**SAFETY-LICENSING ASSESSMENT OF NASAP REACTOR
CONCEPTS AND FUEL CYCLE FACILITIES**

by

W. C. Lipinski, F. G. Prohammer, J. B. van Erp, and W. B. Seefeldt*

June 1978

MASTER

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3.4.1 Introduction

These assessments of the safety/licensability of reactor concepts are based on information supplied by the NASAP characterization contractors in their updated responses to the data package for NASAP Rolling Report II. The assessment of the LMFBR includes information from a characterization contractor on alternate fuel cycles but does not include information provided by a characterization contractor on plant-related safety issues.

The information provided by the characterization contractors was supplemented by assessments provided by the U. S. Nuclear Regulatory Commission (Report No. NUREG-0364, "Preliminary Review of Alternative Reactor Types and Fuel Cycles," October 1977) and by ANL.

3.4.2 Summary of Results

The reactor concepts can be divided into groups based on safety/licensing considerations and the associated time dependent potential for commercialization. The presently deployed Light Water Reactor, the Spectral Shift Reactor, the Heavy Water Reactor, and the High Temperature Gas Cooled Reactor are all considered to be licensable in commercial sizes, or with minimum additional effort. The Tandem PWR/HWR concept utilizes PWRs and HWRs which are independently licensable, but the optimization of the fuel configuration for use in both the PWR and HWR would require safety review and should not present problems which would prevent licensing. The Liquid Metal Fast Breeder Reactor, the Light Water Breeder Reactor, and the Gas Cooled Breeder Reactor have received licensing review but there are safety issues which must be resolved prior to commercialization. The Molten Salt Reactor, Accelerator Driven Reactor, Gaseous Core Reactor, and Fusion-fission Hybrid Reactor appear to require such development of technology and construction of demonstration facilities on a path to commercialization that they would not be expected to be available in commercial sizes until well beyond the year 2000.

Safety and licensing experiences in the fuel cycle have been most extensive for low enriched uranium and little difficulty is foreseen for the licensing of future facilities. Some adjustments may be needed in controlling radon present in tailings piles from milling operations, the design of private enrichment facilities which are not now licensed, and in defining the licensing problems of waste management.

Considerable experience has accumulated, mostly on a small scale, for fuels containing major concentrations of plutonium (usually with uranium). This experience has encompassed principally conversion and fuel fabrication, but also includes operations at a commercial fuel processing facility by Nuclear Fuel Services (NFS). Licensing problems arising from scaleup to commercial practice are considered resolvable in the short term.

Virgin thorium and non-recycled uranium-233 have been used successfully on a small scale in several segments of the fuel cycle, notably conversion and fabrication. Licensing of small facilities would be reasonably routine and environmental effects nominal. Scaleup is not likely to involve major licensing difficulties.

The use of recycled thorium and recycled uranium-233 introduces a range of new problems from the safety and licensing point of view. High levels of gamma activity and the presence of radon will probably impose new requirements on confinement to meet occupational and environmental regulations.

3.4.3 Reactor Concepts

3.4.3.1 Light Water Reactor (LWR)

The present designs of LWRs are licensable, and well-established regulations and practices exist. A new fuel type or fuel cycle would require further development and safety review.

A fuel rod design for the extended burnups to about 50,000 MWD/MT batch average must address several technical problem areas, e.g., the higher percentage release of fission gases at higher burnups, corrosion of the outside surface of the cladding, pellet-clad interaction, and structural stability (elongation and bowing of rods). Fission gas release will pose the most difficult problem. In order to accommodate the fission gases without increasing the internal pressure on the fuel tube beyond design limits, modifications to the fuel rod design will be necessary. The most straightforward modification is to increase the length of the fission gas plenum. However, if an alteration of core internals is to be avoided, increases in fission gas plenum length appear to be limited. Another alternative is the use of annular instead of solid pellets by providing an additional volume for fission gas storage in the central hole and by reducing the operating temperature of the fuel which tends

to lower the release of fission gas. However, for equivalent energy release compared to solid pellets, the burnup of annular pellets must increase even further.

An extensive development program is required before high batch average burnups can be achieved on a large scale commercial basis. For batch average burnups of 50,000 MWD/MT, the average burnup of the highest duty fuel rods is in the 60,000-MWD/MT (solid pellets) to 75,000-MWD/MT (annular pellets) range which is considerably higher than for current PWRs (about 37,000 MWD/MT). The present knowledge of the irradiation behavior of UO_2 or ThO_2 for exposures of this magnitude under PWR operation conditions is quite limited. Fission gas release as a function of burnup and temperature, pellet-clad interaction, dimensional stability, and corrosion behavior are some of the phenomena which need to be investigated. The required in-pile irradiation experiments will be costly and time-consuming. The irradiation experiments must be followed up by post-irradiation examinations, as well as theoretical efforts to correlate the experimental data and to develop analytical models for the design of fuel rods.

The basic types of fuel-cycle-related reactor R&D which must be performed for the thorium fuel cycles include data base development, reactor components development, and reactor/fuel cycle demonstration. The purpose of the data base development R&D is to provide information required for the design and licensing of reactors operating on the subject fuel cycle. Such data-based development, particularly, consists of physics verification and the establishment of safety-related fuel performance characteristics. Physics verification experiments typically consist of critical experiments which provide a basis to demonstrate the ability of analytical models to predict such important safety-related parameters as reactivity level, coefficients of reactivity, and control poison worths. Safety-related fuel performance R&D consists of such aspects as fuel rod irradiations, in order to establish in-reactor performance and discharge isotopics, special reactor experiments to establish such parameters as in-reactor swelling, densification, centerline temperature and fission gas release, and the performance of the fuel during anticipated operational transients. Another aspect of fuel-cycle-related R&D is the reactor/fuel cycle demonstration. This demonstration includes the initial core physics design and safety analysis, which identifies those changes in

reactor design necessitated by the denatured thorium fuel cycles and any resulting changes in design bases events.

A physics verification program is needed to provide the required data both to predict important safety-related physics parameters and to demonstrate the accuracy of such predictions as part of the safety analysis. The first aspect requiring attention is the development of improved cross sections for thorium and for isotopes in the thorium depletion chains, such as U-233, U-234, and Pa-233. Cross-sectional information for such elements has been largely neglected in the past and is believed to be much more uncertain than the corresponding cross sections of isotopes present in uranium-based fuel cycles. Resonance integral measurements must be performed for denatured fuels at both room temperature and at elevated temperatures. These experiments are very important to both the accurate calculation of safety-related physics characteristics and also in establishing the quantities of plutonium produced during irradiation. The second aspect of the physics verification program consists of a series of critical experiments. Experiments should be performed for each of the fuel types under consideration, i.e., for thorium enriched with denatured U-235, denatured U-233, and plutonium, and should preferably be performed both at room temperature and at elevated and moderator temperatures. These experiments serve as a basis for demonstrating the ability of analytical models to predict such safety-related parameters as reactivity, power distributions, moderator temperature reactivity coefficients, boron worth, and control rod worth.

Another major area of data-base development consists of the establishment of safety-related fuel performance information such as transient fuel damage limits, thermal performance for both normal operation and with respect to LOCA margins in stored heat, dimensional stability (densification and swelling), gas absorption and release behavior, and fuel cladding interaction. Transient fuel damage experiments are needed to provide information on the performance of denatured thorium fuels under the more rapid transients possible during anticipated operational occurrences and other postulated accidents.

The effects of alternate fuel cycles on plant safety and licensing require examination of safety criteria and dynamic analyses of design basis events. Appropriate safety criteria, such as acceptable fuel design limits and

limits on maximum energy deposition in the fuel, must be determined. Changes in core physics parameters that result in alternate fuel loadings and the implications of these changes on reactor design and safety need to be identified. For example, changes in fuel and moderator temperature reactivity coefficients, boron worth, control rod worth, prompt neutron lifetime and delayed neutron fraction must be addressed since they can have a large impact on NSSS performance and safety. The effects of alternate fuel cycles on the dynamic system responses should be determined for all transients required by NRC Regulatory Guide 1.70. It will also be necessary to determine the implications of denatured fuel cycles on plant operation and load change performance, to determine whether the response of plant control and protection systems is altered.

Based on the fact that two small LWRs have been licensed and operated satisfactorily with $\text{ThO}_2\text{-UO}_2$: Indian Point I (PWR) and Elk River (BWR), the technology and licensability outlook for alternative fuels of the oxide form is favorable.

In developing a schedule for completing the LWR R&D, it can be assumed that the required fuel cycle/reactor demonstration could be performed in an existing LWR, i.e., it would be unnecessary to construct a reactor specifically for the purpose of demonstrating denatured fuels. With this assumption, the required demonstration program could be initiated within seven years and completed within 11 years of program commitment. Data base development R&D costs of \$40M are estimated.

3.4.3.2 Spectral Shift Controlled Reactor (SSCR)

The Spectral-Shift-Controlled Reactor (SSCR) consists basically of a standard pressurized water reactor (PWR) plant in which the conventional reactivity control system is replaced with spectral-shift control. The heavy water concentration is changed from about 60% at the beginning of a cycle to about 2% at the cycle end. Compared to a conventional PWR, the SSCR requires additional equipment for accomplishing the dilution of the heavy water in the coolant during a cycle, for upgrading and storing the heavy water, and for utilizing primary system leakage because of the high cost of D_2O and the tritium radiological hazard associated with D_2O spillage. The tritium production and tritium concentration in the coolant of a SSCR is considerably higher (about 20 times) than in a conventional PWR.

Because of the closeness of the SSCR design concept to that of a conventional PWR, no new fundamental problems related to safety, licensing, and environmental impact are expected. However, there are a number of safety/ licensing related design aspects which will require evaluation:

Since soluble boron is not utilized in the SSCR, warm-water accidents in the SSCR are expected to be less severe than in the conventional PWR. Since the coolant in the SSCR at end of cycle is essentially pure water, similar consequences of cold-water accidents are anticipated at end of cycle in the SSCR; however, because of the absence of boron and the hard neutron spectrum at beginning of cycle, cold-water accidents early in cycle are also major items which must be explored.

The effectiveness of the safety-injection system, which injects highly borated light water during accidents and anticipated operational occurrences, must also be evaluated; this system may be less effective in the SSCR because of the presence of light water in the injected borated water, and might effect the prevention of return-to-power after steam line break accidents.

Due to the harder neutron spectrum resulting from the presence of D_2O in the coolant, a reduction of the effectiveness of the control rods might be anticipated. However, the thermal diffusion length is longer in D_2O and the control rods are "seen" by neutrons from greater distance. As a consequence of these competing effects, the control rod worth is comparable to that of a PWR. The Doppler coefficient at the beginning of a cycle is greater than for a PWR because of the larger fractional neutron absorption in the fertile material.

The loss-of-coolant accident for a SSCR is not expected to be more severe than for a PWR when the three-batch, 30,000-Mwd/MT fuel management is used. With the higher burnup/5-batch strategy (50,000 MWD/MT) the higher fission gas release and increased fuel rod internal pressure will cause earlier bursting of the cladding which might impair cooling the fuel rod. On the other hand, the generally lower fuel temperature or stored energy in the fuel of an annular pellet makes it easier to cool the fuel rod. An evaluation of these compensating effects will be

required to assess the behavior of the high burnup fuel rods under LOCA conditions; at any rate, the LOCA performance of such high burnup fuel is expected to be similar to the fuel in the conventional PWR.

The longer mean path length of the neutrons in the D_2O -containing coolant causes a neutron fluence at the wall of the reactor vessel which is about 30% higher than for a PWR. The resulting increase of the nil-ductility transition temperature by about 20°F at the end of the vessel life appears to be tolerable.

Inadvertent injection of unborated light water into the coolant system will cause an increase in reactivity when the D_2O concentration is high. This accident is similar to unborated water addition in the PWR. It appears that with similar controls, as in the PWR, the D_2O dilution can be prevented in the SSCR or its consequences mitigated.

In order to minimize the tritium radiological hazard during refueling, a reference procedure has been developed which is a variation of the standard refueling procedure and is based on the concept of exchanging the H_2O/D_2O coolant with H_2O before opening the reactor vessel prior to refueling. In order to salvage as much D_2O as possible, the coolant will be drained and the water level will be lowered below the reactor vessel flange to the centerline of the primary coolant pipes. The remaining coolant volume will be flushed out to one of the holdup tanks with H_2O from the refueling water tank using existing pumps and pipes of the shutdown cooling system as well as some additional hardware. Subsequently, the reactor vessel will be filled with H_2O to the vessel flange and the refueling procedure will be continued. The susceptibility of this refueling scheme to procedural violations requires evaluation.

Tritium control, as it affects operating personnel exposure and consequent constraints on space and equipment access, would need a detailed evaluation.

For the evaluation of the safety aspects of a SSCR, it is expected that the safety criteria which are applied to the evaluation of a conventional PWR will be used.

The main safety-related item which requires R&D effort is the SSCR refueling procedure. The success of the procedure hinges on the successful replacement of the D_2O containing rest-volume of coolant in the reactor vessel by light water from the refueling water tank. The effectiveness of this coolant displacement process requires experimental verification. The degree to which the D_2O rest volume can be displaced directly governs magnitude of the tritium radiological hazard resulting from evaporation of water from the refueling pool during the refueling operation.

It is not expected that safety/licensing considerations will impose any significant constraints on the commercial introduction date. Due to the general similarity of the SSCR to a conventional PWR, the resolution of the specific SSCR safety/licensing items is anticipated to be accomplished within a reasonable time after submittal of a detailed design to NRC for review.

In support of safety and licensing, certain fuel-cycle-related R&D would be required. The estimated cost of these activities is about \$10,000,000 over a two-year period.

3.4.3.3 Heavy Water Reactor (HWR)

The Canadian experience with construction and operation of CANDU reactors provides a favorable outlook on plant safety. Although the Canadian safety and licensing criteria and practices, and design implementation of criteria and practices, have not been compared in depth to those which would be required of a CANDU reactor licensed in the United States, it is reasonable to expect that the CANDU reactor fundamental design is licensable in the United States.

Among the characteristics which could potentially require detailed evaluation to establish acceptability are:

The adequacy of pressure tubes using a material (zirconium-niobium alloy) not approved for pressure boundaries by the ASME, and subjected to the full neutron flux, would require evaluation. The probability and consequences of pressure tube failure and the potential for damage propagation to adjacent tubes would require evaluation. The impact of limited in-service inspectability of pressure tubes, and the use of rolled tube joints, would also require evaluation.

The acceptability of the current emergency core cooling system, particularly the redundancy and diversity aspects, would require evaluation. The acceptability of the approach using the moderator coolant system as a dependable emergency core cooling system would also require evaluation.

The impact of the small positive power coefficient in the operating range would require evaluation. The reactivity would also require evaluation.

The acceptability of containment designs, including containment liner material, would require evaluation.

The impact of tritium levels on personnel exposures and consequent restraint of inspection and maintenance require evaluation.

Potential refueling accidents would require evaluation.

The use of computers in control, including refueling operations, would need review.

The listing above is not intended to imply that these aspects have not been fully and satisfactorily considered in the CANDU design; they are merely indications of areas where the CANDU design is believed to differ from present LWR practice in the United States, and are therefore areas which the NRC staff would give early attention. There are features of the CANDU design which tend to be positive in the safety sense, including the location of control and safety absorbers in the low-pressure moderator, the provision of separate channels (pressure tubes) for each fueled channel (making the introduction of emergency cooling water potentially more predictable), and the limited installed reactivity of the system. All factors need to be considered in a determination of safety adequacy.

There is also an important aspect of the U.S. licensing evaluation by NRC which is worthy of mention; it is the fact that the great bulk of knowledge and experience in the design of CANDU systems, including safety experimentation, has been developed by an agency (AECL) of the Canadian Government. It is probable that an application for construction of a CANDU system in the

United States would depend substantially on the availability of such information to the NRC staff.

The following are areas which have been identified as requiring development effort to support the HWR design. Where possible, an estimate of the time span and cost for these development programs is provided.

Safety Related Physics Parameters - critical experiments designed to measure such parameters as core reactivity, coefficients of reactivity, power distribution, and control absorber worth should be performed for slightly enriched uranium fuel. Such a program, including fuel fabrication, the experiments themselves, and evaluation, would take about 1.5 years and cost \$1,000,000 (sufficient information of this type may be available in Canada).

Fuel Testing - demonstration fuel assemblies, containing an initial enrichment sufficient to achieve the discharge burnups anticipated for the slightly enriched HWR, should be irradiated in test reactors such as NPD. These experiments would provide a measure of the power peaking anticipated in the slightly enriched uranium fuel cycle, and demonstrate the performance of such fuel up to the burnups anticipated and for power changes which occur during refueling. A typical fuel testing program might take five years to complete; one year for experiment design and fuel fabrication, three years for irradiation, and one year for post-irradiation examination and evaluation. The cost of this program would be on the order of \$2 million.

Advanced Pressure Tube Design - the reference NASAP HWR is expected to operate at pressures comparable to those of the PWR in order to increase net station efficiency and reduce the effective capital cost of the HWR. An experimental program to demonstrate the integrity of the pressure tubes and rolled joints at these higher operating pressures will have to be performed. This program would be similar in nature to tests already performed on CANDU pressure tubes and would be intended to extend the range of validity to experimental information to the higher pressures anticipated for the reference design. A one- to two-year time

period is anticipated for these experiments, depending on the amount of modification to existing experimental equipment required. This program is estimated to cost approximately \$500,000.

Thermal Hydraulic Tests - the empirical correlations used to establish the margin to burnout in CANDU reactors probably covers the range of pressures currently employed. This information will also have to be extended to the higher pressures anticipated in the reference NASAP design. Approximately one year should be sufficient to extend existing burnout correlations to these higher pressures. Approximate cost for this testing is \$250,000.

3.4.3.4 High Temperature Gas Cooled Reactor (HTGR)

While small HTGRs (Peach Bottom I and Fort St. Vrain) have been licensed in the past, several significant issues are not yet resolved in licensing the large HTGRs. Based on the NRC staff responses to licensing reviews of the past HTGR commercial projects (with steam generators) for Delmarva Power and Light Company and the Philadelphia Electric Company, and the ongoing GASSAR-6, the following issues are outstanding: seismic design and analysis, structural design practices for graphite, in-service inspection, emergency cooling, allowable temperatures to limit fuel failure during operation, accident probability and consequences evaluation, fission product transport for reactor siting evaluation, plant response to ATWS events, design verification programs for prototype components, oxidation of core support structure, preoperational vibration assessment for reactor internals, and design of radial flow reheater. Additional safety/licensing issues which may have to be treated result from design/cost improvements: fuel particle coating and fuel element, primary coolant system, unlined containment, and PCRV penetration and closures.

A program to design and develop a commercial gas turbine HTGR (GT-HTGR) power plant has been underway at General Atomic Company (GA) for several years. The GT-HTGR comprises a graphite moderated core, fueled with fully enriched U-235, and cooled by helium that is circulated by turbocompressors located in separate loops within the primary coolant system. The issues which merit the greatest licensing attention and which are unique to the GT-HTGR are: safety analysis of the consequences of a spectrum of turbo-machine failures, safety analysis of generator failures, safety analysis of spectrum of internal pressure equilibration accidents, evaluation of flow

resistance assumed for passive non-nuclear primary system components during cooldown, thermal barrier design to withstand pressure transients which accompany gas turbine accidents, safety classification of primary system components, and maximum area for a PCRV depressurization accident.

These matters appear amenable to resolution, but some require research and development. The licensing outlook is favorable.

Overall development costs are estimated to be \$150M expended over a period of 12 years. Development costs associated with the safety/licensing of the HTGR concept are included in the overall development cost estimate, but will be broken out as a separate item for subsequent versions of the NASAP Rolling Report.

3.4.3.5 Light Water Breeder Reactor (LWBR)

Since the reactor plant itself is basically a PWR, it can be readily concluded that the LWBR is fundamentally licensable from a reactor safety viewpoint. Significant changes in design of pressure boundaries, heat transfer equipment, emergency cooling systems, containment arrangements, etc. are not anticipated. The core-related safety aspects, including the aspects of nuclear stability (particularly reactivity coupling between seed modules), power and temperature coefficients, control stability and adequacy, provisions for accident prevention, the potential for recriticality events during meltdown accidents, and core thermal margins would require detailed evaluation to determine acceptability of a given design. It is not presently clear if the LWBR concept can be scaled to larger-size reactors without significant design changes and additional R&D. The acceptability of fuel movement as a means for reactivity control, and the prevention of misoperation and accidents related to this characteristic, would need to be established early in the design development because of the importance of this approach to neutron economy, internal structures, and safety requirements.

The NRC staff and the NRC Advisory Committee for Reactor Safeguards (ACRS) performed a safety review of the Shippingport reactor using an LWBR core, and found the design acceptable from a safety viewpoint when operated according to the plans of DOE-Division of Naval Reactors. This tends to produce a favorable outlook regarding the licensability of LWBR cores when used in PWRs, but the previously mentioned unknowns and uncertainties regarding extrapolation to large-size reactors moderates this outlook.

3.4.3.6 Liquid Metal Fast Breeder Reactor (LMFBR)

The Enrico Fermi reactor was a licensed reactor and the FFTF has been evaluated by the NRC staff and the ACRS and found acceptable for construction. An additional NRC review prior to operation of the FFTF is currently underway. The licensing evaluation of the CRBR conducted thus far indicates that the state of technology and experience should result in a safe design, but the particular design proposed has not yet been found satisfactory. The major concerns that have not been resolved include instrumentation to detect core abnormalities which could lead to accidents, inspection of the reactor system to provide continued confidence in the system integrity, decay heat removal system reliability, and containment design to withstand low probability accident consequences. All of these topics are being addressed in the United States and other countries, and there is a favorable outlook that they can be satisfactorily resolved.

The alternative fuel cycles have an effect on the plant design, safety, and licensability. Of significant importance for licensing is the demonstrated technology; this is presently concentrated on the mixed oxides of uranium and plutonium. Besides accident prevention, core-disruptive accidents, and accident energetics in particular, will continue to represent the key safety issues in the licensing of the LMFBR. The following assessments are made relative to fuel type and licensability:

Relative to accident prevention, no significant safety implications of other fuel types relative to the reference (U,Pu) oxide fuel appear to be present.

The probability for extensive core disruption would not appear to be significantly different for the (U,Pu) oxide and (U,Pu) metal fuel types. For postulated unprotected accidents, preliminary considerations suggest that it becomes more difficult to argue for limited-core damage with carbide fuel than for the oxide and metal fuel types.

If extended fuel dispersal by sodium bond vaporization can be demonstrated, the (U,Pu) metal fuel appears at least as favorable as the (U,Pu) oxide fuel relative to the generic accident energetics issues

(recriticality and fuel-coolant interaction). In particular, the fuel-coolant interaction becomes a nonproblem for this fuel type.

Despite the somewhat higher sodium void worth and lower Doppler coefficient, the (U,Pu) metal fuel appears to have an advantage over the oxide and carbide fuel types relative to the large core positive sodium-void coefficient problem. From a safety point of view, the anticipated earlier dispersive characteristic of the (U,Pu) metal fuel may decrease the need for considering heterogeneous designs. The differences in void worth and Doppler coefficients between (U,Pu) oxide, carbide, and metal fuel types would not appear to have significant safety implication by themselves.

There are safety concerns with the (U,Pu) carbide fuel, because of less favorable characteristics relative to timely fuel dispersal and benign fuel-coolant interactions, as compared with the oxide and metal fuel types. In particular, the potential for a pressure-driven recriticality event would appear to be more difficult to rule out for this fuel type.

Based upon an ex-vessel post-accident heat removal approach with an explicit core retention device (passive or active), the metal fuel would appear to have advantage because of its low melting point and high thermal conductivity, if the question of oxidic fission product migration can be solved by material and design studies. With the inherent retention in the concrete approach, a moderate advantage appears with the oxide fuel because no additional heat or gas are generated by fuel reactions.

While the existing data base is considerably less for the (U,Pu) metal fuel, the magnitude of effort required to resolve key safety issues for the oxide and metal fuel types is considered to be about the same. This is because the lack of data base for the (U,Pu) metal fuel is largely offset by less demand on safety test facility requirements. Relative to (U,Pu) carbide fuel, considerably more detailed testing may be required in view of considerably less favorable safety characteristics as compared to the oxide fuel. The test facilities being planned for oxide fuel should also be adequate for carbide fuel.

With respect to alternate fuel types (thorium based) all the high thermal conductivity fuels (carbide and metal) would appear to have less favorable safety characteristics as compared to the oxide fuel. No significant differences between the latter fuel type and the (U,Pu) oxide fuel are anticipated. Again the test facilities planned for the reference oxide fuel should also be adequate for the alternate fuels.

3.4.3.7 Gas Cooled Fast Reactor (GCFR)

The NRC staff and the ACRS have performed a licensability review of a demonstration size GCFR. In August 1974, the NRC staff published a "Pre-Application Safety Evaluation Report" for the GCFR concept which reached a general conditional conclusion that the GCFR demonstration plant proposed in concept by General Atomic is potentially capable of being designed and constructed so that it can be operated without undue risk to the health and safety of the public. This general conclusion is similar to the general conclusion reached by the staff on the LMFBR demonstration plant, but the information base is significantly greater for the LMFBR plant.

The conditions identified in the NRC staff safety evaluation of the GCFR were specific in many areas, and their satisfactory resolution could cause substantial design changes. The major safety considerations which were considered as unresolved were:

- acceptable power density levels and thermal margins,
- definition of depressurization accidents,
- definition of core disruptive accidents,
- diversity in reactor shutdown systems,
- adequacy of core cooling systems, and
- containment design bases for combustible gas, plutonium aerosols, core disruptive accidents, and post-accident heat removal systems.

The Advisory Committee on Reactor Safeguards issued an "Interim" letter report on the GCFR in November 1974 which left open the decision of "licensing feasibility." The ACRS report indicated the following concerns: core cooling reliability, common mode failure potentials for circulators, valve

reliability in the primary circuit, fuel damage effects on core cooling reliability, containment design bases, core disruptive accidents, reactor vessel and containment response to core disruptive accidents, reliability of shutdown systems, reactor vessel failure mechanisms, in-service inspection of the reactor vessel liner and the impact of liner loss, engineering data on components, and the maintenance of design flexibility in view of the many outstanding issues. It should be noted that the GCFR concept has safety advantages of a strong vessel (PCRVR) for containment of accidents, and does not have large void coefficients characteristic of present LMFBFR designs.

Since 1974, General Atomic has been developing information relative to licensing concerns expressed in the "Pre-Application" review. Progress on this topic was briefly presented to the ACRS and the NRC staff in July 1977, but the NRC staff has no present plans to review this newly available information. The NRC is, however, reviewing a site near Amarillo, Texas for suitability for construction of the 300-MWe prototype GCFR by Southwestern Public Service, a member of Helium Breeder Associates.

Based on the licensing concerns of the NRC and ACRS, General Atomic has pursued a program to resolve the outstanding safety/licensing issues as follows:

Other fast and thermal reactor systems can provide a combination of built-in thermal capacity and/or natural circulation cooling, which avoids the need for rapid restoration of forced circulation of primary coolant to prevent fuel overheating. Studies are continuing on other GCFR design arrangements which provide sufficient natural circulation cooling if coolant pressure is maintained.

Studies of the commercial size GCFRs show that the overall safety characteristics of the GCFR are expected to improve as plant size increases. Reactivity and kinetics parameters of GCFR cores ranging from 300-1500 MWe have been evaluated and results indicate that as plant size increases, the Doppler coefficient becomes more strongly negative and fuel, cladding, and coolant specific worths decrease significantly, so that whole-core cladding and coolant worths for the 1500-MWe core are only slightly higher than those for the 300-MWe core.

Larger plants require a lower core enrichment which reduces the burnup reactivity swing.

Larger plants have a larger reactor coolant inventory which will result in slower depressurization rates for a given flow restrictor area.

As plant size increases, the number of main cooling loops generally increases which should improve RHR reliability.

Similar to the LMFBR program, the commercial introduction of the GCFR requires assurance that the potential consequences of a core disruptive accident are not severe. Programs are underway to provide such assurance prior to commercial introduction of the GCFR.

The GCFR Safety Test Program includes a series of out-of-pile tests to resolve phenomenological uncertainties followed by in-pile tests which will serve as the tool for verification of analytical techniques for assessing the consequences of core disruptive accidents.

The out-of-pile Duct Melting and Fallaway Test (DMFT) program is being conducted at Los Alamos Scientific Laboratory to determine the major phenomena related to cladding melting and refreezing and assembly duct melting during a protected loss-of-flow accident. The DMFT program is currently in the heater development phase.

The Gas-Reactor In-Pile Safety Test (GRIST-2) program was initiated in 1976 with the Idaho National Engineering Laboratory (INEL), ANL, and GAC being the major participants. Using the Transient Reactor Test Facility (TREAT) upgrade facility, the tests will be capable of simulating fuel behavior in rod bundles during unprotected loss-of-flow or transient over power accidents. During 1977, the conceptual design of the helium loop systems was established.

The major safety R&D costs are associated with the DMFT program and the GRIST program. The DMFT program is scheduled to be completed in 1980 at a total cost of \$2M. The GRIST program is scheduled for initial testing to begin in 1983 with a test period of five years at a total cost of approximately \$50M.

Overall, the NRC regards the licensability of the GCFR as not well-established. All major issues will be faced in the licensing review of the GCFR demonstration plant. No constraints on the commercial introduction date due to safety/licensing considerations are foreseen.

3.4.3.8 Molten Salt Reactor (MSR)

The fuel cycle for the MSR can be modified to increase proliferation resistance by using denatured uranium as the fuel. Two fuel cycles are being considered for denatured molten-salt reactor (DMSR) systems. The first of these is a break-even breeder cycle ($CR = 1.0$) which requires only Th and U-238 feed after its initial loading, but which also requires a full-scale, on-line, in-containment fuel processing facility for fission-product cleanup to sustain its performance. The alternative DMSR fuel cycle is based on the same reactor design but substitutes fuel addition for fission-product removal to sustain operation in a converter mode. The reactor is fueled initially with denatured U-235 (20% enrichment), and all subsequent additions are limited to denatured U-235 and/or depleted U-238. The thorium concentration is allowed to decline slowly with time (via Th-232 conversion to U-233) to allow for the buildup of U-238 from subsequent additions of denatured fuel. During the first five years, some depleted U-238 would have to be added to maintain denaturing as the dominant fissile uranium nuclide gradually shifted from U-235 to U-233. The subsequent additions of denatured U-235 would bring in more than enough U-238 for denaturing, and the fissile U isotopic enrichment would gradually decline. At the end of reactor life (30 yr at 75% capacity factor), the fuel charge would be removed for storage and eventual disposal.

The most important feature of the once-through fuel cycle is the absence of the on-line chemical processing facility for fission-product cleanup. Although this fuel cycle eliminates chemical removal of fission products, it by no means eliminates all chemical treatment of the salt. Since the primary circuit will occasionally be opened to the atmosphere for maintenance, and since the fuel salt has limited tolerance for oxide contaminants (as from atmospheric oxygen and moisture), a hydrofluorination facility for oxide removal is almost certainly a requirement. This facility would have to permit no more than occasional bathwise treatment of a fraction of the fuel salt and could be relatively simple to operate and maintain. The reactor

operator would also be required to maintain close surveillance and regulation of the redox potential of the salt and its general chemical composition. All of these requirements would be much less demanding than fission-product processing. The continued addition of (denatured) fissile material and the simple chemical treatments required to maintain the fuel in operating condition will probably require replacement of a portion of the fuel charge during the 30-year operating life of the reactor. About 7% of the fuel charge must be removed to storage in "waste" tanks or perhaps in the reactor fuel drain tank.

Many items, such as safety and environmental considerations, reactor technology status and needs, and the potential schedule and cost for the reactor development are nearly independent of the fuel cycle.

The most distinguished characteristic of the MSR system in general, and from the standpoint of reactor safety and licensing, is that the fissionable fuel is in fluid form (molten fluoride salt) and is circulated throughout the primary coolant system, becoming critical only in the graphite moderated core. The primary system coolant serves the dual role of being the medium in which heat is generated within the core, and the medium which transfers heat from the core to the primary heat exchangers. Thus, the entire primary system is subject to both high temperatures (700°C at core exit) and high levels of radiation by a fluid containing all the daughter products of the fission process. Because of the low fuel salt vapor pressure, however, the primary system design pressure is relatively low. The entire primary coolant system is analagous, in terms of level of confinement, to the cladding in a solid fuel reactor. Two additional levels of confinement will be provided in the MSR, in accordance with present practice.

It is the breach of the primary coolant system boundary, resulting in a spill of highly radioactive salt into the primary containment, which will probably provide the design basis accident. Possible initiators of this accident include pipe failure, missiles, and pressure or temperature transients in the primary coolant system. Work to date indicates that these initiators are highly unlikely, but this is a prime area of concern for future safety R&D. In the event of a salt spill, a redundant system of drains will act to channel the salt to the cooled drain tank. The primary system containment, defined as the

set of hermetically sealed, concrete-shielded equipment cells, would probably not be threatened by such a spill, but cleanup operations would be difficult. Under accident shutdown conditions, the fuel material would be led to the "ECCS" (represented by drain tank cooling), rather than vice versa. The reactor and containment must be designed so that the decay heated fuel salt reaches the drain tank under all credible accident conditions.

Neutronically, the MSR core is well behaved. The amount of excess reactivity which must be controlled (and is available under accident conditions) is small, less than 1% $\delta k/k$. Because the prompt temperature reactivity coefficient of the fuel salt is negative and the delayed neutron fraction is unusually low, the power reacts rapidly to dampen temperature fluctuations. This response is large enough to limit any credible rapid introduction of reactivity. The small positive void coefficient and the extremely high boiling temperature of the fuel salt ($\sim 800^\circ\text{C}$ above normal operating conditions) makes voiding an unlikely accident initiator. Also, in a voided region, the main heat source is lost along with the cooling capability. Thermally, the MSR core is, for short transients at least, an inherently coolable geometry--there is no large amount of heat stored in solid fuel which must be removed rapidly after shutdown to avoid meltdown. The decay heat of the fuel salt, however, must eventually be removed by the drain tank cooling system.

Aside from the question of chemical reprocessing, the design for a DMSR differs from that for an MSR in some significant ways. The low core power density and the goal of a 30-year graphite life tends to shift the graphite development effort toward greater irradiation stability and away from low gas permeability. The once-through DMSR also requires added chemical studies to clearly define solubility limits for materials that would build up during the 30-year fuel cycle. Some aspects of the reactor safety work, particularly those affected by the lower power density, would likely receive different degrees of emphasis.

Areas which have been identified as requiring safety R&D include: guidance studies which will include probabilistic risk assessment of postulated accident scenarios, fission product behavior, primary systems material, primary system inspection, component and systems technology (under accident conditions),

safety instrumentation and controls technology, remote maintenance technology, and safety technology of processing and waste storage handling. It should be noted here that two problem areas, those of excess tritium diffusion and Hastelloy N embrittlement, may not be as serious as previously thought, but will in any case be treated in the safety R&D program.

The problem of developing a primary coolant system which will be reliable, maintainable (under remote conditions), inspectable and structurally sound over the plant's 30-year lifetime will probably be the key factor in demonstrating ultimate safety and licensability.

The MSR would be the first molten salt reactor to go through the NRC licensing process. (The MSRE was constructed as an AEC test facility and did not require formal licensing.) Many of the regulations and precedents set for solid-fueled reactors would not apply to fluid-fueled reactors, and it is estimated that two to three years would be required to establish safety guidelines for this new reactor type.

It is estimated that the R&D safety program will cost \$15M over the first ten years.

3.4.3.9 Accelerator Driven Systems

Three types of accelerator fuel and power producers are being considered:

LAFR - Linear Accelerator Fuel Regenerator,

LADR - Linear Accelerator Driven Reactor,

LAFP - Linear Accelerator Fuel Producer.

All three of these systems employ either natural uranium, low-enriched uranium, or thorium as the fresh fuel charge. Studies to date indicate that most efficient operation results from employment of fuel containing uranium at about 2% U-235 content.

LADRs and LAFRs operate without reprocessing, while the LAFP cycle requires reprocessing. The LADR employs an accelerator "on-line" with the driven reactor. This is to be contrasted with the LAFR and LAFP, where the accelerator is "off-line" and services a number of reactors. The LAFR cycle is compatible with LWRs currently in service and makes the maximum use of existing LWR technology.

The LAFR increases the fissile content of 2% LEU (U-235/U) to 3.2% (U-235 + Pu) by conversion of U-238 to Pu. The exposure of the fuel in the LAFR corresponds approximately to 3,000 MWD/T. The plutonium produced in the fuel rods is both diluted by uranium-238 and contaminated with fission products.

After being burnt to 30,000 MWD/T in an LWR, the fuel (originally generated by the LAFR) is returned to the LAFR for regeneration with an exposure of 3,000 MWD/T. A second 30,000 MWD/T cycle would then take place in the LWR. At this stage, the spent fuel (which has gone through 60,000 MWD/T burnup for power production and about 6,000 MWD/T for regeneration) is highly contaminated with fission products.

In the LADR, plutonium production and utilization take place in situ. The fuel never leaves the reactor until the end of the cycle when it is ready for storage and disposal. Fuel can be burnt in the LADR to a lower fissile content than in the LWR once-through cycle. The discharged fuel has a very high burnup with a consequent increase in the fission product inventory and hence very high radioactivity. An advanced reprocessing technology would be required to extract the plutonium from such high burnup fuel.

Operation of the LAFP in conjunction with LWRs or HWRs will require reprocessing and fuel fabrication involving plutonium or U-233 generated by the LAFP. If U-233 is produced, it would be consumed along with thorium, and presumably plutonium would be mixed with uranium for subsequent burning. A denatured fuel cycle could be supported if U-233 is made by irradiating thorium alone in a LAFP. Denatured fuel could then be made through mixing this U-233 with natural uranium and thorium within a safeguarded fuel cycle center.

The accelerator driven systems are at present conceptual studies with major safety issues still to be identified as the concepts are refined. Some issues of current concern are:

In addition to the hazard potential of fission products and actinides, there are the radioactive spallation products in the target material and the activation in the linear accelerator.

Since the subcritical blanket is driven by the proton beam, alternate and diverse ways of shutting down the beam will have to be demonstrated.

Since no provision is made for control rods in the blanket, it must be demonstrated that the blanket is designed to remain subcritical under all burnups, material changes, and geometric configurations that are physically possible, or alternatively a reactor shutdown system may be incorporated.

Because both the target and blanket generate large amounts of heat during operation, and both have decay heat, the issue of emergency core and target cooling in a loss-of-coolant accident must be resolved. Electrical power requirements for reliable shutdown heat removal for the LAFR may pose a special problem. Because of the possibility of short unscheduled accelerator-outages, and the frequent downtime needed for fuel replacement and shuffling, the shutdown heat removal system will be correspondingly challenged.

If a pressure-tube concept is employed in the LAFR blanket, consideration should be given to methods for detection and mitigation of the consequences of pressure-tube failure. In particular, protection against failure propagation should be provided.

For the fuel regenerator, the buildup of actinides may require that for source term guidelines, site suitability include specific dose guidelines for the actinides above and beyond the guidelines given in 10CFR100.

Of central concern is the blanket meltdown accident and its potential role in the reactor licensing process for the fuel regenerator. For

an accelerator fuel regenerator based on light or heavy water, containment issues may be associated with steam and hydrogen explosions, fuel-coolant interactions, and fuel-concrete interactions. Examples of design safety features which may be considered for the LAFR at least in its early stages are: a core catcher (or appropriately, blanket catcher) and filtered venting systems.

The ability of fuel cladding materials to sustain long burnups in the power reactor ($\sim 60,000$ MWD/T) and the irradiation cycle of the regenerator will have to be demonstrated.

A unique feature to be considered in the licensing process is the fact that there are two interacting systems: the linear accelerator and the target-blanket system.

The above considerations may present an unnecessarily pessimistic view of design features required to ensure safety of the LAFR, though adoption of a conservative view at this stage is the prudent course. On the positive side, the low power density in the LAFR and the pressure-tube design which reduces likelihood of a loss of coolant that might cause extensive meltdown of fuel could serve to relax limits on LAFR designs for safety purposes.

However, the Linear Accelerator Fuel Regenerator does contain unique features in comparison to currently licensed light water reactors. Consequently, prudence dictates that an initial Demo Plant accelerator fuel regenerator design contain sufficient safety margins to account for uncertainties in those areas for which public health is at risk.

The LAFR concept raises safety questions associated with the use of LAFR fuel in an LWR. Namely, the pre-irradiation of LWR fuel in the LAFR to increase the enrichment will result in a non-uniform fuel enrichment distribution because of LAFR gradients. This non-uniformity will have to be considered in the PWR safety analysis based on power peaking factors. If the LAFR-generated fuel is used in existing PWRs, a power derating may be necessary to accommodate the uncertainty factors associated with a non-uniform enrichment distribution.

It is estimated that the R&D to support safety/licensing activities will cost \$100M and will span 15 years.

3.4.3.10 Gaseous Core Reactor

The mixed flow gaseous core reactor is at present a conceptual study, with major safety issues still to be identified as the concept is refined. Some current concerns are:

The effects of moisture leaking into the primary circuit because of heat exchanger failure. This would increase pressure, moderation, and power in the core, constituting a positive reactivity insertion.

Leakage of the entire fuel and fission product inventory into the primary containment vessel in the event of loss of long-term primary loop integrity.

Reliability of the primary pressure boundary materials under severe service conditions, i.e., temperature, pressure, and corrosive nature of coolant.

Definition of the dynamic effects of transients on an interlocking series of coolant loops such that stable and controllable conditions are maintained.

Operation of the primary and auxiliary systems to assure criticality safety under all flow conditions.

Primary system in-service inspection and use of considerable amounts of equipment under conditions requiring remote operation and maintenance.

Acceptability of on-site reprocessing and cleanup resulting in amounts of radioactivity available in forms possibly more prone to release than at existing commercial nuclear power plant sites.

Definition of depressurization accidents.

Definition of core disruptive accidents.

Diversity in reactor shutdown systems.

Adequacy of core cooling system.

Tritium generation and control.

Actinide concentrations in fuel, blanket, and process streams.

Accident probability and consequence evaluation.

Research programs to address problem areas related to safety and licensing of the gaseous core reactor concept can be identified as follows:

Containment methodology and evaluation.

High-temperature materials development and testing.

Corrosion-resistant materials development and testing.

Analysis and testing of complex fuel coolant flow behavior (including flow stability and pressure transients).

Development of remote handling equipment.

Development of the fission product and fuel reprocessing systems.

Systems control characteristics.

Core kinetics (space-dependent treatment needed).

Measurement and analysis of reactivity feedback coefficients.

Development of analytical tools to examine postulated accidents, such as loss of circulators and system rupture.

The experimental and analytical research effort required to address these areas is significant and considered to be an integral part of the efforts required for safety/licensing leading to commercial feasibility. As such, many of the safety-related areas are intimately connected with operational problem areas and it is difficult to assign a cost for the safety-related portion only. Approximately \$200M is estimated to be required over a period of 15 years for basic research and development. Approximately 30% of that cost, or \$60M, can be interpreted as applicable to safety-related R&D over the same period of time.

Although safety and licensing considerations to be resolved are significant, it is not felt that these impose a limiting constraint on commercial introduction of a gaseous core reactor. The combination of operational and safety/licensing R&D with the construction and successful operation of various demonstration facilities suggests an introduction date in about the year 2000.

3.4.3.11 Fusion-fission Hybrid Reactor

Most of the safety/licensing aspects of fusion-fission hybrid reactors will actually be generic to fusion. The fission blanket can be designed to be very subcritical, virtually eliminating a broad range of safety concerns associated with criticality accidents.

Liquid lithium may be utilized as a reactor coolant and/or tritium breeding material for any deuterium-tritium fusion driver. Accidents affecting the blanket or coolant vessels could cause this lithium to be released. The liquid lithium so released is hot and chemically very reactive. It can damage those components that it contacts directly, such as the primary vacuum vessel and shield, structural supports, and magnet components. At the design coolant temperature, it would ignite spontaneously in air and would also react vigorously with water or concrete. Lithium-concrete fires can cause major structural damage directly, as well as the damage to reactor components and containment caused by resulting overpressure and missile-generation. Under accident conditions, it is possible that solid lithium compounds might also produce liquid lithium metal.

Radioactive materials in a hybrid power plant may be found in four forms: tritium, actinides, activation products, and fission products. Routine and accidental release of tritium will be a major safety consideration in any fusion system. Tritium may be released by permeation at high temperatures, or by leakage through valves and seals. Accidental failure of tritium-containing components including fuel storage vessels, getters, cryogenic distillation columns, and fuel extraction and injection lines, may release large amounts of tritium. The biological hazard due to tritiated water is much greater than that due to tritium gas. Deuterium and tritium, as hydrogen isotopes, can

become explosive on contact with air, particularly if they are released as liquids.

The fission modules containing actinides and fission products must operate safely in conjunction with unique fusion systems. Of concern are initiating events leading to loss of containment or reconfiguration of radioactive materials. Although criticality appears not to present a serious safety concern, substantial decay heat will be generated in the blanket. Under accident conditions affecting the ability of the coolant to remove heat, the integrity of the blanket consisting of actinides, fission and activation products, and tritium may be destroyed. Characterization of the blanket behavior under accident conditions will require extensive analysis. The presence of a large vacuum chamber in the nuclear core adds a new dimension to the safety analysis. The inventory of specific isotopes generated by the fusion neutron spectrum must also be established to evaluate the consequences of accidental release. The reactivity effects of cryogen spills on the blanket will have to be investigated.

The 14-MeV fusion neutron source can result in highly activated structural materials with different characteristics than those found in the fission reactor, depending on the design. Coupled with mass transport in the cooling and vacuum systems, this can result in occupational exposure to plant personnel, especially during maintenance operations. Control and disposal of radioactive waste materials and decommissioning procedures will require additional analysis.

Fusion driver concepts using strong magnetic fields will require continued research to quantify biological hazards and establish appropriate regulations. Forces and eddy currents, as well as electromagnetic radiation, generated by rapidly changing magnetic fluxes should be examined.

All fusion concepts require large influxes of energy to achieve the conditions necessary for fusion burn. Energy storage and switching associated with normal and superconducting magnets, neutral beam injectors, inertial confinement drivers (lasers, electron and ion beams), etc. must be examined for initiating events resulting in an explosive release of energy. Release of cryogenic fluids may present hazards unique to fusion-driven nuclear systems, such

as sudden resistive heating of superconducting magnets, spilling and vaporization of liquid helium and hydrogen isotopes, thermal shock, and energetic vapor interactions. Failure of the vacuum vessel may lead to energetic missiles and primary containment failure.

The various fusion driver concepts and the specific safety issues discussed above introduce a number of new requirements on design basis type accident analysis used for the current generation of reactors. Much of the existing fission technology may be relevant in considering fission blanket design criteria.

Analysis of control rod malfunctions can be eliminated. All hybrid reactor concepts specify a subcritical blanket, with no need to control excess reactivity. Large overpower transients in fusion energy production are highly unlikely.

The power density in hybrids is typically less than in LWRs and liquid metal cooled systems and can be designed to provide decay heat cooling by natural convection in loss-of-flow accidents. The actual response will be design dependent, particularly for gas-cooled concepts. The potential for critical reconfiguration of fuel during a LOCA, steam ingress accident, or vacuum chamber failure is also design dependent. Most hybrid systems proposed to date physically eliminate the possibility of criticality regardless of reconfiguration geometry. The possibility of reconfiguration must be addressed independently of criticality consideration. It is possible to operate a hybrid design nearer to criticality and to rely on engineered safeguards similar to LMFBR designs. However, this would compromise a major safety advantage hybrids have over LMFBRs. The hybrid also has a distinct advantage in that blanket modules driven by the fusion reaction can be isolated in the cooling system to minimize the effects of loss-of-coolant accidents.

Loss-of-coolant flow and loss-of-coolant accident analyses will have to be extended to other hybrid systems, particularly cryogenic superconducting magnets. Failure in these or other energy storage systems may represent initiating events resulting in missile generation, loss of containment, etc.

Loss of on-site power will result in the immediate loss of the fusion driver for several concepts, particularly laser or beam driven inertial confinement reactors. Magnetic concepts would also lose neutral beam heaters and current to magnets, resulting in a plasma dump to the vacuum chamber first wall. Superconducting currents will need an active trip to interrupt plasma confinement, and it may be advisable to include a passive impurity injection system to provide plasma cooldown. The vacuum pumping system and tritium processing stream will also have to be investigated for problems caused by loss of power.

Steam generator tube rupture or steam line breaks will have to be investigated for higher tritium inventories in hybrid concepts, especially at liquid metal/water interfaces.

Fuel handling and processing accidents must include blanket modules as well as tritium systems. Some modules may require extensive disassembly or cutting for transportation and processing or storage. The geometry and composition of these vary with design, and maintenance, handling, shipping, and processing requirements must be established before accident analysis can begin.

Unique effects due to excess load, or loss of load, turbine trip, etc. are not known at present and will have to be analyzed.

A vigorous research and development program to understand materials properties under high and low temperature, intense radiation, and cyclic loading conditions must be pursued to qualify materials for hybrid reactors. Tritium handling techniques under normal, maintenance, and accident conditions must be developed and demonstrated. The behavior of the fission blanket under accident conditions must be carefully examined. The consequences of coolant disturbance accidents should be determined and the capability for safe recovery demonstrated. The characteristics of liquid lithium spills should be investigated so preventative and controlling measures may be developed. The safety of large superconducting magnets and extremely powerful laser systems must be demonstrated.

The above unique safety problems associated with fusion hybrid concepts and their impact on design basis criteria need resolution for licensing of hybrid facilities. The effort still needed to develop this concept is extensive and may require resources equal to those expended on the LMFBR in the last two decades.

It is estimated that the safety R&D program will cost \$200M over 15 years.

Most of the constraints placed on the commercial introduction date due to safety/licensing considerations are generic to any deuterium-tritium fusion reactor concept. A number of safety research facilities are required, including those for tritium systems demonstration, superconducting magnet reliability development, and materials irradiation. A Tritium Systems Test Assembly is expected to begin operation at Los Alamos in 1980. Several large superconducting coils are presently planned for construction, none of which are dedicated to safety research. Several fusion materials irradiation facilities are planned, the largest one to be completed at Hanford in the early 1980's. Some modification of these pure fusion facilities may be warranted for hybrid applications. Assuming a three-year operation time before information from these facilities can be factored into hybrid systems designs, and another eight to twelve years for design, licensing, and construction of a demonstration plant, it appears that a very optimistic date for commercial operation would be after 2000.

3.4.3.12 PWR/HWR Tandem

The tandem concept is based on utilizing fuel discharged from a PWR as the fuel charge for a HWR. The basic safety/licensing issues then pertain to those of the PWR and HWR individually, except for the feature of geometric incompatibility of PWR and HWR fuels.

The reference Tandem Cycles considered for either refabricated or reconfigured fuel require no modification of the PWR, and thus do not alter its safety or licensability. Two modifications to PWR fuel might, however, be contemplated to facilitate the incorporation of reconfigured fuel in the HWR. These would be expected to require design modifications or operating restrictions in the PWR due to safety-related considerations.

The first of these modifications might be to employ segmented fuel in the PWR to facilitate the reconfiguration of the short assemblies employed in HWRs of the CANDU type. Axial gaps in the active fuel column would be necessary to accommodate fastening devices and to form a plenum for the fission gases.

Since axial peaking is increased, a corresponding reduction in achievable reactor power might be required to preserve the margin to such accidents as LOCA inherent in the more conventional unsegmented PWR fuel design; a more detailed evaluation of segmented fuel will be required to ascertain the actual magnitude of this problem.

A second modification to the PWR which might be considered is to increase the diameter of the PWR fuel rods to that employed in the HWR, again to facilitate obtaining a fuel bundle more typical of current PHWR designs through reconfiguration. Changes in fuel pin diameter and number of pins would affect the power distribution in the core and would require a safety analysis to ensure that PWR licensing requirements are met.

The following safety considerations pertain to the HWR for the Tandem PWR/HWR concept:

The HWR void coefficient of reactivity is positive in both the tandem and natural uranium cycles and is no more adverse for tandem fueling than for natural uranium fueling.

The use of reconfigured fuel will require a rather extensive redesign of the HWR to accommodate the altered geometric configuration. For refabricated fuel, the design bases are also changed sufficiently so that some degree of modification will also undoubtedly be required, although much less extensive than in the former case. Power peaking increases by about 20% with tandem fueling and hence a power derating or a modification to the overall plant design may be required.

It may also prove necessary to include gas plenums in each of the HWR fuel segments to accommodate fission gas production, since cycle lengths are much longer and hence fission gas production greater. If such plenums are required, some increase in axial form factor is expected similar to that obtained for PWR fuel, which could also necessitate some power derating.

Additional control rods would have to be added to compensate for the more highly absorbent tandem fuel.

Modifications to the shutdown cooling system probably will also be required to accommodate the increased decay heat generation.

It is anticipated that a detailed safety analysis for tandem fuel would identify additional areas where the design basis is sufficiently altered from that of natural uranium fueling to necessitate plant modifications. Consequently, it is concluded that while safety-related considerations do not appear to impose a fundamental constraint on tandem fueling, these considerations will generally preclude tandem fueling in existing commercial HWRs and will necessitate the use of modified HWR designs. The main safety/licensing issue involves the refabrication or reconfiguration of PWR fuel for use in an HWR.

3.4.4 Fuel Cycle Facilities

3.4.4.1 Mining

Source materials for supporting all reactor concepts consist only of uranium and thorium. Mining operations *per se* are not licensed.

Uranium

Safety evaluations and environmental effects have been presented in NRC publications WASH-1248, NUREG-0002, -0016, and -0216. The principal environmental effects are the release of radon from ore bodies and surface disturbances from the removal of ore. Due to the remoteness of mining activities, population exposures have been low and judged generally to be acceptable. Methods to restore strip mined areas are available and their efficacies are being studied and evaluated.

Thorium

Most thorium is obtained as a byproduct of the mining of other metals, principally rare earths. As in the mining of uranium, there is some exposure of workers to airborne radon.

3.4.4.2 Milling

Uranium

At the end of 1975, fourteen conventional mills were operating routinely in the United States. Mills are licensed by the NRC under 10 CFR Part 40 of the Code of Federal Regulations. The principal environmental consequences occur from radon emissions from mill tailings. These have been evaluated in NRC publications WASH-1248 and NUREG-0002.

Thorium

Environmental impacts associated with thorium milling and refining would be similar to those relating to the uranium industry. The principal environmental concern is the release of radon from in-process material and tailings piles. Thorium milling is governed by 10 CFR Part 40 of the Code of Federal Regulations, similar to uranium milling. It is expected that license conditions would be comparable, and could be met without excessive difficulty. Except for quantities, the safety and licensing aspects of non-traditional fuel cycles in these areas are unaffected.

3.4.4.3 Conversion

Uranium

The uranium product from the mill needs to be converted to a chemical form suitable for its intended purpose. For ultimate use in LWR reactors, uranium is converted to the hexafluoride for subsequent enrichment in a gaseous diffusion plant. Two commercial conversion plants are licensed by the NRC. The NRC staff believes that existing regulations are suitable to permit continued licensing.

Conversion of UF_6 following enrichment is practiced routinely today as the lead step at fuel fabrication plants which are licensed by NRC. Environmental effects have been described in WASH-1248 and NUREG-0002. Existing regulations are probably adequate for continued licensing.

For use in reactors fueled with natural uranium, the uranium is converted to oxide. The steps are simple ones, and environmental effects are expected to be low. The operations would probably be performed at fuel fabrication plants which are subject to NRC licensing actions. No licensing problems are envisioned for natural uranium, regardless of the fuel cycle variation.

In the case of recycle or highly irradiated and denatured uranium-233, conversion to UF_6 is not likely. If practical, however, the process would require shielded facilities and attention to decay chain daughters. Licensing should be possible without excessive delay based on similarities to the Midwest Fuel Recovery Plant (MFRP).

Thorium

Conversion of thorium is not practiced on a large scale. The oxide required is easily obtained and environmental effects should be low. Attention must be given to control of possible emissions of radon. Licensing should not be a serious problem.

Plutonium Oxide and Mixed Oxide (Plutonium plus Uranium)

The conversion of plutonium (or mixed uranium-plutonium) to oxide following reprocessing has not been practiced on a large scale, except for military needs. Small-scale operations for applications such as FFTF have been licensed. Environmental impacts have been described in NUREG-0002. Licensing

proceedings for a large-scale facility (EXXON) have now been terminated. These had not progressed sufficiently to state that major problems do or do not exist.

The recycle of spiked (contaminated) plutonium or mixed uranium-plutonium will involve new technological problems, and significant licensing difficulties may develop.

3.4.4.4 Enrichment

Present enrichment facilities for uranium 235 in natural or recycle uranium are operated by DOE and are not licensed. Environmental effects have been described in WASH-1248. The NRC is developing regulations for enrichment facilities but these have not yet been published. It is likely that regulations for new plants will require considerable upgrading from current practices but no particular difficulties are envisioned.

3.4.4.5 Fuel Fabrication

Uranium 235, low enriched (oxide)

Large-scale facilities licensed by NRC for supporting the current LWR industry exist. Environmental effects have been described in WASH-1248 and NUREG-0002. Existing regulations should permit continued licensing.

Uranium-235, high enriched (oxide)

Small-scale facilities licensed by NRC currently exist. Environmental effects for large-scale operations are judged to be somewhat greater than those for low enriched uranium, but licensing should not involve significant difficulties. Fabrication involving dilution with thorium requires added precautions.

Thorium (oxide)

Thorium is fabricated on a small scale in conjunction with high enriched uranium-235, and is a licensed operation (see above). Attention must be given to possible radon emissions.

Recycled thorium (together with recycled uranium-233) has associated significant gamma activity and radon emissions. Most operations will need to be remote and highly automated. The limited experiences to date do not provide assurances that licensing difficulties will be readily overcome.

Uranium-233 (oxide)

Freshly separated uranium-233 has been fabricated with virgin thorium on a small scale. Licensing commercial facilities will require attention to details, but these are considered resolvable in the short term. Fabrication with uranium-238 should involve no unusual difficulties.

Recycled uranium-233 has associated gamma activity requiring shielded and remote operations, regardless of diluent (thorium or uranium-238). Environmental effects and licensability have not been defined.

Plutonium (oxide)

Small-scale facilities have been licensed to produce mixed uranium-plutonium oxides for experimental fuels and for FFTF. The environmental effects for LWR fuels have been described in NUREG-0002 and for FBR fuels in WASH-1535. Licensing actions on one large-scale facility (Westinghouse for LWR fuels) were initiated and subsequently suspended as a result of the Carter initiative of April 1977. Licensing problems (other than safeguards) are considered resolvable on a short-time scale.

The fabrication of denatured plutonium (containing sufficient fission products) will pose new safety problems requiring shielding and off-gas controls. No unusual problems are likely to persist, however.

3.4.4.6 Reprocessing

LWR Fuel

Three commercial facilities have been constructed under appropriate NRC licenses, one of which operated (Nuclear Fuel Services). The NFS plant, as constructed and operated, would not meet today's escalated regulations. The General Electric plant was not successfully commissioned, and licensing proceedings on the Barnwell Nuclear Fuel Plant have been suspended as a result of the Carter initiative. Licensing proceedings for the construction of a fourth plant (EXXON) were initiated and subsequently suspended as a result of the Carter initiative. Safety analysis and environmental reports were prepared by the applicants of the latter two facilities, but final actions have not been taken by NRC. Generic environmental and safety evaluations of both facilities were presented in NUREG-0002. The NRC staff believes that licensing questions for LWR fuel reprocessing can be answered in the relatively short term.

LMFBR Fuel

Experience with the reprocessing of LMFBR fuels has been limited to efforts on the pilot scale and on individual steps. The nature of the operations is comparable to that for LWR fuels. Because radioactivity levels and plutonium concentrations are higher, added attention must be given to confinement systems. Improved technology is probably needed for handling radioactive off-gases to meet license requirements. Environmental effects have been presented in WASH-1535.

Thorium-based Fuel

Much of the technology for the processing of thorium-based fuels has been developed by ORNL, General Atomic Company, and Allied Chemical Company under DOE sponsorship. The first commercial UO_2 - Th_2 fuel (Indian Point) was processed at the NFS plant, a licensed facility.

The nature of the processing will depend on the nature of the fuel (e.g., graphite blocks in the case of the HTGR) and its associated heavy metal constituents (e.g., plutonium, uranium-233, -235, and -238). Actual process steps used will determine off-gas compositions and concentrations, and hence the kinds and extent of confinement systems needed. It is not now possible to define with confidence the licensing problems.

3.4.4.7 Spent Fuel Storage

The NRC staff is preparing a generic environmental statement on spent fuel storage and, based on its work to date, there does not appear to be a reason to believe that the licensing of spent fuel storage facilities will require the development of drastically new technologies or procedures. The staff is preparing regulations governing the design and operation of such facilities. As a result, the staff anticipates being able to license Spent Fuel Storage facility Away From Reactors (SFS-AFR) in the time frame required to permit existing reactors to continue operation without shutdown because of lack of spent fuel storage capacity.

3.4.4.8 Waste Management

Both DOE and NRC have large programs underway in the waste management area. NRC's program is described in SECY 77-10, Reassessment of Nuclear

Waste Management Program. The dominant focus in both programs is the present LWR cycle. While other fuel cycles may vary some, it is not apparent that the waste management plans for the LWR cycle will not be adequate for other cycles. NRC plans to study the differences between the LWR cycle and the thorium fuel cycle.

DOE is considering disposal of spent fuel assemblies from LWRs in its overall waste management program and it appears that permanent or long-term disposal of spent fuel is just as feasible as the storage of high-level wastes from reprocessing.

3.4.4.9 D₂O Production

There are no commercial heavy water facilities in the United States at this time and it is not clear how the production of such material would be managed and controlled. A generic evaluation of the environmental effect of these facilities might be required if spectral shift or heavy water reactors were to be deployed and new legislation might be required for adequately bringing such an industry into the commercial sector. No major licensing problems based on safety are anticipated.