ORNL R&D ON ADVANCED SMALL AND MEDIUM POWER REACTORS: SELECTED TOPICS*

James D. White
and
Donald B. Trauger

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Oak Ridge National Laboratory†
Building 3500
P.O. Box 2008
Oak Ridge, Tennessee 37831-6010

ABSTRACT

From 1984-1985, ORNL studied several innovative small and medium power nuclear concepts with respect to viability. Criteria for assessment of market attractiveness were developed and are described here. Using these criteria and descriptions of selected advanced reactor concepts, an assessment of their projected market viability in the time period 2000-2010 was made. All of these selected concepts could be considered as having the potential for meeting the criteria but, in most cases, considerable R&D would be required to reduce uncertainties. This work and later studies of safety and licensing of advanced, passively safe reactor concepts by ORNL are described. The results of these studies are taken into account in most of the current (FY 1989) work at ORNL on advanced reactors. A brief outline of this current work is given. One of the current R&D efforts at ORNL which addresses the operability and safety of advanced reactors is the Advanced Controls Program. Selected topics from this Program are described.

NUCLEAR POWER OPTIONS VIABILITY STUDY

The objective of the Nuclear Power Options Viability Study (NPOVS)[1,2] was to explore the viabilities of several nuclear electric power generation options for this country in the 2000-2010 time frame. Innovative concepts were identified for which proponents claimed marketability at this time when studies indicated an expected increased demand for new electrical energy capacity. Criteria for future market viability were developed and used for

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assessment of these advanced small and medium power reactors. As shown later, these criteria (and related design characteristics) emphasized safety features, cost, operability, constructability, regulation, research needs, and market acceptance.

GROUND RULES AND THEIR SIGNIFICANCE

To facilitate useful study, NPOVS concentrated on a carefully selected, limited group of concepts by developing and applying ground rules. The following three ground rules were selected:

1. The nuclear plant design option should be developed sufficiently that an order could be placed in the 2000-2010 time period.

2. The design option should be economically competitive with environmentally acceptable coal-fired plants.

3. The design option should possess a high degree of passive safety to protect the public health and property and the owner's investment.

Ground Rule 1 determines the time period of interest. It was assumed that, if orders of additional nuclear power are placed before the year 2000, they will be filled by current or slightly modified designs, primarily of LWRs. By the turn of the millennium, the anticipated demand for power may permit consideration of advanced reactor concepts and their associated advantages. For the present the concept must be supported by an active and capable industrial proponent with a current program. It was considered very difficult, perhaps even impossible, for a proponent to obtain funds, complete a design, conduct R&D, build a demonstration plant or its equivalent, and demonstrate satisfactory operation by 2010, unless design work is already underway. The concept must have no major feasibility problems or major questions that must be resolved by long-term, high-risk R&D prior to commercial acceptance.

Ground Rule 2 stated that for a concept to be viable, it must be economically competitive. The measure chosen is the most likely and perhaps the only major alternative, the coal-fired power plant. Since the cost of coal and its transportation vary widely with location, this ground rule is somewhat site dependent. The most favorable situations for coal might eliminate some or all of the nuclear concepts. However, other problems such as mining, acid rain, and carbon dioxide buildup could become dominant by the time period considered in this study.

Ground Rule 3: Although licensable plants are considered adequately safe by NRC and the nuclear industry, passive safety provides additional protection that is independent from engineered devices and from human intervention or management. The added protection of the owner's investment through passively safe designs may enhance the acceptability of advanced concepts as viable power options. Passive safety should help overcome intervenor's objections, public apprehension, and utility hesitation. These features also may simplify plant operation for the owner-operator. Thus, passive safety, enhanced beyond that of the present safety philosophy of primarily diverse and redundant engineered (active) systems, may provide an ingredient to help revitalize the nuclear industry.

The concepts selected are advanced and have various degrees of innovation when compared with current concepts. For convenience, the selected concepts were classified in the traditional way by their coolants and respective generic names:

- Light-water reactor (LWR)
  - PIUS (process inherent ultimate safety). Promoted by Asea-Atom of Sweden
  - Small BWR (boiling-water reactor). Promoted by General Electric Company (GE)
- Liquid-metal reactor (LMR)
  - PRISM (power reactor inherently safe module). GE advanced concept supported by DOE
  - SAFR (sodium advanced fast reactor). Rockwell International Corporation advanced concept supported by DOE
  - LSPB (large-scale prototype breeder). Electric Power Research Institute-Consolidated Management Office (EPRI-CoMO) concept supported by DOE
- High-temperature reactor (HTR)
  - Side-by-side modular concept (small helium-cooled reactor concept that has the core and steam generator in separate steel vessels in a side-by-side configuration). Supported by DOE and promoted by Gas-Cooled Reactor Associates and industrial firms.

CRITERIA FOR VIABILITY

The criteria developed by the NPOVS study were as follows: (Some elaboration is offered for Criteria 1, 5, and 7 since they relate specifically to safety and licensing.)

1. Public Risk - The calculated risk to the public due to accidents is less than or equal to the calculated risk associated with the best modern Light Water Reactors (LWRs).

This is a fundamental public safety criterion. To implement it strictly, a probabilistic risk assessment (PRA) employing acceptable methods and data bases would be necessary for each new concept and for the "best modern LWRs." However, other approaches based on judgment can be useful. Compliance with this criterion is essentially a prerequisite for licensing.

2. Investment Protection - The probability of events leading to loss of investment is less than or equal to 10^{-4} per year (based on plant cost).

3. Economics - The economic performance of the nuclear plant is at least equivalent to that for coal-fired plants.

4. Design - The design of each plant is complete enough for analysis to show that the probability of significant cost/schedule overruns is acceptably low.

5. Certification - Official approval of a plant design must be given by the U.S. Nuclear Regulatory Commission to assure the investor and the public of a high probability that the plant will be licensed on a timely basis if constructed in accordance with the approved design.

This criterion addresses concern for delays and associated risk for fully designed or replica plants. Its prime concern is with the licensing process, including potential further changes in requirements and regulations. Today's cumulative experience with licensing is extensive and should be sufficient to permit the introduction of one-step licensing at the completion of design. Verification of quality control during construction, of course, would be required.

6. Marketability - For a new concept to become attractive in the marketplace, demonstration of its readiness to be designed, built, and licensed and begin operations on time and at projected cost is necessary.

7. Competence of Owner/Operator - The design should include only those nuclear technologies for which the prospective owner/operator has demonstrated competence or can acquire competent managers and operators.

For the operation of a new or substantially different concept to be satisfactory, utility plant managers and operators must have acquired an adequate background and experience with the technology, equipment, maintenance, and plant surveillance. For operation, simulator training has proven effective for current power plants, and simulators would be necessary tools for new concepts. Where the concept, such as the small BWR, derives from a prior system, this criterion should be relatively easy to meet.

Characteristics which augment the criteria and provide further guidance to designers are divided into two categories, essential characteristics and desirable characteristics. The essential characteristics involve construction costs and lifetime projections, investment risk, cost for reliable and safe operation, availability of financing and other resources, and public acceptance. The desirable characteristics that are related but not readily determined quantitatively are: practical research, development, and demonstration requirements; ease of siting; load-following capability; resistance to sabotage; ease of waste handling and disposal; good fuel utilization; ease of fuel recycle; technology applicable to breeder reactors; high thermal efficiency; low radiation exposure to workers; high versatility relative to applications; resistance to nuclear fuel diversion and proliferation; on-line refueling; ease of decommissioning; and low visual profile.

The logic for using a level of safety equivalent to that of Light Water Reactors (LWRs) as the standard of Public Risk (Criterion 1) was twofold. First, we considered conventional LWRs to be safe. Second, we observed that a different concept could be compared with an LWR on the basis of specific properties or components. Such comparisons include reactivity effects, stored energy, thermal capacity to absorb decay heat, temperature limits for fuel and cladding, and security of primary systems containment. Although such comparisons are not a substitute for a probabilistic risk analysis (PRA), they can provide quantitative means for comparative evaluation until the data base for component reliability and system integrity are adequate to perform an effective PRA.

The Competence of Owner/Operator (Criterion 7) is important, as illustrated by the Three Mile Island 2 and Chernobyl accidents as well as by the long, costly outages experienced by many nuclear plants. Errors in management and by operators will always be of concern since the human factor cannot be totally eliminated.

Small and medium power reactors offer benefits of potentially simpler systems and greater automation. The latter is particularly important since parallel units in a common facility are postulated to require automation. Judgments about licensability will focus on the design and safety features of these small, multi-unit reactors. As the overall complexity of the reactor station is reduced by smaller and more passive designs, the designer needs to ensure that the operational problems will be correspondingly reduced.

Multi-unit plants require standardization within the station complex. Extension of standardization to all station units of a given concept offers advantages for both licensing and factory assembly. The U.S. Nuclear Regulatory Commission [3], in their 1985 policy and planning guidance on advanced nuclear power plants have addressed this issue.

A companion to the reactor standardization policy is a preapproved siting policy. The time gained in the construction schedule by referencing an approved standard design could well be lost in a dispute over the adequacy of a proposed site. It should be possible to gain site approval in advance of applying for a construction permit; some regulations governing early site reviews have been adopted by the U.S. Nuclear Regulatory Commission.

The advent of smaller units has led to a concept of licensing by demonstration; i.e., to subject the first reactor of a concept to unusual stress and thus show its capability to accommodate potential accident initiators. There is great merit in demonstration units to identify problems in design and construction as well as to obtain licensing experience. However, there are limitations to licensing based primarily or totally on demonstration.[4] Not all safety claims or hypothetical accident sequences can be demonstrated; substantial analysis will still be required. Also, a license may be required for the test prototype. The demonstration tests would be complex and expensive. The test module may have to be sited remotely because of the potential risk of test failures. Savings in analysis may be minimal or even negative once the design needs for a successful set of tests are defined.

At the time the NPOVS report was prepared, little information was available from PRA studies for small and medium power reactors. We noted, however, that the designers of passively safe concepts responded to safety considerations in the following ways:

- Many small and medium power reactor concept proponents, relying as they do on passive safety features to prevent adverse effects of accidents, claim that nuclear safety-grade equipment can be limited to the nuclear island.

- Proponents of some of the concepts believe that minimal or no containment can be justified because of a lack of credible severe accident sequences. In fact, some of the proposed passive decay-heat-removal systems would be precluded by the use of conventional containments.

- Some proponents believe that a safety demonstration plant would greatly facilitate licensing.

- There is considerable support for the proposition that very rare accident precursors, with frequency below some particular value such as 10^{-7} per reactor year, need not be considered as design basis events. However, current experience and PRA methods may not be adequate to establish such values.

The use of performance-based regulation to replace the present prescriptive systems should be considered as a long-term objective. The concept can contribute to plant simplification (and reduced cost) while retaining a high degree of protection against public risk. The objectives are similar to those for many small and medium power reactor designs. Several of the following actions could be included in such an initiative:

- Adoption of passive safety systems to replace or supplement active safety systems. The use of passive systems makes verification simpler in that safety becomes more deterministic and less probabilistic.

- Performance standards can be applied to the plant's response to certain accident initiators such as an earthquake of a specified intensity or a pipe break of a particular timing and size. A combination of test and analysis can then be used to determine that a severe accident will not result.

- As experience is gained with the application of performance standards of limited scope and in the use of PRA, greater weight can be placed on the use of PRA to verify the overall achievement of safety goals.

- The response of plants to actual challenges to safety systems (Licensee Event Reports) can be analyzed to verify that the PRA is soundly based.

Current nuclear regulations require that there be a containment system, independent of reactor design, to mitigate the release of an arbitrary fraction of the reactor's fission products independent of reactor design. It is noted that this fraction is probably much greater than the actual release that would be experienced in most accidents; however, the regulation is intended to be conservative. Containment features for confinement of small and medium power reactors, particularly those designed without leak-tight containments, may require extensive research and demonstration to convince regulatory bodies that the proposed safety measures are adequate. At the time of the NPOVS, the following research and development areas were identified:

- Development of quantitative risk criteria for advanced reactors.

- Consideration of the significance of passive safety features to risk reduction.

- Determination of the frequency of rare events that would constitute a lower limit for design basis.

- Appropriate treatment of source term and containment for very safe designs.

- Appropriate focus on safety and risk reduction in the development and application of standard designs.

The NPOVS also addressed the question of market acceptability for new nuclear technologies. Case studies and interviews with public utilities, public utility commissions (which regulate electric rates), and interest groups were utilized to explore the market acceptability for new technology. From this research, a set of major issues was identified that is likely to be at the core of the acceptability question for new reactor technologies. It was concluded that for a new technology to be acceptable in the U.S.A. after the turn of the century, three necessary but not sufficient conditions would be required; these are:

- A projected need for new baseload capacity

- A narrowing of the gap in construction costs between environmentally acceptable fossil and nuclear plants; and

- The absence of a third, more environmentally and economically acceptable option for baseload power to compete with nuclear.

Even if all three necessary conditions are satisfied, there is no guarantee that nuclear options will be chosen. There is a further set of facilitating conditions that would substantially improve the position of nuclear technologies within the market. These include improvements in the following areas:

- Stability of the regulatory environment

- Improved accuracy and reliability of load-forecasting techniques

- Improved cost controls in nuclear construction and operation, including standardized or turnkey plants; and

- Demonstrated technical feasibility of new nuclear reactors.

Stability of the regulatory environment has been identified as important, but, in general, we received the impression that if economic incentives are strong enough, regulatory difficulties will eventually be overcome.

In summary, the features of advanced small and medium power reactors offer substantial potential advantages in safety and licensing. Smaller amounts of decay heat per reactor core, when combined with fuels and structures having higher temperature capability and improved ability to dissipate thermal energy in upset or accident conditions, make possible the innovative designs for current High Temperature Gas-Cooled Reactor (HTGR) and Liquid Metal Reactor (LMR) concepts. These designs include methods for dissipation of decay heat more directly to the atmosphere or to the earth and, in the case of Boiling Water Reactors (BWRs) and Process Inherent Ultimately Safe Reactors (PIUS), to large bodies of water. The safety features of these designs permit the following changes: eliminating conventional containments and engineered safety systems, clustering modules, using common control and power-generating units, and extensive shop fabrication, all of which result in cost savings. Some of these cost-saving changes partially offset the increased cost of the smaller-sized units that results from loss of the economy of scale enjoyed by larger units. Except for shop fabrication, the changes entail new issues that must be resolved in licensing a lead plant. Although the proposed designs may increase the margin of safety, they will require new methods for review at the outset. Licensing revision or reform appears necessary if these otherwise attractive units are to be economically competitive in the near term.

CURRENT DOE-SPONSORED R&D ON ADVANCED SMALL AND MEDIUM POWER REACTORS AT ORNL

The lessons learned from the NPOVS and later studies are integrated to the extent possible into the current R&D programs at ORNL.

We have four areas of activity funded by the Department of Energy in Advanced Reactor Technology: (1) Modular High-Temperature Gas-Cooled Reactor (MHTGR) Technology, (2) Strategic Technologies for Advanced Liquid Metal Reactor (ALMR) Concepts, (3) R&D on advanced LWRs, and (4) Advanced Fuel Cycle Technology. The first three of these areas are outlined here to give the reader a perspective of the kinds of work involved.

MHTGR TECHNOLOGY DEVELOPMENT

With regard to MHTGR technology development, extensive planning has taken place over the past several years culminating in a technology development program covering fuel, fission products, graphite, and metals behavior. These plans have been established by DOE, ORNL, and industry in direct response to specific MHTGR design and licensing requirements. Careful prioritization of technology development needs has led to rigorous cost control. The required experimental technology development program is now well-defined and underway.

Fuel and fission products technology

Fission product retention within the fuel coatings is the key ingredient to achieving passive safety in the MHTGR, and although results to data have been favorable, additional data are needed to demonstrate and validate that expected performance is achieved. Fuel fabrication process development is required to confirm that economic fuel fabrication processes will produce quality coatings which retain their integrity during fabrication and reactor irradiation. Fission product transport studies are needed to conclusively demonstrate adequate retention of fission products within the reactor system during postulated conditions.

Current work in fuel and fission product behavior is designed to update and validate behavior models. This work uses data from several national and international facilities to update/validate models of failure rate for standard and defective fuel particles under normal and accident conditions; fractional release of metallic and gaseous fission products under normal and accident conditions; plateout of metallic fission products on graphite and metallic components; and liftoff/washoff under accident conditions.

Fission product behavior work includes bench scale testing to evaluate: chemical forms of fission products, effects of alloys and surface conditions on plateout and reentrainment, and characterize dust and effect of dust on fission product transport behavior. In pressurized loop facilities, work is being done to evaluate: effect of shear ratio and of differential pressure during depressurization, coolant chemistry/moisture effects, effects of blowdown duration, steam quality and Reynolds number on fission product behavior. Models will be updated based on this bench scale and pressurized loop testing. Then the updated models will be validated in the French COMEDIE loop.

Graphite technology

Graphite constitutes a major volume fraction of the MHTGR core region. The graphite technology program effort emphasizes obtaining physical and mechanical property data under all postulated conditions so as to provide for reliable component design. Important determinations for design and licensing support are irradiation creep effects and oxidation characteristics.

Key areas of current work are physical and mechanical properties, dimensional stability under irradiation, fracture mechanics and steam corrosion. Work in these areas is needed to provide design engineers with better data for design, to provide safety engineers with better statistical properties for PRA analysis, and to provide quality assurance data sufficient for licensing purposes. Dimensional stability and creep data come from irradiation capsule experiments in the ORNL High Flux Isotope Reactor (HFIR) and HFR-Petten. Corrosion data come from bench scale tests and pressurized loop experiments.

Metals technology

The metals technology program for the MHTGR relates primarily to mechanical property data required for reliable component design and to ensure licensability. With regard to the reactor vessel, emphasis is on irradiation effects under normal MHTGR conditions and on material property information pertinent to accident conditions. Long-term testing is needed to complete the materials property data base for the steam generator, hot duct, and core internals; key measurements involve fracture mechanics, mechanical properties, radiation effects, environmental and corrosion effects, and weld properties.

Physics validation

This work demonstrates the capability of physics methods and codes to predict important characteristics: core criticality, temperature coefficient of reactivity, control rod worth, power distribution, moisture worth, burnup swing, Pu buildup and B_4C worth. Data from previous experiments are being analyzed and will be supplemented with new data from the AVR. Supplemental experiments in other facilities may be used.

Shielding analysis

Primary tasks are: (1) calculation of maintenance dose rates after shutdown; and (2) calculation of neutron fluence to reactor vessel system. The first task is to be done by Bechtel. The second task is being done by ORNL. Experimental work is planned to supplement calculations. The fluence to the vessel head by neutron streaming is very difficult to treat analytically. Experimental work in this area will reduce uncertainties.

STRATEGIC TECHNOLOGIES FOR ADVANCED LMR CONCEPTS

Work in several strategic technologies is funded by DOE with emphasis on ALMRs. Most of this work, however, is applicable to advanced concepts of gas-cooled and light-water cooled designs.

Advanced instrumentation

Advanced instruments under development at ORNL include an improved neutron flux monitoring system featuring high-temperature operation and advanced electronics. A prototype system is now undergoing testing. An automated noise diagnostics system has also been developed an is undergoing prototype testing in the Fast Flux Test Facility (FFTF).

Shielding

The present shielding program is a cooperative and co-funded effort with Japan. The cost of shielding materials can be reduced by as much as a factor of 5 by using boron carbide (B_4C), reducing design margins due to improved analysis techniques and uncertainty predictions, and optimizing localized shielding for individual components and work areas. Particularly for modular designs, feasibility of design options can hinge on the use of advanced shielding techniques and materials. Specific feasibility issues include:

- Interim storage of fuel within the vessel
- Vessel diameter reduction to allow barge/rail shipment
- Dose criteria due to activation of secondary heat-removal system
- Location of in-vessel nuclear instrumentation

Verification of these improved shielding materials and development of methods is underway. A four-year joint Japanese/American Shielding Program (JASPER) to accomplish these objectives is being carried out at ORNL's Tower Shielding Facility. Six major experiments are planned, with the first two having been completed. These technology improvements have broad application in all advanced reactor systems as well as space nuclear systems.

In addition to this major program, ORNL provides direct design support in shielding design to the PRISM design teams.

High-temperature materials and structural design

ORNL has been and continues to be DOE's leading center of excellence in Materials and Structural Design Technology. To build on this expertise and transfer the benefits of this technology to industry, DOE has recently added a new $19 million High-Temperature Materials Laboratory at ORNL.

A major element of ORNL's LMR Materials and Structural Design program over the last decade has been the development of a new alloy, modified 9Cr-1Mo steel. The development is now almost 90% complete. Modified 9Cr-1Mo is approved for application under ASME Section VIII, Div. 2 (fossil and non-safety-grade nuclear application); and approval under ASME Section III, N-47 (Nuclear Applications) has been requested. Modified 9Cr-1Mo is already manufactured in the U.S., Japan, and France. International acceptance of this alloy has been excellent; it is now in service in fossil plants in Europe and Japan, as well as in the U.S. Substitution of modified 9Cr-1Mo for alternate materials in the SAFR steam generator and intermediate heat exchanger was estimated by Rockwell International to save more than $20M per 350 MW(e) module. Remaining activities for modified 9Cr-1Mo include: (1) development of test data to optimize weld procedures and establish weldment properties; (2) completion of mechanical properties tests; and (3) development and validation of design methods, rules, and criteria for application in the LMR service environment. The level of effort required to complete these activities over the next four years is more modest than that carried out in previous years, but is essential to achieve the benefit of the technology. The Japanese Atomic Power Company (JAPC) expressed an interest in supporting this work and providing their database on modified 9Cr-1Mo in return for the U.S. database. Negotiations between JAPC and DOE have led to a three-year program to begin in FY 1989. In addition, modified 9Cr-1Mo has potential for application in space nuclear power systems.

ORNL's role in materials and structural design involves more than just new alloy development. A number of critical materials and structural design issues remain, involving traditional materials and design methods. Structural failures resulting from creep and fatigue have occurred in both fossil and earlier LMR plants. Current LMR design methods and criteria must be improved to preclude such failures in the future.

Nondestructive testing technology is under development at ORNL for remote in-service inspection of components in a sodium environment. Inspection of components such as steam generators and intermediate heat exchangers in situ can save downtime and enhance the system reliability and availability.

Robotics

A primary motivation for utilizing robots in nuclear power plants is the reduction of personnel radiation exposure to "as low as reasonably achievable" (ALARA) as recommended by the NRC. The exploitation of advanced robots for hazardous operations will contribute substantially to achieving this goal. Robotic access reducing the number of human entries to sensitive locations has the potential for an increased frequency of monitoring and inspection, resulting in enhanced overall safety and reduced personnel exposure. Furthermore, the development and utilization of robots can potentially improve plant availability by permitting some maintenance tasks within containment to be done remotely during power production. It is believed that the number of tasks which could be accomplished this way is large enough to substantially reduce outage durations.

ORNL is leading and coordinating a team effort to pursue the development and deployment of advanced robotic systems capable of performing surveillance, maintenance, and repair tasks in nuclear energy facilities. A cooperative five-year plan is being pursued. In addition to ORNL, the team involves four major universities and their respective industrial partners:

- University of Florida, Odetics
- University of Michigan
- University of Tennessee, Combustion Engineering, Remotec
- University of Texas, Martin Marietta Aerospace

Annual joint demonstrations of the team's progress are planned, with the first demonstration to take place in December 1988 at ORNL.

Reliability, availability, maintainability (RAM) data

This program has also become international with the addition of 50% Japanese funding and participation under the Centralized Reliability Data Organization (CREDO). CREDO gathers (1) detailed engineering data on components, (2) failure data, and (3) operating data from U.S. and Japanese facilities.

The purpose is to establish and maintain a well-documented, centralized, comprehensive source of RAM data for use by LMR designers in probabilistic risk assessments and other analyses in support of design and licensing and in establishing inspection and maintenance practices to achieve high plant reliability and availability. The evolving data base provides information needed to properly design and license LMR systems. Presently the data base includes data on 20,000 components, 1,800 events/incidents, and 1.2 billion component operating hours. Space nuclear power systems will also benefit from the data base.

ADVANCED CONTROLS PROGRAM

Modern automated reactor control systems can enhance the economic competitiveness of advanced reactors by increasing their operational reliability, availability, and maintainability; improving their safety and licensability by reducing challenges to the plant protection systems; and significantly reducing the manpower needed to operate the plant.

Through on-line monitoring and surveillance of equipment performance, rapid detection and response to equipment malfunction, and reduction of operator errors, automation can significantly reduce the occurrence of abnormalities in plant operation and prevent abnormalities from becoming accidents.

Automation is critical to the economic competitiveness of multimodular plants. Next-generation plants must adopt modern digital control technology to make use of these new automated and intelligent control strategies and systems. Today's power plants are based on time-tested analog equipment and controls. Thus the new advanced control systems and strategies must first be demonstrated and tested to provide the necessary confidence to regulators, manufacturers, and power plant operators.

To achieve a practical design for an automated control system for advanced nuclear power plants, ORNL is integrating activities in: (1) control system design, architecture, and components; (2) artificial intelligence for adaptive, predictive, and self-learning control strategies and expert systems for operator support; (3) integrated human-system engineering for allocating responsibilities between humans and computers, and for optimal control complex design; and (4) plant simulation, software development and validation, and system reliability improvements.

These integrated activities are being conducted in the Advanced Controls (ACTO) Program. The goal is to provide a national center of excellence in research, development, and testing of nuclear control systems. This program

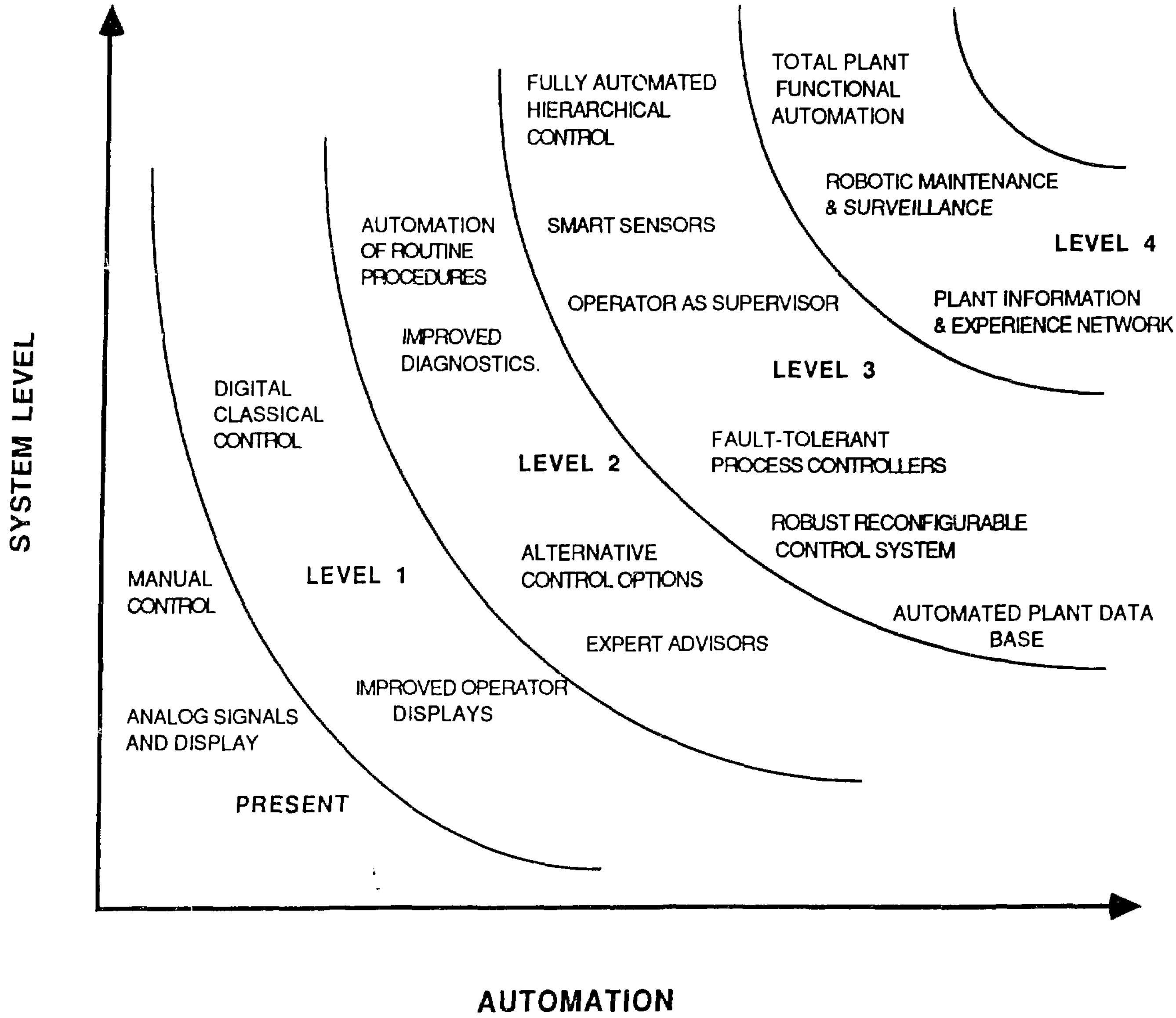

Fig. 1. Nuclear plant control automation with evolving technology.

will provide an integrated environment to support the rapid, confident design and testing of advanced control systems that will assure improved operability, reliability, and safety for advanced reactors.

The transition from today's nuclear control systems (with some analog control at the subsystem level and significant operator integration) to future designs (with complete automation under human supervision) will occur in phases. The transition may be described in terms of four levels as shown in Fig. 1. The first phase of improvement is automated data management at a plant, which is now actually occurring to a limited extent in U.S. LWRs and is under study in the U.S. LMR Program. Also in this level will be some replacement of today's analog controllers with more reliable digital controllers performing basic proportional-integral-differential control. EPRI is sponsoring some of this work at existing LWR sites.

Level 2 will be automation of routine procedures like startup, shutdown, refueling, load changes and certain emergency response procedures. Significant assistance will be given to the operator in the form of expert systems and control room displays of plant status. Control strategies will be predetermined choices selected from hierarchical, optimal, robust, multivariate options. Advanced LMR concepts being studied fit within this phase.

Level 3 is a significant advance toward automation with capability for full automation of all hierarchical levels of control. The operator's role will be to interact with and monitor the intelligent, adaptive supervisory control system. Smart sensors will validate their own signals and communicate with robust, fault-tolerant process controllers. The process controllers will be able to reconfigure the control logic to meet the operational objectives selected by the supervisory control system. Control strategies will be adaptive, uncompromised by nonlinear effects in the processes, and very robust to off-normal conditions. Plant designs will be completely automated, with plant data bases available to the control system and the operator. Operational experience of all plant systems and components will be tracked in an automated data base, and the control system will recommend maintenance schedules and outages to the operator. Good human performance modeling will permit optimum allocation of function decisions so as to keep the operator motivated and informed about plant status.

Level 4 is total functional automation of the plant through an intelligent control system aware of the entire operational status and in interactive communication with the operator to keep him apprised concerning operational status, any degraded conditions, likely consequences of degradations, and possible (recommended) strategies for minimizing deleterious consequences. By this time plant designs will have many automated and robotized functions including maintenance and security surveillance. The control system will be an integral part of not only the total plant design but also the national network of commercial power plants. The control system computer will learn from the network information concerning other plant and component operational experience and will alert the operator if that experience is relevant to current operations.

To provide the necessary national leadership in the design of advanced control systems, this program will support four major kinds of activities:

- Demonstrations of advanced control system design features using current developments in control theory, automation, artificial intelligence, information management, modeling and simulation, and man-machine interaction research.

  -- The Advanced Controls Program will provide national leadership in control system design by demonstrating examples of advanced control system designs for nuclear reactors. These demonstrations will be carefully designed to show how state-of-the-art research can be used to help accelerate the transition to fully automated control. For example, we developed and demonstrated in a prototype design this year a new, promising (easier, faster) technique for designing a hierarchical, distributed control system using multivariate optimal control theories for non-linear systems with uncertain dynamics.[5,6] Preliminary testing of the prototype design indicates that this technique leads to improved ability to: (1) adapt to changes in plant performance, (2) accommodate plant component modifications (as in plant aging or component replacement), and (3) perform well even in the case of noisy plant signals. The next demonstration will occur late this year and will be a prototype advanced, automated control system design for a feedwater system for an advanced LMR. The feedwater train is a complex system that is the origin of incidents causing a significant fraction of lost plant availability in conventional LWRs.

  These and other demonstrations in following years will help transfer to the reactor industry the benefits of the latest proven advances in control systems strategy, control system and whole-plant simulation, computer-aided software engineering for control systems design, man-machine interaction modeling and analysis, and the other technologies being used within the program.

- Establishment of a design environment that allows designers to formulate and test various control strategies for the plant of interest quickly and economically.

  -- The program will provide a centrally located, user-friendly design environment for control system designers within the DOE community. The environment will consist of four parts: (1) networked, intelligent, computer workstations into which have been integrated software tools, graphics capabilities, on-line design guidance, on-line documentation and interfaces to the large plant simulation capability at ORNL; (2) plant/component models and databases useful for control system design and plant simulation; (3) information resources concerning advanced control system strategies for automated control; and (4) man-machine interaction models and guidelines for designing control system interfaces with operators. This year, we developed and demonstrated a prototype of a unique intelligent workstation for control system design featuring artificial intelligence and graphical interfaces.[7,8,9] The prototype provides the ability to easily, rapidly produce, simulate and evaluate certain ALMR control system strategies and techniques.

Also during the year, we developed and linked a prototype model of a human operator to an ARIES-P simulation of PRISM in FY 1988.[10,11,12] This model enables a rough simulation of operator performance during ALMR operational upsets. The model accounts for cognitive functions of the operator as well as training, timing, and probable error rate.

In the area of advanced controls R&D, we developed and evaluated (using EBR-II data) several advanced multivariate optimal control strategies for ALMR systems.[13] Showed that certain advanced control techniques are better than classical proportional-integral techniques (more robust with respect to changes in plant performance due to fouling, etc.).

- Testing and validation of advanced control system designs by simulation.

  -- The ability to simulate an entire plant in real-time is critical to the design of a fully automated plant. The program will provide this simulation capability to the technical community. State-of-the-art advances in computer architectures, software engineering, very high-level languages, area networking, artificial intelligence, and database management will be integrated into a whole-plant, real-time nuclear power plant simulation capability. In FY 1988, we procured and installed a parallel processor and several computer workstations in the program computer laboratory to enhance our testing and simulation capability.

- Guidance in control software and hardware specifications.

  -- The program will provide standards, guidelines, and specifications for control software and hardware. ORNL will acquire and develop tools and methods for generation of large, standardized software programs needed for automation of nuclear reactors. Methods for locating errors in software programs will be acquired and developed, and software verification and validation procedures will be utilized.

R&D ON ADVANCED LWRs

Two efforts are underway at ORNL on Advanced LWRs. These are concerned primarily with requirements development and review for a large ALWR and preliminary investigations on more advanced developmental LWRs. In the EPRI requirement development work, emphasis is on man-machine interface systems (including control systems) for a large ALWR in the DOE/EPRI ALWR Program--but the work probably will be applicable to smaller ALWR concepts. The ORNL work consists of team participation in the development and review of requirements, review of codes and standards for digital control and protection systems and development of software verification and validation plans. The developmental LWR activity is directed to investigate advanced passive LWR features beyond those currently being considered in the ALWR program.

In a related activity, ORNL has the lead role in the Advanced Neutron Source (ANS) design. ANS is a major new research reactor project directed at establishing the world's leading center for neutron scattering research. The ANS work is funded by the DOE Office of Basic Energy Sciences. The facility will have very high thermal and cold neutron fluxes, state-of-the-art neutron-scattering facilities, isotope production capabilities, and materials irradiation positions.

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