LICENSING OF HTGRs IN THE UNITED STATES

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ABSTRACT

The licensing history of the high-temperature gas-cooled reactor (HTGR) in the United States is given historical perspective. The experience began with the licensing of the Peach Bottom Atomic Power Station and extends to the continuing experience at the Fort St. Vrain Nuclear Generating Station. Additional experience was obtained from the licensing reviews in the mid-1970s of the large HTGR plants that were to be built by Philadelphia Electric Company and Delmarva Power and Light. Also, information was provided by the licensing review of the General Atomic standard plant by the U.S. Nuclear Regulatory Commission (NRC) at about the same time. These experiences are summarized in terms of the principal design criteria that were required by the regulatory authority for each project. These criteria include specification of the design basis accidents that were postulated for the plant safety analysis. Several technical issues raised by the NRC during their review of the large HTGR are presented.

The licensing requirements for the Fort St. Vrain plant have changed since the operating license was issued. These have arisen from new requirements for all reactors (e.g., fire protection, security, and Three Mile Island accident) and from operational experience. The effects of the Three Mile Island accident on the Fort St. Vrain licensing requirements have been minimal.

A look at the future of HTGR licensing in the United States suggests an increased use of quantitative safety requirements as well as the associated probabilistic assessment methodology. This should help to heighten awareness in the regulatory authority of the large safety margins inherent in gas-cooled reactor technology. General Atomic has used this methodology to evaluate the HTGR relative to the light water reactor (LWR) in meeting some of the criteria proposed by the NRC's Siting Policy Task Force in Report No. NUREG-0625. General Atomic is working with Gas-Cooled Reactor Associates, a utility organization, to carry out a pre-application review program with the NRC in which it is expected that a number of the generic safety issues can be resolved prior to the next application for a construction permit for an HTGR.

1. INTRODUCTION

This paper addresses the licensing or regulatory compliance of the high-temperature gas-cooled reactor (HTGR) in the United States. It is concerned with the public safety aspects of the HTGR and not with the environmental impacts of routine operation. The HTGR is characterized by a moderator and core structure that is largely graphite, ceramic fuel material, and use of helium gas as a primary coolant. The helium, flowing in a closed loop, transfers heat to boiling water in a steam generator that supplies steam to a power conversion cycle that is typical of modern, fossil-fired, steam power plant technology. Two power plants of this type have been built and operated in the United States: the Peach Bottom Atomic Power Station [40-MW(e)] in Pennsylvania and the Fort St. Vrain Nuclear Generating Station [330 MW(e)] in Colorado.

The licensing history of the HTGR in the United States began with the issuing of a construction permit to Philadelphia Electric Company for the Peach Bottom plant in 1962. This plant was constructed and first operated in 1967. The Peach Bottom plant was operational until 1974, when it was shut down for de-commissioning. Information obtained from the operation and post-operational examination of the plant contributes to the experience base to support licensing of future HTGRs. (See Ref. 1.)

The Fort St. Vrain Nuclear Generating Station (FSV) was authorized for construction in 1968. This plant has been in operation since 1974. The plant has yet to reach its design power output of 330 MW(e) because of technical licensing difficulties as described below. The operational history of FSV is described in Ref. 2. The FSV plant embodies many of the design features appropriate to large HTGR power plants, so its operational history will provide important information for future HTGR licensing activities.

Additional licensing experience was accumulated in the 1970s with the applications for construction by Philadelphia Electric Company and Delmarva Power and Light to build twin-unit power plants of 1100-MW(e) and 770-MW(e) unit capacity, respectively. These applications were carried through the issuance of safety evaluation reports by the staff of the U.S. Nuclear Regulatory Commission (NRC). These reports identified technical licensing issues to be resolved prior to operation of the plants. In 1974, General Atomic Company (GA) submitted a safety analysis report (GASSAR-6) for NRC review that provided a safety evaluation of the nuclear steam supply system of a standard design, 1160-MW(e) unit for generic approval by the NRC. The review was curtailed when GA ceased commercial HTGR activities in 1975. The NRC issued a draft safety evaluation report on this standard design in 1977. This report identified several additional technical licensing issues that would need to be resolved in any future HTGR licensing activities.

The remainder of this paper is devoted to discussing the evolution of principal licensing criteria applied to the HTGR, the unresolved technical licensing issues, and the prospects for new requirements and approaches in the future.

2. COMPARISON OF PRINCIPAL LICENSING CRITERIA

It is instructive to review the evolution of licensing requirements for HTGRs in the United States by examining the principal licensing criteria that were imposed at the times when the particular plant designs were submitted to the regulatory authority for review or were finally approved for construction and operation by that authority. In order to understand these licensing criteria, it is first necessary to summarize the principal design features of each plant that bear upon the criteria. Table 1 is a summary of these design features. The evolution in design features has been to locate the entire primary coolant system within a prestressed concrete reactor vessel (PCRV) embodying redundant structural members. This vessel, in turn, is surrounded by a rather conventional, concrete containment building designed for a low rate of leakage under the pressure produced by postulated accidents. The evolution in core cooling to assure safe shutdown has been from use of the main cooling loops to incorporation of dedicated decay heat removal cooling loops. The trend in control of release of fission products from the fuel elements is toward use of fuel particles that are coated with impervious layers of ceramic material to provide the primary fission product release barrier. Other special safety features of the design are treated in the discussion of principal licensing criteria.

Licensing criteria are those design and operational requirements that assure that the nuclear power plant will operate consistent with a minimum level of protection of the health and safety of the public. The principal criteria selected for presentation are those illustrating an evolutionary trend in U.S. licensing requirements for HTGRs. These are restricted to major design features and assumptions. There has also been an increase in requirements for quality assurance and in-service inspection and surveillance and in sophistication in engineering methodology. Design criteria for resisting severe natural phenomena, such as earthquakes, have become more demanding. These latter requirements are applicable to all nuclear power plants and are not unique requirements for the HTGR.

The principal licensing criteria used by the regulatory authority for HTGRs in the United States are presented in Table 2. The design basis accidents that are postulated require demonstration that 10CFR100* dose limits are not exceeded offsite. The large HTGR plants and the FSV plant also require the postulation of a single failure of safety-related equipment concurrent with the postulated accident. All of the plants require demonstration that the reactor can be safely shut down subsequent to the postulated accident.

*Code of Federal Regulations, Title 10, Part 100.

The trend of the radioactive source term for reactor siting has been to use more conservative releases as the plant size has increased. The Peach Bottom plant was licensed using a release of fission products from the primary coolant system that was based on a postulated sequence of events judged to be highly unlikely. The FSV plant was licensed assuming not only a total loss of forced circulation cooling, which is very unlikely, but also a release rate from the fuel that exceeds the rate at which experimental evidence indicates that the fission products can diffuse out of the fuel material. The siting source term for the large HTGR used an even more conservative model for the case of fuel particle coating failure with temperature and gave no credit for the time-delayed diffusion of fission products out of the fuel material. The result of the licensing criteria imposed by the regulatory authority on the large HTGR was to require an exclusion area boundary radius and containment building leak rate not very different than that required for an LWR of the same thermal power capacity.

The capability to provide decay heat removal subsequent to an interruption of helium circulation by the main loop helium circulators varies from plant to plant. The trend is to the employment of independent, diverse cooling loops for decay heat removal in order to reduce the probability of common mode failure of the core cooling function. The Peach Bottom plant, with flow upward through the core, had the capability of adequate core cooling by natural circulation of the helium through the steam generators. The FSV core does not have the capability for natural circulation cooling of the core. However, the PCRV liner cooling system is capable of limiting core temperatures so that, given a permanent loss of forced circulation, the offsite radiation doses are well within regulatory limits. Fort St. Vrain is the only nuclear power plant in the United States specifically designed and licensed to meet 10CFR100 guidelines with a postulated loss of convective core cooling. For the large HTGR, a low-leakage containment building with an internal, re-circulating filter system maintains offsite doses within regulatory limits in the event of a permanent loss of forced circulation cooling.

Moisture ingress into an HTGR is an accident unique to this type of reactor. The design features of all the U.S. plants provide moisture detection and isolation of the leaking loop as well as dumping of the water from the steam generator of that loop. This approach is conceptually unchanged from Peach Bottom through the large HTGR. The regulatory authority has consistently required that these actions be performed with automatic, safety-related equipment, although the calculated consequences of these accidents are small compared with other design basis accidents.

The design basis depressurization accident (DBDA) for the Peach Bottom unit was postulation of a rupture of a primary coolant pipe outside the reactor vessel similar to the loss-of-coolant accident for an LWR. With the enclosure of the primary coolant system in a PCRV at FSV, the DBDA became a depressurization through a penetration closure with the flow area limited by structurally independent flow restrictors. This assumption was also applied to the large HTGR, except that for the FSV reactor double closures were employed whereas single closures were specified for all of the large HTGR penetrations. This accident provides the basis for the containment building design pressure as well as the pressure forces acting on

TABLE 1
PRINCIPAL HTGR PLANT DESIGN FEATURES

PLANT	DESIGN FEATURES								
	FUEL ELEMENT CONFIGURATION	REACTOR PRESSURE VESSEL	CORE FLOW DIRECTION	HELIUM CIRCULATOR	STEAM GENERATOR CONFIGURATION	REACTOR BUILDING	PRIMARY CIRCUIT FISSION PRODUCT CONTROL	DEDICATED DECAY HEAT REMOVAL SYSTEM	REACTIVITY CONTROL
PEACH BOTTOM UNIT 1 [40 MW(e)]	CYLINDRICAL, FULL-LENGTH, CENTRAL FUEL COMPACTS	CYLINDRICAL, STEEL WITH ELLIPTICAL ENDS; MULTI-LAYER METAL INSULATION	UP	ELECTRIC DRIVE OIL-LUBRICATED BEARINGS; EX-VESSEL	U-TUBE AND DRUM; OUTSIDE REACTOR VESSEL	STEEL, PRESSURE-RESISTING CONTAINMENT, INERTED DURING OPERATION; RE-CIRCULATING FILTER SYSTEM	INDIVIDUAL FUEL ELEMENT PURGE; HELIUM PURIFICATION SYSTEM	NO; PONY MOTOR ON MAIN CIRCULATOR	SOLID ABSORBER ROD, PNEUMATIC ACCUMULATOR INSERTION: PLUS ELECTRIC DRIVE RODS; PLUS THERMALLY INITIATED, GRAVITY DROP RODS
FORT ST. VRAIN HTGR [330 MW (e)]	HEXAGONAL PRISMATIC BLOCKS 79 cm LONG, 36 cm ACROSS FLATS, CONTAINING BONDED FUEL PARTICLES IN RODS	PRESTRESSED CONCRETE, SINGLE-CAVITY STEEL LINER; FIBROUS INSULATION	DOWN	STEAM DRIVE WATER-LUBRICATED BEARINGS, IN-VESSEL	HELICAL COIL WITH REHEAT COIL; IN-VESSEL	CONFINEMENT BUILDING WITH FILTERS TO VENT	COATED FUEL PARTICLES; HELIUM PURIFICATION SYSTEM	NO; PELTON WHEEL DRIVE ON MAIN CIRCULATOR	SOLID ABSORBER RODS, GRAVITY INSERTION; PLUS ABSORBER PELLETS IN RESERVE SYSTEM
LARGE HTGR [PHILADELPHIA ELECTRIC 1160-MW (e) PLANT, DELMARVA 770-MW (e) PLANT, AND GENERAL ATOMIC 1160-MW (e) STANDARD PLANT]	HEXAGONAL PRISMATIC BLOCKS 79 cm LONG, 36 cm ACROSS FLATS, CONTAINING BONDED FUEL PARTICLES IN RODS	PRESTRESSED CONCRETE; MULTI-CAVITY STEEL LINER; FIBROUS INSULATION	DOWN	STEAM DRIVE; WATER-LUBRICATED BEARINGS; IN STEAM GENERATOR CAVITY OF VESSEL	HELICAL COIL WITH REHEAT COIL IN STEAM GENERATOR CAVITY OF VESSEL	CONCRETE, PRESSURE-RESISTING CONTAINMENT BUILDING; RE-CIRCULATING FILTER SYSTEM	COATED FUEL PARTICLES; HELIUM PURIFICATION SYSTEM	YES, 3 DEDICATED CORE AUXILIARY COOLING SYSTEM LOOPS	SOLID ABSORBER RODS, GRAVITY INSERTION; PLUS ABSORBER PELLETS IN RESERVE SYSTEM

TABLE 2
PRINCIPAL HTGR LICENSING CRITERIA

PLANT	LICENSING CRITERION OR DESIGN BASIS ACCIDENT				
	SITE SUITABILITY SOURCE TERM - 10CFR100	DESIGN BASIS DEPRESSURIZATION ACCIDENT	DESIGN BASIS MOISTURE INGRESS	DESIGN BASIS REACTIVITY ACCIDENT	LOSS OF MAIN LOOP NORMAL CIRCULATION
PEACH BOTTOM UNIT1	RELEASE TO CONTAINMENT FROM RUPTURED PRIMARY COOLANT LOOP. LOSS OF CORE COOLING.	FAILURE OF ONE PRIMARY COOLANT LOOP PIPE. COOLING ON OTHER LOOP.	10 LB/SEC STEAM GENERATOR LEAK WITH PRIMARY COOLANT LOOP FAILURE. CONTAINMENT IS NOT OVERPRESSURIZED.	SINGLE ROD WITHDRAWAL AT MAXIMUM RATE	COOLING BY: 1. PONY MOTOR DRIVE ON HELIUM CIRCULATORS. 2. NATURAL CIRCULATION OF HELIUM. 3. VESSEL COOLING COILS.
FORT ST. VRAIN HTGR	UNRESTRICTED CORE HEAT-UP NORMALIZED TO TID-14844 SOURCE TERM. DEPRESSURIZATION OF PRIMARY COOLANT THROUGH HELIUM PURIFICATION SYSTEM. CONTINUING LEAK AT LEAK RATE OF PCRV.	FAILURE OF DOUBLE PENETRATION CLOSURE. AREA LIMITED BY FLOW RESTRICTOR. COOLING ON MAIN LOOPS.	90 LB/SEC STEAM GENERATOR LEAK. MOISTURE MONITOR SYSTEM FAILURE. NO FLAMMABLE MIXTURES. ONE RELIEF TRAIN AVAILABLE.	ROD PAIR (SINGLE DRIVE) WITHDRAWAL AT MAXIMUM RATE	COOLING BY: 1. PELTON WHEEL DRIVE ON HELIUM CIRCULATOR. a. FEEDWATER b. EMERGENCY FEEDWATER c. FIRE WATER 2. PCRV LINER COOLING, NORMAL AND AUXILIARY COOLING METHOD.
LARGE HTGR (PHILADELPHIA ELECTRIC, DELMARVA, AND GENERAL ATOMIC STANDARD PLANT)	UNRESTRICTED CORE HEATUP NORMALIZED TO TID 14844 SOURCE TERM. BLOWDOWN OF PRIMARY COOLANT TO CONTAINMENT.	FAILURE OF SINGLE PENETRATION CLOSURE. AREA LIMITED BY FLOW RESTRICTOR. COOLING BY CORE AUXILIARY COOLING SYSTEM.	90 LB/SEC STEAM GENERATOR LEAK. MOISTURE MONITOR SYSTEM FAILURE. NO FLAMMABLE MIXTURES. ONE RELIEF TRAIN AVAILABLE.	ROD PAIR (SINGLE DRIVE) WITHDRAWAL AT MAXIMUM RATE	COOLING BY: 1. CORE AUXILIARY COOLING SYSTEM. 2. LINER COOLING, WHICH PROTECTS PCRV STRUCTURE

reactor vessel internal structures. The radiological consequences of this accident for the FSV plant and the large HTGR were estimated to be small compared with those calculated using the site suitability source term.

Postulated control rod withdrawal accidents for HTGRs are similar to those for LWRs. For the FSV reactor and the large HTGRs, where the control rods are inserted by gravity, the maximum credible rate of reactivity insertion is determined by the maximum speed of a single drive mechanism. For the Peach Bottom reactor, where the control rods were driven in from the bottom, the drop of a control rod out of the core was made incredible by the design of the mechanism. For all of the plants, the maximum rate of reactivity insertion due to water ingress is always less than that calculated for the rod withdrawal accident.

In summary, from the licensing of the Peach Bottom plant through the licensing review of the large HTGR, no new generic accident was required to be postulated. However, treatment of some of the details of these accidents has evolved as discussed above.

3. REGULATORY REQUIREMENTS DURING FORT ST. VRAIN OPERATION

Subsequent to the safety evaluation of the FSV plant that formed the basis for its operating license, a number of new requirements have been imposed as a result of technical problems in the plant, the fire at the Brown's Ferry plant, the accident at Three Mile Island (TMI), and some new, general regulatory requirements.

Pelton wheel drives on the helium circulators are used to provide motive power for helium circulation when an adequate steam supply is not available. Early in the plant operation, cracks were found in these wheels and in a shaft coupling. As a result, the wheel material was changed and the allowable shaft speed was reduced for use of the water drive.

Excessive heating of the top head of the PCRV was observed in the vicinity of the control rod drive/refueling penetrations. This was due to an unexpectedly high rate of local helium convection in the penetrations. The control rod drive assemblies were removed and modified to baffle this convective flow and reduce the excessive heating.

Temperature fluctuations with time at the core outlet were observed. This resulted in regulatory restraints on power level and the institution of a diagnostic study to determine the cause of and remedy for these fluctuations. The study has resulted in the hypothesis that the fluctuations are caused by variable bypass flow in the space between the fuel elements. It is thought that the flow varies because of fluid-pressure-induced, radial motion of the fuel elements that, coupled with thermally induced motion, causes a periodic change in the space between fuel elements. A design to remedy this situation resulted in the installation of radial restraint devices at the top of the core during the first refueling. At this writing, tests of the effectiveness of these devices have yet to be performed. The NRC has approved the plans for these tests up to 70% of reactor design power.

A commitment has been made by the owner (Public Service Company of Colorado) to upgrade the helium circulator service system to improve its reliability.

Detection of some errors in the plant accident analysis has resulted in the NRC restricting plant operation to 70% of the design capacity pending some plant modifications and NRC approval of a revised analysis. The plant modifications are the addition of booster pumps to the fire water system to increase helium circulation when using fire water to drive the Pelton wheels and changes that reduce to 2 hr the time by which the PCRV must be depressurized given a postulated, permanent loss of forced circulation cooling. At this writing, the NRC has approved the plant modifications, but they have yet to approve the plant operation at power levels above 70% of design capacity.

Some misrouted cables were found about the time of the Brown's Ferry plant fire. Because of these occurrences, the plant was upgraded by correcting the cables, improving the fire protection system, and adding a new plant system, the Auxiliary Cooling Method (ACM). The ACM provides an independent means of providing cooling water to the PCRV liner cooling system to limit offsite radiation doses given a permanent loss of forced circulation cooling. This system is effective even if a fire destroys the cables in the main cable-spreading area of the plant because all of its essential components are remote from this area.

The plant security requirements have been made more stringent since first operation, resulting in some architectural changes and increased staffing.

The NRC has required some plant protection system setpoints to be reevaluated to better account for instrument calibration error and drift. This has been required of a number of the other nuclear plants in the United States.

The impact of the requirements of the Short Term Lessons Learned from the Three Mile Island accident has been minimal for the FSV plant compared with that for some of the operating LWRs. Modifications to the plant have included the addition of some shielding around filters in the reactor building to protect the operators and some upgrading of radiation-monitoring capability. In addition, some administrative and procedural changes are required in the emergency preparedness system to provide independent services for each pair of circulators that operate in each of the two helium loops. This is in response to an incident that occurred in 1978 in which a small amount of radioactive helium was released to the environment through the circulator service system. The committed change will allow isolation of one reactor cooling loop in response to faults while allowing for cooling on the other loop.

Concern about oxidation of the graphite core support structure due to moisture in the helium has led to the installation of removable surveillance specimens in the core support structure. These will be removed and analyzed according to a schedule when refueling is performed.

4. UNRESOLVED LICENSING ISSUES

A number of outstanding issues would have to be resolved with the NRC in any new application to construct an HTGR power plant. These issues are: (1) the issues from the Philadelphia Electric and Delmarva reviews that were left for resolution until after issue of the construction permit; (2) additional issues identified by the NRC in their Interim Safety Evaluation Report on the General Atomic standard plant (GASSAR-6); and (3) problems identified in the operation of the FSV plant. In addition, there are potential licensing issues for the HTGR as a consequence of the lessons learned from the TMI accident. The impact of that accident is discussed in Section 5.

The unresolved issues identified in the safety evaluations of the proposed large HTGR plants are listed in Table 3. These issues are discussed below.

The issue of design criteria for graphite structures relates to the stress levels used for design of reactor vessel internals and fuel element structural components for the postulated plant conditions. A consultant to the NRC (Franklin Institute Research Laboratories) has recommended the use of more conservative criteria than GA has used in the past. Studies are being done by GA that include experiments to provide a technical basis to resolve this issue.

The core seismic response issue concerns the verification of the methodology employed to predict the mechanical response of the HTGR core and the core supports to earthquakes. Work is being performed at GA to develop and verify the computer codes used in this design analysis. This work is both analytical and experimental.

The in-service inspection and testing of the pressure-retaining components of gas-cooled reactors will be specified in Section XI, Division 2, of the ASME Boiler and Pressure Vessel Code, which is under development. An NRC staff member participates on the subcommittee that is carrying out this work. At the time of the large HTGR reviews by the NRC, the utilities committed to future compliance with the code after it was developed. The NRC reserved acceptance of the commitment pending development of the final code.

The pre-operational vibration testing of reactor vessel internals was not well-defined at the time of the safety evaluations of the large HTGR. The comparable requirements for LWRs are specified in Regulatory Guide 1.22. General Atomic has done considerable work since that time to specify an appropriate test program for the large HTGR. This work has not been reviewed by the NRC.

The issue of anticipated transients without scram (ATWS) has yet to be resolved in the United States for LWRs, although it appears to be close to resolution at this writing. The NRC has not defined the criteria that would be applied to gas-cooled reactors. However, it is thought that the criteria would be similar to those for LWRs. A preliminary analysis of the response

TABLE 3
TECHNICAL REGULATORY ISSUES FROM PREVIOUS REVIEWS

PHILADELPHIA ELECTRIC AND DELMARVA LARGE HTGRs

1. DESIGN CRITERIA FOR GRAPHITE STRUCTURES
2. CORE SEISMIC RESPONSE
3. IN-SERVICE INSPECTION AND TESTING
4. PRE-OPERATIONAL VIBRATION TESTING OF REACTOR INTERNALS
5. ANTICIPATED TRANSIENTS WITHOUT SCRAM
6. CONFIRMATION OF THE CONTAINMENT DESIGN BASIS
7. LONG-TERM BEHAVIOR OF METALLIC COMPONENTS OF THE PRIMARY COOLANT SYSTEM

GENERAL ATOMIC STANDARD PLANT (GASSAR-6)

8. THERMAL-HYDRAULIC PHENOMENA DURING SAFE SHUTDOWN COOLING
9. LOW PROBABILITY ACCIDENT DEFINITION

to anticipated transients of a large HTGR with failure of control rod motion was prepared and submitted for NRC review as part of the Delmarva application. The report provided some evidence that these postulated events could be accommodated by the HTGR design within the LWR criteria of that time. The subsequent change to low-enriched fuel with its increased negative temperature coefficient of reactivity should further improve the HTGR response.

Confirmation of the containment design basis is a requirement in establishing the HTGR plant response to depressurization accidents. The time-dependent containment pressure response depends upon the mixing of helium and air in the containment. In addition, the local temperature response of the containment depends upon the behavior of jets of helium issuing from the rupture of the primary coolant system. Subsequent to the NRC review, both GA and Los Alamos Scientific Laboratory have done work to develop computer codes to better predict these phenomena.

The long-term behavior of metallic components in the primary coolant system remains an issue because of the lack of experience with these materials in the HTGR coolant environment for a 40-year plant life. Laboratory testing at GA continues to produce data to resolve this issue.

In the transition from cooling on the main loops to cooling on the core auxiliary cooling system, there is an issue of local high temperatures of reactor internals due to the transition to laminar flow when pressurized and also to the formation of hot, rising jets of helium above the core when pressurized. Better computer modeling of these phenomena is being developed at GA, and Oak Ridge National Laboratory is performing experimental studies of the formation and dispersion of hot jets.

In the review of the GA standard plant, the NRC raised the issue of the need for consideration of combinations of low probability accidents, such as a depressurization accident combined with steam generator failure. General Atomic has subsequently performed the Accident Initiation and Progression Analysis (AIPA) study (Ref. 3), which estimates the frequency of occurrence of accident sequences. It is expected that these results in conjunction with increased acceptance of this methodology by the NRC can be used to show that combinations of low probability failures are sufficiently low that they need not be considered to be design basis accidents for the plant.

The core fluctuation problem at FSV is expected to be an issue in any future HTGR licensing activity. General Atomic is continuing to study this problem including the use of experiments. Success of the core radial restraint devices at FSV would do much to alleviate concern with this issue.

Oxidation of graphite reactor vessel internals at FSV is expected to create an issue in the future for licensing large HTGRs. General Atomic is studying the use of graphites having a higher resistance to oxidation as well as the use of circulator service systems having greatly enhanced reliability against water ingress to the primary coolant system compared with that at FSV.

5. IMPACT OF THREE MILE ISLAND ACCIDENT ON FSV

Last year all operating nuclear plants in the U.S. were requested to respond to 32 items for compliance that were derived from ideas generated by the TMI accident. Of these, six were not applicable to FSV because it is not an LWR. Nine of the items were judged to be already in compliance with no change required. The remaining 17 required some plant, operational, or administrative change.

The plant design changes include additional shielding to protect operators from radioactive material in the reactor building, safety classification of emergency feedwater flowmeters, relocating the radiochemistry laboratory to protect its occupants, increasing the range of some radiation monitors, and providing an onsite technical support center to be used in the event of an accident. Emergency planning and plant staffing and procedures were improved, including the use of shift technical advisors in the control room. The FSV plant has received NRC approval for the shift technical advisor to be on 1-hr call rather than onsite as required for all of the LWRs. This is in recognition of the HTGR's inherent, slow response to disturbances. Additional areas still being negotiated between Public Service of Colorado and the NRC include the location of an emergency operations center, the distance range of environmental monitoring, and evacuation planning. The NRC has approved an evacuation radius of 5 miles for the FSV plant versus 10 miles for large LWRs and an iodine ingestion pathway planning radius of 30 miles versus 50 miles for large LWRs.

6. FUTURE CONSIDERATIONS

It is expected that the licensing process in the United States will include increasing reliance on the methodology of probabilistic risk assessment (PRA). This trend was established well before the TMI accident. However, the occurrence of the accident has caused unusually deep scrutiny of the U.S. regulatory process, which has resulted in recommendations for more use of PRA to account for multiple failures as well as to define "how safe is safe enough." There is little doubt that use of these techniques will make the licensing process more rational. In order to make PRA most useful, it is necessary that quantitative safety goals be established. A number of groups in the U.S., including the NRC, are working on this task.

General Atomic is developing general design criteria and positions on NRC Regulatory Guides for the HTGR and hopes to get the NRC to review these in the near future. These positions are derived to take into account the inherent safety features of the HTGR. General Atomic is working with Gas-Cooled Reactor Associates, a utility organization, to initiate a review program with NRC. For these features to be recognized as a way to provide the maximum benefits, however, requires that the NRC eventually recognize established siting criteria for the HTGR that are commensurate with its relative safety margins compared with LWRs. The results of the AIPA study, when compared with the results of the Reactor Safety Study, provide some measure of the relative safety margins.

In August 1979, the NRC published the "Report of the Siting Policy Task Force" (NUREG-0625) with recommendations for reform of U.S. reactor siting policy for LWRs. If adopted as recommended, reactor site distance parameters would be the same for all reactor plants regardless of their inherent safety characteristics or engineered safety features: a minimum distance to the exclusion area boundary; a minimum emergency planning distance, maximum population density, and distribution criteria; and minimum stand-off distances for external hazards.

The report states that the "siting principles" in the study are not "directly applicable" to the gas-cooled reactor, and therefore it may be possible to develop a less restrictive policy for the HTGR. The principles in the report if applied to the HTGR are viewed by GA to be overly restrictive in view of the inherent safety characteristics of the HTGR. General Atomic plans to comment to the NRC staff as the new siting policy develops and to recommend that the policy be specific to reactor type and power capacity. To establish a technical basis for this activity, analyses were performed to compare the acute and latent effects of a core melt sequence for a 1000-MW(e) PWR, as characterized by the release scenarios of the Reactor Safety Study (Ref. 5), with those of core heatup sequences of a comparably rated HTGR (Ref. 3). The analyses assumed that the site parameters for distances and population densities described in NUREG-0625 were employed for both types of reactor plant.

For additional comparison purposes, analyses were performed for both reactor types assuming reference U.S. site parameters typical of existing and planned U.S. reactor sites through the year 2000 rather than the NUREG assumptions. While the PWR analysis assumed evacuation out to 25 miles, the HTGR analysis conservatively assumed evacuation out to only 1.6 miles. Nevertheless, the results indicate that (1) the event for the HTGR would not result in any acute fatalities regardless of the site parameters; (2) the NUREG parameters would reduce the acute fatalities for the PWR case by about one order of magnitude at an event frequency of 10^{-7}/reactor-year; (3) the NUREG parameters would produce no detectable differences in latent fatalities for the PWR relative to the U.S. reference site; and (4) even with the reduced extent of evacuation assumed for the HTGR, the number of latent fatalities predicted for the HTGR using the NUREG parameters is similar to those calculated for the representative U.S. site. It is concluded that the NUREG recommendations are not effective in reducing the risk of latent effects for either type of reactor plant and that the NUREG parameters appear to be excessively conservative for the HTGR.

7. SUMMARY AND CONCLUSIONS

The history of licensing of the HTGR in the United States has been one of changing requirements due to changing design concepts, changing plant size, and changing level of detail of the review by the regulatory authority. A number of outstanding licensing issues have been identified which must be resolved with the NRC in future HTGR applications. These issues have been under study by GA and others since they were identified, and the prognosis for their future resolution is believed to be very good.

The TMI accident has badly shaken the U.S. regulatory process. The effects on the HTGR concept appear to be minimal, judging by the FSV experience and by study of the final report on lessons learned from TMI (Ref. 4).

General Atomic expects an increasing use of PRA in the future regulatory process. This trend should result in heightened awareness of the large safety margins inherent in the HTGR compared with other reactor types.

The outlook for future licensability of the HTGR in the U.S. is very good.

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