# Task CH6: Graphite-Moderated Reactors

The Fort St. Vrain HTGR and DOE's N-reactor at the Hanford Reservation in Washington State are the only

graphite-moderated power reactors operating in the U.S. This task, outlined in Chapter 6 of NUREG-1251,[1] called for the staff to assess the HTGR concept (with emphasis on Fort St. Vrain) against the issues raised by the Chernobyl accident: operations, design, containment, emergency planning, and severe accident phenomena. Because the N-reactor is not licensed by the NRC and is under the authority of DOE, the implications of the Chernobyl accident for the N-reactor are to be assessed separately by DOE and others.

## ITEM CH6.1: GRAPHITE-MODERATED REACTORS

This item consists of two recommendations that are evaluated separately below. **ITEM CH6.1A: THE FORT ST. VRAIN REACTOR AND THE MODULAR HTGR DESCRIPTION**

At Fort St. Vrain, a helium coolant is used that is pressurized to 700 psi and flows downward through 1/2-inch diameter holes in a fully ceramic (graphite) core. The reactor core and the entire primary coolant system,

including steam generators and helium circulators, are enclosed in a pre-stressed concrete reactor vessel that, through use of inner and outer penetration seals and in conjunction with a filtered and vented confinement building, satisfies the NRC's general design criteria for reactor containment.

The MHTGR concept will use a fuel and reactor design that is derived from the Fort St. Vrain reactor. However, the reactor will be contained in a steel pressure vessel and the helium circulator and steam generator in a connected second steel vessel rather than full enclosure of the primary system in a single pre-stressed concrete reactor vessel. Its safety approach is based on an inherent negative power coefficient and selection of the reactor power density and vessel size such that decay heat can be removed passively from the exterior wall of the vessel during postulated accidents. Decay heat would be removed by natural convection airflows that are adequate to preclude fission product release from the fuel or unacceptable damage to the reactor vessel or to other vital reactor systems. The reference MHTGR plant would consist of four such modules and would produce a total of 550 MWe.

This item called for the staff to coordinate licensee preparation of a PRA for St. Vrain as part of the implementation of the Severe Accident Policy; the Chernobyl lessons were to be factored in the PRA.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue considered is to be a licensing issue.

## CONCLUSION

The only features that the 330 MWe Fort St. Vrain reactor, the MHTGR, and the Chernobyl design have in common are the use of a graphite moderator and gravity-driven control rods. A limited Fort St. Vrain PRA and further experiments with structural graphite were considered before the Chernobyl accident. While the Chernobyl events supported the need for such work, the imminent termination of the operation of Fort St. Vrain removed that need. The issues raised by the Chernobyl accident have not caused any new concerns about HTGR severe accident phenomena. Thus, this licensing issue was dropped from further consideration.

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[1] NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vols. I and II) April 1989.

## ITEM CH6.1B: STRUCTURAL GRAPHITE EXPERIMENTS DESCRIPTION

There is a need to determine the impact of cracking of a graphite fuel block at Fort St. Vrain on confidence in the long-term reliability of graphite as a structural material in an HTGR reactor core. In an extreme scenario, graphite structural failure could conceivably allow the core to drop away from the control rods, causing a reactivity accident. This issue called for the staff to complete an earlier study on the combined effects of thermal and mechanical loads on structural graphite. This study would provide an improved understanding of graphite behavior.

The staff will examine PGX graphite specimens for the interaction of thermal and mechanical stresses in the same configuration used in prior H440 graphite experiments (i.e., smooth rings, uniform internal heating with diametrically opposed loads) and perform tests to include a notch in the PGX graphite. This will permit examination of the sensitivity of the behavior of PGX structural components to combined thermal and mechanical stresses when a stress riser is present.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue considered is to be a licensing issue.

## CONCLUSION

Since Fort St. Vrain (FSV) indicated it intends to terminate operation in 1990, additional experiments on FSV graphite structural integrity will not be done. Graphite structural integrity for the MHTGR will need to be

established by the applicant to support licensing this design. The staff will review the proposed MHTGR graphite structural criteria and its supporting basis as part of any application submittal. Therefore, no additional work is planned at this time on this issue.

## ITEM CH6.2: ASSESSMENT DESCRIPTION

Administrative control and operational practices at Fort St. Vrain, although generally similar to those of LWRs, originally contained some differences believed to reflect the unique features of the HTGR concept. In recent years, however, changes have been made to bring plant operations much closer to those of LWRs. A program to upgrade the TS is currently underway that will result in administrative controls that are comparable to those of LWRs. The Fort St. Vrain reactor also must meet the same or equivalent requirements as those for LWRs with respect to quality assurance, equipment qualification, external events, physical security, fire protection, radiation protection, and operator training and qualification.

Two important differences between HTGRs and LWRs with respect to operational safety are the slower response of HTGRs to plant transients, because of low power density, and their increased margin to fuel failure, because of the fully ceramic core. These differences formed the basis for permitting less prescription in some administrative procedures and are considered to enhance overall safety. MHTGR designers are proposing

a design that uses inherent and passive safety features and fully automated plant control systems that will minimize the need for operator action to ensure safety, thus reducing the importance of the man-machine interface to reactor safety. This issue called for the staff to review this approach and include its findings in an SER on the MHTGR.

In pursuing this issue, the staff is expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety. Therefore, the issue considered is to be a licensing issue.

## CONCLUSION

The staff assessed the areas of operations, design, containment, emergency planning, and severe accident phenomena and found that the implications of the Chernobyl accident have generated no new licensing concerns for HTGRs; general conclusions and those pertaining to specific areas are the same as those

for LWRs. In performing its assessment, the staff reviewed the existing information related to these areas and concluded that programs underway or being considered adequately satisfy any concerns that could be generated because of the Chernobyl accident. Thus, this licensing issue has been resolved.