the depletion of fuel (and burnable poison). The only noticeable effect would then be a correspondingly small change of performance output, such as modified energy conversion efficiency. Consequently, near-base-load thermal power reactor operation would support nuclear safety in this circumstance, whereas significant load-following operation could introduce larger perturbations through associated coolant temperature and xenon transient effects that might induce a reactor trip, without control rod movement.

A qualified reactor operator would make small adjustments to the control absorber mechanical stop positions on a routine, periodic basis, but without physical access to the control rods, similar to the procedures presently used with submarine reactors. Care in the mechanical and electrical design of control rods and their drive mechanisms is essential to ensure fail-safe operation and to prevent inadvertent rod withdrawal events and the development of possible rod ejection forces from pressure gradients under any circumstances.

A safety shutdown system would be provided to insert negative reactivity rapidly by dropping the control rods into the core (Note: in the Canadian regulatory environment, dropping the control rods might not be credited as a safety system action since the rods are not physically separate from the control system and are, therefore, used for two purposes). This system would be triggered by abnormal signals for several core parameters, but particularly including neutron flux values and their rate of change. An additional independent shutdown system involving poison (boron) injection would likely be provided, but might be initiated manually.

Safety analysis would need to address the potential for reactivity insertion transients initiated by overcooling events, such as steam outlet header rupture and inadvertent primary pump start-up. Systems relying completely on natural circulation cooling have intrinsic protection against the latter accident event.

5.2 CORE COOLING

The provision of adequate cooling of the reactor core for an extended period of autonomous operation imposes additional safety design requirements to ensure that the core always stays covered by coolant. Thus, a sufficiently large inventory of coolant must be available to absorb the heat generated by the fuel and a means provided to deliver it to the core during postulated accident events that disrupt the normal heat transport pathways.

Design features that enhance the ability to cool the core include:

- a low power density to provide a large fuel heat transfer surface area and a large temperature margin to fuel failure,
- a large primary coolant inventory to provide a slow response to transients,
- relatively low primary system operating temperature and pressure to provide large safety margins, reduced stored thermal energy and slower response to Loss Of Coolant Accident (LOCA) events,
- a thermalhydraulic arrangement that facilitates natural coolant circulation while avoiding the potential formation of vapour-lock flow barriers,
- a passive decay heat removal system, particularly one that does not require active initiation, and
- limitation of LOCA events by using only small diameter piping penetrations on the RPV and locating them well above the top of the reactor core.

5.3 FISSION PRODUCT CONTAINMENT

Defence in depth requires the provision of multiple barriers to the release of hazardous fission by-products from the fuel. In general, these barriers include the fuel matrix itself (UO_2), the fuel cladding, the primary heat transport circuit boundary and a surrounding containment structure. Autonomous reactor operation for a specified time interval requires that all containment barriers are initially intact and that no events are foreseen that could compromise barrier integrity within this duration.

Concerning the fuel cladding containment barrier, special mention must be made of the high-temperature ceramic cladding used with TRISO coated-particle fuel that provides excellent containment behaviour in high temperature gas cooled reactors, even though fuel based on this principle is not presently applicable to PWRs. A similar subdivision of the fuel cladding containment is found in the CARAMEL-type [7] fuel plates used in certain French research reactors.

The primary cooling circuit in a PWR is a high-integrity, pressure-resistant system that will contain any fission products released from the fuel in an accident until the internal pressure exceeds the values that would actuate the pressure relief devices. A simple, compact primary system will be easier to qualify and inspect and to protect from seismic events and external hazards. The RPV penetrations should be as few as possible and of small diameter. All primary system openings would be kept sealed for the duration of autonomous operation.

A small power reactor would have a separate containment structure acting as an additional, leak-tight barrier to the release of fission products to the environment and capable of withstanding the excess pressure that would develop during a design-basis LOCA event in the primary system.

6. THE INTEGRAL PRESSURIZED WATER REACTOR (IPWR)

6.1 THE INTEGRAL REACTOR DESIGN PRINCIPLE

In an ***integral*** or "unitized" reactor configuration, all the primary coolant is kept in the same vessel as the reactor core and the primary-to-secondary heat exchanger or steam generator is immersed in the primary coolant. The discussion of integral reactors in this paper is limited to reactors with primary cooling systems that are sealed in a reactor pressure vessel (RPV), that operate at a pressure significantly above one atmosphere and that use light water as coolant.

A wide range of IPWR core concepts have been studied based on technology from both large PWRs, which typically operate with the primary system pressure at 15-16 MPa with no bulk boiling in the core, and large Boiling Water Reactors (BWRs), which usually operate at a lower pressure of 7 MPa with substantial boiling. Some of the main technological differences arise in the fuel assemblies, which are enclosed in flow boxes in BWRs and use an open-lattice arrangement in PWRs, and the control absorbers, which typically use cruciform control blades in BWRs and rod clusters in PWRs.

Many IPWR designs are neither pure PWRs or pure BWRs, but combine features of both systems as discussed in reference 8 for the specific case of the KWU-NHR 200-MW_t district heating reactor. However, BWR-based designs that use bulk boiling are sensitive to coolant density fluctuations and are generally considered unsuitable for mobile applications, such as marine propulsion. A listing of many of the proposed IPWR designs and their intended applications is provided in Table 1 and discussed briefly in Section 7.

6.2 ORIGIN OF THE IPWR CONCEPT

The IPWR concept is an outgrowth of widespread interest in commercial nuclear ship propulsion in the late 1950s and early 1960s. Economic evaluations of commercial marine propulsion reactor designs based on several distinctly different reactor concepts (e.g., loop-type PWR, gas-cooled, organic-cooled, etc.) concluded that none were competitive at that time with conventional fossil-fuelled propulsion systems, at least in sizes up to 20-30,000 shp (shaft horsepower).

The search for a plant design with reduced capital cost naturally led to a simplification of reactor design by merging the heat exchanger and RPV components to minimize the amount and complexity of the interconnecting piping and thereby achieve a more compact, lighter system. These considerations gave rise to the 60-MW_t Vulcain reactor concept, which originated at BelgoNucléaire in Belgium in 1959 and grew into a joint Anglo-Belgian research program with the participation of the UKAEA (United Kingdom Atomic Energy Authority).

Concerning the choice of design basis for Vulcain, reference 9 notes:

> "... the integral concept, in which the major components of the primary circuit are incorporated in the reactor vessel, was regarded as essential for significant improvement in terms of weight and cost."

In the United States meanwhile, the Babcock & Wilcox Company (B&W) began looking at an integral reactor concept called the Integral Boiler Reactor, also in 1959 [10]. These studies evolved into an integral reactor design concept in 1962 known as CNSG-I (Consolidated Nuclear Steam Generator) based on evaluations of cost, weight and size of the various alternatives. The CNSG-I design formed the basis for the FDR reactor [11] for the German NS Otto Hahn merchant nuclear ship project which was built by a B&W-Interatom consortium between 1964 and 1968.

6.3 GENERIC IPWR DESCRIPTION

In the ideal IPWR system, all the primary reactor cooling system components including the pumps (if required) and pressurizer are incorporated within the single primary reactor pressure vessel. A simplified drawing of a generic low-power IPWR concept that uses natural circulation without any primary pumps is shown in Figure 2.

The natural-circulation IPWR design shown in Figure 2 is similar in its basic operation to small, pool-type natural-circulation reactors, like SES-10. The main difference is the higher operating temperature and pressure of the IPWR and its enclosure within an RPV and a pressure-retaining containment vessel as a result.

Reactor Core

The reactor core of an IPWR generally uses proven technology from loop-type PWRs or BWRs to minimize cost. Typically, the fuel assemblies are square arrays of Zircaloy-4-clad, enriched UO_2 (< 5% ^{235}U) fuel rods with some burnable poison rod (Gd_2O_3) positions.

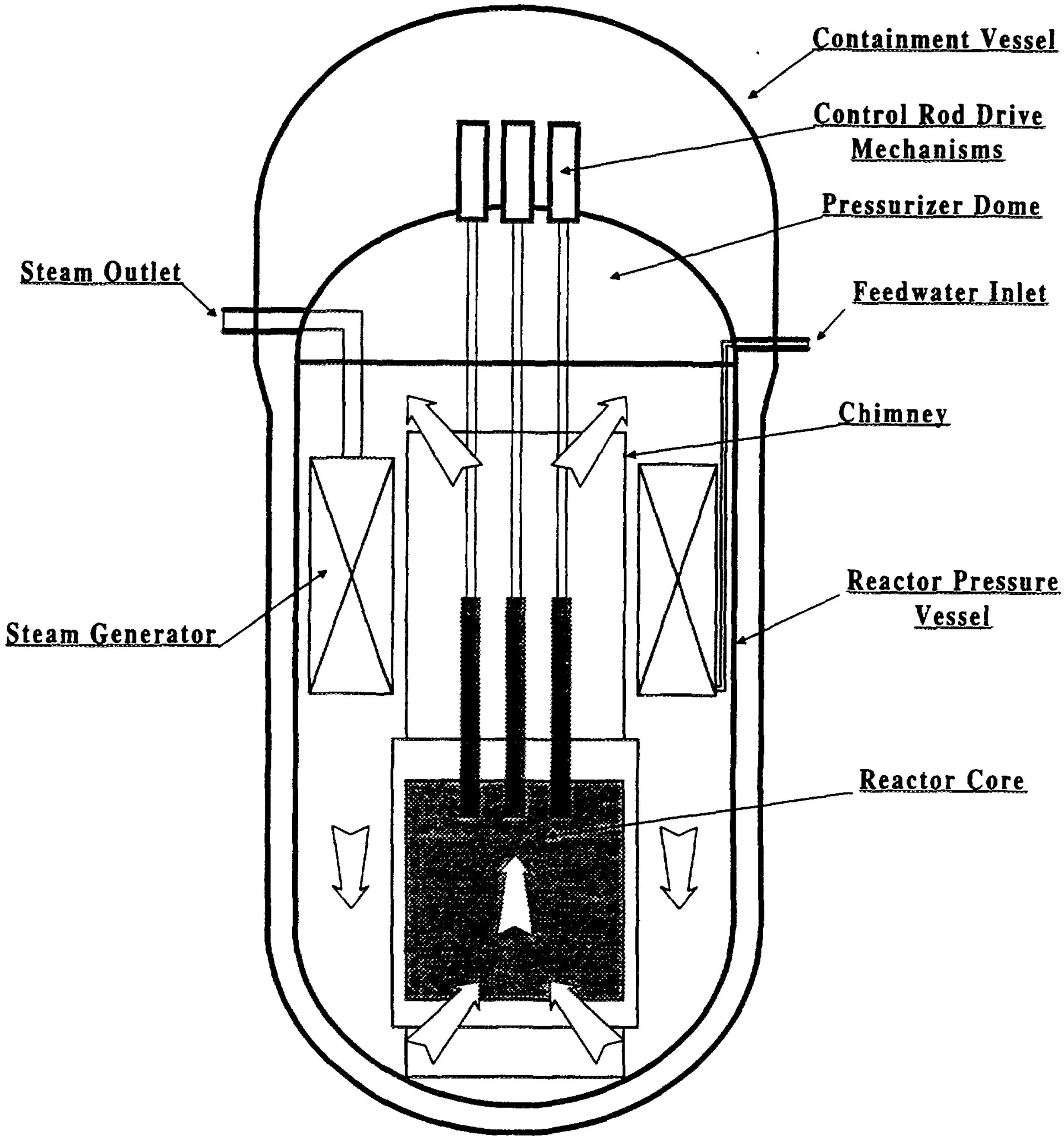

Figure 2. Natural Circulation IPWR

Control absorber rods or blades are associated with the fuel assemblies and their drive mechanisms are located on the RPV top closure head. Safety shutdown is achieved by control rod drop. A boric acid injection system is usually provided as an additional shutdown system for use in emergency situations.

Thermalhydraulic Arrangement

In a typical IPWR, light water coolant enters the base of the core, is heated as it passes upward over the fuel and continues upward through a chimney that provides flow separation from the downcomer region. Self-pressurization is achieved by providing a vapour space at the top of the vessel, such that the vapour phase is in equilibrium with the liquid phase at the core mean outlet temperature. An overpressure of inert gas may be added.

Primary coolant circulation pumps (if needed) are often located on the top RPV closure head for easy access. Primary coolant passes downward over the steam generators in the space between the chimney and the RPV wall and returns to the core inlet. Overall, the thermalhydraulic arrangement enhances natural circulation of the primary coolant.

The primary coolant circulation pumps are typically internal, canned axial pumps, of the glandless, vertical mixed-flow type. Some alternate IPWR arrangements have the pumps located on nozzles or "stems" attached to the lower portion of the RPV, as in the NS Otto Hahn design, and often in the cold leg of the primary circuit.

Two types of steam generators have been used with IPWR designs, both based on once-through designs with the secondary fluid generally on the tube side. The first type, is a helical-wound, cross-parallel-flow, Benson-type steam generator, usually divided into two or more independent sections and wound into a common packet. The second type uses multiple, vertical, straight-tube modules arranged in a circle inside the downcomer annulus.

For applications involving electricity generation, the steam generators deliver superheated steam to the turbines, eliminating the need for steam drums, driers and steam separator equipment. Since the secondary pressure is less than the primary pressure, the steam generator tubes will operate under compression, which provides added resistance to stress-induced corrosion cracking.

Control System

The control scheme for an IPWR is based on maintaining the primary pressure constant and independent of the load in a load-following mode as introduced through the steam generator. The steam generator has a very low heat capacity compared to the entire primary circuit, so that the primary pressure can be kept constant by the control system over the whole load range.

Containment

Although the RPV for an IPWR is larger than that for a loop-type PWR of the same power output, the surrounding containment structure and the reactor building that encloses it are smaller for the IPWR. Indeed, a characteristic feature of many IPWR designs is the use of "dual" pressure vessels as shown in Figure 2, where the containment vessel is close fitting and follows the contours of the RPV, providing clear evidence of Defence In Depth.

A low-volume containment structure that is closely fitted to the lower portion of the RPV ensures that the core will not become uncovered during a LOCA event.

In some larger IPWR designs, the containment vessel may consist of a prestressed reinforced concrete vessel with a steel liner, which eliminates the need to perform difficult welds on thick steel shells. The containment systems for large IPWRs usually include a pressure-suppression system consisting of an arrangement of condensation chambers that are partially filled with water. The advantage of a wet containment system is that the pressure increase will be lower, leading to a lower design pressure and/or smaller containment dimensions.

In certain cases, the IPWR reactor containment vessel is itself immersed in a large pool of low-temperature water that serves several functions in a simple, passive manner:

- provision of biological shielding, especially for fuel handling and storage,
- provision of an ultimate heat sink for decay heat removal from the primary circuit by natural circulation,
- pressure suppression of releases from low-probability severe accidents, and
- scrubbing/filtering of released fission products except for noble gases.

The main factor determining the required response time for operator intervention is the thermal capacity of the ultimate heat sink, i.e., the volumetric water inventory of the pool.

Reactor Auxiliary Systems

The reactor auxiliary systems are similar to those found on other PWRs and typically include: a primary water volume control and inventory system, a primary water purification system, radioactive liquid and gaseous effluent treatment systems, and a ventilation system. At low power levels, many of these systems may be required only on an intermittent basis and would be valved out during periods of autonomous operation.

6.4 IPWR ADVANTAGES

Partly as a consequence of its origins in the competitive civilian marine propulsion sphere, the IPWR design strives for simplicity, cost-effectiveness, maintainability and self-sufficiency in a compact, lightweight and rugged package. Notably, the search for economy in the IPWR design supports enhanced safety (e.g., natural circulation).

The use of self-pressurization based on nuclear heating is desirable to avoid the expense of an auxiliary electric power system, pressure control equipment and the potential for an accident event involving a break of the piping between an external pressurizer and the RPV. Also, the volume inside the RPV serving as an internal pressurizer for an IPWR tends to be larger than that for an external pressurizer with a loop-type PWR and, thus, provides a slower pressure response to transients. Moreover, the integral configuration is more comprehensible to the operator because the water level in the pressurizer always corresponds directly to the amount of water above the core.

The IPWR is amenable to shortened construction schedules because the RPV and internal components can be shop-fabricated, pre-assembled and tested under optimum conditions. In particular, the IPWR configuration eliminates the need for difficult on-site welding work on the primary system. This feature leads to improved quality assurance and reduced construction costs.

The simple, compact arrangement of the IPWR makes it easier to shield efficiently; less shielding is required as a result of the elimination of gamma sources within the primary piping and separate components of the reactor cooling system. Also, the fast neutron dose to the RPV is extremely low because there is a large water gap between the reactor core and the vessel wall. Consequently, radiation embrittlement of the RPV can be ignored. For the same reason, the RPV will present a reduced radiation hazard at reactor decommissioning.

The IPWR configuration is much easier to protect against seismic events on land or against mechanical shock loads in a marine propulsion application. Also, since the RPV has few protrusions and requires no major on-site welding work, it is well suited to siting below ground level. In turn, a low building profile would permit more aesthetic structures and ease licensing concerns for environmentally sensitive areas. Moreover, RPV location below ground provides protection against external hazards.

6.5 IPWR SAFETY ATTRIBUTES

Intrinsic LOCA Resistance

The most significant safety advantage of the integral reactor arrangement is its lack of large primary coolant pipes. The only possible primary piping connections are for coolant makeup, letdown, purification or pressure relief. Even for large IPWR designs these connections would be less than about 8 cm in inside diameter, so that the worst pipe break would produce a relatively slow depressurization. This allows more time to detect the accident and initiate the safety system response.

In addition, primary piping connections for IPWR designs are generally made as high as possible on the RPV, through the top closure head or just below the flange joint on the lower portion of the RPV, so that the core remains covered for as long as possible.

Large Primary Coolant Inventory

The IPWR has a large inventory of primary coolant that is contained entirely within the RPV and, thus, is immediately accessible to the core fuel. A large coolant inventory provides a large heat sink and a correspondingly long response time during accident events to initiate protective measures.

The large primary inventory mitigates the effects of a LOCA event by keeping the core cooled and covered during a blowdown without the activation of a safety system. In a similar manner, the large inventory also serves as a thermal energy storage or absorption medium during a loss of heat sink event, such as a sudden feedwater pipe break.

An additional advantage of a large primary coolant inventory coupled with a smaller secondary coolant inventory is that the magnitude of a cold water reactivity insertion event on primary pump start-up is greatly diminished.

Enhanced Natural Circulation and Passive Shutdown Cooling

The vertical arrangement of the steam generator above the core level permits natural circulation of the coolant through the core and a primary circuit thermalhydraulic design with low flow resistance. Also, any nucleate boiling in the core creates a pronounced buoyancy-induced enhancement of this natural convection flow. Consequently, reactor shutdown decay heat can be passively removed from the core through the steam generators entirely by natural circulation.

In addition, the IPWR configuration eliminates the multiplicity of high points that exist in some loop-type PWR piping arrangements where vapour can collect and potentially block a natural-circulation flow path.

Self-Regulation

Like other LWRs, the IPWR core arrangement has negative reactivity coefficients (i.e., coolant temperature, coolant void and fuel temperature) that promote stable power output.

However, because the IPWR operates with the coolant in the core close to the saturation temperature, a small amount of transient local overheating quickly produces a significant amount of void generation (nucleate boiling) in the core which immediately induces a large negative reactivity response. This feature reduces the core power rapidly in a loss-of-heat-sink event.

Response to Steam Generator Tube and Steam Line Rupture

Since the steam generators in an integral reactor are located inside the RPV and can be isolated individually (if there are more than one in the design), there is no need to reduce the primary system pressure in the event of a steam generator tube rupture, unlike the situation in current large loop-type PWRs.

Steam line breaks are generally not serious accidents since the small amount of secondary water boils off rapidly. The resulting reactivity transient induced by the temporarily enhanced cooling is minor because the primary coolant inventory is large.

In some IPWR designs for district heating applications, the pressure is maintained at a slightly higher value in the secondary circuit than in the primary circuit so that a breach of the heat exchanger tube would not create an immediate leakage of the primary fluid.

7. SURVEY OF IPWR DESIGNS

A partial listing of many of the IPWR designs and concepts that have been considered in various countries over the past 36 years is provided in Table 1 as compiled from numerous information sources.

The entries in Table 1 are arranged alphabetically according to country of origin and in ascending order according to thermal power output within each country group. In this table "multipurpose" refers to designs that have been considered for electricity generation, district heating and water desalination. For these cases, the entry listed in the "output" column usually provides typical data for hybrid applications, such as electricity (MW_e) plus district heating (MW_t) or water desalination (m^3/d).

Although Table 1 may be incomplete or contain some out-of-date information, certain observations are apparent:

(i) The IPWR concept has enjoyed a widespread level of technical interest in many countries over a long period of time. In recent years, this interest is particularly strong in Russia.

(ii) The range of applications considered is extremely broad and includes marine propulsion (particularly France and Japan, in recent years), district heating (especially China), and cogeneration or multipurpose use (Russia).

Table 1. Listing of IPWR Designs and Concepts

Country	Designation	Application	Thermal Power (MW_t)	Output	Status/Comment	Reference
Argentina	CAREM	multipurpose	-	15-150 MW_e	concept	12
Belgium/UK	Vulcain	marine propulsion	60	-	early concept	9
China	NHR-5	district heating prototype	5	-	operating, Beijing	13
	NHR-200	district heating	200	-	design being licensed	14
France	SCORE	marine propulsion, electricity	10	<2 MW_e	concept	15, 16
	CAS-48	marine propulsion:Rubis/Amethyste	48	-	~6 units operating	17
	THERMOS	district heating	100	-	concept	18
	CAP-70	electricity	-	70 MW_e	concept	19
Germany	FDR	marine propulsion: NS Otto Hahn	38	460 kW_e + 11000 shp	decommissioned	11
	IPWR-38	multipurpose	38	6.7 MW_e + 10000 m^3/d	concept	20
	IPWR-138	multipurpose	138	19.5 MW_e + 40000 m^3/d	concept	20
	KWU-NHR	district heating	200	-	concept	8
	IPWR-220	multipurpose	220	38.5 MW_e + 60000 m^3/d	concept	20
	EFDR	electricity	275	100 MW_e	concept	11
Japan	DRX	marine propulsion	0.75	150 kW_e	concept	21
	MRX	marine propulsion	100	-	concept	22
	ISER	electricity	645	210 MW_e	concept	23
	SPWR	electricity	1100-1800	350-600 MW_e	concept	23, 24
Russia/CIS	GAMMA	electricity, prototype	0.22	6.6 kW_e	operating, Moscow	25
	ELENA	district heating, electricity	3	70 kW_e	concept	25
	ABV-1.5	cogeneration	12	1.2 MW_e + 5 MW_t	concept	26
	AST-30B	district heating	30	-	concept	27
	ABV-6	multipurpose	48	12 MW_e / 6 MW_e + 14 MW_t	concept	27
	ABV	multipurpose	60	9 MW_e + 14000 m^3/d	concept	28
	ATETS-80	multipurpose	250	80 MW_e	concept	28
	AST-500M	district heating	500	-	concept	29
	ATETS-150	multipurpose	536	150 MW_e + 70000 m^3/d	concept	30
	ATETS-200	multipurpose	690	200 MW_e + 60000 m^3/d	concept	28
	B500 SKDI	electricity	1350	515 MW_e	supercritical pressure concept	31
	CHPP	cogeneration	1650	500 MW_e + 522 MW_t	concept	32
	VPBER-600	electricity	1800	630 MW_e	concept	28
Switzerland	SHR	district heating	10	-	concept	33
UK/US	SIR	electricity	1000	320 MW_e	concept	5
US	NCIR	electricity	-	10 MW_e	concept	34
	CNSG-IVa	marine propulsion	314	120000 shp	concept	10, 35
	CNSS	electricity	1200	400 MW_e	concept	36

(iii) The thermal power outputs considered in the various IPWR designs span a broad range from 0.22 to 1800 MW_t. Of the 36 reactor concepts listed, about 50% belong to a low-power category ≤ 100 MW_t, including seven very-low-power designs and concepts that are ≤ 12 MW_t.

(iv) Despite the large number of IPWR designs and concepts listed, very few have been actually constructed. The exceptions are very-low-power experimental prototype units in China and Russia (≤ 5 MW_t) and marine propulsion systems of intermediate power levels (< 50 MW_t) in France and Germany.

The very low power range ≤ 12 MW_t includes NHR-5, SCORE, DRX, GAMMA, ELENA, ABV-1.5 and SHR. All of these small IPWR designs use natural circulation of the primary coolant without pumps and incorporate passive safety features such that some degree of limited autonomous reactor operation may be achievable.

8. CONCLUSION

Small power reactors may offer attractive solutions to a wide range of energy needs provided overall costs are competitive with alternate power sources. Operation of small power reactors with limited on-site nuclear infrastructure and operating staff would reduce generating costs, if the necessary technical resources and support can be obtained from centralized sources when needed and if the cost of this external support is distributed over a large number of identical field application units. Such operation requires a small reactor design incorporating a high degree of safety autonomy, so that there is a long transition time to any state requiring operator intervention, and continuous remote supervision.

Among the existing power reactor designs capable of producing electricity and steam, the IPWR appears well suited to a wide range of applications at low power levels. Although it has been widely studied, the present implementation of IPWR designs is mainly for low-power prototypes and marine propulsion systems. At very low power levels (≤ 10 MW_t), IPWR systems using natural circulation of the primary coolant appear to be technically capable of achieving safe, autonomous operation for limited periods. However, regulatory issues arise concerning the supervision of pressurized systems of any type. Widespread use of small IPWRs with limited on-site supervision would benefit from the establishment of streamlined reactor licensing requirements and acceptance criteria based on international consensus, as well as prior experience with similar modes of operation with unpressurized pool-type reactor systems of comparable capacity. This paper reflects the personal opinions of the authors.

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NEW SAFE REACTOR, BUT NOT FIRST OF A KIND ENGINEERING

M. NURDIN
National Atomic Energy Agency,
Jakarta, Indonesia

Abstract

NEW SAFE REACTOR BUT NOT A FIRST OF A KIND ENGINEERING is a new reactor concept to fulfil the need on Small Reactor for power generation, both for electricity and for co-generation.

Nuclear reactor system of this concept in certain degree has similar design compared to the established and successful reactor systems now in operation; so the material used for the same function and purpose is not the same.

The strategy or choice adopted in achieving this concept will be automatically shown by the inspiration or philosophy of "not to re-invent the wheel."

Based on the above mentioned strategy, a certain degree of experimental verification and justification are of course needed/necessary to know better the deviations and the differences from the existing nuclear reactor concepts and further to anticipate of course precisely engineering behaviour of the proposed concept.

Physical and engineering discussion on the proposed concept are main objectives of this paper in which most of the scope and objectives of this IAEA TCM on Small Reactors with Minimized Staffing and/or Remote Monitoring are elaborated. They are discussed in such a way to give the technical and economical background of the proposed concept.

I. INTRODUCTION

Indonesia has taken its second 25 years development plan since 1994, now is looking forward to improve its energy situation and its economy.

As can be seen in Figure 1., Indonesia is an archipelagic country consisting more than 13 thousand islands and having almost 200 million population.

Due to geographic distribution of resources of energy and lack of uniformity of population concentration in the country, the demand for electricity leads Indonesia to consider that the nuclear energy is the energy for the future.

For the most populated and developed island, the size of the nuclear power station is beginning from 600MWe, but for the other small islands and for some isolated areas, small power reactors are compatible.

To fulfil such demands, since 1991, Indonesia has carried out the complete Feasibility Study and Site Investigation for its first nuclear power plant and beginning from 1994, Bid Invitation Specification has also been undertaken. Both studies will be completed next year, and in Figure 2., time schedule for the first studies are illustrated.

THE MAP OF INDONESIA

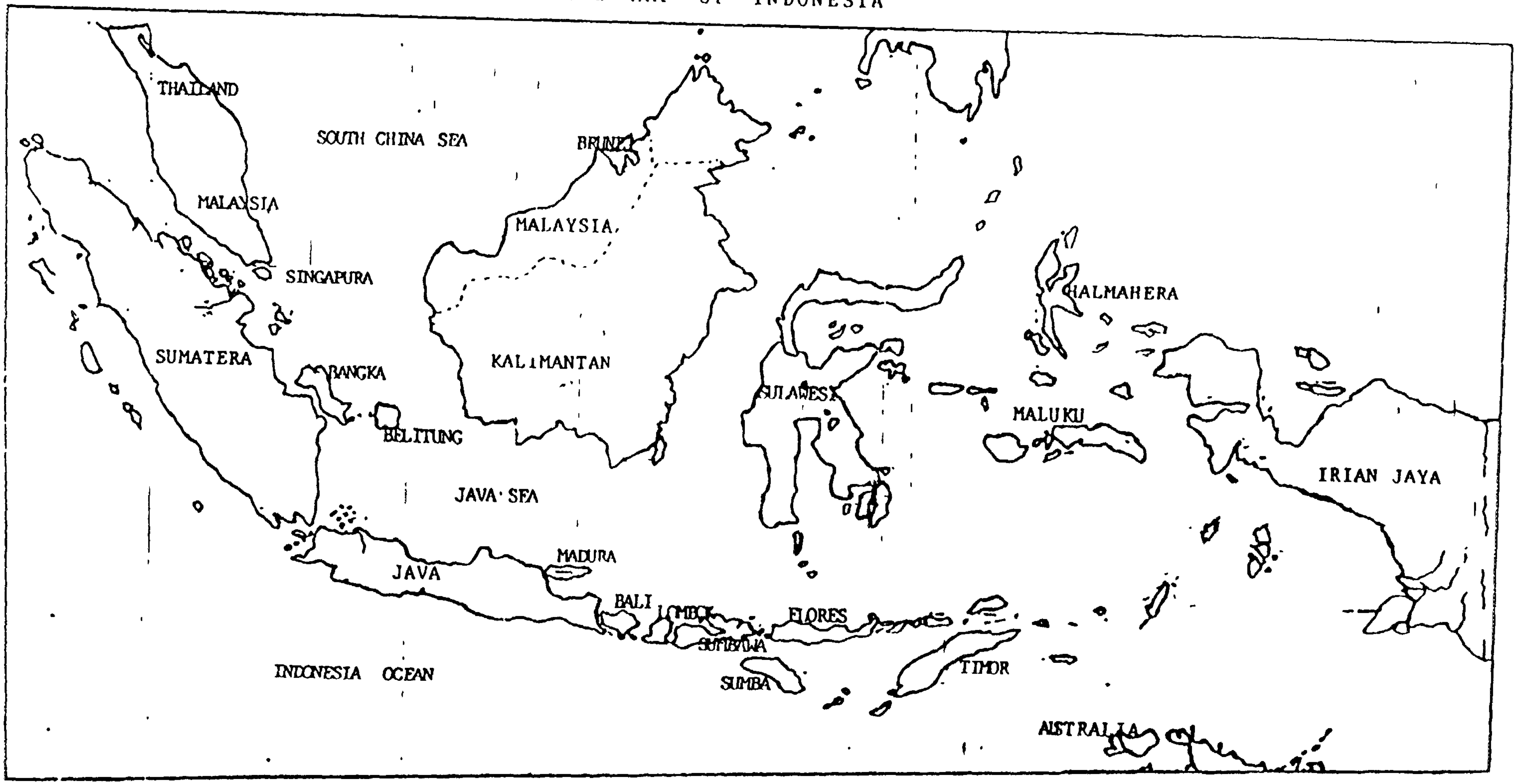
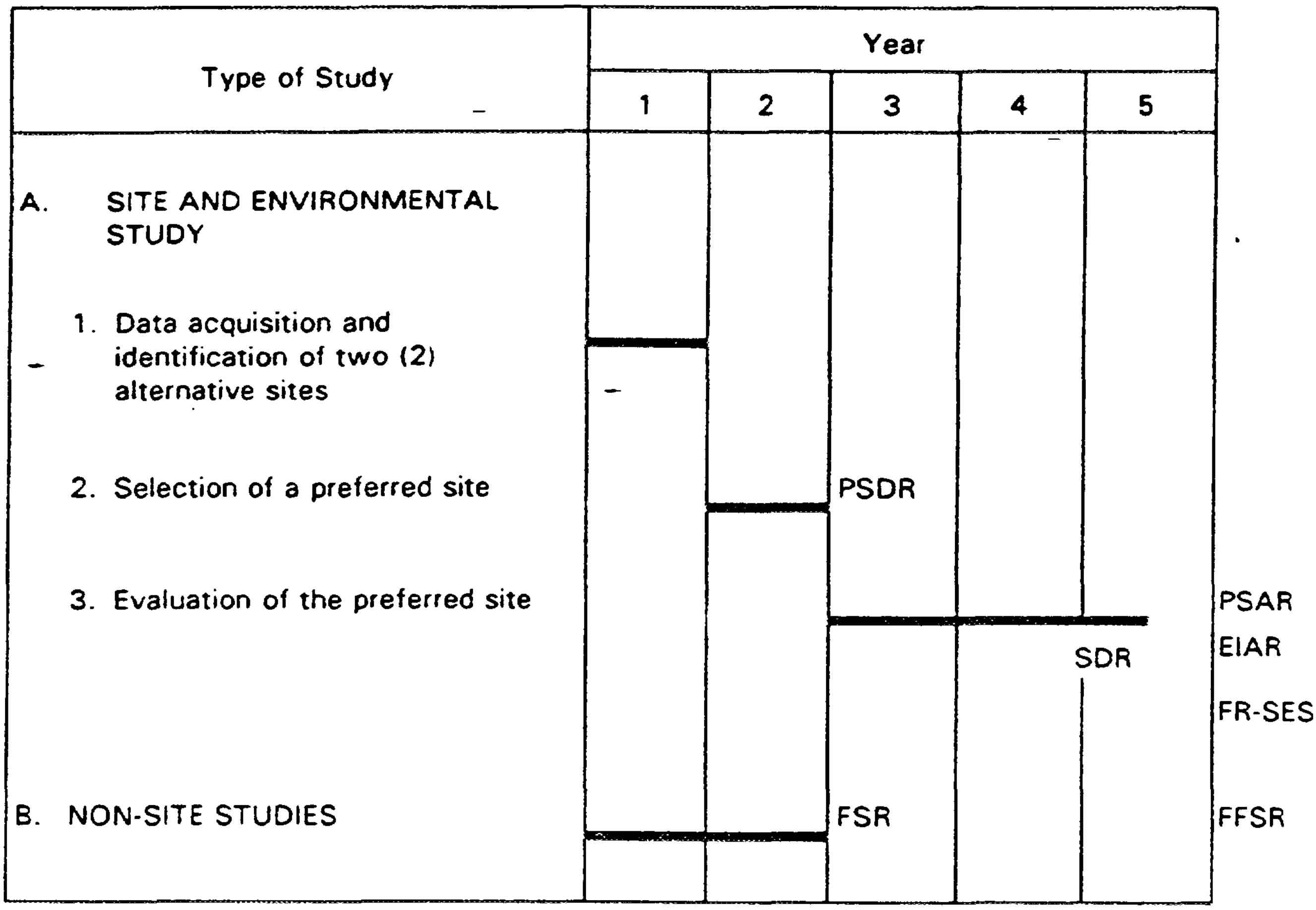


FIG. 1.



PSDR - Preliminary Site Data Report
FSR - Feasibility Study Report
SDR - Site Data Report
PSAR - Preliminary Safety Analysis Report (Site Part)
EIAR - Environmental Impact Analysis Report
FR-SES - Final Report of Site and Environmental Study
FFSR - Final Feasibility Study Report

FIG. 2. Time schedule of the feasibility study for a nuclear

The study for small nuclear power reactor has been initiated since late eighties, using light water and helium gas as the coolant, and since 1990, BATAN (National Atomic Energy Agency) as a responsible agency in this matter has set up a team dedicated to study related aspects of a small (mini) nuclear power reactor using light water as the coolant.

A group studying High Temperature Gas Reactor for enhance oil recovery in 1987 has become a team dedicated for high temperature reactor since 1993.

Along the line of mini reactor, the study on liquid metal reactor (Pu-burner) has also been taken place. The activity of this group was firstly communicated at IAEA meeting in Djerba Zarzes, September 6-10, 1993. the philosophy "not re-inventing the wheel" is well adopted. Most of technical or engineering aspects used in this concept are coming from proven and experienced reactor systems, so that, due to time frame, it is expected that this concept challenges the needs for small reactor in many island and isolated areas in Indonesia.

Regarding to the philosophy adopted in exploring this concept, it enables us to consider that this concept does not belong to the first of a kind engineering. To realize the prototype or the first reactor of this concept, Indonesia recommends an international cooperation under the IAEA frame work.

II. ENGINEERING CONCEPT OF THE REACTOR

From thermodynamic point of view, the higher enthalpy will give higher efficiency; and to do so, normally the thermodynamic conditions of the working fluid is translated directly to the operating condition of the primary loop of heat transport system.

High temperature and pressure steam in the Pressurized Water Reactor's (PWR) system are below the temperature and pressure of the primary coolant, on another word, to have better efficiency, we are obliged to increase the temperature and pressure of the primary system. This is the origin of the problem, where to handle complication come up, as a result of reactor operating conditions, the system becomes sophisticated, and in the same time, safety and safety related items increase and the choice of material used becomes more and more difficult.

The most recent examples is the phenomena of stress corrosion cracking found at the vessel head of some PWRs. The remedy was made by replacing material around the penetration, from inconnel 600 (T600) to inconnel 690 (T690) and the same change was also proposed-envisaged for Steam Generator piping.

The system envisaged for small islands and for isolated areas taking into account the conditions suggested by International Atomic Energy Agency (IAEA) Technical Committee Meeting (TCM), the system must be simple, be easy to operate and be easy to maintain and repair. To obtain this goal, the system has to have minimum safety and safety related items, but it does not mean that the plant safety is not adequate, compared to the existing reactors in operation.

From the considerations and criteria mentioned above, the system should always be capable of producing high temperature steam, it means that the primary coolant temperature is high and even higher than in PWRs. To minimize the pressure and to keep the function of the plant for energy production as economic as possible, lead us to use liquid metal as a coolant. Since the boiling temperature of the liquid metal is relatively high, the reasonable modification can be carried out, i.e. that the primary coolant pressure is atmospheric pressure and the temperature can be adjusted according to the type of the coolant, fuel element used, steam quality recommended and the objectives of the installation as a whole.

Another important factor is the capability of the reactor to accommodate the transient. Whatever it is the cause of the transient, the core will not produce any unstable of any power excursion. To enhance the leakage of neutron during increase of the power, both at normal or abnormal condition, the core design must have the annular part. This annular part must be able to absorb the excess neutron coming in, and if the reactor becomes bigger and bigger, this part can be used to facilitate the heat sink, in case of emergency.

The primary circuit is isolated from the secondary ones by using an intermediate heat exchanger, so the heat (energy) is transferred from liquid metal to the same liquid metal at another side. Intermediate loop is designed as such, if there is a leakage, the primary liquid metal will never come to the intermediate loop, so that the contamination of the intermediate liquid metal is expected not to occur.

III. CHOICE OF COOLANT, MODERATOR AND FUEL ELEMENT

1. **Reactor Coolant:**

To have high temperature at atmospheric pressure conditions, unitization of liquid metal as the reactor coolant is inevitable and represent a proper solution.

So far, liquid metal used as the reactor coolant is liquid sodium; it's use is well known internationally. It's chemical, physical and engineering properties are recognized in detail, in which, those properties impose many restrictions.

Considering the problems imposed by the unitization of liquid sodium, another liquid metal having physical, chemical and engineering properties affordable in technical point of view, should be introduced.

In this case, liquid metal lead having no reaction with the air, transparent and from neutronic point of view has very low neutron absorption (isotope Pb 208) is interesting, to be used as the reactor coolant. It's melting point is around 320C and it's boiling temperature is about 1500C.

Using the inlet coolant temperature of 360C, according to the design, the outlet coolant temperature can range between 400C and 750C. It depends on the objective and on fuel element used, the outlet temperature of the liquid lead can be adjusted. In this case, desalination does not belong to the objective; and the high pressure and temperature steam can be used for electricity generation and for enhancing oil recovery.

Assuming that in many islands and isolated areas in Indonesia, the main objectives are only in the need for electricity and potable water; the reactor inlet temperature must be adjusted to enable us to use the steam produced by the installation. In this case, mix of lead and bismuth is inevitable, making the alloy of Pb-Bi, the melting point will reduce. Lead bismuth entectic melts at 125C and various tritectics have melting point in the mid ninety Celsius range.

With inlet coolant temperature around 150C, it can produced the outlet coolant temperature beginning from 250C to 450C.

The second scenario is much more realistic compared to the first one, if the reactor is intended to supply electricity and to produce desalted water.

Dealing with the material that will be used in the reactor core, it's pure concentration on the degree of impurity must be well verified, in order the possibility of contamination can be reduced as much as possible.

2. **Reactor Moderator:**

There are two types of moderators, the first type is liquid moderator like light water, heavy water and another type is the solid one like graphite.

According to the operating conditions derived from the possible objectives elaborated above, utilization of liquid moderator is impossible, and it is automatically that the more relevant one for these objectives is graphite.

Graphite material is an effective moderator and can be used in high temperature environment. The compatibility between graphite and liquid lead is very amazing, where the solubility of lead in graphite is very low.

Graphite block having the hole in the centre, as the annulator part, will of course contain many holes (smaller than annular part) to accommodate fuel channel, control rods, instrumentations, etc. Since temperature maximum of the operating condition is much less than 1000C, it is not necessary to be cooled.

The core structure is similar to the CANDU core structure, but for this reactor system, the core is vertical, where the absorber material can be inserted by gravity force and passive safety features can be explored as much as possible.

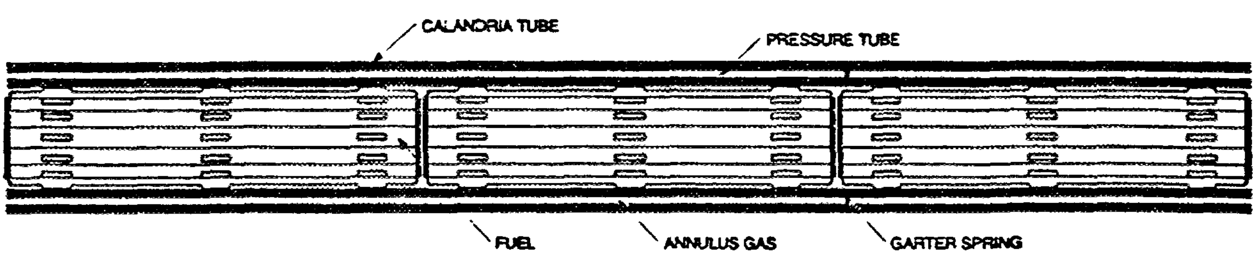
3- Reactor Fuel:

As explained in the above paragraph, fuel element should be introduced into the fuel channel, and this can be carried out through the upper part of the reactor core.

The pitch of the fuel may be the same or greater that in CANDU, but the length of fuel bundle is the same. The fuel assembly design adopted is coming from CANDU6 fuel design; where in one bundle exists 37 rods of fuel, and each rod has the same outer dimensions. It is worth to mention here that the cladding thickness may be different.

The number of fuel bundle per each fuel channel is the same, but they are surely less than 12 bundles as used in CANDU; and they are not subject to daily refuelling methods. Fuel management patterns will be defined through burn up calculation. One third of the fuel channel will be unloaded and loaded at the end of a cycle and due to the fuel management pattern defined before, out-in shuffling is undertaken. The fuel bundles in the fuel channel will look like as follows:

FUEL CHANNEL DETAIL WITHIN CALANDRIA



In this concept, there is no annulus gas, garter spring and calandria tube unifies with pressure tube. The main reason for this simplification is that the operating pressure of this system is atmospheric pressure and it has graphite moderator, which in case of leakage, there in no negative effect in the neutronic point of view and besides that the solubility of lead in the graphite is very low.

The choice of the fuel meat is determined by the conditions imposed to the system in the aspect of reactor operation, maintenance and safety. These conditions oblige the fuel element to have the following features:

- excellent prompt negative coefficien.
- good performance record.
- largely used and good availability.

Triga reactors have been used worldwide, both in developed and in developing countries, so they have been operated in more than a dozen countries. Indonesia has operated Triga reactor since 1964 at Bandung Reactor Centre.

Triga is a research reactor, operate at low temperature and at atmospheric pressure, and so far, they have developed-exported 5 types of fuel elements, namely:

- F 104 using 8.5% uranium content
- F 106 using 12.0% uranium content
- F 20/20 using 20.0$ uranium content
- F 20/30 using 30.0% uranium content
- F 20/40 using 45.0% uranium content

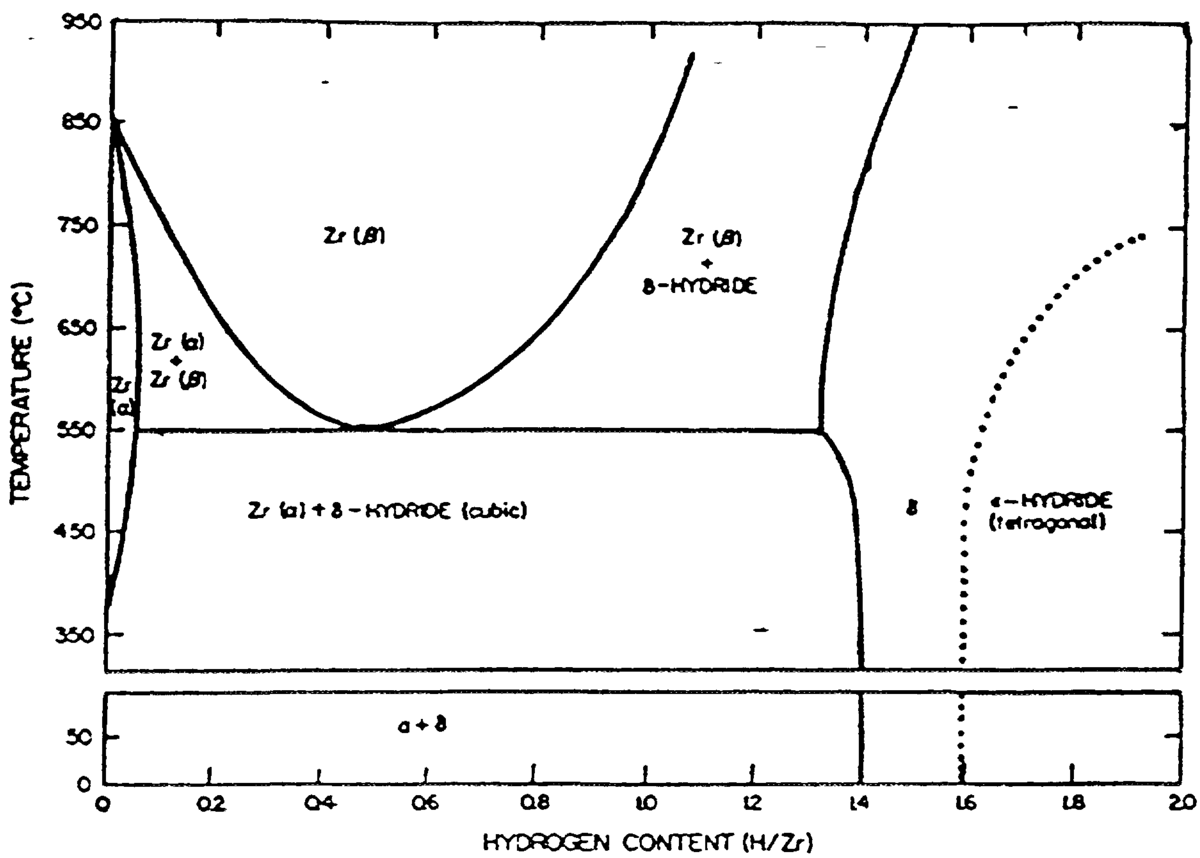
The first 4 types are in the form of fuel rod, and the last one is in the form of cluster, consisting of 16, 25 and 36 fuel rods. The dimensions of a cluster is more or less the same with outer dimension of MTR plate type fuel.

Triga fuel use UZrH as the fuel meat, the hydrogen bound in the atom Zirconium oscillate, and when epithermic neutron comes into head on collision with that hydrogen atom, the neutron will lose its energy and become thermal neutron and can enhance the fission reaction.

When the power increases, the hydrogen oscillates faster, at that situation if a neutron come and head on collision takes place, the neutron gets some more energy, so it becomes faster and faster. This fast neutron will lower the probability to have fission reaction, and so, the power decrease.

Prompt negative coefficient of Triga fuel is well known, and for fuel having higher uranium content, they use Erbium as the burnable poison to compensate excess reactivity.

Potential application of Triga fuel for power reactor (at high temperature) can be justified through the phase diagram of the fuel, as below:



From the phase diagram illustrated above, at around 1000C, Triga fuel having hydrogen to zirconium ratio equal 1.6, the fuel still stable; and mechanical properties at high and low temperatures are still coherence.

IV. CONCLUSION

1. As seen in this study, almost all sub-systems have been utilized in the other reactors that have already been in operation since many years.

2. Utilization of lead liquid metal and lead bismuth liquid alloy is still limited, verification is necessary.

3. Since the condition of reactor operation is determined by the objectives, choice of the coolant between lead liquid metal and lead bismuth liquid alloy can be derived according to the objectives taken into action.

4. Configuration of the whole system can also be obtained from many exercises and with the references to the existing or to the proposed concepts already discussed in many meetings and seminars.

5.. International cooperation will enable the realization of the project, share on the risks and on the market available can facilitate the task.

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A FIVE MW NUCLEAR HEATING REACTOR

ZHANG DAFANG, DON DUO, SU QUINGSHAN
Institute of Nuclear Energy Technology,
Tsinghua University,
Beijing, China

Abstract

The 5 MW Nuclear Heating Reactor (NHR-5) developed and designed by the Institute of Nuclear Energy and Technology (INET) and has been operated for four winter seasons since 1989. During the time of commissioning and operation a number of experiments including self-stability, self-regulation and simulation of ATWS etc. were carried out. Some operating experiences such as water chemistry, radiation protection, and environmental impacts and so on, were also obtained at the same time. All of these demonstrate that the design of NHR-5 is successful.

1. Introduction

The 5MW Nuclear Heating Reactor (NHR-5) developed and designed by Institute of Nuclear Energy Technology (INET) has been put in operation for four winter seasons The construction of NHR-5 began in March 1986, the civil egineering was completed in september 1987 and the erection of NHR-5 were finished in April 1989 The initial criticality of NHR-5 was reached in Nov 1989 and full power operation in Dec of the same year

In order to expand the utilization of NHR and to improve its economical competition, the operational experiments of cogeneration - heat and electricity and refrigeration for air condition using nuclear steam from NHR-5 were carried out in 1992

The mile stone of NHR-5 is listed in Table 1

Table 1 The mile stone of NHR-5

Beginning of construction	Mar 1986
Completion of civil engineering	Sep 1987
Completion of erection of reactor	Apr 1989
Beginning of commissioning	May 7, 1989
Initial fuel loading	Oct 9, 1989
Initial criticality	Nov 3 1989
Full power operation	Dec 16 1989

The operational practice shows the NHR - 5 has excellent operation and safety features and a high availability of 99% The parctice also shows the NHR-5 is easy to start up and to be operated The operation results demonstrated that the NHR-5 has fully reached the design requirements and the main design parameters Table 2 gives the main operation parameters in comparison with the design values In the table the operation temperature of the reactor inlet is higher than the design value which shows the reactor has a larger natural-circulation capability

Table 2 Main operation parameters of the NHR-5

	Design value	Operation value
Reactor thermal power	5MW	5MW
Reactor		
Outlet temperature	186°C	186°C
Inlet temperature	146.6°C	151°C
Pressure	1.37MPa	1.37MPa
Intermediate circuit		
Primary heat exchanger		
Outlet temperature	142°C	144°C
Inlet temperature	102°C	100°C
Flow rate	107t/hr	97 t/hr
Intermediate heat exchanger		
Outlet temperature	75.2°C	80°C
Inlet temperaure	142°C	144°C
Flow rate	64 t/hr	67 t/hr
Heating grid		
Outlet temperature	90°C	84 °C
Inlet temperaure	60°C	56 °C
Flow rate	143t/hr	152t/hr

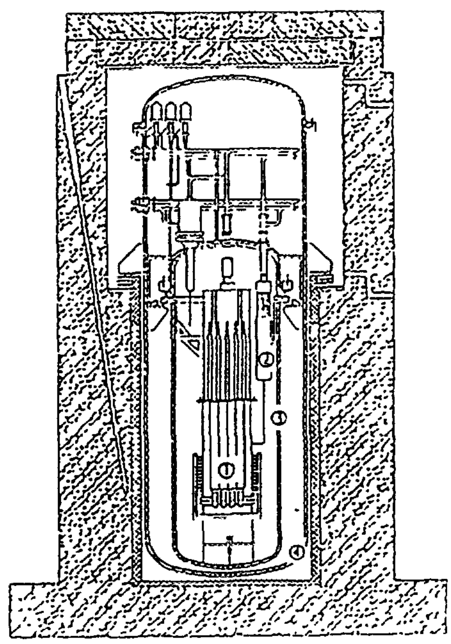
2. Description of NHR-5

The NHR-5 is the first vessel type heating reactor in operation in the world. It is an integrated vessel type light water reactor cooled by natural circulation with self-pressurized performance.

2.1 Structure of NHR-5

Integral and natural circulation

The core and main components of primary circuit are housed within a reactor pressure vessel (RPV) The reactor core is located at the bottom of a hanging barrel, underneath the hanging barrel, a secondary support is placed in the bottom of the vessel There is a long riser above the core outlet to enhance the natural circulation capability There are four primary heat exchangers in the downcomer between the riser and the vessel wall. The reactor core is cooled by natural circulation and the carried heat is transferred to the intermediate circuit vir primary heat exchangers



①core ②primary heat exchanger
③RPV ④containment

Fig 1 The NHR-5 structure with dual vessel

Dual pressure vessel

The dual pressure vessel is adopted in the design of NHR-5 The reactor pressure vessel is designed for an operation pressure of 1 5MPa Outside the RPV, a second metalic vessel-containment(quard vessel) is mounted The design pressure is 1 5MPa at the temprature of 177°C. The gap between the RPV and containment is very small All RPV penetrations are located higher than core outlet atleast 3m, and there are no large -bore pipes

All of these measures can void and mitigate serious consequences which result from the loss of coolant accident. If the RPV had been broken at its bottom the core can also be covered with water The Fig 1 shows the reactor structure with dual vessel The main parameters of dual pressure vessel is listed in Table 3

Table 3 The main parameters of dual pressure vessel

Pressure vessel		
ID	m	1 8
Total height	m	6 5
Working pressure	MPa	1 5
Working temprature	°C	198
Material		22g
Lining thichness (Braze welding)	mm	~ 6
Thichness of cylinder	mm	90
Total weight	t	35
Containment(quard vessel)		
ID	m	2 8
Total height	m	9 5
Thichness of wall	mm	20
Design temperature	°C	177
Design pressure	MPa	1 5
Material		16MnR
Weight	t	29

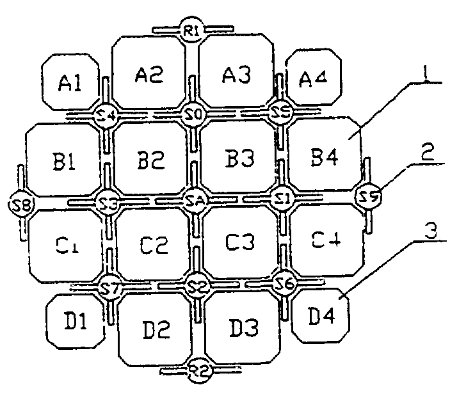
Self-pressurized system

A space above the coolant level inside the RPV is as a self - pressurized space (~1m) The pressure inside the RPV is depends on initial partial pressure of nitrogen and saturate vapour pressure corresponds to the core outlet temperature in the pressurized water operation mode Due to the nitrogen partial pressure existing the coolant can be kept subcooling in the core outlet This is called pressurized water operation mode

2.2 Overall Arrangement of Reactor Core

Reactor core

The core cross section of NHR-5 is shown in Fig 2 In the core there are 12 fuel assemblies with 96 fuel rods and 4 with 35 fuel rods The fuel rod with cladding of Zircaloy-4 has an active length of 690mm and a diameter of 10mm The nuclear fuel is uranium dioxide with an enrichment of 3% The total amount of UO_2 loaded in the core is 0 508 tons



❶ Assembly with 96 fuel rods ❷ Control rod
❸ Assembly with 35 fuel

Fig 2 The cross section of NHR-5

Control of reactivity

The reactivity is controlled by a combination of fuel rods containing fixed burnable poison of 1 5% Gd_2O_3, movable absorption rods(boron carbide) and negative reactivity efficient In the core there are 13 control rods which are all driven by a new hydraulic driving system. The control rod driving system consists of three parts: an actuating loop outside the containment, 13 hydraulic step cylinders in the core and two control units (combine valves). The control rods can be dropped into the core by gravity when the reactor shutdown is needed. Boron injection system as a standby shutdown system is initiated by pumps or pressurized nitrogen during event of ATWS.

2.3 Main Heat Transfer System

The main heat transfer system is composed of three circuits, i.e. the primary circuit, the intermediate circuit and heat grid. The intermediate circuit is singl loop which connects with the primary circuit and heat grid via the four primary heat exchangers and 2 intermediate heat exchangers. 4 primary heat exchangers are divided into two groups in parallel operation, which are merged single loop through isolating valves. The operating pressure in the intermediate circuit is higher than the primary circuit which can keep the heat grid free of radioactivity. Heat generated in the core is transferred to the heat grid via the intermediate circuit. The main heat transfer system is shown as Fig.3.

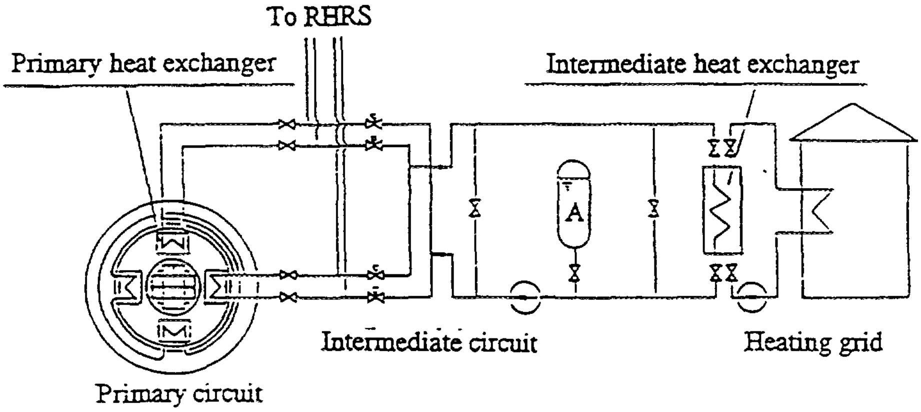


Fig. 3 Main heat transfer system of NHR-5

2.4 Residual Heat Removal System

The residual heat removal system (RHRS) of the NHR-5 consists of two independent trains which assigned to two groups of primary heat exchangers. There are three natural circulation cycle for each train. Figure 4 shows the schematic system diagram of the RHRS. After reactor shut-down the decay heat will be transferred to the intermediate circuit via the primary heat exchangers. Then the heat carried is going to a vaporizer located at a high position in the reactor hall. This is the first natural circulation cycle. The second one consists of the vaporizer,air cooler and related pipes and valves. Finaly, the decay heat can be discharged to the atmosphere via the air cooler located on the roof of the building by natural convection of air

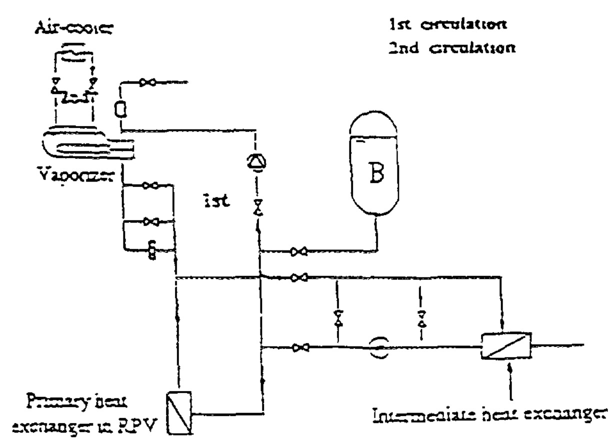


Fig 4 Schematic system diagram of RHRS

3. Operatinal Experience of NHR-5

3.1 Reactor operation condition

Start up
Star up of the NHR-5 is a process from cold condition to the expecting operation state by means of nuclear heating itself. During the start up process three things have to be made i.e. to set up initial partial pressure of nitrogen in RPV.to limit the rising temperature rate less 50°C/hr in primary circuit and to keep the coolant level at a certain range in RPV. Fig. 5 shows the start up process with the fullexternal load.

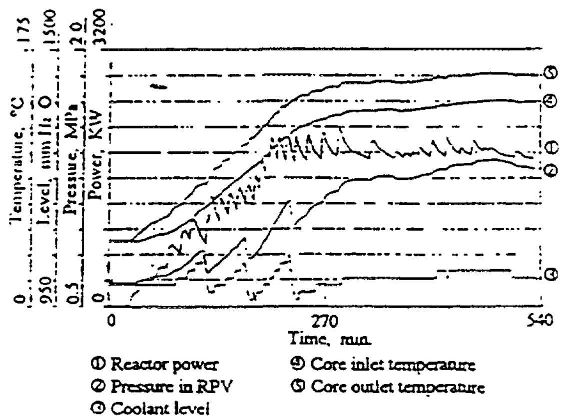


Fig. 5 The start up process of NHR

Feeding nitrogen and water into RPV
Nitrogen and water into the RPV to compensate their loss caused by various reasons(mainly sampling) are made up from time to time for keeping the normal operation condition of NHR-5.

As a result of feeding gas into RPV, the reactor power increases with increase in pressure and comes to a peak, after that it begins to decrease and finally reaches a new steady state. The result is given in Fig. 6. In the process of this experiment. the reactor power increased 5.7%. the core inlet and outlet temperature rised 1.1°C and 1.4°C respectively. The variation of reactor power indicates there is a certain void content in the core at operation condition.The reactor has similar behavior when the water feeds into the RPV.Water fed into the RPV via purification system is reheated by a regenerative heat exchangers. and enters to the downcomer and then into the core. Due to the coolant level rising and fed water temperauture being less than the coolant temperatuthe the core inlet temperature has slight decrease and the pressure increases. For the both reasons of pressure rising and temperature decreasing the reactor power increase. The experimental results are given in Fig. 7.

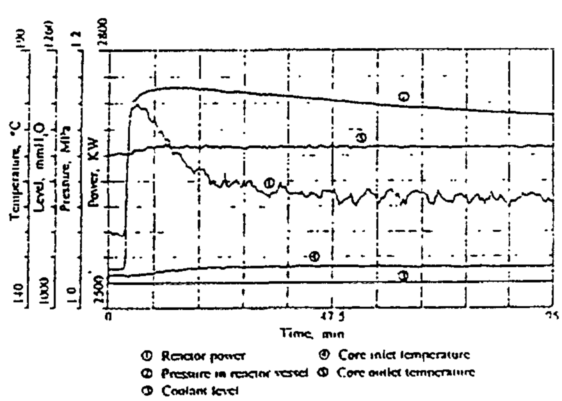


Fig.6 feeding nitrogen into the RPV

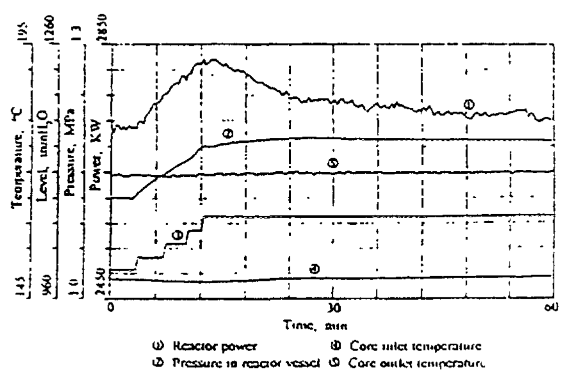


Fig. 7 Feeding coolant water into the RPV

Self-pressurized performance
A space in the upper part of the vessel is used for the self-pressurized space. Total pressure in RPV is formed by both nitrogen partil pressure of 0.3MPa and saturate steam partial pressure of 1. 17MPa which correspond to the core outlet temperature of 186°C

The change of the total pressure in RPV caused by various transient conditions are smooth and small, which result from the large coolant inventory and large self-pressurized space. For example, when the external load changed in 60% the total pressure in RPV only changed in 5%. The change in total pressure is by reasons of both changes of core outlet temperature and coolant level.

High operation availability
As a heating reactor, the NHR-5 is only operated in winters. The operation availability of the NHR-5 is evaluated by comparing the actual operated days with the planned operation days. From December 1989 to March 1993, the NHR-5 has been operated for more than 9330 hours. The average avaibility of heating operation was about 99%. There were four times of unexpected reactor shutdwon caused mainly by failure of electric power supply and auxiliary systems during the four winters operation. Each duration of reactor shutdown was less than 4 hours, so space heating was not affected very much due to the great heat capacity of the heat grid. In spite of the fact that the NHR-5 is the first vessel type heating reactor, it has reached a high availability of heating operation.

3.2 Radiation Protection and Environmental Impacts

Specific radioactivity of water in three loops
During operation the water radioacivity level in the primary circuit, intermediate circuit and heating grid have been regularly monitored. The radioactive back-ground of potable water in this area is about 0. 10 Bq/l. The radioacivity level in the water of the intermediate circuit and the heating grid are as low as that of the potable water. In the primary circuit the specific radioactivity of coolant is at the level of 2.5×10^2-2.7×10^3Bq/l. The nuclide analysis showed there were no fission products in the coolant. From the point of view of radioactivity isolating the intermediate circuit performs a perfect function to keep the heating grid free of radioactivity.

r-exposure rate
The distribution of r - exposure rate in the NHR -5 building is reasonable. A large part of the building have very low r-exposure rate near the background level. A higher r-exposure rate is found outside the biological shielding where the regenerative heat exchanger of the primary purification system is placed here. A local shielding with lead has to be added to reduce the r-exposure rate.

Effluent
During normal operation, the gaseous effluent radioactivity level is at the same level as that of the background. The nuclides analysis indicated that there was no artificial nuclide in the effluent. the nuclides in the effluent are natural ^{40}K and Radon daughters.The amount of waste water, produced from operation and maitenance is about $10.2m^3$ in four years.

Collective dose
The collective dose for all operators in each heating period are also very low, and are indicated in Table 4.

Table 4 Collective dose for all operators in each heating period

Period	Collective dose (mSv-man)
1989.11-1990.3	2.4
1990.11-1991.2	3.2
1991 11-1992.3	11.4

In addition there are many items regularly monitored on the onsite and offsite, such as γ-exposure rate, gross β - radioactivity level of aerosal and liquid effluent the sample of water, soil, air, plants and so on

All measuring data indicate the NHR-5 operation do not cause any change in radioactivity level in this area

3.3 Water chemistry of NHR-5

In consideration of the features of NHR-5 low temperature, low power density, and refueling interval being longer than PWR, and by reference to the operation experience of nuclear powered ship " Otto Hahn" a water chemistry system different from the PWR and BWR is adopted in the operation of NHR-5 This water chemistry system is in neutrol water, not to contain boron and not to add hydrogen in premary coolant, and oxygen removed by chemical additive (C_2H_4)

The results of monitoring and analysis show the dissolved oxygen can maitain the level of 40 ppb and pH value of 6-7 Table 5 listed the analysis results and the specification of primary coolant

Table 5 Specification and monitored results for primary coolant

iterm	specification	analysis results
dissolved oxygen	<50ppb	30-40 ppb
pH(25°C)	6-10	6-7
F	< 100ppb	< 50ppb
Cl	< 100ppb	< 50ppb
Cr	< 10ppb	< 0 1ppb
Fe	< 10ppb	< 0 05ppb
Na	< 5 ppb	< 5ppb
Cu	—	< 0 2ppb
NO_3	—	< 5ppb
NO_2	—	< 5ppb
total solid	< 1ppm	< 0 5-1ppm

The nitrate and nitrite are less than 5 ppb in the coolant at any operation conditions This concentrate is too low to cause metal structure to corrode So nitrogen used as covered gas is feasible for NHR-5

In order to effectively decrease the dissolved oxygen level in the primary coolant, the followings will be considered in the future operatio to remove the oxygen from the makeup water, to add additive into the primary circuit continously and to exhaust the air from the nitrogen supply lines, especialy during replacement of the nitrogen cylinder

3.4 Operation of Intermediate Circuit

To maitain isolating fuction

The pressure in the intermediate circuit is higher than that in primary circuit in order to keep isolating function The RHRS is a part of intermediate circuit during reactor in normal operation In this case the pressure in the intermediate circuit depends on the pressure of pressurized tank (A) in this circuit When the intermediate circuit is isolated the isolating condition depends on the pressure of the pressurized tanks (B and C) installed in the RHRS The two pressurized tanks are connceted to the RHRS by small bore valves and pipes The advantage is that the isolating function can be kept after large loss of water in the intermediate circuit

Detection of leakage rate for the intermediate circuit

The changes in water level in three pressurized tanks (A、B and C) is used for detecting the leakage rate. This method is applicable for the steady operation state. The operation practice indicates the normal leakage rate is about 1 l/hr.

3.5 Operation of RHRS

The reactor residual heat is removed by a passive residual heat removal system which connects to the intermediate circuit. There are two independent trains of the RHRS which composed of three natural circulation loops. (See Fig. 4)

Hot standby condition

When the reactor is operated in normal condition, the RHRS is working at the hot standby condtion. In this case the vaporizer of RHRS and primary heat exchangers work in parallel and a very small flow of the intermediate loop passes through the vaporizer to prevent freezing in the air-cooler. In order to set up the second circulation-vaporization and condensation, the air in the shell side of vaporizer has to be discharged at its high temperaure.

Direction of natural circulation

When the RHRS is put in operation the primary heat exchanger and vaporizer will work from parallel mode to series mode. So the direction of water flow must change in the vaporizer or the primary heat exchangers, which is dependent on the temperature distribution in this system. In gerneral, if reactor is operated at high power level the direction of natural circulation will be the same as in primary heat exchanger, if reactor at low power level, the direction will be the same as in vaporizer. The experimental results indicated two circulative direction has same capacity to remove the decay heat from the core.

From the experimental results it is indicated that the natural circulation of the RHRS can be reliably established and the direction of natural convection in the intermediate circuit did not effect the decay heat removal.

Capability test of RHRS

According to the principle of thermal energy balance the heat removal capability of RHRS is measured at a steady operation state of NHR-5. The heat generated in reactor core should be balanced by the heat loss, the heat cooled by purifaction system and the heat removed by the RHRS. A heat removal capability of 116KW was measured at the average temperature of 166°C in the primary circuit. This value is more than the design value of 75 KW for each train.

In addition the RHRS can be operated at a temperature lower than 100°C, and has a certain capability to remove the residual heat from the core. It shows the reactor can be cooled down to the cold shutdown condition by the RHRS only.

3.6 Operational status of control rod driving system

The control rod driving system was satisfactory for starting up, regulating reactor power and reactor shutdown during the past operation. The full travel time for dropping into the core is less than 2 seconds.

Owning to use of the temperature compensation device in the hydraulic driving system, to adjust the flow rate at high temperature is not neccesseary indeed.

The ultrasonic position indicator were also satisfactory for indicating the position of control rod under the pressurized water operation mode. The ultrasonic indicator system can not work under two phase flow or the condition with a interface of gas and liquid. Therefore, the correct position of control rod would not be indicated in this system as the loss of pressure or fast pressure reduction inside RPV.

4. Safety Features Experiment of NHR-5

In the course of commissioning and operation, a number of experiments have been carried out to demostrate the feasibility and safety of the vessel type heating reactor concept. In these experiments there are no any external interference of the operators.

4.1 Self-regulation Feacture

The self-regulation experiment has been performed to investigate the reactor self-regulation ability to follow the change of heating load. The heating load can be varied by means of changing the flow rate through the intermediate heat exchangers.

The flow rate through the intermediate heat exchangers was from 8 t/hr to 35 t/hr, then back to 8 t/hr. This value corresponds to a heating load change from 1.5 MW to 2.5 MW, a variation of about 66% Figure 8 shows the behaviour of NHR-5 following the heating load change. The reactor power caused by the self-regulating mechanism automatically to vary after 90 seconds, reached a new power level to match the heating load within 30 minutes. The moderator temperature coefficient plays a main role in this process The experimental results show that the NHR-5 has a very good self regulation ability to follow a load change without any operator action.

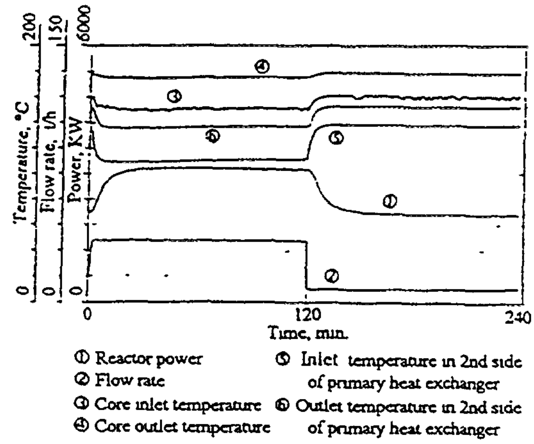


Fig.8 The feature of self-regulation at NHR-5

4.2 Self-stability Feature

The self-stability experiment was performed in order to investigate the response of reactivity insertion. In this experiment the reactor was operated at a power of 2.5 MW, then a step insertion of 2 mk reactivity was introduced. Figure 9 indicates the variation of reactor parameters. At the beginning of the transient the reactor power increased rapidly due to the extra reactivity and reached a maximum relative value of 1.18 in 100 seconds.Then the reactor power began to decrease due to the feedback of negative reactivity coefficient and came to a new relative power level of 1.08 in 30 minutes. The core inlet and outlet temperatures added an increment of 3.8°C and 4.2 °C correspondingly The reactor pressure increased with a Δp of 0.102 MPa

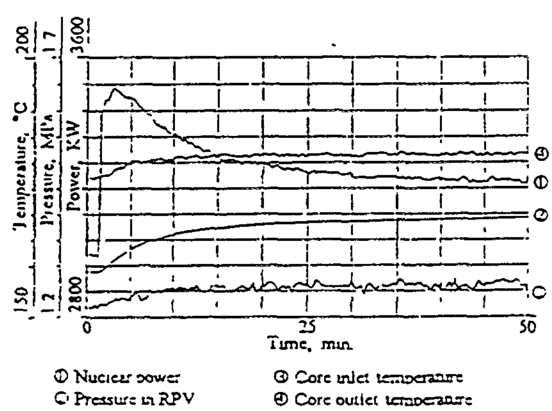


Fig. 9 The feature of self-stability at NHR-5
(Inserting reactivity of $2\times10^{-3}\Delta k/k$)

4.3 Experiment for ATWS

In order to study the safety behaviour of the NHR-5 in 1990 an experiment has been carried out.

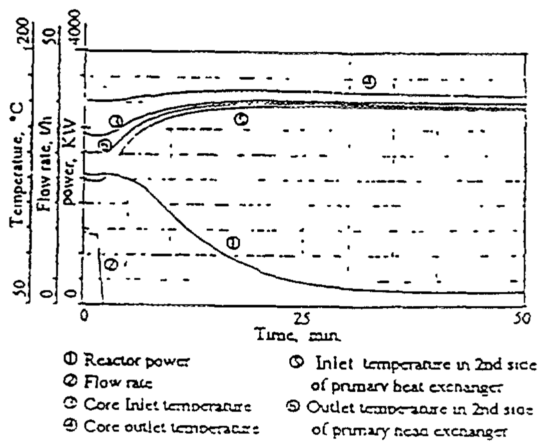


Fig 10 The transient of loss heat main heat sink without scram

which simulated the ATWS. i.e a loss of the main heat sink followed by the failure of all 13 shutdown control rods.

In this experiment, the intermediate heat exchangers was isolated at a reactor power of 2 MW, and noneof the shutdown rods was inserted. Figure 10shows the power variation observed together with the changes in temperaure and pressure of the reactor. The power decreased as a consequence of feedback of the negative temperature coefficient to a stable value of about 0.2 MW in about 30 minutes. The inlet and outlet temperature of the reactor core rised by 20.4°C and 4.7° C respectively. The temperature variation is not serious at all. The primary system pressure rised by 0.23 MPa. The result of the experiment demonstate that the NHR-5 has excellent inherent and passive safety features. The ractor will be shutdown passively even in the described ATWS case.

4.4 Residual heat removal under the interruption of natural circulation in the primary circuit
When a loss of coolant accident (LOCA) occurs in the primary circuit, the water level inside the RPV will decrease. Due to the integrated arrangement of the primary circuit and all penetration of small pipes located at the upper part of the vessel the reactor core will never be uncovered. But as a result of the water level decrease the natural circulation of the primary circuit might be interrupted. In this case the residual heat of the reactor will be transported by vapor condensed at the uncovered tubes of the primary heat exchangers.

To demonstrate the capability of residual heat removal under LOCA conditions a special experiment was carried out at the NHR-5 in March 1992. After reator shut down the water in the reactor vessel was discharged by opening the valve to the blowdown tank. The water discharge rate was $1.6m^3/hr$ and an amount of $2.4m^3$ water was drained off. The water level in the reactor vessel decreased below the entrance of the primary heat exchanger and the natural circulation was interrupted. In this case the residual heat removal was mainly realized by condensation of the vapor Due to the discharge of $2.4m^3$ water the partia pressure of nitrogen reduced from 0. 29 MPa to 0 022MPa. so that the water subcooling of the reactor outlet temperature decreased from 12 °C to 2 °C

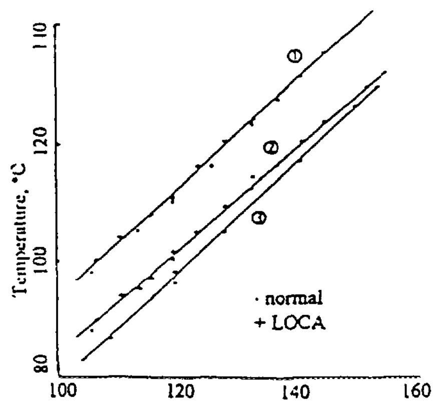


Fig. 11 The comparison of normol and abnormal operation codition ofNHR-5

The reduction of sub-cooling enhansed the vaporization - condensation process. Figure 11 shows the comparison of the residual heat removal capabilities during LOCA conditions and under the normal operation. From the results of the comparision it can be shown that the procedures of both LOCA and normal operation are almost the same. The decay heat can be reliably removed by means of vapor condensation on the primary heat exchanger under LOCA conditions.

5. Summary

During four winters of NHR-5 heating operation. the reactor has been known as a valuable tool for a number of experiments on operation behaviors and safety features The operational and experimental results have successfully demonstrated the inherent and passive safety characteristics of NHR-5 It was proven that the design concept and technical measures of NHR are suitable to meet the requirements for distric heating in northern cities. congeneration and air condition in the middle cities of China. as well as the seawater desalination

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SAFETY CONCEPT AND OPERATION CONTROL APPROACH IN THE DESIGN OF SMALL NUCLEAR REACTORS

Yu.D. BARANAEV, Yu.I. OREKHOV,
Yu.A. SERGEEV, N.I. SOSNIN
Institute of Physics and Power Engineering,
Obninsk

K.V. ZVEREV
Minatom, Moscow

Russian Federation

Abstract

Operating experience and manifold feasibility studies reveal promising market potential for Small Reactors (SR) in remote areas of Russia. A number of SR designs ranged from few MWth to several dozens MWth are proposed by designers for power or heat production and cogeneration. Some of them are at the detailed design stage and ready for practical implementation.
Safety concept and operation control approach of SRs are discussed in the paper using Floating Nuclear Power Plant VOLNOLOM-3 design as a typical example.

1. Small Reactor Design Activities

Long term operating experience of Bilibino Nuclear Power Plant /1/ (four water-graphite reactors EGP-6 of 48 MWth each) at Chukotka peninsula (North Siberia), as well as results of manifold site specific feasibility studies revealed that Small Reactors (SR) have promising market potential in remote isolated regions of Russia and can be considered as a viable alternative to fossil fuel energy sources. About 90 sites in the fuel deficient regions in the North and North - East of Russia were found prospective for detailed assessment of technical and economic aspects of Small Nuclear Power Plants construction.

Taking into account this favourable conditions for small reactor application, a number of SR designs ranged from several MWth to several dozens MWth were proposed by designers for power or heat production and cogeneration. These designs are based on the experience gained in construction and operation of existing small and large power reactors, as well as ship propulsion, space, and research reactors.

Most of the proposed SRs are at the preliminary design stage. But there are several SR designs advanced to a practical implementation. Detailed design has been completed for Floating NPP (FNPP) VOLNOLOM-3 equipped with two ABV /2/ reactors of 38 MWth and for water-graphite reactor ATU-2 /3/ of 125 MWth for second stage of Bilibino NPP. Conceptual design of pool-type heating reactor RUTA /2/ of 20 MWth has been completed. Technical and economic assessments of construction of Nuclear Heating Plant using four RUTA reactors of 55 MWth are in progress now. The design activities for Floating NPP with two ice breaker reactors KLT- 40 /2/ of 160 MWth were started this year and scheduled to finish the basic design in 1997.

The experience gained in SRs operation and design showed that the objectives of a better operability and enhanced safety of SR can be easier achieved in the design providing minimal operators actions under normal conditions and minimal prompt operators intervention in case of an accident. Exclusion of requirements for any operators actions under normal operating conditions and inherent provisions for long grace period (up to few days and even more) for all credible accidents are considered as a desirable objectives in SR design approach.

The FNPP VOLNOLOM-3 design is considered in more details to clarify the design approach aimed at development the SRs capability for minimal operators actions in a reactor control process.

2. Reactor Safety Concept

Detailed design of the Floating Nuclear Power Plant (FNPP) VOLNOLOM-3 using two integral pressurized water reactors ABV was completed in 1994 and licensing process has started. The ABV reactor characteristics are presented in Table 1.

Table 1. ABV reactor design characteristics

Parameter	Value
Rated reactor capacity	38 MWth
Electric output	6 MWe
Heat production capacity	12 Gkal/h
Refuelling (batch) period	4 - 5 yr.
Primary circuit	
pressure	15.4 MPa
temperature (inlet/outlet)	245/327 ^{0}C
water flow rate	85.4 kg/s
water inventory	7.6 m^3
Secondary circuit	
steam pressure	3.17 MPa
steam temperature	290 ^{0}C
steam flow rate	14.7 kg/s

The principal design solutions which define high level of reactor safety and reliability are:

- integral arrangement of the primary circuit and natural primary coolant circulation,
- self-protection and self-control properties as well as inherent safety characteristics,
- usage of engineered safety features including passive ones,
- usage of primary coolant pressure and temperature driven devices for direct safety systems actuation,
- simplified reactor and safety systems design.

Design and neutronic characteristics of the reactor core ensure negative power, temperature, and void reactivity coefficients. As a result, self-limitation of the reactor power at the reactivity accidents and transients without scram takes place. Reactor self-control properties enable to change reactor power in the range of 20 - 100% of rated power (No) at a rate up to 0.5%No/s without control rods displacement and just for automatic control of feed water flow rate.

Increased primary coolant inventory, natural primary coolant circulation and negative reactivity feedback slow down accident progression and ensures rather long grace period for engineered safety systems actuation.

The ABV reactor is equipped with the following engineered safety systems:

Reactor shut - down is carried out by means of deenergizanion of control rod drives and subsequent gravity driven control rods insertion to the reactor core

Emergency heat removal system has 5 independent trains Two of them supplying cooling water to the steam generators from pressurized tanks are passive gas pressure driven ones.

Emergency water injection system for reactor flooding under LOCAs is designed as active one because of certain size limitations in the FNPP. It includes 3 high pressure and 2 low pressure pumps Water recirculation system for long term decay heat removal is also provided It is essential that even in case of all emergency heat removal and water injection systems failure under LOCA the reactor core uncovery starts only after 3 hours of the accident initiation

The ABV reactor is equipped with the low pressure boron injection system and reactor cavity flooding system. Both are non-automatic and non-safety graded systems and designed for severe accident management

Two ABV reactors in the FNPP VOLNOLOM-3 are located in separate steel containments withstanding internal pressure up to 0.6 MPa.

3. Safety Analysis and Control Concept

The results of safety analysis taking into account single failure criteria demonstrated that for all design basis accidents (DBA) the ABV reactor safety was ensured without reliance on any operators actions. Practically no on - site and off - site consequences take place at the DBA

The relevant general results of the beyond design basis accident (BDBA) analysis are presented in Table 2. They show that for most of the realistic BDBA scenarios it is possible to cope with the accidents without reliance on the operator actions for a very long (several days) period Only for a few low probable (commutative frequency $< 10^{6}$ per year) scenarios operator actions are required in 5-12 hours But even in that case operator actions are rather simple and unambiguous This actions can be carried out from control room or from local equipment control terminals located at the FNPP in such a way to guarantee operating staff exposure in accident conditions, well below the permissible level.

The most important task of the operators for severe accident prevention is restoration of at least one heat removal train. The simplest but not a single way to do this is reactor cavity flooding to provide better conditions for heat transfer to radiation shielding tank water Usage of this heat sink ensures decay heat removal for a practically infinite period of time. Only two valves should be open remotely from control room to actuate corresponding passive gravity driven system

Consideration of the ABV reactor operability and safety actually revealed a good potential for operating staff minimization. But final decisions for this has to be made taking also into account required high level of reactor and NPP reliability and availability. It is extremely important for the FNPP VOLNOLOM-3 designed for siting in remotely isolated regions where it can be and probably will be the one and the only long term power and heat source In such conditions the NPP staff has to be ready to carry on required maintenance and contingent repair procedures without reliance on local manpower and industrial capabilities. These reasons force designers to be very careful approaching to FNPPs staffing minimization.

The FNPP VOLNOLOM-3 operating shift staff schedule as it is in the design documentation is presented in Table 3 A part from that, provision is made for day-time staff consist of 2 engineers, 2 technicians, and 4 workers These figures demonstrate how real staffing formation practice looks like for current FNPP design

Detailed assessment of operating and maintenance procedures shows there is a certain design margin in staffing formation for VOLNOLOM-3. But very strong and clear arguments are required to overcome existing conservatism based on feedback from current NPPs operating experience.

Table 2. General Results of the ABV Reactor Beyond Design Basis Accidents Analysis

Initial Event Considered	Additional Safety Systems Failures	Required Prompt Operator Actions to Cope with the Accident	Grace Period
2 control rod clusters (CRC) withdrawal at reactor start-up	Scram failure	NN[1]	VL[2]
All (6) CRCs withdrawal at reactor start up	Scram failure	NN	VL
1 CRC withdrawal at 100% power	Scram failure	NN	VL
1 CRC withdrawal at 20% power	Scram failure	NN	VL
Stop feed water at 100% power	Scram failure	NN	VL
Stop feed water at 100% power	Scram + active trains of emergency heat removal systems failure	NN	VL
	Scram + all trains of emergency heat removal system failure	Restoration of one heat removal train	4 - 5 h
Primary pipeline rupture at 100 % power	Two trains of high pressure and two trains low pressure water injection systems + all trains of emergency heat removal systems failure	NN	VL
	All trains of high and low pressure water injection + all heat removal systems failure	Restoration of one injection train	5 - 6 h
	All trains of high and low pressure water injection failure	Restoration of one injection train	10 - 12 h

[1] NN - not necessary
[2] VL - very long

Table 3. FNNP VOLNOLOM - 3 operating shift schedule

Location at the FNPP	Field of Responsibility	Graduation Status	Number of Persons
Control Room	Head of shift	E[1]	1
	Reactor control	E	2
	Electric systems	E	1
	I&C	E	1
	Radiological safety	E	1
Equipment Compartments (local posts)	I&C	T[2]	1
	Radiological Safety	T	1
	Electric Systems	T	1
	Turbine/Generator	T	1
		M[3]	1
		W[4]	1
	Chemical systems	W	1
	Ship auxiliary systems and structures	W	2
		Total	15

[1] E - engineer
[2] T - technician
[3] M - mechanic
[4] W - worker

More promising way for that is staffing reduction due to joining operator functions. Preliminary assessment showed that it could result for VOLNOLOM-3 in a reduction of staff in the plant operating shift by half.

Construction and operation of small reactors with minimized staffing and/or remote monitoring is not only a technical issue. Psychological problems including public acceptance is also very important. Step by step approach would be the best way to achieve public acceptance in the area in question. Therefore, discussing at current stage staffing reduction, instead of staffing minimization and/or reactor remote monitoring, would be mote practical. In any case it is very important and actual to elaborate coherent international approach to this problem.

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SAFETY OBJECTIVES AND DESIGN CRITERIA FOR THE NHR-200

Dazhi XUE, Wenxiang ZHENG
Institute of Nuclear Energy and Technology,
Tsinghua University,
Beijing, China

Abstract

The construction of a nuclear district heating reactor (NHR) demonstration plant with a thermal power of 200 MW has been decided for the northeast of China. To facilitate the design and licensability a set of design criteria were developed for the NHR, based on existing general criteria for NPP but amended with regard to the unique features of NHR-200. Some key points are discussed in this paper.

1. GENERAL SAFETY REQUIREMENT

For a nuclear district heating reactor (NHR) it is necessary to locate it near the user due to the necessary way of heat transport (hot water or low pressure steam). This means that a NHR is surrounded by a populous area. Using evacuation as an essential element in the ultimate protection of the public can thus become impractical. The safety for a NHR has to be ensured with its excellent inherent features and passive safety. In all credible accidents the radioactive release from a heating reactor has to be reduced to such low levels that off-site emergency actions, including sheltering, evacuation, relocation and field decontamination will be not necessary. In the Technical Report on "The Design of Nuclear Heating Plant" [1] issued by the Chinese National Nuclear Safety Administration (NNSA) it is stated explicitly that no off-site emergency actions such as sheltering, evacuation etc. are allowed for a NHR. In other words, the maximum accident should be no more serious than a level 4 event by the International Nuclear Event Scale. For a typical site in northern China a maximum individual dose of 5 mSv will result from an activity release of 4.7×10^{13} Bq I-131 equivalent via a stack of 50m height. It is indicated that a release of 3.7×10^{13} Bq I-131 can be the limitation for the maximum credible accident.

It is a fact that the existing reactors have become much safer due to various measures, with backfitting and upgrading, especially learning from TMI and Chernobyl accidents. Also, there is a trend that future plants will be, or will have to be, better in terms of CDF than the best of the existing ones, to be achieved by evolutionary design or/and innovative improvements. For convenience, the following figures of CDF can give an idea of what safety has been achieved and of what is the target for the next generation of NPP.

	CDF (1/reactor•year)
The best of the existing NPP	10^{-4}-10^{-5}
The NPP coming on-line by the year 2000 or later	10^{-5}-10^{-6}
The innovative designs	$<10^{-7}$

For a NHR, the safety requirement in terms of CDF has reached the top of the safety target if a figure of much less than 10^{-7} of CDF is achieved.

On the other hand, there is a serious challenge to the economy for a NHR. The capacity of a NHR can not be as big as that of NPP due to the limitations of heat transport. The economic thermal power is in the rang of 200-500MWt. Moreover, the load factor is also much lower than that of a normal NPP.

It is obvious that to meet the safety requirements, and to lower the capital investment are the major concerns in the design of a NHR. The only solution is to have a design with inherent safety characteristics and passive safety as much as possible instead of the complex engineering safety features. In addition, the high end-user efficiency of district heat application instead of electricity generation with low end-user efficiency is also a key point to improve the economy for a NHR.

2. DESIGN CRITERIA

The design criteria for conventional nuclear power plants already exist. Most of them are suitable for the NHR, but some of them have to be modified due to more stringent safety requirements and unusual design approaches. In order to have a design basis and to enable the safety regulatory body (NNSA) to evaluate systematically the design of the NHR-200, a set of design criteria for NHR is being drawn up and reviewed by a team organized by NNSA. Since no large scale operational experience is available at this time, the design criteria are not a complete nor a official set of regulations. It will be issued as a technical document. Some of the major points of this criteria are discussed as follows.

(1) Operation categories

Usually 4 operation categories are classified for a conventional NPP. Among them Category III and IV are accident conditions. The accidents of Category IV are the most serious ones in the sense of DBA. But in recent years addressing beyond design basic accidents has been required by various regulatory bodies. The French H procedures are perhaps the most comprehensive ones.

In order to meet the enhanced safety requirements as well as the position of the Chinese regulatory body, an operation Category V is added beyond the 4 categories in our design criteria. Generally, the operation Category V is such an accident condition after a DBA (Category III or IV). Apart from the assumption of a single failure, there is an additional failure assumed to occur. Therefore, Category V consists of events with lower frequency than those for Category IV. The typical events are: loss of off-site power followed by an assumed failure to scram combined with a stuck open safety valve; break of reactor vessel in its lower part or pipe break of coolant purification system followed by failure of isolation due to failure of two isolation valves; intermediate loop break followed by failure of isolation. The deterministic analysis is conducted with realistic parameters. The measures for mitigation of such accidents should be reliable. But a grace period can be taken credit of. The acceptance criteria of this Category is that the release of radioactive substance into the environment is not allowed to disrupt normal life beyond the non-residential area (the plant).

The limiting doses for individuals of the population during each operational Category are much less than that for a conventional NPP. They are listed as follows:

- Category I (Normal operation) 0.1 mSv/a
- Category II (Anticipative operation events) 0.2 mSv/event
- Category III (Rare accident) 1.0 mSv/event
- Category IV (Limited accident) 5.0 mSv/event
- Category V (Additional Operation Category) 5.0 mSv/event

(2) Thermohydraulic design criteria

In order to reduce the radioactive release from the fuel elements in case of an accident, the thermohydraulic design criteria for a NHR are more rigorous than those for conventional NPPs.

a. In respect to fuel element damage the differences between NPP and NHR are listed below:

	NPP	NHR
Category I and II	No additional fuel damage	No additional fuel damage
Category III	Fuel damage should be limited in a small part of all fuel elements	No additional fuel damage
Category IV	Fuel damage possibly occurs with a large amount of all fuel elements	Fuel damage should be limited to a small part of all fuel elements
Category V	NA	Small as above

b. Correspondingly the DNBR must stay above the limit value in operation Category I and II for conventional NPP, but also in Category III for the NHR. The same requirement for fuel temperature is that the maximum temperature at the center of the fuel element at the hot spot never reach the melting temperature in the operation Category I and II for conventional NPPs, but also in Category III for the NHR.

c. For conventional NPPs, the average temperature of the fuel clad at the hot spot has to be below the embittlement temperature of 1204°C in case of a LOCA. But for the NHR it is required that the reactor core is always covered by coolant in case of LOCAs. Thus the temperature will be far less than the above limit.

(3) Containment

As a final barrier against fission product release, a containment system is one of the important Engineered Safety Features (ESF) in a current nuclear power plant. It consists of a containment structure and several systems to maintain the integrity of contaminant during accident scenarios. This system is very expensive. Recently, along with the development of advanced nuclear power plant, especially for more innovative reactor designs, the concept of a containment is also getting development. For example, a vented confinement concept

[2] is provided for a small or middle size modular high temperature gas cooled reactor instead of the gas-tight pressurized containment for the current generation LWRs due to the exclusion of the possibility of a fission product release from coated particle fuel elements in case of an accident. Another example, for the Safe Integral Reactor (SIR) developed by the UK and the USA [3], the integrated arrangement of the primary coolant system makes it possible that the containment is a compact one.

Based upon the definition given by 10CFR50, the primary reactor containment means the structure or vessel that encloses the components of the reactor coolant pressure boundary, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. For the systems connected to the reactor vessel, the reactor coolant pressure boundary is up to and including the second isolating valve. Since the NHR-200 is an integrated arrangement of the primary coolant system, a small compact containment is adopted which meets the above definition. Moreover, this containment has the further important function that it ensures the reactor core being always covered by coolant in all pipe break accidents, even in the case of a small break in the lower part of the reactor vessel.

The reactor building serves as a secondary confinement. The main function of this structure is the protection against external events. It also provides a subatmospheric enclosure to collect the leakage from the compact containment during LOCA.

(4) Special credit for NHR-200

Since the safety systems adopted for the NHR-200 have many differences compared with those in normal NPPs, these differences have to be reflected regarding the requirements for the support systems. Some special credits are discussed as follows.

a) Emergency power system

For a NPP, apart from two independent off-site power supply connections, there is an emergency power system with two or three separate trains equipped with a quick-starting diesel unit for each to supply power to all redundant safety-related systems, such as ECCS, containment cooling and spray system, residual heat removal system and the related support systems in case of loss of off-site power. But for a NHR some safety-related systems, such as ECCS, containment cooling and spray systems are not necessary, and some safety-related systems, such as the shut down and residual heat removal systems are passive systems. Therefore, the emergency diesel generators are no more necessary. Nevertheless, there are two diesel generators in the design of NHR-200, but they are not classified as safety related. In order to enhance the reliability of stand-by power supply, one of these two diesel generators is classified as seismic Category 1.

b) Component cooling water (CCW)

In the design of NHR-200 the loads of CCW are cooling of coolant purification system, condenser of liquid waste treatment system, and cooling of the control rod drive system. Among them there are no safety-related systems, therefore, the CCW is classified as a non-safety related.

c) Heating, ventilation and air conditioning system (HVAC) for the control room

Since a large release of radioactive material into the reactor building is impossible under all credible circumstances, the HVAC is not classified as safety related.

3. SPECIAL RADIOLOGICAL PROTECTION ISSUES FOR NHR

Since a NHR has to be located near the user, some radiological protection issues which are different from the case of NPPs have to be investigated. They are discussed as follows.

1) Site criteria

The features of current site criteria for NPP shown by the American code 10 CFR100 are as following:

1) The evaluation is based on an assumption of a core melt accident.
2) In case of the maximum hypothetical accident, emergency actions, including sheltering, evacuation and field decontamination, are adopted in order to assure that no individual receives a whole body dose in excess of 0.25Sv which would result in acute injury. Also, the accumulative dose received by the population is limited to a reasonable value.
3) An exclusion area and a low population zone are necessary. Also, a distance from the reactor to population center has to be kept in order to meet the above requirement.

More than 6000 reactor-years of NPP operating experience with only two significant accident shows that the above principles are correct. NPPs are quite safe but core melt with subsequent release of appreciable quantities of fission products is still considered credible (The frequency of core melt is considered as 10^{-5} per reactor year) [4]. Preparation of an off-site emergency action including evacuation is necessary. A site for NPP should be far from population center, saying larger than 25 km.

For the NHR it is impossible to require a large area adjacent to the plant with sparse population or a site far from a city. It means that distance is not a protecting factor any longer. The actions of evacuation, relocation and field decontamination are no longer practical emergency measures to protect the population from over-exposure. The public is protected only by the safety features of the NHR. Safety is achieved by adopting more inherent safety features, they have been presented in many papers [5, 6]. The frequency of core melt for a NHR is much less than 10^{-7} per reactor-year [6, 7] which can be considered negligible in practice. In this way core melt is no longer considered as a design basis. The site criteria for NPPs can not be used in case of the NHR. For a NHR the recommended dose limit for the maximum design basic accident is 5 mSv without any emergency action.

Regarding high population, high utilization factor of land and the unit power of 200-500MW for one NHR, which is one magnitude smaller than that for a NPP, a non-residential zone of 250m in radius and a physical isolation zone of 2km in radius are proposed. During the lifetime of the NHR, development in the physical isolation zone should be restricted in terms of population and large scale public facilities. This physical isolation zone is only functioning as an isolation between the plant and the public to reduce the interference with each other during normal operation as well as during abnormal conditions.

2) Liquid effluents

For a NHR site it is better to have no restriction on liquid effluent release. In most cases, there is no suitable receiver of liquid effluent near a proposed site for a NHR.

Therefore, the principle of treatment and disposal of liquid waste for a NHR should be different from that for a NPP. For a NPP the distinguishing features are: large amounts of liquid waste (more than 10,000 m^3/a), a moderate degree of decontamination (depends on the amount of salt content, evaporation or demineralizing approaches that are used respectively; the target of decontamination is 10^{-8}-10^{-7}Ci/l). After treatment, the liquid effluent is mixed with circulating cooling water and released to a river or sea.

For a NHR the amount of liquid waste is much less than that for a NPP (300 m^3/a is expected). The proposed principle is increasing the decontamination factor (the target of concentration after treatment is 10^{-10}Ci/l), and then reusing the decontaminated water as much as possible. For the remains of usage it can be used as a make-up of plant cooling water or evaporated in a natural evaporative pond, even drained to a city sewage network.

3) Protection against pollution of the heating grid

Differing with a nuclear power plant, the NHR will be connected with the user through the heating grid. Therefore, protection against pollution of the heating grid is extremely important. In the design of the NHR, an intermediate loop is adopted to separate the heating grid from the radioactive primary loop. Moreover, the pressure of the intermediate loop is kept higher than that of the primary loop in all conditions, so that it ensures that leakages, if there are any, are always from the intermediate loop to the primary loop, never vice versa. In addition, pressure and radioactivity of the intermediate loop are monitored continuously. An isolating device is also installed in order to quickly isolate the intermediate loop from the primary loop in case of occurrence of a large leakage. These measures ensure that the contamination of the intermediate loop is very low. The pressure of the intermediate loop is also kept higher than that of the heating grid. This arrangement not only makes heating grid operation easier but also favors keeping the water quality of the intermediate loop.

The limits of radioactive concentrate are: 10 Bq/l for the intermediate loop; 0.37 Bq/l for the heating grid.

4. SUMMARY

Most of the design criteria for conventional NPPs are suitable for the NHR, but some of them have to be modified due to more stringent safety requirements and due to the unusual design approaches. A set of design criteria for the NHR has been drawn up for facilitating the development of NHRs in China. Meeting these criteria means that off-site emergency plans for protecting the population would not be necessary.

The evaluation of the design of the NHR-200 which meets the proposed design criteria shows that the design of NHR-200 has attractiveness both in safety characteristics and economy. The proposed design criteria have to be updated along with the accumulation of practical experience with NHRs.

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TECHNICAL OUTLINE OF A HIGH TEMPERATURE POOL REACTOR WITH INHERENT PASSIVE SAFETY FEATURES

R.S. HART
Atomic Energy of Canada Ltd,
Mississauga, Ontario,
Canada

Abstract

Many reactor designers world wide have successfully established technologies for very small reactors (less than 10 MW_{TH}), and technologies for large power reactors (greater than 1000 MW_{TH}), but have not developed small reactors (between 10 MW_{TH} and 1000 MW_{th}) which are safe, economic, and capable of meeting user technical, economic, and safety requirements. This is largely because the very small reactor technologies and the power reactor technologies are not amiable to safe and economic upsizing/downsizing.

This paper postulates that new technologies, or novel combinations of existing technologies are necessary to the design of safe and economic small reactors. The paper then suggest a set of requirements that must be satisfied by a small reactor design, and defines a pool type reactor that utilizes lead coolant and TRISO fuel which has the potential for meeting these requirements.

This reactor, named LEADIR-PS, (an acronym for LEAD-cooled Integral Reactor, Passively Safe) incorporates the inherent safety features of the Modular High Temperature Gas Cooled Reactor (MHTGR), while avoiding the cost of reactor and steam generator pressure vessels, and the safety concerns regarding pressure vessel rupture.

This paper includes the description of a standard 200MW thermal reactor module based on this concept, called LEADIR-PS 200.

1. INTRODUCTION

1.1 BACKGROUND

Reactor vendors have achieved success with very small reactors, for example SLOWPOKE, TRIGA, PULSTAR and MAPLE-X, and with large power reactors, for example CANDU, PWR, and BWR. They have however failed to produce commercially viable small reactors (say in the 10 to 1000 MWth range). This is because the very small reactor technology is not amiable to safe and economic scale-up, while the large reactor technologies cannot be economically downsized.

This leaves a very large potential market (the thousands of small Pacific Islands, the Canadian and Russian north and desalination facilities in the middle east, for example) without an economic nuclear energy option.

For nuclear energy to make a substantial contribution to serving the energy needs of these areas, significant improvements must be made in many areas, including those of safety, environmental impact, operations cost, maintenance cost, security cost, capital cost and high temperature capability. This will likely require the implementation of novel technologies, ranging from reactor technologies to control and construction methods.

1.2 SMALL REACTOR REQUIREMENTS

If small reactors are to be commercially viable, they must, on a specific output basis, challenge the capabilities of large modern water cooled reactors; for example, their capital cost per MW_{th} output, operation and maintenance cost per MW_{th} output, and risk to the public on a per MW_{th} basis must not be significantly greater than that of the large plant.

With this objective in mind, the following are suggested requirements for small nuclear power plants.

1. Inherent Shutdown: inherent characteristics that will achieve reactor shutdown under any accident condition <u>without</u> the use of any <u>active</u> detection or shutdown mechanisms.
2. Passive Decay Heat Removal: The removal of decay heat by natural and passive means, <u>without</u> the use of any active detection or operating mechanism,
3. Eliminate Severe Accident Scenarios: Eliminate real and <u>perceived</u> beyond design basis events for which there are not transparent, inherent or passive solutions. For example, pressure vessel rupture, graphite burn, sodium water/air reaction.
4. Low Environmental Impact: Plant discharges of all types, including chemicals and radioactive isotopes must be minimal. A comprehensive waste management scheme (low, medium and high activity) must be included.
5. No safety dependence on Operator: power plant safety should not be dependent on operator action, and should be immune to malicious or incompetent operator action.
6. Low Operating, Maintenance and Security Costs: The nuclear plant must be no more demanding in any aspect of normal operation than a small fossil-fired station. Small size

plants, particularly those without steam raising equipment, should be capable of unattended remote operation.

7. Broad Application Capability: The technology and concepts employed should serve a broad range of applications ranging from low temperatures (district heating for example) to high temperatures (steel making for example).
8. Broad Size Range Capability: The technologies and concepts employed should allow the construction of reactors covering a broad size range (say 10MW(th) to 1000MW(th)) thereby facilitating standardization of technology.
9. Volume Construction: The plant must be fully modularized, and designed for "production line" fabrication with minimal on-site construction activity.

1.3 LEADIR-PS

This outline identifies a reactor concept with the potential of meeting these requirements, called LEADIR-PS, and develops a specific configuration for a 200MW thermal version of this concept called the LEADIR-PS 200.

The LEADIR-PS power plant concept is the result of taking a fresh approach in determining what is needed to produce a safe and economical power plant design that will meet the requirements of the public, the utility and the regulatory.

Early studies concluded that the nuclear heat transport systems of a small reactor could not operate significantly above atmospheric pressure, if all design requirements were to be satisfied. It was noted that all of the very small reactors currently available, and generally regarded as "inherently safe" (SLOWPOKE, TRIGA, MAPLE-X) are pool type reactors.

It was also recognized that problems and unknowns tended to increase, almost exponentially, as the number of novel features increase. Hence, to minimize the development effort necessary, LEADIR-PS utilizes established technologies to maximum extent feasible, and the minimum number of novel technologies.

Many reactor configurations are possible, for example lead coolant and fuel contained within a zirconium pressure tube of the CANDU type, in combination with a graphite moderator. This however raises many concerns, for example, lead zirconium reaction, zirconium graphite reaction, graphite cooling requirements, etc. Very little technical information exists with respect to most of these issues. LEADIR-PS therefore uses only lead and graphite in the core (external to the fuel), and an established fuel design technology which prevents the fuel from contacting the coolant. This minimizes the range of development work required, allowing a focus on the performance of the lead coolant, and on lead/graphite interaction. Considerable literature exists on the latter.

2. OVERVIEW

It is assumed that the "new generation" reactor would be a fission reactor, and be firmly based on established technologies, possibly merging one or more current concepts.

A brief tabulation (Table 1) identifies the key features of Advanced Reactor Concepts now being developed. The modular high temperature gas-cooled reactor (MHTGR) excels in most areas except capital cost; the relatively high capital cost results from the low power density in the MHTGR pressure vessel (about 20% that of the AP600), and the cost of helium confinement systems. A second tabulation of the materials utilized for coolant, moderator and fuel in current or recent reactors was compiled (Table 2). Various combinations of coolant, moderator, and fuel were considered.

The above efforts identified the potential for a lead (or lead alloy) cooled, graphite moderated reactor utilizing the TRISO coated fuel kernels. Essentially, a MHTGR without pressure vessels, utilizing lead coolant.

Reactors based on this concept, and incorporating inherent shutdown and passive safety features are called LEADIR-PS, an acronym for LEAD-cooled Integral Reactor – Passively Safe. A possible configuration for LEADIR-PS, discussed in more detail in Section 3 and Section 4, is presented in Figures 1 and Figure 2.

A key feature of LEADIR-PS, shared with the Modular High Temperature Gas-Cooled Reactor (MHTGR) under development by General Atomics, is that radionuclide releases are prevented by retention of the radionuclides within the fuel particles under all design basis events without operator action or the use of active systems. Thus, the control of radionuclide releases is achieved primarily by reliance on the inherent characteristics of the coolant, core materials, and fuel. Specifically, the geometry and size of the reactor core, its power density, coolant, and reactor vessel have been selected to allow for decay heat removal from the core to the ultimate heat sink through the natural processes of radiation, conduction and convection, while the negative temperature coefficients of the fuel and moderator assure reactor shutdown.

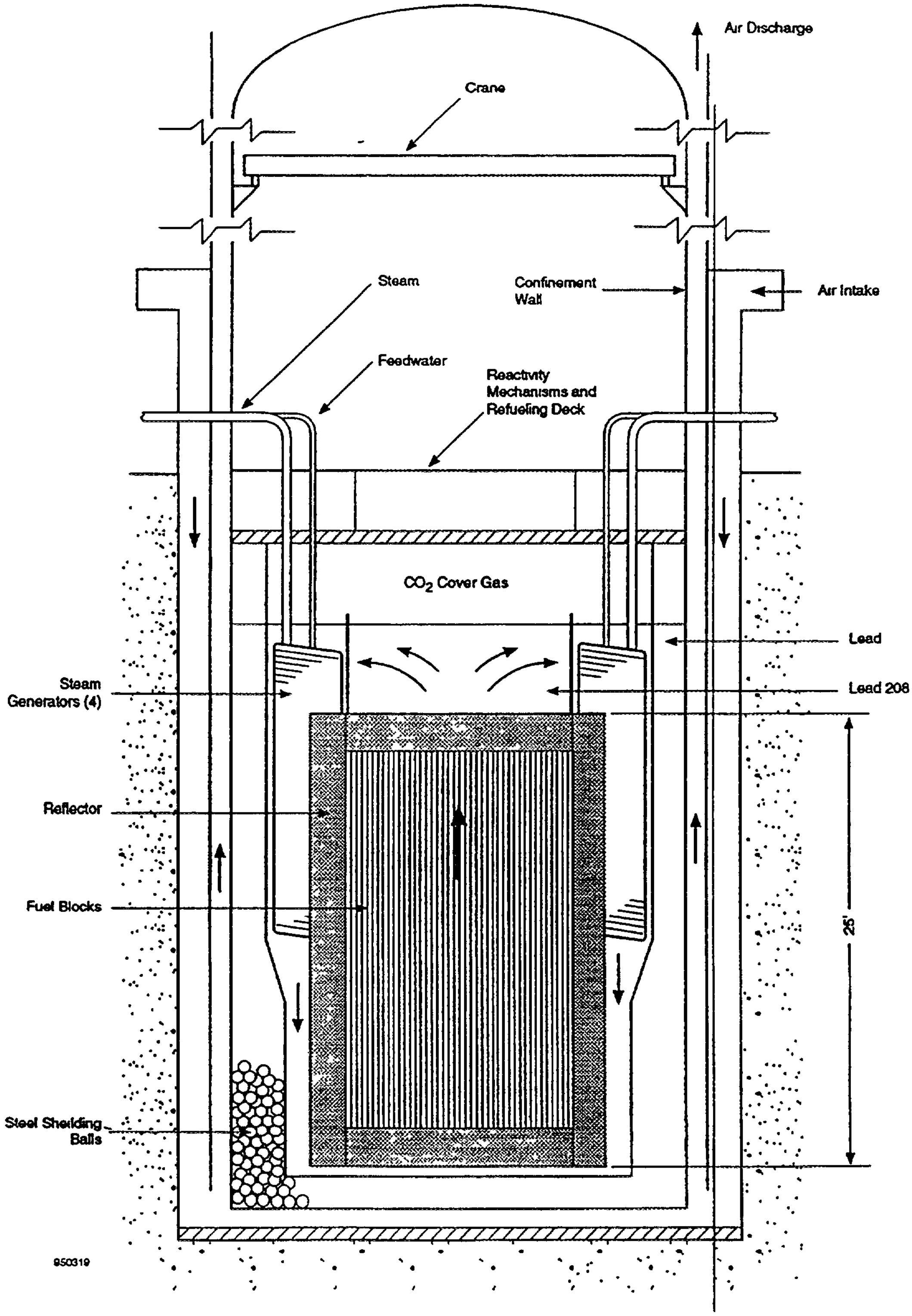

Figure 1: LEADIR-PS 200 Reactor Section

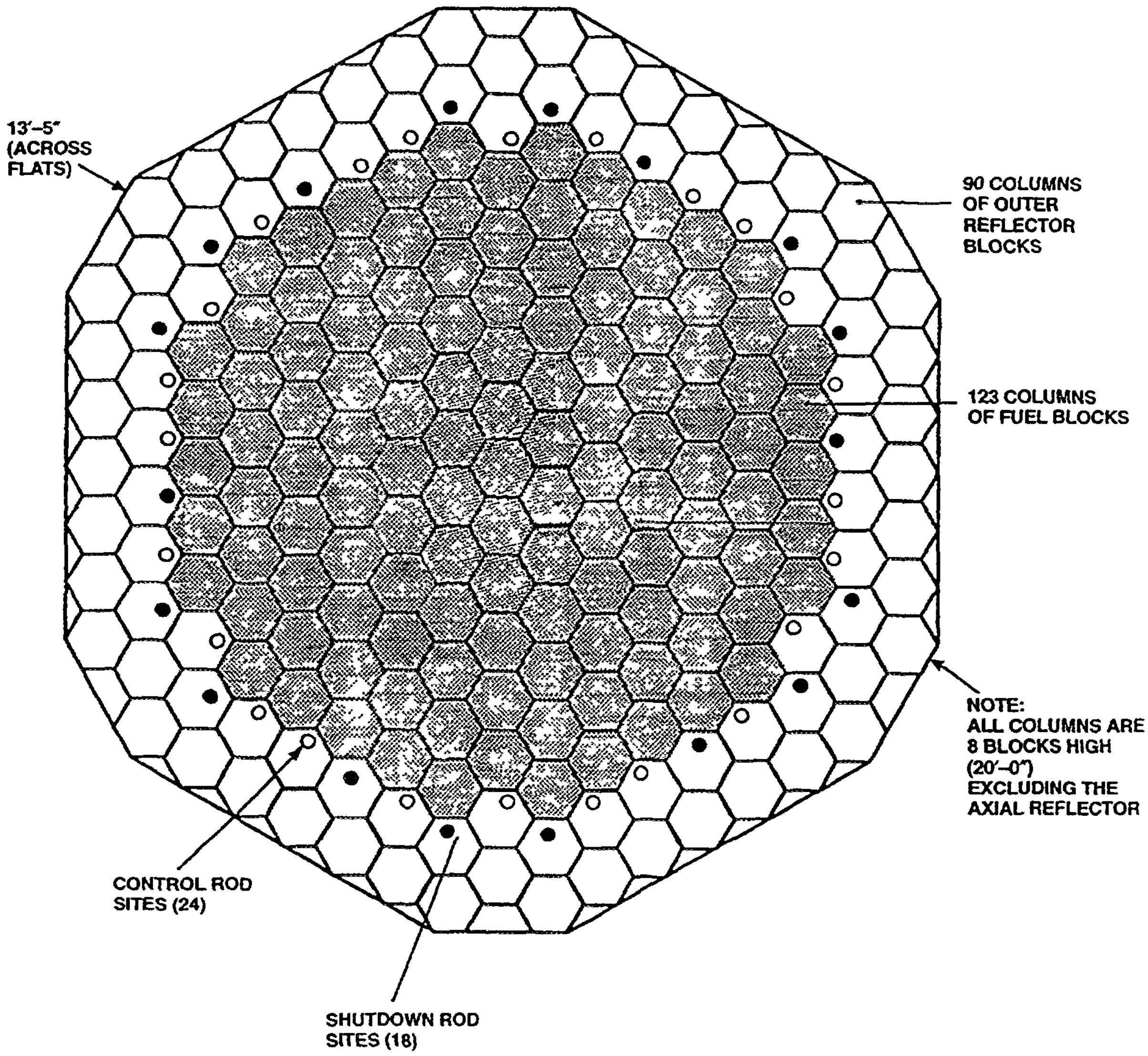

Figure 2: LEADIR-PS 200 Reactor Plan

TABLE 1
ADVANCED REACTOR CHARACTERISTICS

(Relative to Current PWR)

	Inherent Shutdown	Passive Heat Removal	High Temp.	Capital Cost	Operating Cost	Reduced Security Needs	Fuel Cycle	Pressure Vessel
APWR	=	=	=	=	+	=	=	=
AP-600	=	+	=	+	+	=	=	=
CANDU 3	=	=	=	+	+	=	+	+
MHTGR	+	+	+	–	+	+	+	=
PRISM	+	+	+	–	=	=	++	+
+ indicates improved		= indicates same as		– indicates worse				

TABLE 2
MATERIALS BASE FOR CURRENT REACTORS

	COOLANT	MODERATOR	FUEL
PWR	H_2O	H_2O	UO^{2+}
BWR	H_2O	H_2O	UO_2+
CANDU PWR	D_2O	D_2O	UO_2
SGHWR	H_2O	D_2O	UO_2
CANDU OCR	ORGANIC	D_2O	UC
MAGNOX/AGR	CO_2	GRAPHITE	UO_2+
HTR	He	GRAPHITE	UO_2+/TH
LMR	Na	–	Pu/U_2+
USSR SUB	LEAD-BISMUTH	?	UO_2+
+ indicates enriched uranium			

However, unlike the MHTGR which utilizes pressure vessels to house both the steam generating equipment and the reactor, and helium coolant at high pressure, LEADIR-PS incorporates the reactor and heat removal equipment in a pool of lead coolant at near atmospheric pressure.

LEADIR-PS thereby avoids the cost and safety concerns related to pressure vessels, and any prospect of the burning of graphite core materials by maintaining these materials submerged in the lead coolant.

LEADIR-PS 200, a standard 200 MW thermal reactor module based on the LEADIR-PS concept, is discussed in this paper.

3. DESIGN BASIS

3.1 REACTOR OUTPUT

Symbiosis of reactor thermodynamics and physics is necessary to achieve inherent shutdown and passive heat removal for all postulated events. Preliminary calculations indicate that this can be achieved with reactor outputs in the range of 1000MW thermal, utilizing the LEADIR-PS concept.

For conservatism, a reactor of 200MWth was selected for consideration in this paper. This is an appropriate output for many market applications.

3.2 REACTOR ARRANGEMENTS

Two established reactor arrangements, (the Prismatic core and the Pebble Bed core) are possible incorporating the LEADIR-PS concept, each with unique advantages and disadvantages. A feature common to both core configurations in the necessity of holding the moderator and fuel elements down, since graphite is buoyant in the lead coolant. Key factors to be considered in selecting the reactor configuration include the cost of the coolant, and the isotope conversion rate for the coolant in the reactor core and neutron capture cross section of the coolant. The prismatic and pebble bed reactor arrangements are discussed below.

Prismatic Core: A prismatic configuration similar to that adopted by General Atomics for the MHTGR is presented in Figures 1 and Figure 2; the reactor core consists of columns of hexagonal graphite blocks containing coolant passages (Figures 3), and as appropriate fuel, housed within a lead filled steel vessel.

The arrangement utilizes force-assisted coolant circulation to minimize the quantity of coolant in the core. This also maximizes the flexibility of the concept; for example, ordinary lead or lead-bismuth alloy could be used as coolant if the cost of lead 208 (see section 3.3) proves to be too high. On the other hand, if lead (208) proves economic, the design can be modified (increased coolant passage diameter) to provide natural circulation of the coolant under all operating conditions.

Pebble Bed Reactor Core: A pebble bed reactor core arrangement, would be similar to the Siemens/KWU HTR concept, consisting of a bed of graphite pebbles, some of which contain fuel particles, surrounded by graphite reflector. In the case of LEADIR-PS however, the fuel pebbles float in the coolant, and new and recycled fuel pebbles are added to the bottom of the reactor, and irradiated fuel removed from the top (opposite to the THTR-300) during semi-continuous or batch refuelling.

The relatively low inter-pebble forces due to the buoyancy of the pebbles in combination with the lubricating properties of lead facilitate the use of control devices within the pebble bed.

In the pebble bed arrangement about 25% of the pebble bed volume is occupied by coolant. This provides sufficient flow area for natural circulation under all operating conditions. However, detailed thermohydraulic analysis is necessary to confirm the performance of this arrangement under all (normal and accident) conditions.

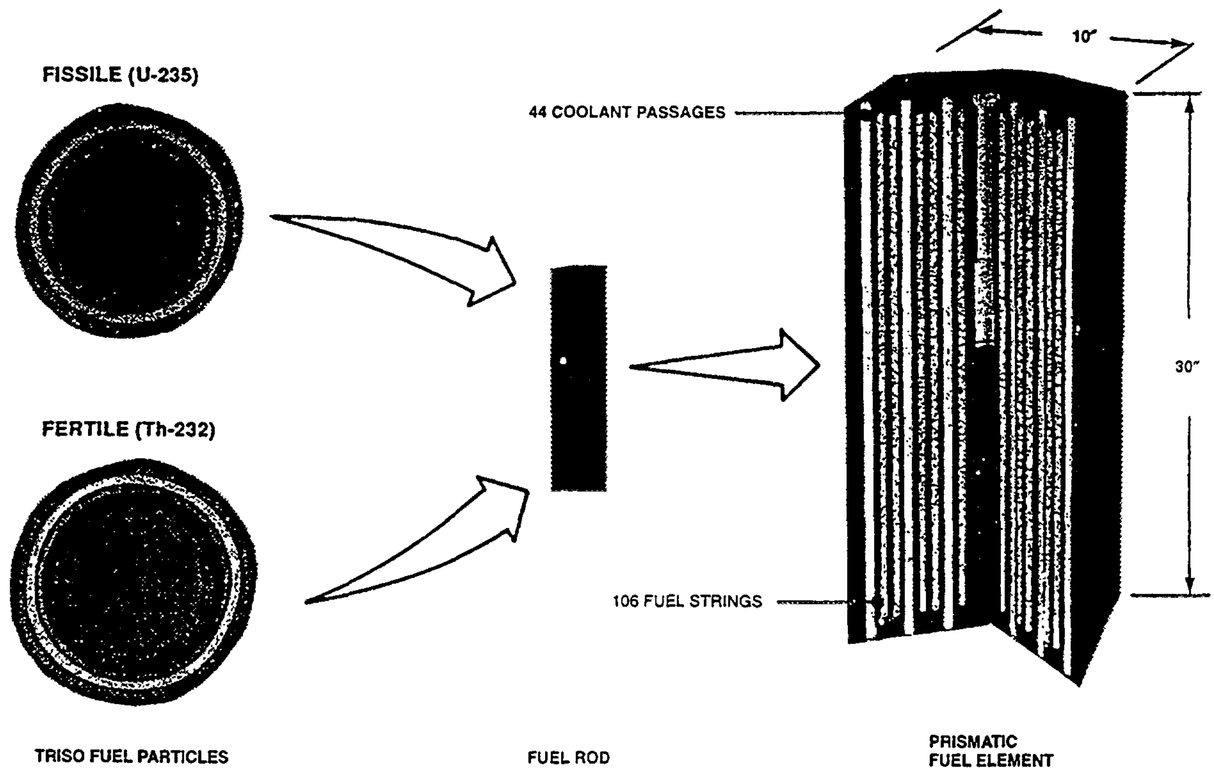

Figure 3: LEADIR-PS 200 Fuel

An annular core configuration, as employed with the prismatic MHTGR core, for reactors with an output above about 600 MW_{TH} also appears feasible with the pebble bed core.

Reference Design: The prismatic core was selected as the basis of this preliminary development and evaluation of the LEADIR-PS concept. This was largely due to the relatively simple analysis, particularly thermohydraulic analysis, afforded by the prismatic arrangement, and the larger data base available for the prismatic MHTGR design. The reference design utilizes periodic off-power refuelling; however, the configuration can also accommodate on-power refuelling (semi-continuous or batch), even if force assisted circulation is used.

3.3 MATERIALS CONSIDERATIONS

General: It is necessary to identify materials for coolant, moderator, and fuel, and for the structure and components of the reactor and reactor systems, which are compatible under all operating conditions. Basic compatibility was established for the combination of lead coolant, graphite moderator, and TRISO fuel, and for lead and the reactor and reactor systems, structures and components. Specifics are discussed below.

Coolant: Various lead and lead alloy coolants were considered. The principal advantage of alloying is a reduction in melting point; lead bismuth eutectic for example, melts at 125°C compared to 327°C for pure lead. Various tritectics have melting points in the mid-ninety celsius range. For this study, pure lead coolant was selected, again, based on the desire for simplicity; alloys such as those containing bismuth add complexity. Lead coolant has a high temperature capability (boiling point of 1555°C at atmospheric pressure). The solubility of graphite in lead is very low while the solubility of silica-carbide is negligible. The lead (208) isotope is the preferred primary coolant due to its very low neutron absorption cross-section (about 1×10^{-3} barnes); it is also a non-moderator. These factors combine to yield a near zero reactivity void coefficient in the well thermalized graphite moderated core.

Lead (208) constitutes about 52% of most naturally occurring lead deposits, and is therefore abundant; however, an economic isotope separation method is necessary to ensure the economic viability of utilizing lead (208) as a coolant., There is also the potential for obtaining lead (208) from specific actinide decay chains; for example the thoria chain yields essentially pure lead 208, although quantities are small.

Moderator: There is extensive experience with graphite moderated reactors. For example Magnox, AGR, RBMK, and HTR in the US and Germany. Hence, graphite performance in high radiation fields, and at a variety of temperatures is established. Graphite is an effective moderator, and does not chemically react with molten lead. The solubility of lead in graphite is very low (about 0.01% by weight at 1000°C for example). A particulate graphite addition system may be necessary to limit graphite displacement from the reflector and fuel blocks.

Fuel: The fuel and fuel element design is derived from that of the MHGR, and illustrated in Figure 3. The TRISO fuel is protected from the lead by the graphite fuel element structure. However, there is no chemical reaction between molten lead and the silica-carbide coating of the TRISO fuel particles, and the solubility of silica-carbide in lead is negligible.

The adapted fuel element design requires a substantial amount of graphite, which contributes to fuel cost and waste disposal costs, if utilized on a "once through" basis. The possibility of removing the fuel pellets from spent fuel elements, and reuse of the fuel element graphite blocks was investigated briefly and is considered feasible. Initial LEADIR-PS200 reactors should operate on the "once through" principle; the technology to recycle the graphite fuel element blocks can be developed later.

Structures & Components: There are a variety of materials available for the structures of the reactor, and for the systems components. These are discussed in more detail in Appendix A.

3.4 POWER CONVERSION

Two basic options exist for electricity generation; a conventional steam turbine driven generator, or a closed cycle gas-turbine driven generator; these are discussed below.

Steam Turbine Generator: This approach is relatively complex and costly, but yields high thermal efficiency (above 40%) in the generation of electricity. Since this paper focuses on the reactor concept, a conventional steam turbine generator is assumed.

Closed Cycle Gas Turbine Generator: A preliminary evaluation of closed cycle gas turbine power module performance utilizing CO2 was completed.

Power turbine, circulator, and generator, operating at 10,000 rpm are housed within a pressure containment, with a turbine inlet pressure of 900psia. The power module is about 1.5m diameter by 4m long, and produces 30MW(e) with an electrical conversion efficiency of about 30%. Exhaust heat is rejected to a low temperature energy user (district heat, water desalination, etc.) at 100°C. Electrical frequency is reduced to 50 or 60 cycle by a solid state converter.

Two such power modules are required for LEADIR-PS 200. The closed cycle gas turbine is very attractive, facilitating modular construction, module exchange maintenance and overhaul, and remote (unattended) operation.

The gas turbine power module could be developed and tested in parallel with and independently from the reactor development program, and adapted when proven.

3.5 DESIGN METHODS

The design process utilized published information on the General Atomics MHTGR to the extent feasible. Physics analysis to confirm the characteristics of the Lead (208),and other lead and lead alloy coolants, and the fundamental neutronic and thermohydraulic core behaviour was completed.

Simple axia symmetric heat transfer models were used for temperature distribution calculations; the accuracy of many of the physical lead and lead alloy properties (viscosity, coefficient of expansion, etc.) used are questionable; refinements in the design may be required.

4. PLANT DESCRIPTION

4.1 OVERVIEW

LEADIR-PS 200 is a standard reactor power module that produces 200MW thermal. It incorporates inherent and passive features that preclude the release of significant activity under any credible situation, without the action of active systems. For example, LEADIR-PS 200 can withstand any combination of loss of forced coolant circulation, loss of normal heat sinks, and reactivity excursion from full power without fuel temperatures exceeding a level at which significant incremental fuel particle failure would be observed.

A section through the LEADIR-PS 200 reactor is shown in Figure 1.

The reactor core, consisting of an array of fuel block columns incorporating TRISO fuel, surrounded by graphite reflector blocks (see Figure 2) is submerged in a pool of Lead (208 isotope) coolant (Figure 1). Four steam generating coils are located around and above the reactor core. During normal on power operation coolant pumps assists natural convection in circulating the coolant downward through the steam generating coils, and hence upward through the reactor core.

The primary lead coolant pool is surrounded by a secondary lead pool consisting of ordinary lead, contained within the reactor vessel structure. The secondary lead pool and steel shielding provides a short term heat sink for the most severe design basis events, allowing the volume of the primary pool to be minimized; it also reduces heat loss during normal operation.

It is necessary to achieve a balance between minimizing normal operating heat losses from the reactor, and assuring sufficient reactor cooling under accident conditions (loss of all active heat sinks for example). In the configuration analyzed, a layer of solid lead forms on the outer wall of the secondary lead pool during plant operation, thereby reducing normal heat loss while retaining heat rejection capability under accident conditions. A natural convection air cooling system surrounds the reactor vessel, to cool the reactor cavity and maintain concrete temperature at an acceptable level.

The inner and outer walls of the reactor vessel are steel, connected by reinforcing webs; each wall is fully capable of containing the lead coolant in the event of a rupture in the other wall. These walls, together with the steel balls in the innerspace, provide neutron shielding for the concrete.

4.2 REACTOR CORE

A cross-section of the prismatic reactor core is shown in Figure 2. It consists of 984 fuel blocks in an arrangement of 123 columns, 900 radial graphite reflector blocks arranged in 90 outer columns, and 426 axial graphite reflector blocks, with one block located at each end of each fuel and reflector column.

The fuel block structure is shown in Figure 3. The fuel blocks are similar but smaller than those utilized in the MHTGR; the width of the blocks is reduced to 10″ from 15″, and the main coolant passage diameter is increased from 5/8″ to 3/4″; the ratio of fuel to moderator is maintained, while allowing larger coolant passages to reduce coolant circuit pressure drop. The graphite radial and axial reflector blocks have the same external dimensions as the fuel blocks. The columns of reflector blocks that contact the fuel blocks each contain 12 coolant passages. Many of the radial reflector blocks contain control rod, shutoff rod, or heater locations. The axial

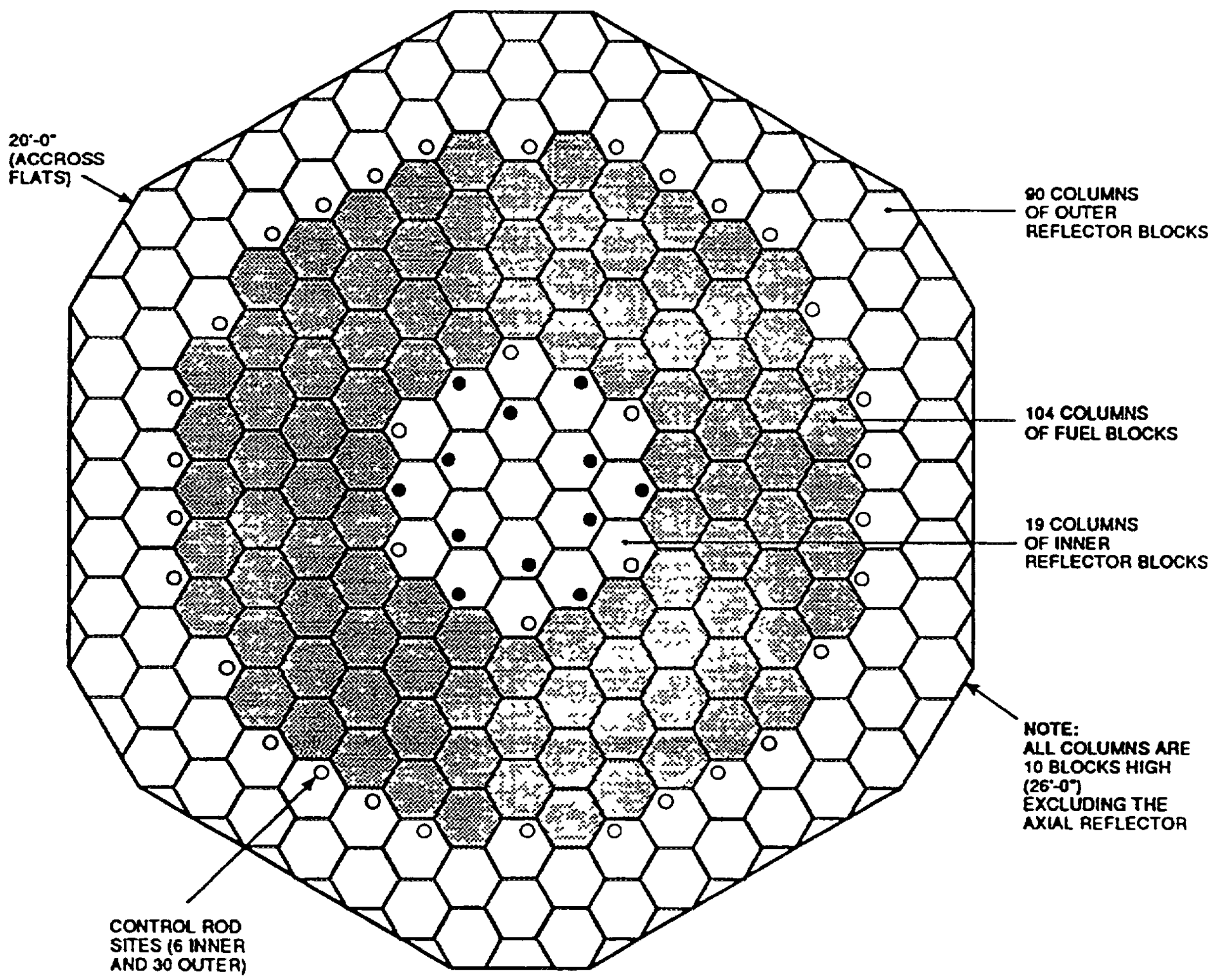

Figure 4: LEADIR-PS 500 (500 MW_{TH}) Reactor Section

reflector blocks contain coolant, control rod, or heater passages that correspond to the fuel or reflector blocks of each column.

The suitability of the LEADIR-PS concept to reactors of larger output was assessed, and a reactor output limit of about 1000 MW_{TH} established. Reactors with outputs above about 600 MW_{TH} require an annular core configuration (similar to MHTGR). The central reflector blocks provide additional heat capacity to accommodate postulated accident conditions, and locations for the control and shutdown rods necessary for reactor control and shutdown.

An arrangement of the core for a 500 MW_{TH} reactor, utilizing an anular core is shown in Figure 4.

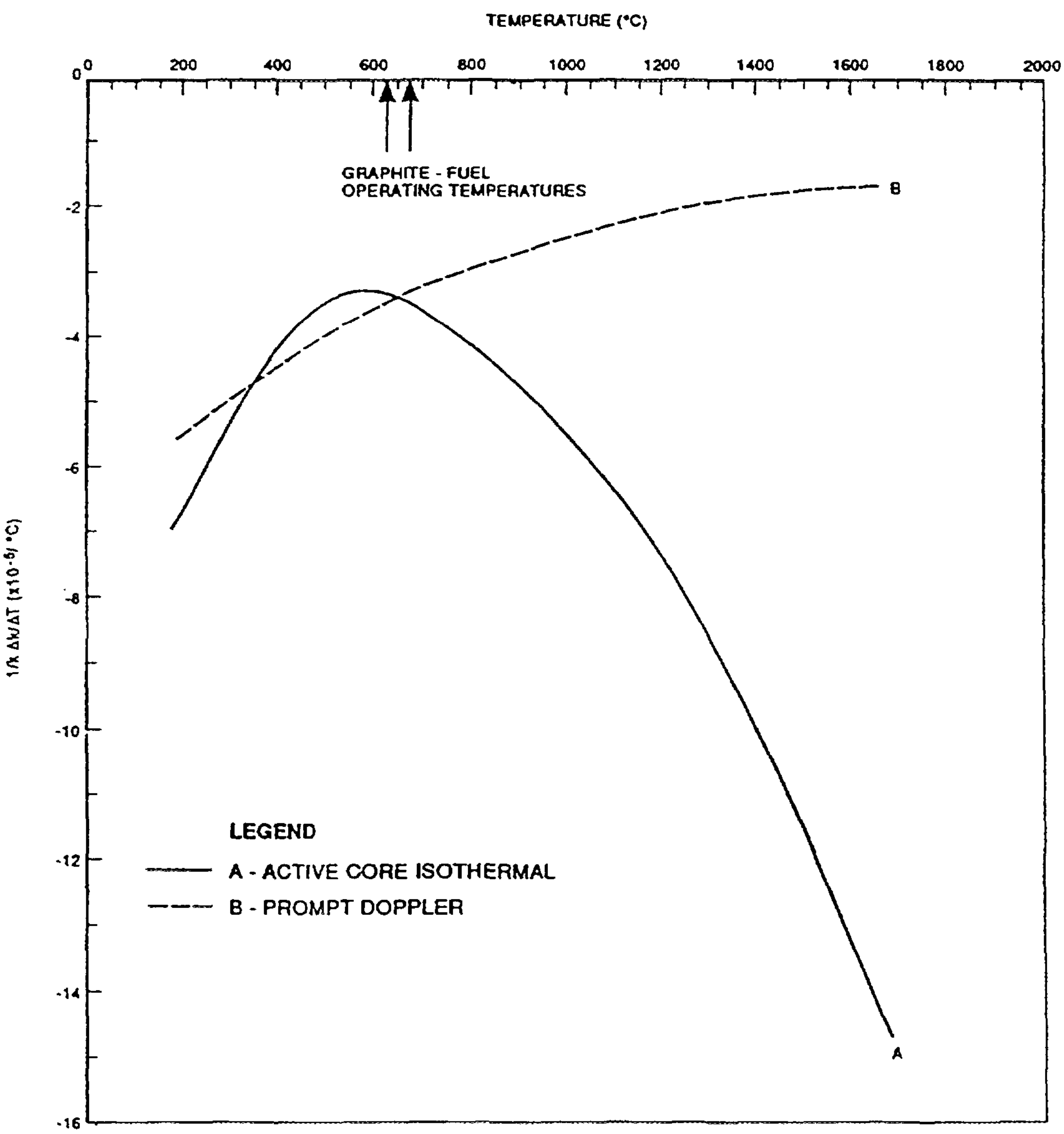

Figure 5: Temperature Coefficient - Equilibrium Core

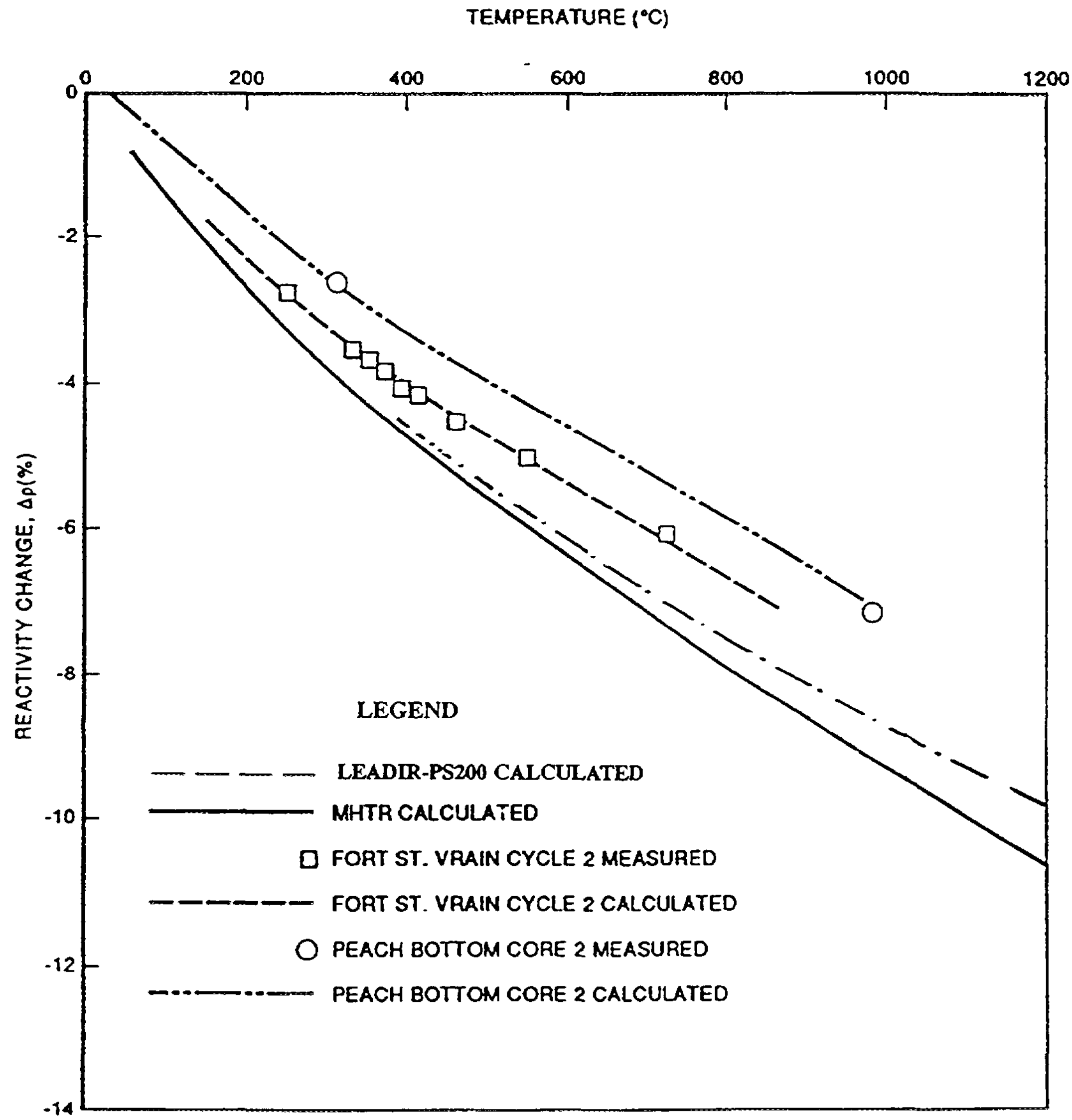

Figure 6: HTGR Temperature Reactivity Effects

4.3 PROCESS CONDITIONS

The reactor coolant (Load 208) conditions were selected so that the temperature leaving the steam generator coils was comfortably above the melting point of lead, in a range of relatively low viscosity. The core outlet temperature was selected to be high enough to provide good electricity conversion efficiency using gas turbine generators, but low enough to assure that the fuel and graphite elements operated well below the experience limit of HTG reactors. The key process parameters are presented in Table 3.

Table 3

Key process parameters

Lead Temperature	–	Core inlet	360°C
		Core outlet - max.	730°C
		- avg.	700°C
Lead Flow	-	Core total	11000 lb/sec
	-	Coolant passage (each)	2.2 lb/sec
Coolant Pump Power			1000 Kw
Steam Temperature			550°C
Feedwater Temperature			200°C

4.4 OPERATION

The variable speed coolant circulating pump varies coolant flow as a function of reactor power, thereby maintaining a constant core coolant inlet temperature over the normal power operating range. Feedwater flow to each of the steam generators is also varied as a function of reactor power to maintain a constant degree of super heat in the steam.

Independently controlled isolation devices are provided on the feedwater line and steam line to each steam generator.

Incidence during which the primary lead coolant pool solidifies are very infrequent. However, in the event that the lead coolant solidifies during a cold shutdown, coolant flow is re-established by providing steam at 350°C to the steam generators, or by the use of electric heaters that can be positioned in the reflector.

4.5 REACTIVITY CONTROL

Reactor control is provided by 24 control rods located in the outer reflector. A backup shutdown system consisting of 18 shutdown rods is also provided;

In the event that all normal reactivity control systems fail, the negative temperature coefficients of the graphite moderator and fuel (Figures 8 and 9) shut the reactor down well before the integrity of the fuel is threatened.

4.6 HEAT REMOVAL

Heat removal is normally via the steam generators utilizing the normal plant feedwater system. Two of the steam generators are provided with feedwater from an independent system for decay heat removal.

In the event that normal heat removal systems fail, heat is transferred to the reactor cavity cooling system via a combination of convection, conduction, and radiation. In the event that the cavity cooling system is unavailable, decay heat is transferred to the surroundings, primarily via radiation.

In the event that overcooling of the steam generators should occur to the extent that they "freeze", windows retained by a medium melting point material located between the steam generators near the top of the primary pool will open when the coolant temperature reaches 1000°C thereby allowing the coolant to bypass the steam generators and maintain natural circulation. If purely natural circulation is utilized, the flow geometry can be arranged so that the windows are simple openings, without incurring steam generator flow bypass during normal operation.

Should coolant circulation down the outside of the outer radial reflector blocks somehow be prevented, natural circulation is established upward through the fuel columns, and downward through the reflector columns.

The system is also designed such that, in the very unlikely event of both the primary and secondary pool walls failing, the core remains covered by lead and fuel cooling is assured.

4.7 REFUELLING

For the reactor core configuration selected, about 1/3 of the fuel blocks are replaced every 2 years during the refuelling/maintenance outage.

In advance of refuelling, the reactor is placed in secure shutdown condition via the insertion and securing of all shutdown and control devices. With the temperature stable at an average of 340°C and reactor decay power below 0.5%, the automated refuelling assembly is installed above the refuelling hatch in the mechanisms deck. The refuelling assembly includes a circulation and cooling system for the CO_2 reactor pool gas blanket, and fuel manipulator arm. Except for a cooling system to permit operation above 330°C and the use of CO_2 in place of He, this system is conceptually similar to that employed on the MHTGR.

Alternately, semi-continuous on-power or batch refuelling can be accommodated. Preliminary calculations show that if the fuel blocks of each column connected by a tie rod, are employed the coolant bypass of the core when a fuel column site is vacant is acceptable, even when force assisted circulation is employed.

The lead coolant serves as an excellent lubricant during the refuelling operation, and allows for sufficient clearance between the prismatic blocks to assure refuelling capability. Even under severe seismic conditions, the lead between the prismatic blocks precludes damage due to column impact.

During refuelling, a portion of the fuel removed from the reactor core is returned to the core, together with new fuel elements, while the remainder is placed and sealed into interim storage flasks. The cylindrical flasks which accommodate a column of six fuel elements, are then placed in sealed fuel storage pits located to one side of the reactor. A natural convection air cooling system removes decay heat from the fuel located in the fuel storage pits. After two years this fuel is transferred to a central irradiated fuel management site, or part to the next refuelling process, and the fuel storage pits are reused.

4.8 MAINTENANCE

LEADIR-PS 200 is designed to operate for extended periods (2 or more years) without maintenance. Major maintenance, much of it by "change-out" is completed during scheduled maintenance/refuelling outages every two years. This work is completed by a dedicated maintenance team that, together with the necessary equipment and tools, travels from one LEADIR-PS 200 plant to another to complete the maintenance outages. Assuming a 2 week maintenance outage, a single maintenance crew could service 40 or more LEADIR-PS 200 units. Equipment overhaul and major services are provided by a central maintenance facility.

LEADIR-PS 200 does make provision for unscheduled maintenance, should component failures occur. For example; facilities are provided to allow the removal of a steam generator or control rod with the reactor at power. The reactor coolant pump can be removed with the reactor operating at reduced power. Unscheduled maintenance, which is minimal, is provided by a separate maintenance crew, that is available to a large number of LEADIR-PS 200 plants.

4.9 OPERATION

LEADIR-PS 200 plants utilizing steam turbine generating equipment will likely require the 24 hour per day presence of an 'operator'. This operator will have the capability to shut the reactor down, and to take a number of protective actions, but no authority to start the reactor, or otherwise control the unit.

Control of LEADIR-PS 200 plants will be provided via a central operations facility, capable of monitoring and controlling 20 or more LEADIR-PS 200 units. This facility would also co-ordinate an emergency response team that could respond to any LEADIR-PS 200 accident condition (team member to be available, but not on duty at anytime). Members of the maintenance crew will be present during reactor startup following a maintenance outage. These people will be specifically trained to assist the central control and monitoring group during this procedure.

4.10 SECURITY

Extensive automated security systems will be provided in LEADIR-PS 200. These will detect the unauthorized entry of persons onto the premises, and detect any unauthorized attempt to enter the LEADIR-PS 200 buildings. The security system will provide data to both local authorities (such as police) and to the central control and monitoring facility.

5. SAFETY FEATURES

5.1 GENERAL

The safety features of the LEADIR-PS 200 are dominated by the safety characteristics common to HTGRs as well as features unique to the particular configuration of the LEADIR-PS 200 module. The general safety characteristics are dominated by the inherent characteristics of the coolant, core materials, and fuel as described below.

- LEAD COOLANT — Lead coolant has several advantages including a boiling point well above the assured shutdown temperature of the core; therefore minimal coolant level measurements are required and pump cavitation cannot occur. Further, there are no significant reactivity effects associated with the lead (208) (it is essentially transparent to neutrons) and no chemical reaction between coolant and fuel or moderator is possible.

- GRAPHITE CORE — The strength of the graphite core and the stability of the ceramic fuel coating at high temperatures result in a wide margin between operating temperatures and temperatures that would result in core damage. Further, the high heat capacity and low power density of the core and the heat capacity of the coolant pools result in a very slow and predictable temperature transients.
- COATED FUEL PARTICLE — The multiple ceramic coatings surrounding the fuel kernels constitute tiny independent pressure vessels which contain fission products. These coatings are capable of maintaining their integrity to very high temperatures in the 1600° to 1800°C (2910 – 3270°F) temperature range. Zirconium carbide coatings are capable of even higher temperatures in the 2200°C (4000°F) range.

The physical configuration of the reactor power module assures decay heat removal by passive means in the event that all normal heat sinks are lost, without the action of the operator or any active system, for all credible events.

The design allows the complete separation of the Nuclear Steam Plant (NSP) from the conventional plant; the NSP does not impose any safety demands on the conventional plant beyond those of a typical fossil-fired station.

5.2 RESPONSE TO POSTULATED EVENTS

LEADIR-PS 200 does not pose a safety concern to the public for any credible event; events that are a risk for other reactor types are non-concerns. Coolant channel blockage for example, although very unlikely, does not have significant consequences. Even if all coolant channels in a fuel column were blocked peak fuel temperatures in the block (at full power) would only reach about 1000°C, well below the threshold for fuel failures.

Steam generator tube ruptures are also accommodated. Immediately following the ruptures significant steam generation occurs; however, the lead coolant quickly solidifies in the region of the rupture and steam releases approach those due to the flashing of feedwater only. There is no lead-water reaction, and no public safety concern results.

The location of the reactor core below grade, and submerged in lead also makes the core impervious to most external events. Even an aircraft crash with an ensuing fire would not pose a threat to the public.

5.3 RESPONSE TO TRANSIENTS

LEADIR-PS 200 has a graceful and safe response to all anticipated transients. For example, an overcooling event (as could be caused by loss of feedwater control or spurious opening of steam relief valves in combination with control system failure) causes the core inlet temperature (normally 350°C) to fall; as the freezing point of 327°C is approached the coolant viscosity increases, coolant flow decreases, and in the absence of any control system action, the negative temperature coefficients of the fuel and moderator reduce reactor power. Heat removal is maintained by natural convection.

6. THE FORMULA FOR SUCCESS

6.1 DEVELOPMENT

To minimize development cost and time, maximum use of existing technologies and expertise is required. Hence, cooperation between countries and institutions with relevant experience, for example, with graphite moderator, lead coolant, TRISO fuel, and plant and equipment design is essential. There are two principal areas requiring development, both related to the coolant. These are:

a. To identify a process and estimate the cost of producing lead 208. If this proves to be prohibitively expensive, the concept remains viable with the use of ordinary lead coolant, but requires some modification.

b. To identify the effects of radiation on lead 208, and establish methods of chemistry and isotopic control.

6.2 MASS PRODUCTION

The small reactor will not be economical if produced and operated in small numbers (say 5 or 10 units). A significant population of small plants of near identical design is required. This offers a number of essential advantages:

a. The development, design, and licensing costs are distributed over many units.

b. The economics of mass production and volume construction are realized.

c. The economy of remote monitoring and operation achieve their potential. For example, a single control centre could operate 20 or more units located in diverse locations.

d. Economies of maintenance is realized. For example, if the small reactor requires a maintenance outage of two weeks every two years, a skilled crew could complete the on-site maintenance of 20 or more units a year (i.e. maintain 40 or more units).

e. Central service and overhaul facilities can economically and efficiently service reactor components, on a volume basis, affording maximum economic benefit.

f. Refuelling can also be completed by a dedicated crew, serving 40 or more units if refuelling were required every 2 years.

g. Waste management and disposal can also be efficiently coordinated.

h. Long term research, development, problem solving, and product advancement can be shared by a large user base.

The situation is somewhat analogous to the aircraft industry. A few 737s are not economical, but many 737s operated by many airlines are. Similarly, the economies of volume production and operation of small reactors can be realized even if they are in diverse locations, or even in difference countries. LEADIR-PS is amieable to volume production and operation.

7. CONCLUSIONS AND RECOMMENDATIONS

Confirmation of the market for small reactors is necessary before expending significant effort on their development. Given the limited financial resources of most nuclear vendors, this must go beyond traditional market studies, and firmly establish the intentions of potential users. This is best accomplished by interested users forming and supporting, a buyers organization. This organization would set requirements for the small reactor (including comprehensive economic and performance requirements) conceptually similar to the EPRI requirements for Advanced Light Water Reactors, and co-ordinate and promote the international co-operation necessary to develop a viable small reactor. Support by potential users throughout the process should consist of both finances and expertise. The expert contribution of potential buyers could be substantial, if their respective nuclear development programs were co-ordinated and directed at a common goal.

The LEADIR-PS design which utilizes a novel combination of established technologies and a minimum of new technology to meet the suggested small reactor requirements should be seriously evaluated by the potential small reactor users.

Ferrous alloys have demonstrated varying compatibility with liquid Pb, Pb-Li and Pb-Bi eutectics. In general, ferritic stainless steels such as HT-9 and Fe-9Cr-lMo exhibit superior corrosion resistance in these environments relative to austenitic stainless steels like 316. While exposed to flowing (0.35 l/min.) Pb-17Li at 550°C [1], HT-9 possessed a uniform corrosion rate of 20 um/year. The dissolution rate activation energy was estimated to be 92.5 kJ/mol. Ferritic steels are usually attacked in a uniform manner under these conditions [2], with some intergranular attack occasionally observed. At temperatures greater than 370°C, the weight loss for HT-9 is lower than for Fe-9Cr-lMo [3].

The behaviour of austenitic stainless steels in liquid Pb-Li is quite different. A porous ferrite surface layer forms on the steel, depleted in Cr, Ni, and Mn [2,4,5]. The thickness of the ferrite layer is a function of temperature, cold work, and grain anisotropy. Strong Ni and Cr depletion in the ferrite layer has been observed [4,5]; Ni content falls from 12 % to 0.5-1 I, and Cr content is reduced from 17 X to 3-5 %[4]. It should be noted that the selective leaching of Cr is generally not observed for ferritic steels. A steady state corrosion rate of 30 um/yr was observed for 316L exposed to 400°C Pb-17Li [4]. In addition, the presence of oxygen in liquid Pb-Li increases the ferrite thickness from 45 um to 70 um, and results in greater Ni depletion [5].

In general, the corrosion resistance of ferritic stainless steels is 5-10 times greater than that of austenitic steels in Pb-Li[2]. For a corrosion limit of 20 um/yr, the peak operating temperature is 500°C for ferritic steels and 410°C for austenitic steels[3]. Consequently, the use of austenitic stainless steels is not recommended [6]. For temperatures in excess of 450°C, ferritic steels require Zr or Ti corrosion inhibitor additives, so their suitability for temperatures as high as 750°C is questionable.

Vanadium, niobium, tantalum and molybdenum and their alloys have demonstrated good corrosion resistance in liquid Pb-Li [8]. These materials have dissolution rates in the range 0.001 to 0.004 $g/m^2/day$ at 645°C [8], which is far superior to the performance of Fe-based alloys. However, the cost associated with many of these materials would make their application impractical for a full-scale reactor.

Several materials have demonstrated acceptable corrosion resistance in liquid lead. Fecr-alloy (Fe,0.2 C,13 Cr,4 Al) presented no visible signs of attack after 551 days in 700°C liquid Pb[9]. In addition, Tantiron (Fe,13.5 Si,0.5 Mn,l C) showed no visible signs of attack after 56 days in 720°C liquid Pb [9]. Mo has shown no detectable attack after 300 days in 800°C Pb [9]. In particular, the alloy Mo-30W demonstrates outstanding corrosion resistance. The solubility of various metals in liquid lead are presented in Table-I for comparison. Iron and Cr are lightly soluble in liquid Pb, but sufficient to cause solution attack. A summary of the corrosion performance of several materials in a liquid lead or liquid Pb-Li environment is provided in Table II.

Table III lists the neutron absorption cross sections for many of the metals described above, as well as their cross section relative to the typical reactor material, zirconium. Materials with a very large cross section relative to zirconium would result in a reduction in the thermal utilization factor 'f' and hence a reduction in N_{ff}. Consequently, Ta, W, V, Mo and Ni based alloys would be impractical choices for a reactor core. From this literature survey, it appears that Fecralloy would provide the greatest promise as a containment material for liquid lead. In addition Tantiron may be an alternate choice. More extensive studies on the applicability of inhibitors such as Ti should be undertaken to determine their affect on the corrosion resistance of these materials.

Table I - Solubilities of Fe, Cr, and Ni in liquid Pb. After [2]

Temperature (°C)	Solubility (ppm by weight)		
	Fe	Cr	Ni
400	0.2-0.08	0.002	1800

Table II - Corrosion performance--of various materials in liquid Pb and pb - Li After (7,8,9).

Material	Comment
<u>In liquid Pb-Li</u>	
Fe-9Cr-lMo	good resistance @ 600°C. limited @ 800°C
HT-9	good resistance @ 600°C, limited @ 800°C
1.4922 (12 Cr,0.5 Ni,0.5 Mn,1 Mo, 0.5 V)	Nj leached out, slightly greater dissolution than V
316L	poor resistance
Mild C-steel	good resistance @ 600°C, limited @ 800°C
Low Cr-steel	good resistance @ 600°C, limited @ 800°C
2-9% Cr steel	good resistance @ 600°C, limited @ 800°C
V, Nb, Ta and alloys	good resistance, dissolution rate < 0.004 $G/m^2/day$
<u>In liquid Pb:</u>	
Hastelloy-N (Ni,16,5 Mo, 7 Cr,5 Fe,1 Si,8 Mn)	severe attach after 4 days at 700°C
Croloy 2.25 (Fe,2.25 Cr,1 Mo,0.5 Mn)	severe attach after 4 days at 700°C
Croloy 2.25 + plasma sprayed Mo	no visible cracks/minimal wt. gain after 56 days at 720°C
Croloy 2.25 + calorizing coating	no visible attack after 56 days at 720°C
Croloy 2.25 + Ti inhibitor	no visible attack after 85 days at 700°C
Fecralloy (Fe,0.2 C,13 Cr,41)	no visible attack after 551 days at 700°C
Tantiron (Fe,13.5 Si,0.4 Mn,1 C)	no visible attack after 56 days at 720°C
Alumina	no wetting or attack after 282 days at 700°C
Silicon Nitride	no wetting or attack after 56 days at 720°C
Mo	no detectable attack after 300 days at 800°C

Table III - Neutron absorption cross sections for various materials [10]

Element	Atomic #	Capture Cross-section (b)	Cross-section relative to Zr
Zr	40	0.182	1
Fe	26	2.50	13.7
Al	13	0.232	1.3
Ni	28	4.54	24.9
C	6	0.0034	0.02
Si	14	0.16	0.9
Mo	42	2.65	14.6
Nb	41	1.15	6.3
Ta	73	22	120.9
W	74	18.5	101.6
V	23	5.06	17.8
Ti	22	6.1	33.5

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Appendix I
DESIGN AND DEVELOPMENT STATUS OF SMALL AND MEDIUM REACTORS 1995

M.A. Al-Mugrabi
International Atomic Energy Agency, Vienna

1. INTRODUCTION

The increased confidence in reactor technology coupled with the argument about the economics of scale have led most of the industrialized countries to design progressively large reactors up to 1600 MWe power capacity blocks. This, as a result has shifted all earlier designs to the small or medium size reactors. Moreover, a medium size plant capacity has shifted to include 700 MWe power units. This allows the SMR range to be suitable to many developing as well as some of the developed countries. Small reactors are being used for heat applications and power generation on a smaller scale in remote areas and for developmental purposes. For the purpose of this technical report, small reactors has been taken as reactors with a thermal power of 400 MWth or less.

Taking as a rule of thumb power units to be 10% of the total grid size, many utilities worldwide could only support a small or mid size plant. Site specific constraint such as the absence of adequate cooling water and the national technological level in many developing countries make the SMR range most suited for power application. Remote areas, isolated from the national grid such as the case in Russia and many remote islands in the South East region of Asia provide suitable condition for smaller power units that could run economically even in the range of 40-100 MWe or less. The assessment of the world market projection for seawater desalination carried out by the IAEA as part of the options identification programme concluded that a sufficient large demand in the years 2015 and beyond will support the installation of sizable desalination capacities. A reasonable part of this, is in the size ranges of 50,000 to 100,000 m^3/d. This range corresponds to 25-50 MWe net power output. The most convenient size of the reactor would be in the small reactor range, both if the reactor power be used totally for desalination or if the reactor is to operate in a cogeneration mode.

A small reactor could have simpler safety systems due to the larger margins, making the implementation of passive safety easier to engineer. Small reactors have also been an important vehicle for the development of new designs.

The need for smaller size reactors in many parts of the world has made the potential market for such system to look large and has encouraged the manufacturers to continue their efforts. In spite of the fact, that the market has turned out to be over estimated, the number of reactors in the SMR range in operation or under construction closely matches the case for large units (Fig.1). These factors have made the SMR area getting a wide attention worldwide. In this paper the main design features and market potential of the SMRs in all three reactors lines namely WCRs, GCRs and LMRs will be discussed. Design and development efforts worldwide will be highlighted.

Design features

Looking at the currently active reactor design concepts, several common basic features could be summarized as follows:

- Low power density.
- simplified configuration.
- Utilization of natural driving forces (passivity).
- Enhanced overall plant arrangement.
- Modular construction and fabrication.
- Reliance on proven technology.
- Low radiation exposure dose.

Light water reactors

These basic design features have been used to develop different design, concepts in the light water reactor (LWR) area that are present today. A large number of these concepts are safety driven. Innovation in residual heat removal, make up systems and the elimination of large LOCA by design are some aspects worth noting. New concepts and the overall simplification of SMR systems are an area that deserve discussion.

Makeup systems

Gravity fed accumulators, steam injectors and pressurized makeup tanks have many variants in the different designs. Accumulators are generally designed for large volume make up water and usually operated under N_2 pressure in assuring high water flow into the Reactor Pressure Vessel (RPV). Makeup tanks are usually operated at the full primary circuit pressure and water injection is gravity driven. Steam injectors are also used. In this case

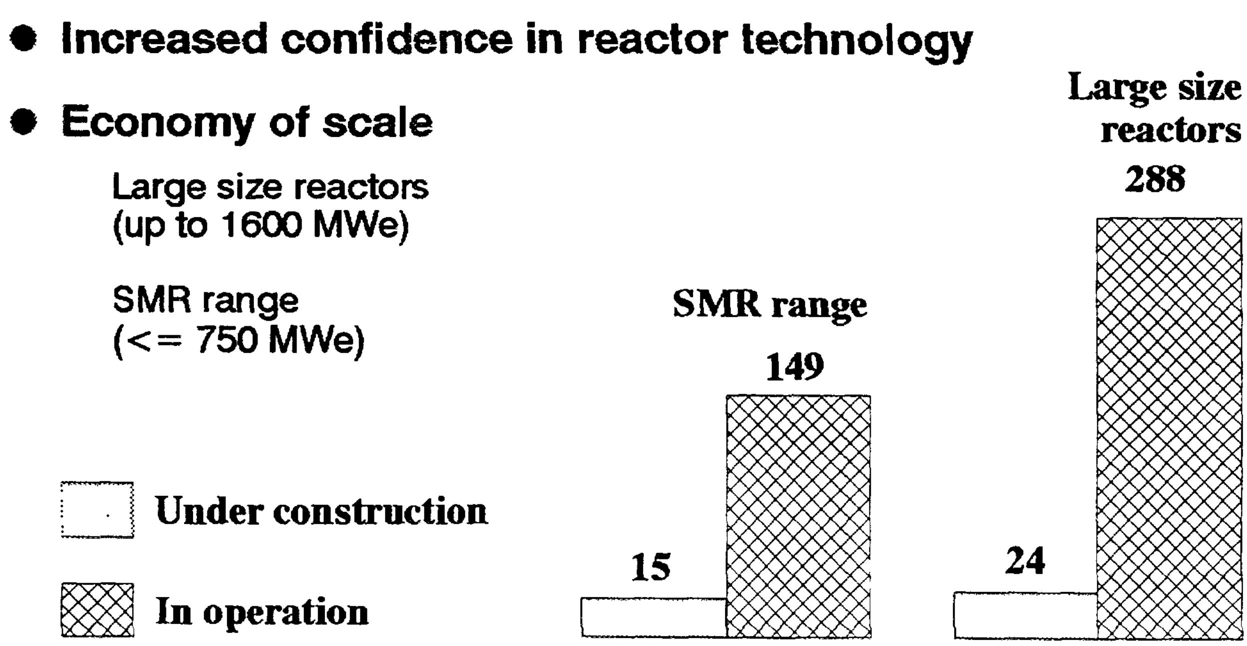


FIG. 1. SMR Design and development status.

steam is used to inject water from a water storage tank. The technique is as old as the steam locomotive industry but is needed here at a much higher pressure and hence their reliability has to be demonstrated. Experimental programmes to test their effectiveness are underway.

Residual heat removal systems

Residual heat can be removed from the primary or secondary circuit and from the containment. There is, thus scope for diversity without being extravagant in the complexity of the systems. There are normally two types of residual heat removal systems (RHRS); systems used during normal operation, and those provided in the safety grade systems. Almost all the SMRs go for passive RHR systems.

Residual heat is usually removed from the primary coolant via a heat exchanger which transfer the heat via natural convection to an external heat exchanger in a larger water storage tank or directly to the atmosphere. In most cases valve operation is required to put the system in operation, but in a limited cases the system is designed for a continuous operation which has a significant effect on efficiency. This has partially overcome by extracting some of the heat in a feedheater.

An other alternative used in some designs is rechanneling part of the primary water to an auxiliary heat exchanger, as it is the case in the AP600 design.

In integral reactor designs, steam generators within the RPV are made use of as a convenient route for residual heat removal. Valve movement in this case is inevitable and hence some redundancy must be established.

Additionally, a bleed and a make-up system could be utilized to remove residual heat from either the primary or secondary circuit. The AST500 and SIR designs make use of such alternative.

Innovation has been exercised by residual heat removal walls of the containment. Steam from the reactor is condensed directly on the walls of the containment or by means of a heat exchanger arrangement. The latter system allows for a full double containment as required in some countries. For practical reason, direct heat transfer through the containment puts a limit on the reactor power level that could be attained with such arrangement. This limits such possibility to the SMR range.

Loss of coolant accident

The first line of defence in depth is prevention. Prevention could start in the design stage. Pipe connection would always run as part of the primary and secondary circuit but if they are kept small in number and size the possibility and consequence of a LOCA would be substantially reduced. BWRs by their basic design have such features. In integrated PWRs the entire primary circuit is enclosed in the RPV with the exception of the CVCS and makeup water piping, achieving the same objective.

Second line of defence is to ensure that core uncovery does not take place. the main feature is to avoid any penetrations in the RPV well above the core level by several meters to provide coolant inventory for boil off and relax the need for fast makeup water in the large quantity.

The other approach is to provide an outer vessel which is either permanently flooded (PIUS concept) or can readily be flooded (guard vessel). The PIUS concept provide for the most innovative design in the SMR range. The primary coolant is isolated from the highly boronated water in the out vessel by hydraulic seals. In case of any significant disturbance to the normal flow the boronated water would always enter the primary system providing large water inventory and causing the scram of the reactor. Other design that used the PIUS concept are the Italian ISIS and the Japanese ISER designs.

Small LOCA in older design was caused by pump seal failure. This has been eliminated by selection of canned rotor pumps. These pumps have a considerable operational experience. If the circulating pumps position is selected in the RPV bottom area to avoid cavitation, their penetration has to be designed to the same standard as the RPV head with regard to integrity and inspectability.

Heavy water reactors

Several reactor design have appeared in the SMR range that use Heavy Water as a moderator (e.g. CANDU 3, CANDU 6, PHWR 500, PHWR 220). The main features of this technology line is:

- The use of natural uranium
- On-power type of refueling
- Very low access of reactivity
- No RPV, but rather pressure tube is used.

In addition to these features, all designs allow for the passive residual heat removal in case of loss of power. CANDU 3 emphasized the assurance of high capacity factor, the maximization of component life and the easy replacement of components.

Gas cooled reactors

The design features of the gas-cooled reactors have centered on the use of ceramic coated particle fuel, and the use of an inert gas as a coolant. The fuel particles are uranium oxide or carbide approximately 0.5 mm in diameter with a multiple layer coating of pyrolitic carbon and silicon carbide forming a micropressure vessel around the individual fuel particle that could withstand around 800 bar of internal pressure. The fuel particles are bound together in a graphite matrix capable to retain fission products up to 1600°C and provide high temperature gas outlet. The operating temperature restriction is mainly due to the limitation on the structural material. New materials for heat transfer systems capable of operating at high temperatures (e.g. 900°C) make them attractive for direct use in gas turbine and high heat process applications. The coolant being an inert gas eliminates any chemical or energetic reaction with core structure.

These features combined with a negative temperature coefficient of reactivity, large heat capacity of the graphite and the large design margins make the reactor safety extremely difficult to challenge.

The HTGR technology line utilizes complete passive system for RHR and shutdown. Residual heat is taken away by radiator surrounding the reactor pressure vessel utilizing natural circulation. Shutdown is accomplished by simply dropping small absorber spheres by gravity force.

Liquid metal reactors

The main design features of this line of technology is the low operating pressure due to the high boiling temperature and high conductivity of the coolant and the ability of the coolant to absorb the heat with insignificant moderation.

Small fast reactors can have totally passive safety systems. The main characteristics of this design are the metal fuel and sodium coolant.

Currently there are four modular small or medium-sized, liquid metal reactors. The Advanced Liquid Metal Reactor (ALMR) former (PRISM), the Modular Double Pool Reactor, the 4S (Super, Safe, Small and Simple) and the BMN-170.

2. DEDICATED NUCLEAR HEATING PLANTS (NHP)

The power range of nuclear heating reactors is generally lower than SMR power reactors. They are rated from about 2 to 500 MWth. Apart from the high temperature reactors, their outlet temperature is aimed mainly at district heating or sea water desalination and does not exceed 130°C. This corresponds to a primary circuit temperature of around 200°C, and a power density ranging from 2 to 60 kW/l.

The smaller size and lower pressure resulting from these requirement leads to simplification of the overall design and allows for the maximum utilization of natural processes. Simplifications have been achieved through a less massive RPV, through integration of the primary circuit in the RPV, and in the safety systems and containment. Further simplifications have been made in the use of natural circulation for normal heat removal (made possible by the large safety margins in the NHP design) and by the use of passive safety systems.

Over a dozen reactor designs are known worldwide, most of which have originated in developing Member States. The economics of these reactors, however, can only be justified in remote regions isolated from a national grid. Only a few of the concepts have been constructed (e.g. AST-500 in Russia, HR5 in China and SLOWPOKE in Canada). As a result there is only little operational experience.

Simplification

Plant simplification imply simplification of system arrangement, operation, maintenance, inspection, and quality assurance. Modular isolation and prefabrication are key design features of the SMR systems. SMRs of the new generation with no exception lay great emphasis on simplification. The use of passive systems leads also to simplification. This is mainly due to the elimination of multiple redundancies and safety grade power supplies. Integral designs eliminate large pipe penetrations leading to further simplification. Hydraulic drive mechanism provided further simplification for some designs (NH200).

Most designs have reduced the number of components (valves, cables, piping) by as high as 80% in the most favourable conditions. Digital electronics and distributed systems provided further simplification and increased reliability.

Traditional control rod drives require a lot of space either above or below the core. There are possibilities to use liquid absorber materials, which do not require the space for rod drives and for in vessel storage when withdrawn. There have also been designs for in vessel mechanical drives (PSR, MRX, HR 200). These eliminate the need to consider control rod ejection which is one of the main, but unlikely, reactivity accident initiators. A more radical solution is in the JAERI SPWR design where liquid filled tubes are used instead of control rods.

The elimination of large primary circuit pipes in integral PWRs allows an easing of the containment specification. Pressure suppression systems for PWRs become feasible and several versions have been proposed.

There are very significant developments in instrumentation and control systems allowing simplification and an increase in reliability at the same time. Many proposals use process computers and digital electronics leading to a complete redesign of the architecture of the control system.

The use of passive safety systems leads directly to simplification in design since it eliminates the need for multiple redundant safety systems with their redundant safety grade power supplies. A system which relies only on gravity for its operation has no problem about the availability of its power supplies and has a reliability determined only by the integrity of its piping and flow channels.

Passive safety

The passive safety approach deserves a separate mention since it is a feature of many SMR designs. The original incentive was to produce designs which could cope with any accident initiating event coupled with the failure of all engineered safety systems. There are thus different degrees of passivity and the recent IAEA document on reactor terminology has gone to some lengths to include all the different types of system for which their designers claim passivity [5].

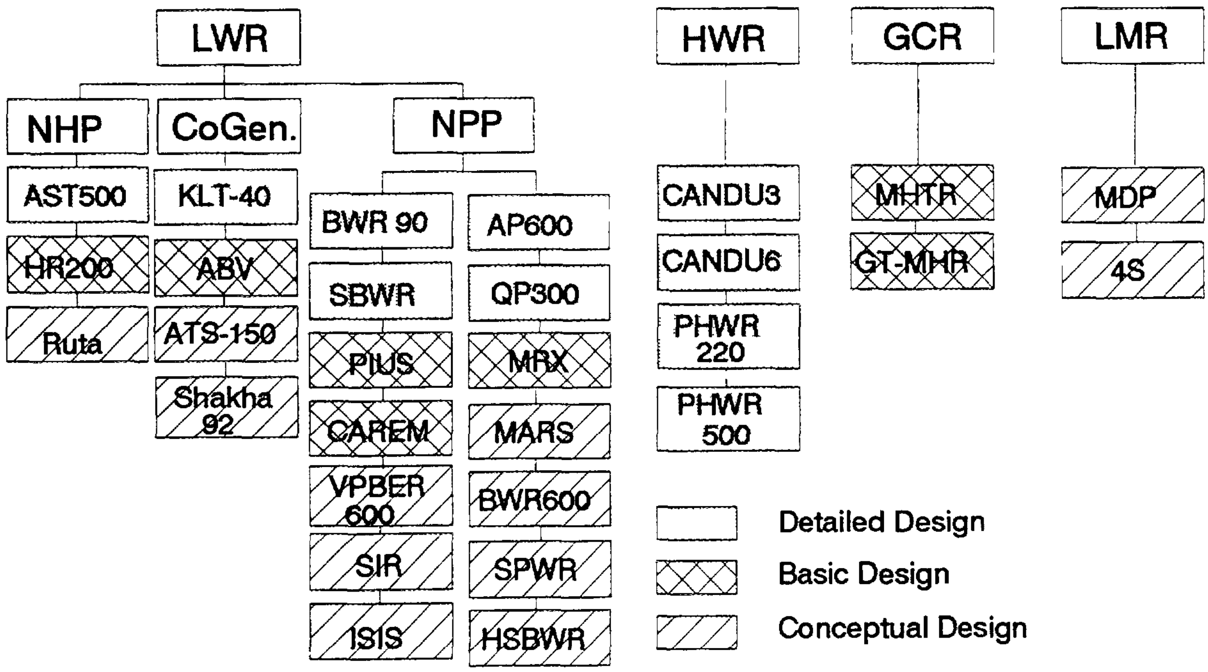
Small and medium size reactors have utilized the reliance on passive principles to attain safety functions. In some cases complete reliance on passive safety systems has been the main incentive for a number of designs (e.g. AP600, PIUS, MHTGR). One of the main design objectives is to be able to deal with any accident initiating event coupled with the failure of the engineered safety systems utilizing solely natural process such as gravity driven and natural convection. This would need no operator intervention for a long, may be indefinite, period. Some reactors have achieved this in their design with their safety systems being passive both in initiation and in operation. Others have relayed on stored energy (e.g. batteries, springs) for process initiation. The important issue here is not the degree of passivity but rather the reliability of such systems and the possible need for redundancy. Experimental programmes for verification of these issues are underway.

SMR development

Nuclear energy is playing an important role in supplying a significant portion of the world electricity demand. In spite of the slowdown or stoppage of nuclear programmes in many countries in the last decade utilization of nuclear power is picking up momentum at various bases in South East Asia, Eastern Europe and the former Soviet Union.

Table 1 provide the main reactor designs in the SMR range being developed and their design status. It could be seen that the development effort worldwide is large with many

TABLE 1. SMR DESIGN STATUS



being in the detailed design stage and some are under construction (Table 2). Other prototype demonstration plants have either started operation or under construction (e.g. MONJU, 10 MWt, HTGR, HR5 in China).

2.1. The SMR market

The current growth of population and energy demand is dominated by developing countries. There are many places and applications where this increased demand will be best met by power plants in the SMR range, due to a small grid system or for application in a remote area or for a special purpose.

The world primary energy consumption amounts to well over 300,000 Peta joules and over half of that is used as hot water, steam and heat. Only a few nuclear power plants are being used for heat applications (district heating, heat for industrial processes, and seawater desalination). Potential nuclear heat applications include enhanced oil recovery, petroleum refining, petrochemical industries, and methanol production from hard coal. The need for potable water in some parts of the world is large, vital for sustaining development, and ever increasing. Clearly nuclear heat and power production could play a major and important role.

Nuclear power at present is used mainly for electrical power generation which only forms 30% of the energy market. There have been numerous studies on the use of SMRs for heat applications rather than electrical generation and some of these studies have shown the SMR option to be viable both technically and economically [2]. Future expansion of nuclear application, beside addressing large power generation demand, may also come from more spread energy market involving smaller units for process heat applications and small scale power generation in remote areas.

2.1.2. SMR projects

With such a range of possible applications in many different parts of the world, a large number of different R & D and design projects have been set up. Fig 1 lists those for which descriptions have been submitted to the latest IAEA review on the subject and indicates the status of their development. LWRs, HWRs, GCRs and sodium cooled reactors all have active development work in various Member States.

Over the past 30 years there have been many market surveys for SMRs. They have shown a potential for sales of a large number of reactors before the turn of the century. These estimates of the market have turned out to be grossly overoptimistic but have encouraged developers to continue their efforts. In spite of a moderate response from the market, there is still a very large development effort continuing but few of the advanced SMR designs have yet been in operation to demonstrate their capabilities. Indeed, few of them have been funded through the detailed design stage to make them ready for construction. They do, however, present a variety of solutions to the problems of reactor design for future designers to draw on and to give an impression to purchasers of the capabilities of current designs, which could be developed to meet their needs. The Agency is currently involved in a study on the market potential of SMRs which is expected to be concluded in 1997.

There is thus a gap between the designs available but not built, and their exploitation in what appears to be a potentially large market.

2.1.3. Bridging the gap

Possible ways of bridging the gap would be for vendors to collaborate on one design to spread the design and development costs and for users to collaborate to define an SMR requirements document for particular applications. There have been some notable vendor collaborations in industrialized countries demonstrating that this is a possible way forward. Requirements documents have been produced for power generation and requirements have been harmonized on a regional basis (Asia, Eastern Europe, Western Europe and North America). Developing Member States having similar technological and financial circumstances could establish their version of requirements for an identified market (e.g. desalination). Such requirements could be taken up in some of the developing projects to enhance their prospects for constructions. An other important aspect to the deployment of nuclear power in developing countries is the development of the required manpower and infrastructure for a successful programme.

2.2. Incentives for development

Small and medium size reactor development has many incentives; some are economic others are safety related. The motivation for these developments has included the need to enhance public acceptance of nuclear power. The simplification of designs should improve the transparency of their reactor safety. Another incentive to SMR development has been its suitability for the implementation of new design approaches. Innovative and evolutionary designs with novel features have been implemented in the SMR range. A passive safety approach has so far been the technology of small and medium reactors. SMRs have particular characteristics which can enable them to be economically viable in spite of losing the advantage of the economics of scale.

The incentives for the development of SMRs can be summarized as follows:

- Simpler design,

  An SMR can be modularised more easily and constructed in a shorter time than larger plants, thus reducing constructions costs (including interest during construction) and generating earlier revenues.

- Increased safety margins leading to a longer grace period,
  Passive safety features simplify the design and attain the required safety objective in a different way compared to large plants with more active safety systems. This could reduce cost and facilitate the presentation of the safety of the reactor to both regulatory authorities and the public.

- Lower severe core melt frequency and minimum accident consequences.

- Better match to grid requirements,
  SMRs can provide a better match to small grids or to a slow growth of energy demand. Taking into consideration the potable water demand and the corresponding energy requirement, a SMR would be a suitable candidate for a developing country starting its nuclear programme.

- Better use of nuclear industry infrastructure and manpower skills in countries with small nuclear programmes.
  One 600 MWe unit every 2 years is preferable to one 1200 MWe unit every 4 years.

- SMRs could open up energy markets,
  SMRs can be used for process heat, desalination, district heating and enhanced oil extraction as well as power generation.

- Lower financial risk due to:
  lower financing requirements per unit,
  shorter and better predictable construction schedule.

2.3. Objectives and requirements for SMRs

Development or deployment of SMRs could take place in a programme under the following general objectives:

1. The size of reactor is appropriate to a geographical location, distribution network or application.
2. It should be economic within the constraints of the other objectives.
3. It must be demonstrably safe and licensable.

These general objectives are applicable to reactors of any size but there are particular aspects of SMRs which help in meeting them.

1. Size. SMRs are appropriate for remote regions with limited load. They are appropriate for utilities with small grid systems. They are appropriate for some dedicated applications such as desalination, district heating or process heat possibly in a co-generation mode.

2. Economics. SMR designs all aim to simplify the design to reduce costs and offset to some extent the economies of scale. Modularisation allows a greater element of factory construction and assembly and is generally less expensive than work on site. It leads to shorter construction times and savings in interest during construction. The reduced capital requirements compared with large plants may well be attractive to purchasers.
3. Safety. Most SMRs make extensive use of inherent safety features and passive safety systems. Such systems are appropriate to SMRs and are harder, if not impossible, to engineer on large reactors. They tend to be simpler than active systems resulting in a simpler safety case and easing the problems of public acceptability.

While objectives provide for general and long-term applicable targets for nuclear reactors of present and future designs, requirements provide more specific, clear and complete statements by utilities in a given country. The requirements are usually grounded on well proven technology and long experience of commercial operation. The design requirements usually take into consideration problems of the past and incorporate new features assuring simple, robust, and more forgiving designs. They also provide for a common ground for regulators and vendors on licensing issues. Well defined requirements agreed upon by regulators, vendors and utilities provide for investor confidence. The design requirements usually cover the whole plant (i.e. NSSS, BoP, safety systems etc.) and provide clear specifications with regard to performance, maintainability and plant economics. Taking into consideration infrastructure and experience, requirements in most developing countries and some industrialized countries are expected to be easily fulfilled by a small or medium reactor.

3. PROGRAMMES FOR SMR DEVELOPMENT

3.1. Current activities in Member States

Nuclear energy plays an important role in supplying a significant portion of the world electricity demand. Reactor generated heat has been utilized in several parts of the world for district heating, process heat application, and seawater desalination. It should be noted here that over 50% of the world energy demand is utilized for either hot water or steam production. Such processes could be carried out more efficiently and cleanly utilizing nuclear energy.

Some South and East Asian countries believe strongly that nuclear power will be a principle source of energy for many years to come. Small and medium reactors form a major part of this activity. The People's Republic of China has a well developed nuclear capability having designed, constructed and operated nuclear reactors. A 300 MWe PWR (QP300) has been in operation for three years and two 600 MWe reactors are under detailed design and site preparation. Longer term plans call for development of a 600 MWe passive reactor (AC600). A 5 MWt integrated water cooled reactor has been built and operated for five winter seasons (since 1989) for district heating. Another purpose of the 5 MW reactor is the development work for other applications such as desalination. Construction of a 200 MWt demonstration heating reactor has been started aiming at start of operation by the year 1998. A 10 MWt high temperature gas cooled reactor for process application is also under construction.

India has adopted a prime policy target of self reliance in nuclear power development, based on heavy water moderated reactors. Five units of the 220 MWe PHWR type are under construction and all are expected to be in operation by the year 1997. An additional four units of the same type and an extra four units of a scaled up 500 MWe type are planned.

Japan has a preference for large reactors on the available sites to maximize the power output from them. There is a very strong and diverse programme of reactor development supported by the big industrial companies, by the national laboratories and by the universities. At least seven different designs are currently being worked on in the SMR range; namely SPWR, MRX, MS 300/600, HSBWR, MDP and 4S. SPWR and the marine reactor MRX are integrated PWRs. The MS series are simplified PWRs. HSBWR is a simplified BWR. MDP, 4S and RAPID are small sodium-cooled fast reactors. Preliminary investigations have shown a high level of safety, operability and maintainability. The economics of these systems are promising and they are expected to form part of Japan's next generation of reactors.

Japan has also a development programme where gas cooled reactors in the small and medium size range are under development. A High Temperature Engineering Test Reactor (HTTR 30 MWt) has been under construction since 1991 at Oarai.

Korea has ten PWRs and one PHWR in operation and has an ambitious programme for the further development of nuclear power. Most of the existing plants are of the large PWR type, but, since April 1984, there has been a policy to install medium size PHWR (~ 700 MWe) to diversify supply and operation. One PHWR is operating, another is under construction and two more are in the stage of seeking a construction permit. In addition, a relatively small 330 MW(th) integral reactor is also currently under development for a cogenaration purpose.

Indonesia has a very rapid growth of population spread over 13,000 large and small islands. There is a clear future potential for reactors in the SMR range. However, the main island has over half the current population and could take a large station; a feasibility study covering this and other aspects of Indonesia's possible nuclear programme has been undertaken. The outcome is in favour of the nuclear power option. 7000 MWe of nuclear capacity is being considered up to the year 2015. Optimal plant size is being looked at and a number of 600 to 900 MWe units are being considered. Indonesia has deposits of tar sands for which extraction based on nuclear heat using HTGRs is being investigated. A programme on public acceptance is being executed.

Thailand has just started a feasibility study on the construction of a nuclear power plant.

The current Russian programme is largely based on 1000 MWe units but the 500-600 MW range is well represented in the development programme. Two units of 600 MWe each are planned in the Far East region of the country for the period 2000-2010. Two others in Karel'ska are planned for the same period.

Russia is a country with a clear scope for the deployment of smaller plants due to its huge land mass with remote communities living in areas with harsh winters. The nuclear energy option seems to have favourable economics compared to conventional sources for application in remote areas, especially for domestic heating. Several reactors of small size (10-30 MW) are planned for construction around the year 2000.

Eastern Europe has VVER units of the 440 MWe size but for the future larger units are considered. In Western Europe, most utilities have opted for large nuclear power plant (1000-1500 MWe) if they have opted for nuclear at all. On the basis of several different national development programmes on SMRs, many innovations using a wide variety of coolants, fuel, containments and safety features have been worked out. More recently, SMR-specific development effort in Western Europe has decreased because of reductions in governmental funding.

In the USA, the AP600 in the SMR range is being supported and aggressively marketed worldwide, in addition to a large reactor design (ABWR). In Canada a perceived need for a simpler, cheaper reactor which could be more easily demonstrated to the public as safe has led to the development of a smaller version of the CANDU line. Design and safety requirements for the next generation of reactors have been identified both in Canada and in the USA by the utilities and governmental agencies. In North America, Medium Size Reactors are expected to supply a significant share of nuclear electricity in the future.

In Argentina the work on Atucha 2 (745 MWe PHWR) is continuing. Argentina has carried out a development effort for the design of a small pressurized water cooled reactor "CAREM". The system has a total power of 100 MWth and it is of the modular integrated type. The basic design of the system is complete and it is currently undergoing detailed design.

North African and Middle Eastern countries have identified a strong need both for electricity and for power for desalination and several of them are looking at the nuclear option. The reserves of fossil fuel are massive in some countries but others rely largely on imports. The water problem is compounded by low rainfall, a rising population with increasing expectations for its standard of living and by a lowering of the water table in the traditional sources under the desert sands. A study for the North African countries of the economic feasibility for nuclear desalination has been completed [3]. A feasibility study for a demonstration facility for seawater desalination in Morocco is expected to start in 1996. In Egypt, a feasibility study has been completed for a medium sized NPP[4].

TABLE 2. SMRs UNDER CONSTRUCTION

Country	Number of units	Name	Type	Power MWe (net)	Expected date of commissioning
Argentina	1	Atucha II	PWR	692 MWe	1998
India	4	PHWR 220	HWR	202 MWe	1998-1999
Republic of Korea	3		PHWR	650 MWe	1997-1998-1999
Pakistan	1	Chashnupp	PWR	300 MWe	1999
Romania	2		PHWR	650 MWe	1996-2002
Slovak Republic	4	Mochovce	PWR	388 MWe	???
Russian Federation	2	AST 500	PWR	500 MWth	1998

From information provided by Member States (see Table 2), it can be seen that several nuclear power plants in the SMR range are under construction around the world. Nuclear power investment on a worldwide basis has preferred large units due to the economy of scale, especially in the industrialized countries. This can be clearly seen from the number of nuclear power plants in operation today (Fig. 1). The number of units currently under construction in the SMR range is in the same range as that of the big power plants. These data show that SMRs could play an important role in many industrialized and developing countries.

Conclusions

Small and medium size reactor design and development is a very active area. The design work has made maximum use of simplification, modularization and prefabrication. No SMR design has not tried to make use of natural forces. The SMR range has been the main vehicle for innovative designs.

Small and medium sized reactors have good potentials for future deployment specially in developing countries both for power generation and process heat application.

Small and medium reactor systems provide an attractive option for a wide range of applications worldwide. The design approach and design characteristics of the SMRs with regard to size, economics and safety appear to provide favourable conditions. Specific requirements on these topics will provide a common ground for the suppliers and interested users to further the discussion on specific design requirements such as performance, operability, maintainability, reliability. For successful deployment, overall cost must be competitive with other alternatives, taken into consideration the main objectives.

Among existing reactor designs, the pressurized water reactor of the integral type seems to be well suited for a wide range of low power applications, including seawater desalination. Integral reactors using natural circulation of the primary coolant and utilizing passive safety systems appear to be technically capable of achieving a high degree of safety and reliability.

An important aspect to the introduction of nuclear power in a developing country is a well planned and executed programme on the development of the required infrastructure according to the objectives of the programme.

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Appendix II
PUBLIC ACCEPTANCE OF SMALL REACTORS

D.S. McDougall
AECL Technologies Inc., Canada

INTRODUCTION

The success of any nuclear program requires acceptance by the local public and all levels of government involved in the decision to initiate a reactor program. Public acceptance of a nuclear energy source is a major challenge in successful initiation of a small reactor program. In AECL's experience, public acceptance will not be obtained until the public is convinced that the specific nuclear program is needed, safe and of economic and environmental benefit to the community.

The title of public acceptance is misleading. The objective of the program is a fully informed public. The program proponent cannot force public acceptance, which is beyond his control. He can, however, ensure that the public is informed.

Once information has begun to flow to the public by various means as will be explained later, the proponent is responsible to ensure that the information that is provided by him and by others is accurate.

Most importantly, and perhaps most difficult to accomplish, the proponent must develop a consultative process that allows the proponent and the public to agree on actions that are acceptable to the proponent and the community

AECL's EXPERIENCE

AECL has decades of experience with power reactors and waste management issues in Canada. Until the 1970s public acceptance of nuclear energy was not an issue and the information generally available to the public was largely of a technical nature. After TMI, as a sector of the public became concerned about the use of nuclear energy, organizations on both the for and against side of the issue sprang up and commenced surveys on public opinion, and also initiated programs to influence public opinion. In the 80s and 90s, pro-nuclear agencies began advertising the positives of nuclear energy to counter the concern raised by the Chernobyl accident, movies such as the China Syndrome, novels and anti-nuclear groups which had succeeded in turning the majority of the population very cautious about use of the nuclear option. In the U.S., increasing regulation and frequent court mandated construction reviews resulted in a number of U.S. utilities losing large sums of money on their nuclear programs. As a consequence, nuclear proponents, including AECL, in a number of countries with established successful programs were forced to defend these programs and gained a great deal of experience in dealing with the public on nuclear issues.

AECL also gained specific experience in dealing with the public on small reactor programs. The heating reactor program built around the 10 MW_t SLOWPOKE Energy System provided direct public experience with small communities (Sherbrooke, Quebec and the University of Saskatoon, in Saskatoon, Saskatchewan). At these sites AECL gained experience appropriate for the subject of this week's session, and experience that is relatively recent, 1989, compared to the siting experience for the large power reactors which took place a number of years ago.

From these contacts, a number of people's concerns have been identified. They are

1) Lack of Control

People fear that they have no control over the nuclear industry. It is seen as large, reporting to government authority beyond the reach of the average individual, and therefore able to act according to its own wishes.
Compounding this existing feeling of non-control, nuclear programs are usually seen as and in fact usually are megaprojects. Megaprojects have a history of perceived megarisks - cost overruns, and often expensive to operate.

2) Safety

Safety is a concern both with regard to the plant itself, and to disposal of nuclear waste. The public fear what they cannot see, smell or touch, and the risks from plant and waste effluent fall into this category.

3) Disaster

A further complication for the nuclear proponent is that human nature is fascinated with disaster. The nuclear bomb, while unrelated to a power plant, has left a legacy of potential disaster - amply reinforced by Chernobyl

DEALING WITH PUBLIC CONCERNS

How do we deal with these concerns? What is our message?

The proponent must ensure that the community understands that it retains control over the siting process. In Canada and the United States and in most of Europe, the public does indeed control the siting process. Therefore, even though the public may not understand the science of nuclear fission, they are able to demand answers to public concerns before the project can proceed.

The proponent must be able to satisfactorily explain the community need and benefit, and the ability of the nuclear option to meet that need. This discussion will include data of an economic nature, security of energy supply, etc.

The message should include a discussion on the specific project advantages such as jobs and technology transfer. In fact nuclear has a very positive message - how should it be delivered?

MESSAGE DELIVERY

Nuclear 's positive message has not been delivered adequately. However a number of lessons have been learned.

To deliver the message the proponent must:

1) Establish a presence within the community. Outsiders are not believed to have the community's interests as a top priority. AECL has found that employees must be moved into the community in order to gain credibility. These employees are seen as truly making a commitment to their new community, and it is appreciated. By moving to the community, the individual also gains a much better appreciation for the true concerns of the citizens of that community.

2) Establish credibility with the local leaders, both the obvious and those 'behind the scenes'. In most communities, a few people are very influential, hard-working and community minded. These are the people who will take the time to understand the issues, and in turn be able to explain the issues to other citizens. Local leaders have more credibility than the proponent's employees.

To help these interested people to understand the issues, it is the proponent's responsibility to:

provide easy to understand literature

give honest answers to the difficult questions - explain the pros and cons and ensure that risk is understood

be available to help potential supporters

bring guest lecturers as required.

3) Establish a relationship with the media that is frank but friendly. The proponent must insist on correcting errors (errors in fact, not opinion) as they occur. The proponent must also learn to understand the role of individuals working in media, gain an appreciation of their deadlines, space constraints and tailor his message accordingly. In other words, treat the media with respect.

4) The proponent must ensure that local staff have sufficient autonomy to deal with issues as they arise. Avoid having to 'check with Head Office". In this way, local staff will be able to deal with last ~~minutes~~ speaking engagements, radio talk show phone-ins, letters to the editor in a prompt manner so as to keep the public aware of accurate information as opposed to rumour. It is AECL's experience that a quick response is preferable to the well-crafted, but delayed response.

5) Nuclear projects represent opportunities and politicians will be interested in exploring the option. In order to gain and maintain political support, the proponent must work with the public to ensure that the politicians receive vocal support from their citizens. It is not sufficient to gain political support and then expect the politician to carry the arguments to completion. The proponent must stay involved throughout the public information program to ensure that the community remains accurately informed.

Most importantly, and underlying all of the above, the public will believe the message, if they trust the messenger.

THE MAIN DIFFICULTIES IN A PUBLIC INFORMATION PROGRAM

The main difficulties in a public information program are:

1. Public apathy. The proponent is interested, the opponents are interested, but experience indicates that the vast majority of the public has too many other issues on their minds to get deeply involved with a nuclear siting issue. Consequently, it is difficult to get a good understanding of the communities' concerns, and therefore difficult to issue information that will meet the needs of the silent majority. As a result it is often difficult to achieve a sufficient level of understanding with the public at large.

Given the reality that only a small minority will ever be truly interested, the approach is to identify those groups whose support is essential and work with them - community leaders, political leaders and businesses/employers who will benefit.

2. Potential for Irrational Debate. Opponents will strive to introduce factors that have nothing to do with the project at hand, e.g.~~s~~ Chernobyl, nuclear weapons.

A public that is frightened of horrific consequence will be unwilling to take even a small risk if they perceive that the nuclear option will put the community at risk. Again the approach is to identify those groups whose support is essential and ensure they understand the merits and demerits of the nuclear program, and the true intent of the opposition. The proponent must strive to educate the media, schools, and community organizations to explain the issues and the objectives of the proponents and opponents.

Which methods are most successful to explain a nuclear program to the public and key groups?

1. Simple, easy to understand literature - pictures, short sentences, facts

2. The proponent must be accessible - set up information centres in town markets (i.e. go to the people)

3. Avoid townhall meetings - they are ineffective for the important question and answer format which allows the proponent to truly understand the issues within the heart of the community.

SUMMARY

Key factors for success must include an understanding on the part of the public that they do have control over the fate of the project. Too often opposition arises because of frustration and anger due to the mistaken belief that the project will go ahead regardless of local input.

In Canada, the Environmental Review Process includes public hearings and no approval will be granted without public hearings. Furthermore no license will be granted until the nuclear regulator is satisfied that the plant is safe, and that the public has had the opportunity to comment on the plant.

A second key factor is the willingness of the proponent to become a member of the community, recruit local champions and support them with visiting scientists, professors, and a public information program that listens and responds.

Thirdly, the proponent must listen and make changes to the program in response to community needs wherever possible.

Finally - be credible: learn the concerns of the community, agree on actions with the community and carry out your commitments.

LIST OF PARTICIPANTS

Al-Mugrabi, M.A.	Division of Nuclear Power, International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria
Baranaev, Yu.D.	Institute of Physics and Power Engineering, 249020 Bondarenko Square 1, Obninsk, Kaluga Region, Russian Federation
Hart, R.S.	Atomic Energy of Canada Ltd, 2251 Speakman Drive, Mississauga, Ontario L5K 1B2, Canada
Hilborn, J.	AECL Chalk River Laboratories, Chalk River, Ontario KOJ 1JO, Canada
Kozier, K.S.	Whiteshell Laboratories, Atomic Energy of Canada Ltd, Pinawa, Manitoba ROE 1LO, Canada
Kupca, S.	Safety Studies and Analysis, National Defense Headquarters, MGengeorge R. Pearkes Building, Ottawa, Ontario K1A 0K2, Canada
Mortimer, M.	Atomic Energy Control Board, P.O. Box 1046, Station B, 280 Slater Street, Ottawa, Ontario K1P 5S9, Canada
Nurdin, M.	National Atomic Energy Agency, J1. K.H. Abdul Rohim, Kuningan Barat, Mampang Prapatan, Jakarta 12710, Indonesia
Othman, I.	Atomic Energy Commission, P.O. Box 6091, Damascus, Syrian Arab Republic
Rousselle, P.	Technicatome, Industrial Division, B.P. 17, F-91192 Gif-sur-Yvette, France

Snell, V.	AECL-CANDU, 2251 Speakman Drive, Mississauga, Ontario L5K 1B2, Canada
Sweeney, B.	CANDESAL Inc., 112 Kent Street, Suite 2010, Ottawa, Ontario K1H 6A3, Canada
Xue, D.	Institute of Nuclear Energy Technology, Tsinghua University, 100084 Beijing, China