GLP-5588

SAFETY ANALYSIS REPORT
USE OF H-451 GRAPHITE IN
FORT ST. VRAIN FUEL ELEMENTS

GENERAL ATOMIC COMPANY
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TOPICAL REPORT EVALUATION

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Reviewed by: Reactor Fuels Section, Core Performance Branch, Division of Systems Safety, March 1979.

Summary of Topical Report

This topical report was submitted by General Atomic Company (GA) via letter (G. L. Wessman (GA) to R. P. Denise (NRC), March 28, 1978). In that letter we were requested to review the report and concur with GA's opinion that the substitution of type H-451 graphite in the Fort St. Vrain (FSV) reactor core for the type (H-327) currently in use is acceptable. The reasons for desiring to substitute H-451 for H-327 graphite in FSV were stated to be as follows:

1. General Atomic is contractually obligated to supply Public Service Company of Colorado (PSC) with fuel elements.

2. It appeared to GA that H-327 graphite, the type currently used, would not be available in the future.

3. General Atomic believes that H-451 graphite is superior to H-327, both in terms of dimensional stability under irradiation, as well as strength. Thus, NRC's approval of H-451 graphite for use in the FSV reactor was solicited so that the reference core H-327 graphite fuel and reflector elements could be replaced with H-451 graphite elements during future reloads; the first potential substitution could begin with reload "segment 9," currently scheduled for early 1981.

The topical report was structured into three major sections: (1) a performance analysis of the H-451 elements, encompassing the nuclear, thermal, and structural design, (2) a safety analysis of the effect of the H-451 elements on accidents that had been considered in the FSV FSAR, and (3) a summary of material property data for H-451 graphite. In general, the approach used was to compare the analytical results based on H-327 graphite with the results obtained with the assumptions of H-451 graphite as the core material. The objective was to demonstrate that H-451 was a better structural material than H-327 and that the use of H-451 would (a) result in negligible changes in the nuclear and thermal behavior of the core, and (b) not result in reduced safety margins or reliability compared to the reference H-327 core.

The core nuclear performance analysis addressed fuel loading and excess reactivity, power distribution, fuel burnup and exposure, shutdown margins and reactor control, and rod withdrawal accidents. The rod withdrawal accident was given special attention because it is identified in the FSV FSAR as the worst-case reactivity initiated accident; control rod ejection accidents are not considered

credible for the FSV reactor and, therefore, the consequences of a rod ejection accident are not analyzed.

The core thermal analysis addressed the fact that the thermal conductivity of H-451 graphite is approximately a factor of 1.2 greater than that of H-327 graphite. The better thermal conductivity was predicted to result in a smaller temperature rise across the graphite web and in lower fuel centerline temperatures. The expected reduction in fuel temperatures was expected also to result in negligible changes in metallic fission product release and a reduction in gaseous fission product release.

The graphite structural analysis report section addressed (a) the mechanical properties of H-327 and H-451 graphites in a comparative way, (b) stress analyses for both types of graphite, and (c) graphite irradiation-induced dimensional changes. The H-451 graphite has higher strength than H-327, but the H-451 fuel element would have higher stress levels. Design stress margins were, however, asserted to be improved by the inclusion of H-451 graphite elements in the FSV core because the improvements in strength were more than required to meet the increase in stress.

The topical report's "safety analysis" section was used to examine events and accidents previously analyzed in Chapter XIV of the FSV FSAR to determine if the substitution of H-451 graphite in FSV reload cores could alter the consequences of postulated accidents. Events given a re-examination included fuel element "malfunctions," loss of normal shutdown cooling, moisture inleakage,

permanent loss of forced circulation, rapid depressurization, and the rod withdrawal event mentioned above. It was concluded in the report that existing FSAR results of accident analyses conservatively bound any perturbations resulting from the introduction of H-451 graphite fuel elements.

Summary of Regulatory Evaluation

During the course of our review of GLP-5588, requests for additional information were responded to by GA in writing (Refs. 2 to 5). Our inquiries and GA's responses will be incorporated into GLP-5588 via amendment.

Our review focussed primarily on six aspects of H-451 graphite performance relative to H-327 graphite performance in the Fort St. Vrain core: (1) material reproducibility; (2) corrosion (oxidation); (3) mechanical and physical properties; (4) irradiation-induced dimensional changes; (5) fission product release and transport; and (6) post-irradiation examination and surveillance requirements. These subjects will be discussed in order below.

Graphite Reproducibility - This has been a continuing concern in HTGR fuel system reviews (Ref. 6) because (a) the properties and performance of graphite materials are strongly dependent upon the nature of the precursor materials and methods used in manufacture, and (b) the graphite used in FSV is not manufactured by General Atomic. Graphite grades H-327 and H-451 are manufactured by the Great Lakes Carbon Corporation; only the machining and fuel loading of

the as-fabricated graphite blocks are performed by General Atomic. Accordingly, we asked GA to indicate how adequate assurance would be provided that (a) the H-451 graphite to be used in future FSV reloads is the same as that subjected to the qualification test program, and (b) if different precursors or fabrication parameters were used for future reload material, the same (or predictable changes in) properties and thermal/irradiation changes would be obtained as with the test-qualified material.

The major thrust of the response to this question (see response to Q231.1a) was that, for FSV reload segments, the H-451 logs will be manufactured from near-isotropic coke taken from the same batch as that used for the manufacture of a preproduction lot that is used as a standard for production of the FSV reload elements and that is to be used for future H-451 production. This same preproduction lot has also been well characterized for properties and irradiation behavior, and the data obtained are used in the design data package for the H-451 fuel elements. The H-451 purchase specification further assures reproducibility of coke structure by requiring the fabrication and testing of the coefficient of thermal expansion (CTE) of calculated coke samples. The CTE test method is specified (Ref. 7). Thus, several steps are taken to ensure that the same coke will be used in the future FSV reload H-451 graphite blocks as was used in the H-451 qualification test program. In addition, the H-451 purchase specification requires that the binder should be a coal tar pitch of the same type used to manufacture prototype, preproduction or production logs of H-451 graphite and that the impregnant shall be a type certified to the purchaser.

We, therefore, conclude that adequate assurance has been provided concerning the reproducibility of the H-451 graphite for FSV, so long as the precursor materials and processing remain unchanged. Should the precursor materials or fabrication process be changed, however (for example, should the current near-isotropic coke supply be exhausted, thereby necessitating the production of a new batch with potentially different qualities), we will require GA to provide evidence to support the performance predictions for the graphite elements produced from the altered materials or process parameters.

A secondary concern regarding the effect of fabrication process on material performance involved the method used to fabricate the fuel rods that are contained in vertical holes drilled into the hexagonal graphite fuel blocks. Two processes have been developed: (1) an out-of-block forming and curing process; and (2) a cure-in-place process, sometimes called "in-block carbonization". The cure-in-place process can result in a stronger bond between the coated fuel particles and the fuel rod matrix; this has been observed to cause tears in the outer pyrocarbon coatings due to differential irradiation-induced shrinkage. On the other hand, the cure-in-place process improves heat transfer between the fuel rod and graphite block because the gap is closed during the curing process. In response to our question (Q231.1b), GA indicated that the cure-in-place process was not being considered at this time for the fuel rods to be used in the H-451 fuel elements, and that if the cure-in-place process were to be adopted in the future, a separate safety analysis and report would be submitted.

Graphite Corrosion - This subject is important from the standpoint of heat generation, production of flammable gas mixtures, potential fission product release, burnable poison oxidation, fuel particle hydrolysis, and effects on graphite physical and mechanical properties (and potential effect, therefore, on core integrity). Our review focused, in particular, on the data base for H-451 graphite, the effect of catalysts on H-451 graphite oxidation rate, and a general comparison of the effects of H-451 versus H-327 graphite oxidation on core performance. One review question (Q231.10), for example, addressed the effect of graphite type on the reaction rate as a function of percent graphite burnoff; this question also addressed the overall effect of H-451 graphite oxidation rate on the predicted accident consequences discussed in the report safety analyses. Another question (Q231.11a) addressed the relationship between graphite reaction rate, fuel hydrolysis rate, and associated release of noble gas fission products.

The responses to the above questions are contained in reference 2. Experimental data on a variety of graphites were cited as evidence that the chemical reactivities of H-451 and H-327 graphites are similar. Parametric calculations were performed to determine the sensitivity of the oxidation effects (hydrolysis rates, for example) to the coolant conditions (such as moisture levels). The results of these calculations are documented in reference 4. Results of the parametric calculations showed that a factor of ten increase in reaction rate decreased the noble gas release due to hydrolysis by 15%, while a threefold decrease in reaction rate increased the noble gas release by about

20%. These are relatively small effects when compared to the assumed order-of-magnitude change in reaction rate. The total amount of graphite oxidized throughout a simulated reactor steam ingress accident, calculated using the OXIDE-3 computer code (Ref. 9), varied by about a factor of two for a tenfold change in reaction rate (our OXIDE-3 review conclusions are reported in reference 10). We conclude, therefore, that the substitution of H-451 graphite for H-327 graphite in the FSV core will not significantly alter the oxidation characteristics and consequences as predicted in the FSV FSAR.

Mechanical and Physical Properties and Stress-Strain Relationships - Section 4 of the topical report listed the materials property data for both the H-451 and H-327 graphites. Our review concentrated on the H-451 graphite data base, the comparison of H-451 graphite properties with those for H-327 graphite, and the on-going and planned experimental program. Those mechanical properties that are identified in GLP-5588 as the ones that determine the element stresses, stress-strength margins, and element deformations are (1) modulus of elasticity, (2) tensile strength, (3) creep properties, (4) irradiation-induced dimensional changes, (5) thermal expansion, and (6) thermal conductivity. As part of our review, we asked GA (Q231.6) to illustrate the relative importance of these properties to the calculation of the above parameters. Dimensional changes are discussed separately below.

GA's response was in terms of the effect of material property changes at the time of maximum stress, for both operating and shutdown conditions, and at the time of maximum dimensional change. This approach was followed because of the

complexity of the effect of material property variations on stress, design margins, etc., i.e. the effect of any specific variation is dependent upon the interaction with various other properties which vary with time. Thus, the maximum operating stress was reported to occur at zero days and was due to startup thermal stresses. At that time there are no creep effects, and any change in creep parameters, therefore, has no effect on stress and stress margins. Although, as indicated in reference 3, at end of life in the element, a +25% variation in the steady-state creep coefficient can result in a +38% increase in oper~~ating~~ating stress, the operating stress at this time is reported to be sufficiently low that even this 38% increase produces a stress of only one quarter the stress during initial startup. We, therefore, conclude, based on a calculation of the maximum stresses during normal operation and using the available H-451 property data, that there is reasonable assurance that the H-451 graphite element design will accommodate the maximum stress with adequate margin.

As indicated earlier, the off-normal case that produces the maximum graphite stress (neglecting, for the moment, seismic considerations) is the maximum-worth rod withdrawal event. This event produces the greatest temperature differences across the web between fuel and coolant holes and, therefore, results in the greatest thermally induced stress. However, as indicated in the response to question Q231.8, the H-451 fuel elements retain considerable margin (~ 1550 psi and ~ 1150 psi in the axial and radial directions, respectively) for the worst case rod withdrawal. These margins are greater than those calculated for the H-327 graphite elements (~ 100 and ~ 700 psi, respectively).

The above-discussed calculations of operating and shutdown stress and strain distributions were performed using the computer programs FESIC and SAFE/GRAPHIT, which were used in the design and analysis of the FSV initial core. As noted in our evaluation of the FULTON plant (Ref. 11), we had not yet reviewed FESIC or SAFE/GRAPHIT, and since these codes have been superceded by more recent codes, we still have not reviewed them. FESIC and SAFE/GRAPHIT have been cross-checked, however, by hand calculations and comparison with these newer codes. Therefore, we find that the use of FESIC and SAFE/GRAPHIT are acceptable in this application.

Regarding the acquisition and current availability of H-451 graphite mechanical and physical property data, a considerable body of such data exists, as shown in section 4 of GLP-558. This information was generated for the most part under DOE-funded programs. The details of the test procedures and results are provided in the quarterly progress reports on the "HTGR Fuels and Core Development Program." These quarterly reports are supplemented by extensive summary reports. H-451 graphite property measurements are continuing in certain areas (notably, creep), but the data acquisition schedule has been stretched out because of the redirection of the HTGR commercialization effort in recent years. The creep tests on H-451 graphite will not be completed until late-1982, whereas H-451 graphite reload segments could be placed in FSV by early 1981. The creep tests will lead the FSV elements in irradiation exposure, however, so that full-exposure creep data will be available over the range of temperatures expected in FSV (see GA response to Q231.21 (Ref. 4)).

We will require General Atomic to provide timely reports on the results and future test projections for the H-451 graphite test program, which includes not only creep tests but other mechanical and irradiation tests as well.

With regard to seismic stresses and analysis and their relationship to H-451 mechanical properties, GA indicated in response to our review questions that H-451 fuel elements would be expected to see increased seismic loading compared with H-327 because of H-451's higher elastic modulus in the radial direction (see modulus data in report tables 4-3 and 4-9). Using simple spring/mass relations, H-451 fuel element impact forces were expected to be 30% greater than that of H-327 elements. Because H-451 fuel elements have more than 65% higher radial strength, however, the seismic structural damage potential for H-451 elements is less than that for H-327 elements. The method of analysis used to determine the seismic stresses in the graphite fuel and reflector elements was essentially the same as that used for the FSV FSAR. Testing of full scale structures under simulated seismic loadings have also been conducted to substantiate the strength of the fuel elements. We conclude that adequate assurance has been provided that the H-451 graphite elements seismic response is at least equal to or better than that of the reference H-327 elements.

The fatigue behavior of H-451 graphite is addressed in the topical report in terms of the homologous stress, which is defined as maximum applied fatigue stress divided by tensile strength. In response to staff question Q130.1, GA asserted that both primary and secondary (thermal) loadings are below the

homologous stress limits, and that no fatigue analysis was necessary, therefore, for the H-451 elements. The response to a subsequent question (Q130.2) on the relationship of homologous limits to percent survival indicated that both secondary and primary limits have been calculated to be well below the stress limits for 99% survival. Cyclic fatigue behavior of near-isotropic graphite is discussed in detail in reference 12.

Irradiation-induced Dimensional Changes - Irradiation of graphite causes damage on an atomic scale (lattice defects) that is manifested in bulk dimensional changes which vary as a function not only of irradiation temperature and fluence, but which are also affected by graphite structural variables. A factor that usually determines the usefulness of a graphite in reactor design is the maximum radial contraction, which influences the stability of a graphite stack. Another factor that required consideration in our review was the dimensional compatibility of H-451 graphite elements with H-327 elements since only one-sixth of the core would be replaced at each refueling. In addition, because of the current theories relating the power fluctuations in FSV to gap sizes between fuel columns and regions, we required an assessment of the potential for component damage that might be caused by gap flow-induced motion and accounting for the different dimensional change behavior of H-451 and H-327 graphites (see Q231.5).

In response to our questions regarding the effects of irradiation-induced dimensional changes, GA pointed out the following pertinent facts: (1) The transition to H-451 graphite fuel elements will take place on a region-to-

region basis. H-451 and H-327 graphite elements will, therefore, not be mixed in the same layer within a region during the transition. (2) The maximum radial shrinkages of the H-327 and H-451 fuel elements are calculated to differ by about 9.1%. Therefore, the characteristics of the power fluctuations (assuming that they are related to opening and closing of inter-regional gaps, as postulated in reference 13) should be unaffected. (3) Potential impact loads calculated for the H-451 elements were 24% higher than the fluctuation impact loads for H-327 elements. Because H-451 graphite has a 66% greater radial strength than H-327, however, the likelihood of damage due to fluctuation-induced loadings is decreased with the substitution of H-451 for H-327 graphite.

We thus conclude, based on the above reasoning, that the substitution of H-451 graphite should not have any measurable adverse effects due to irradiation-induced dimensional changes on the characteristics of core fluctuations or on the overall stability of the core.

Fission Product Release and Transport - The improved thermal conductivity of H-451 graphite relative to that of H-327 is cited by GA as reason to expect a slight reduction in gaseous fission product release. Metallic fission product releases are not so directly assessable, however. In contrast to light water reactors, metallic fission products do not proceed directly from failed fuel to the primary coolant, but must migrate through the fuel rod matrix material and graphite element web to the graphite-helium interface. Metallic fission

product transport from failed fuel particles is thus a multi-step process which is, in part, a function of the sorptivity and diffusivity of the fission product in graphite.

In GLP-5588, GA claimed that the fission product retention characteristics of H-451 graphite are similar to H-327. In general, however, most of the fission product transport data have been obtained on the current reference H-327 graphite, not on H-451. In some cases, no data exist for H-451; for example, there are no data for the diffusion of strontium in H-451 graphite. We, therefore, asked GA to provide additional support for their claims regarding fission product transport in H-451 graphite.

GA responded with some previously unpublished data (on sorptivity of cesium on H-327 graphite; see response to Q231.12). By comparing data in terms of general fit functions (Fig. 2 of response to Q231.12), they showed that there was no significant difference in sorptivity of Cs on H-327 and H-451 graphites.

Data on sorptivity of strontium on H-327 graphite (provided in Tables I through III of the GA response) were compared with data on H-451 in Fig. 1 of GA's response to show that there is reasonable agreement between the sorption isotherms for H-327 and H-451 graphites. Although no data exist on the diffusion of Sr on H-451 graphite, data were presented (Table VII and Figure 3 of GA's response) that indicated that there is no dependence on graphite type. These and other results discussed at length in GA's responses to questions Q231.12 and Q231.22 lead us to conclude that there is reasonable assurance

that fission product transport behavior has been modeled conservatively. As indicated in the response to Q231.22 b, for example, even if the Cs-137 source term is increased by about a factor of seven, the two-hour exclusion-area-boundary bone dose for FSAR "Design Basis Accident No. 2" (Rapid Depressurization Blowdown) is increased by only about 0.5 rem. This is still well below 10 CFR 100 limits and is the only dose significantly affected by the hypothetical increase in Cs inventory.

Post-Irradiation Examination and Surveillance - Although there is a considerable body of experimental data concerning the behavior of H-451 graphite, surveillance, including interim and post-irradiation examinations, is required to confirm the safety analysis of any new fuel design feature. We have in recent months issued several position statements regarding surveillance and PIE of both test and reference fuel in FSV; e.g., Refs. 14 and 15. These statements address, in part, the planned insertion of eight test elements (Ref. 16), composed of H-451 graphite, in reload "segment 7" (currently being loaded into the reactor during the spring 1979). Although we had no formal surveillance requirement for the eight test elements as a condition of this insertion, we have noted (Ref. 14) that safety analyses supported by results of such post-irradiation surveillance may be required for future loads of fuel of new designs. Thus, before final approval of future reloads of H-451 graphite can be granted, we will require (a) results of surveillance examinations on the eight H-451 test elements, and (b) a commitment to perform PIE on future large-scale H-451 reload elements. With regard to the eight test elements

FTE-1 through FTE-8, the currently planned DOE-funded post-irradiation program described in Amendment 2 (Appendix A) of reference 16, should provide a considerable amount of confirmatory information and should provide sufficient information for licensing purposes.

Regulatory Position

We have completed our review of topical report GLP-5588, which is intended to serve as a reference to provide the basis for allowing the substitution of near-isotropic H-451 graphite fuel and reflector elements for the current needle-coke H-327 graphite elements in the Fort St. Vrain reactor. In our evaluation we focused primarily on six aspects of H-451 graphite performance relative to H-327 graphite performance in the FSV core: (1) graphite reproducibility; (2) corrosion (oxidation); (3) mechanical and physical properties; (4) irradiation-induced dimensional changes; (5) fission product release and transport; and (6) post-irradiation examination and surveillance requirements. Based on our evaluation of the information provided in the topical report and the responses to our requests for additional information, we conclude that reasonable assurance has been provided that the substitution of H-451 for H-327 graphite elements in the FSV core will (a) result in negligible changes in the nuclear and thermal behavior of the core, and (b) will not result in reduced safety margins or reliability compared to the reference H-327 core.

Because data acquisition on H-451 properties and thermal irradiation performance is ongoing (via DOE-funded programs involving test-reactor irradiations and test elements to be irradiated in FSV), we will require, as part of any

future application for insertion of reload H-451 elements, that General Atomic Company provide timely reports on test results involving H-451 graphite. Specifically, we will require that reports be provided on the results of (a) the on-going irradiation creep program and (b) post-irradiation examinations and surveillance on the eight H-451 elements that are being inserted as part of reload Segment 7. In addition, should changes be made in the coke or methods of fabrication of the H-451 graphite blocks, we will require GA to provide evidence to support the performance predictions for the graphite elements produced from the altered materials or process parameters.

This evaluation applies only to H-451 graphite to be used in the Fort St. Vrain reactor, because the analyses were performed in terms of the transient analysis for FSV and the comparative effect of H-451 versus the FSV reference H-327 graphite. A separate analysis would be required for any application of H-451 graphite in a high temperature gas-cooled reactor having a design differing from Fort St. Vrain's.

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