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EXPERIENCE ON FUEL AND STRUCTURAL MATERIALS DEVELOPMENT IN THE USA

J. J. Laidler
G. A. Last

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EXPERIENCE ON FUEL AND STRUCTURAL MATERIALS DEVELOPMENT IN THE U.S.A.

Abstract

The United States has conducted extensive LMFBR fuel and structural materials development programs since the mid-1960's. Fuels and materials irradiation tests conducted in EBR-II formed the basis for evaluating the expected performance of FFTF fuel and identified candidate fuels and materials for further full-scale testing in FFTF. The performance of FFTF fuel through the first three years of reactor operation (700 EFPD) has been outstanding. Peak fuel burnup has been limited by swelling of the cold worked AISI 316 austenitic stainless steel ducts rather than by fuel pin behavior. Standard FFTF driver fuel is currently being irradiated routinely to a peak burnup of approximately 100,000 MWd/MTM. Fuel assemblies fabricated with Alloy D9 (titanium modified austenitic stainless steel) cladding and ducts will complete irradiation to an exposure of 155,000 MWd/MTM by the end of 1985, and a large-scale FFTF test program to demonstrate the extended performance capability of fuel assemblies fabricated with the ferritic/martensitic stainless steel alloy HT9 is in progress. Current information indicates that mixed oxide fuel clad with alloy HT9, a modified D9 alloy or dispersion strengthened ferritic stainless steel, enclosed in an HT9 alloy duct, can be expected to achieve the extended burnup goals set for the U.S. program.

Introduction

In the mid-1960's, support for fuel and structural materials development programs in the U.S. increased as part of an expanded LMFBR development program. At that time the focus of the program shifted from metallic fuels to the mixed uranium-plutonium oxide fuel system and the use of austenitic stainless steel for cladding and duct applications -- an approach shared in common with many other countries. The initial goal was to provide an extensive data base for the austenitic stainless steel clad mixed oxide fuel system which would support licensing and operation of LMFBRs in the United States. As the U.S. program for LMR/LMFBR development has evolved over the past twenty years, a period of great change for the nuclear power industry, there have been a number of changes in emphasis in reactor design. Issues addressed have included quality of construction, breeding ratio, proliferation resistance, engineered and passive safety features, construction costs, field vs. shop construction, plant size, and so on. A few parameters in the changing energy equation have remained constant, however, one being the fact that the economics of power generation in a liquid metal cooled fast reactor are improved substantially by development of core systems with high reliability and a long in-core residence capability. Efforts to develop such systems, based principally on the development of improved in-core structural materials, have been conducted in parallel with the aforementioned work on establishment of a comprehensive data base for the reference core system. In addition, the availability of resources in the U.S. program has been such that it was possible to carry on work on alternative or backup fuel systems, for application if and when the attributes of such systems were called for. The U.S. program for fuels and materials development has been a program of active experimentation, with the performance of various core systems concepts subject to rigorous testing under core environmental conditions as nearly prototypic of ultimate service environments as possible.

This paper describes the results of the U.S. LMR/LMFBR fuels and materials development program, with particular emphasis on the results of testing through the first three years (700 EFPD) of operation of FFTF. Ongoing development and testing programs supporting extended lifetime fuels and structural materials will also be reviewed.

Reference Fuel System

The designs of FFTF and the Clinch River Breeder Reactor (CRBR) were based on the use of mixed uranium-plutonium oxide fuel with cold-worked AISI 316 austenitic stainless steel for both cladding and ducts. Initial testing of both fuel and the cladding/duct alloy was conducted in EBR-II in parallel with the design and fabrication of four complete core loadings of fuel for FFTF. Fuel pins were tested in EBR-II under cladding temperature and linear power conditions representative of those anticipated in FFTF and CRBR. Because of the smaller core size and lower flux/harder spectrum conditions of EBR-II, fuel pin lengths and fluence-to-burnup ratios were non-prototypic of larger LMFBRs. Fuel and cladding performance was characterized to burnups in excess of 125,000 MWd/MTM and irradiated fuel pins were tested in the TREAT reactor to characterize the transient performance capability of the fuel. Materials tests were conducted to characterize irradiation-induced mechanical properties changes and the swelling behavior of cold-worked AISI 316 stainless steel. Both fuel pin behavior and materials property data were used to calibrate analytical models and codes used to define the expected performance of fuel assemblies in larger reactors such as FFTF.

The performance of the FFTF reference driver fuel has been outstanding. To date 112 subassemblies (24,300 fuel pins) have been irradiated, with peak exposures reaching 118,000 MWd/MTM and peak fast fluences up to 1.7×10^{23} n/cm^2 (E>0.1 MeV), without cladding breach. A histogram showing the total number of reference oxide fuel pins (including pins which were contained in experimental assemblies) irradiated through FFTF Cycle 5, and the associated burnup levels, is presented in Figure 1. Irradiated fuel pins are distinguished by commercial fabrication source. The performance limits have been clearly defined to be the direct result of swelling of the cold worked 316 SS duct. Fuel assembly core residence time is limited by duct dilation, axial growth and bowing in the core region, which combine to create high assembly withdrawal loads during reactor refueling (dilation and bowing) or potential in-vessel fuel handling and storage problems (axial growth).

An illustration of the burnup limitations imposed by duct dimensional changes is given in Figure 2. This figure shows both _in situ_ length change measurements (performed with the FFTF in-vessel fuel handling machine) and similar measurements performed in the IEM Cell, an in-containment hot cell at FFTF.

The nominal limit on assembly axial growth in FFTF is 0.65-0.75 inch (16.5-19.0 mm), and the data indicate that this limit is reached at a peak fluence of approximately 1.6×10^{23} n/cm^2 (E>0.1 MeV). The corresponding peak fuel burnup is 100,000 to 120,000 MWd/MTM, depending upon the fissile content of the fuel.

The fact that there have been no fuel pin failures in the FFTF reference driver fuel assemblies is noteworthy. The design goal burnup for the reference fuel was 74,000 MWd/MTM, and this is routinely exceeded by 50 percent. Furthermore, EBR-II tests had indicated a substantial failure probability at such burnups. If there is a negative aspect to the results of the program for characterization of reference FFTF driver fuel performance, it is that such dimensional changes have precluded the establishment of significant fuel pin failure statistics in a prototypic core environment. Final analysis and documentation of the results of this characterization program are in progress and will be completed in late 1985.

Improved Mixed Oxide Fuel Systems

It was recognized soon after the discovery of irradiation-induced swelling and enhanced in-reactor creep in the late 1960's that dimensional changes in core structural materials could impose lifetime limits on core components. It was not until relatively high fluences were attained in materials irradiation tests, using both neutron and charged-particle irradiation methods, that the magnitude of the potential dimensional changes became clear. The limitations on burnup capability of systems employing cold-worked AISI 316 stainless steel as the cladding/duct material became the driving force for undertaking the development of new core structural materials in the United States.

The U.S. program for advanced cladding/duct alloy development began in 1973 with EBR-II irradiation and characterization of a large number of candidate alloys. The number of candidates was reduced to two by 1980. One alloy, D9, is a titanium modified austenitic stainless steel which has greatly improved swelling resistance relative to AISI 316 and the second, HT9, is a ferritic/martensitic alloy with excellent swelling resistance characteristics. Both alloys, as well as CW 316 SS, are being subjected to detailed evaluation in the current program. Irradiations are being conducted in the FFTF Materials Open Test Assembly (MOTA) under conditions of active temperature control, to within $\pm 5^{o}C$ or better.

Initial fuel irradiations using these alloys were conducted in EBR-II and extended into FFTF. Experimental FFTF fuel assemblies using D9 alloy cladding and ducts have been undergoing irradiation in FFTF since initial reactor startup. Monitoring of subassembly length charges and withdrawal loads indicates that these assemblies can reach significantly greater burnup levels than are possible with the reference system. The improved swelling resistance of D9 compared to AISI 316 is illustrated by fuel assembly length changes measured in FFTF (Figure 3). Extrapolation of length-change data for the assemblies with D9 ducts indicates a potential to reach fluences in the range of 2.5×10^{23} n/cm^2 (E>0.1 MeV), corresponding to burnup levels of 150,000 to 160,000 Mwd/MTM. Irradiation tests conducted in the TREAT reactor have established the transient performance capability of D9 clad fuel pins and link their performance to the more extensive data base for cold-worked AISI 316 clad fuel pins.

Support for the use of non-swelling ferritic/martensitic alloys increased in the 1980's when greater emphasis was placed on economics and emphasis on system doubling times was reduced. Goal in-core residence times of three to five years were established as a means of minimizing LMR fuel cycle costs. Peak burnups and fluences of 150,000 to 250,000 MWd/MTM and 2.5 to 4.0×10^{23} n/cm^2 (E>0.1 MeV), respectively, are required to achieve this objective. Ferritic/martensitic alloys are the only clear candidate alloys capable of achieving the high fluence objectives without undergoing excessive swelling.

An extensive irradiation test of mixed oxide fuel assemblies using HT9 cladding and ducts is in progress in FFTF. The heart of this program is the Core Demonstration Experiment (CDE) scheduled to begin irradiation in FFTF in early 1986.[1] The goal of the program is to demonstrate at least three year (900 EFPD) in-core residence time capability of this fuel system. Minimum burnup and fluence objectives are 170,000 MWd/MTM and 2.5×10^{23} nvt (E>0.1 MeV). Design analyses indicate that these objectives can be achieved with peak cladding temperatures in excess of 600°C. Tests with HT9 ducts began irradiation in FFTF at the start of Cycle 2, and the first all-HT9 (cladding and duct) mixed-oxide fuel assembly was inserted at the start of Cycle 3. At the time of the Symposium this assembly has accumulated a burnup level of 97,000 MWd/MTM and a residence time in excess of 465 EFPD. Lead tests fully prototypic of the CDE fuel design were inserted in the FFTF core at the start of Cycles 5 and 6 for operation at nominal and 2-sigma cladding temperatures, respectively. All fuel tests incorporating HT9 structural components have functioned without incident, and ongoing materials tests continue to show the superior characteristics of this alloy.

Alternative Fuels

Programs to investigate the potential applications of mixed uranium/plutonium carbide fuels and to extend the burnup capability of metallic fuels have been conducted in support of the LMFBR program. The interested reader is referred to recent review articles [2,3] covering work on these fuel systems.

The highest level of activity on the mixed carbide fuel system was achieved in the mid- and late-1970's. An extensive irradiation test program was conducted in EBR-II to evaluate design parameters and support performance model development. Decreased support for the program as a result of reduced emphasis on breeding gain and doubling time limited the extent of the FFTF testing program. A total of three assemblies are being irradiated in FFTF to evaluate the performance of mixed carbide fuel clad with both AISI 316 and D9 austenitic stainless steel alloys. One of these assemblies was removed from the core during Cycle 6 after having experienced cladding breach in a helium-bonded fuel pin at a burnup of 73,300 MWd/MTM. Further development of carbide fuel will be contingent on a clearly recognized requirement for a fuel system capable of achieving doubling times of 15 years or less in an LMFBR. The irradiation of at least one metal fuel assembly in FFTF, for the purpose of establishing the feasibility of a ternary U-Pu-Zr fuel, is planned to commence at Cycle 9. This fuel alloy offers the promise of enhanced fuel-cladding compatibility.

Future Needs

Current information supports the position that alloy HT9 can meet all of the forseeable requirements for a duct material for fuel assemblies capable of irradiation to fuel burnups of 250,000 MWd/MTM. It is less certain that this alloy can meet cladding requirements for steady state fuel design conditions exceeding $625^{o}C$ to $650^{o}C$. Dispersion-strengthened ferritic alloys are being investigated for use at these higher temperatures. Successful demonstration of the use of these alloys would fully satisfy the anticipated needs of a economically competitive liquid metal reactor (LMR). To the extent that a goal lifetime of three years [peak fluence of 2.5×10^{23} nvt (E>0.1 MeV)] is a viable option, an improved D9 alloy (D9I), containing increased levels of interstitial alloying elements, can be expected to meet all fuel cladding performance requirements. Confirmation of these projections can only be accomplished by irradiation of test subassemblies to target exposure levels.

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[2] WALTERS, L. C., SEIDEL, B.R., KITTEL, J. H., Performance of metallic fuels and blankets in liquid-metal fast breeder reactors, Nuclear Technology 65 (1984) 179.

[3] MATTHEWS, R. B., HERBST, R. J., Uranium-plutonium carbide fuel for fast breeder reactors, Nuclear Technology 63 (1983) 9.

LIST OF FIGURES

REFERENCE FUEL PINS IRRADIATED IN THE FFTF THRU CYCLE 5

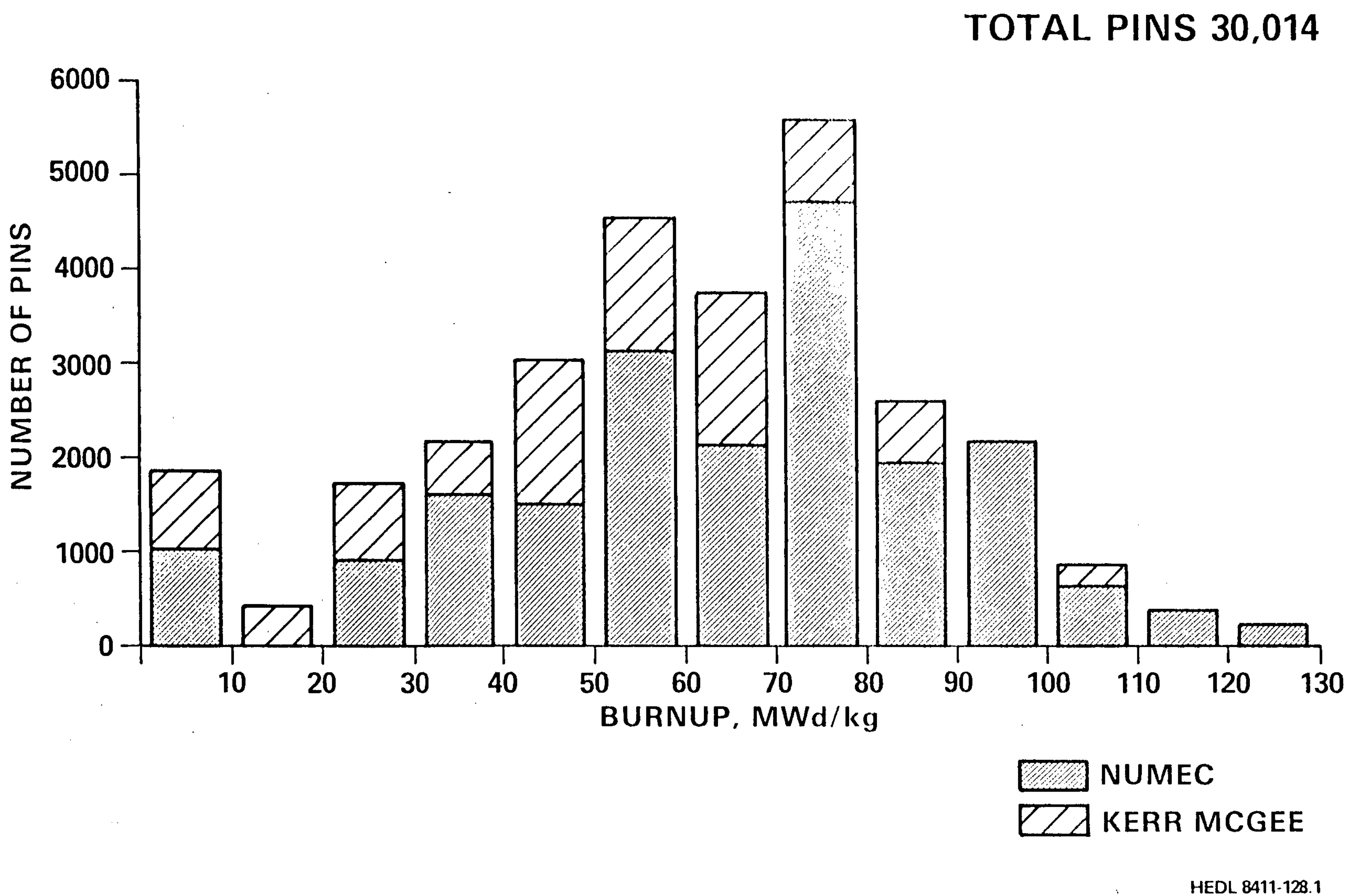

Figure 1. Burnup histogram for commercially-fabricated FFTF reference driver fuel pins.

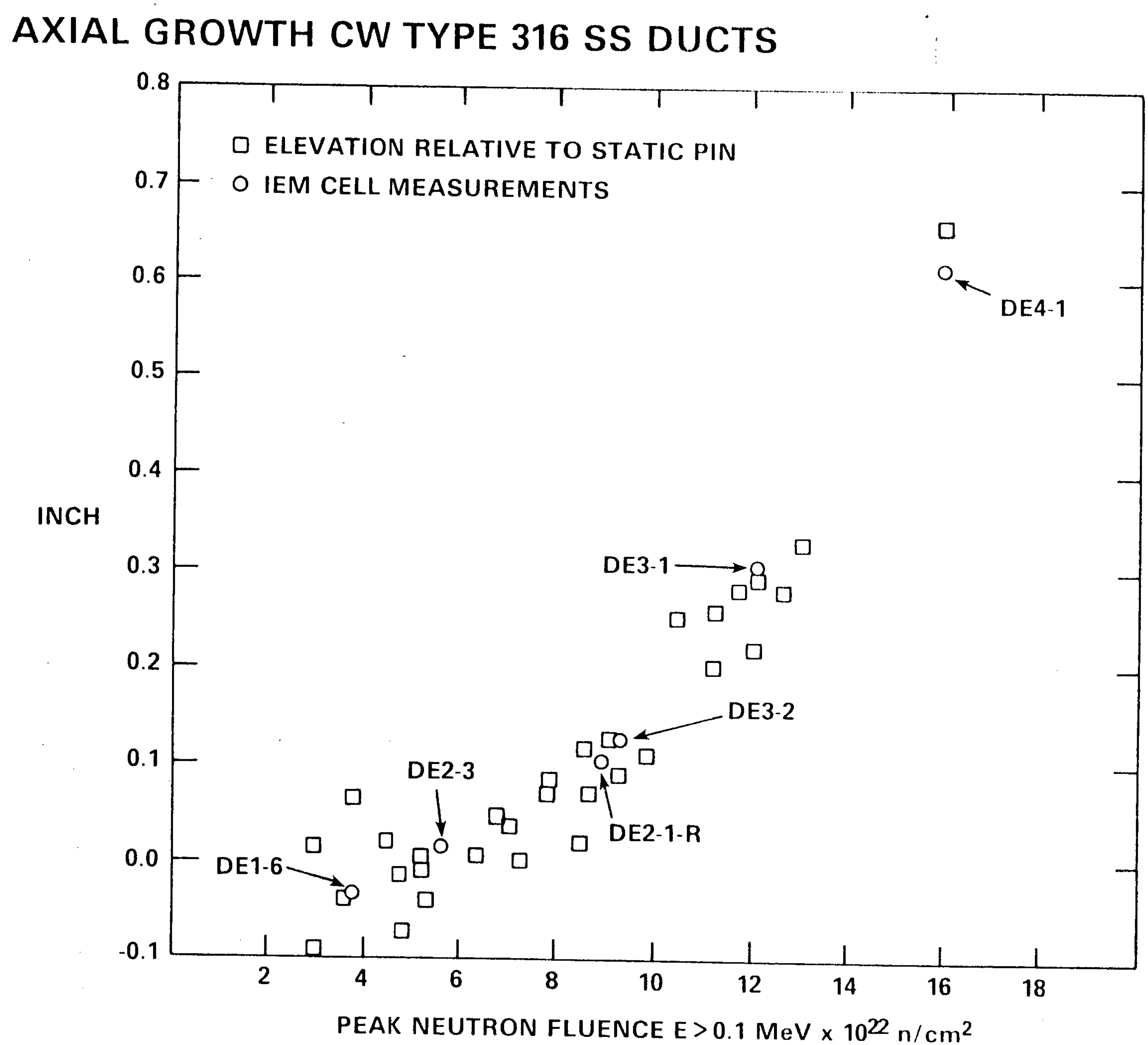

Figure 2. Length increases of FFTF reference driver fuel assemblies as a function of peak fast fluence.

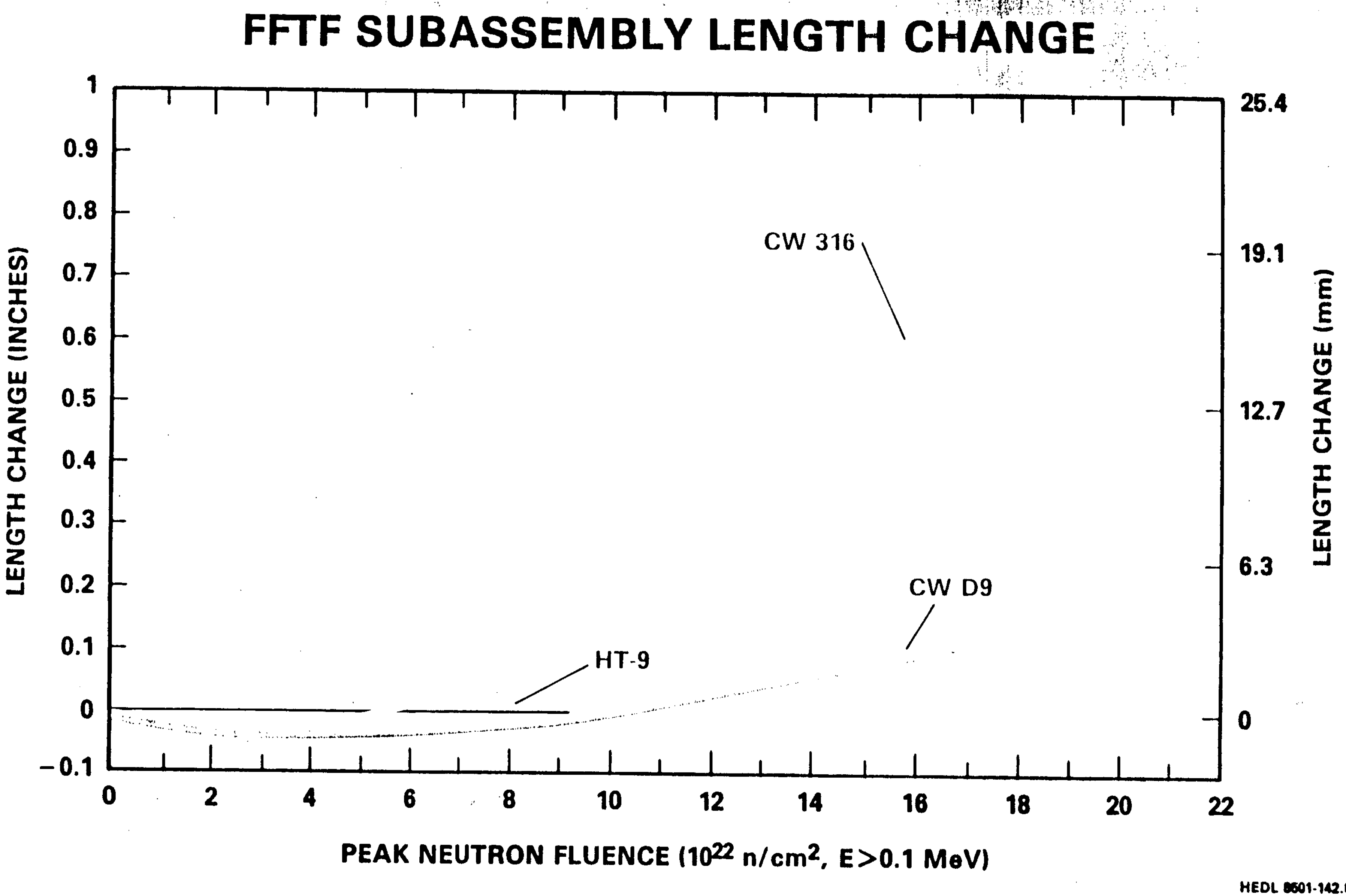

Figure 3. Comparison of length increases for CW 316 and CW D9 assemblies as a function of fluence.