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PHILLIPS PETROLEUM COMPANY
IDAHO FALLS, IDAHO 83401

RESEARCH AND DEVELOPMENT DEPARTMENT
ATOMIC ENERGY DIVISION

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April 16, 1965

Transmittal of SNAPTRAN-1
Test Series Proposal No. 5
Ny-101-65A

Mr. J. F. Kaufmann, Assistant Manager
Technical Operations
Idaho Operations Office
U. S. Atomic Energy Commission
Idaho Falls, Idaho

Dear Mr. Kaufmann:

Transmitted herewith are two copies of the SNAPTRAN-1 Test Series Proposal No. 5 (Broc-13-65A-N). This Test Series Proposal covers the operations to be performed on the SNAPTRAN-1 reactor during transient tests from ambient and elevated temperatures with step insertions of reactivity.

This Test Series has been approved by the STEP Staff, the SPERT-STEP Safeguards Committee, and by me for performance in the IET facility, commencing the latter part of April 1965. This Test Series Proposal is submitted for approval by ID.

Very truly yours,

W. Thomas

acting Assistant Manager
Nuclear Safety Technology
Atomic Energy Division

WENyer:je

cc: J. F. Kaufmann
J. P. Lyon

PHILLIPS PETROLEUM COMPANY
Atomic Energy Division
Idaho Falls, Idaho
April 13, 1965

Inter-Office Correspondence / Subject:

SNAPTRAN-1
Test Series Proposal No. 5
Revision 1
Additional Step Transient Tests
Broc-13-65A-N

N O T E G R A M

To: T. R. Wilson
From: G. F. Brockett:je/WEK B.E.K

Management approval is requested for the performance of a series of transient power excursion tests in which the SNAPTRAN-1 reactor is subjected to step reactivity inputs with the initial fuel temperature ranging from 70°F to 1000°F. The purpose of this test series is to extend the average core fuel temperature range over which the temperature dependent feedback coefficient may be determined from 800°F to 1200°F.

I. Introduction

The previous step test series, performed within the scope of Test Series Proposal (TSP) No. 3, Revision 3, provided information needed to determine the negative reactivity feedback coefficient as a function of average core fuel temperature in the range of 70°F to 800°F. These previous tests were performed using conservative operating limits of strain, pressure, and temperature to confirm the transient behavior of the reactor and to assure that reactor damage as defined in TSP No. 3 and redefined in TSP No. 4 did not occur. Tests performed under TSP No. 3 were limited to a fuel hotspot temperature of 1500°F. On the basis of the data obtained from isothermal furnace tests at Atomics International and from the previous step tests, the upper hotspot temperature can be safely increased to 1700°F while maintaining the former limits on strain and pressure. This temperature can be safely accommodated since the axial distribution equilibrates shortly after the nuclear burst to an average fuel rod temperature below that which results in damage. At the core fuel hotspot temperatures of 1400°F attained during previous tests, less than 5 psig of hydrogen pressure was observed.

The only strain observed during the previous step tests was that resulting from longitudinal thermal expansion of the fuel material. An average core temperature of 1500°F is calculated to produce thermal expansion and hydrogen pressure induced clad elongation sufficient to fill all radial void spaces between the fuel rods and the beryllium reflector material. The results of the previous step tests indicate that step tests can be performed with hotspot temperatures up to 1700°F without exceeding the strain and pressure limits as defined in the previous TSP No. 4. Within this criterion, however, to prevent excessive pressure from hydrogen release following the post-burst relaxation of the axial thermal gradient, the average temperature of the hottest fuel element will not be allowed to exceed 1350°F.

II. Scope

Three or more tests will be performed with the reactor at an ambient initial temperature of about 70°F. These tests will be initiated by step reactivity insertions of ~ 1.40 dollars and ~ 1.50 dollars and will be followed by insertions leading to the highest value which can be accommodated within the strain and pressure limits as defined in TSP No. 4 (viz, fuel clad elongation of 1.2%, circumferential reactor vessel elongation of 0.5% and internal fuel rod pressure of 800 psi) and temperature limits of 1700°F for the hotspot or 1350°F for the average fuel temperature of the hottest fuel element.

Several additional step tests will be performed with the core fuel at initial temperatures above 70°F. This will be accomplished by means of nuclear heat. Under these conditions, the hotspot Δ T (i.e., the temperature rise from starting temperature to the ultimate maximum temperature of the hotspot) must be limited to prevent the average temperature of the hottest fuel rod from exceeding 1350°F. The core fuel hotspot temperature occurs in the center fuel rod at the axial mid-plane. Since during a transient, an axial peak-to-average power density factor of 1.3 exists in the center fuel rod, the hotspot temperature decays following the power burst to the axially equilibrated average center fuel rod temperature. The equilibrated center fuel rod temperature resulting from a given transient depends on the total temperature rise at the hotspot which then reduces to an average center fuel rod temperature equal to the starting temperature of that fuel rod plus 1/1.3 times the hotspot temperature rise. Figure 1 shows the hotspot temperature calculated to give an axially equilibrated center fuel rod temperature of 1350°F plotted as a function of average core starting temperature. The average core temperature attainable within this limit is also shown. It is anticipated that an average core fuel temperature of 750°F will be used for the starting temperature. The tests initiated at this elevated temperature are designed to yield an energy release which will result in an average temperature in the hottest fuel rod no greater than 1350°F. Figure 1 indicates that using the 750°F starting temperature, a maximum fuel hotspot temperature could not be allowed to exceed 1530°F.

III. Reactor Control and Instrumentation

Control of the step transient tests in this series will be achieved in the same manner as for the step tests previously performed under TSP No. 3. The doors to the thermal box which normally surrounds the reactor may be in place to reduce heat loss from the reactor following nuclear heatup. The heaters attached to the thermal box doors may also be used to help flatten the elevated starting temperature distribution prior to initiating a test.

The instrumentation used for these tests will consist of the same instrumentation as that used for the impulse tests as outlined in TSP No. 4. This will include the following:

13 each - Nuclear detectors, ranged from 10^{-3} watts to 10^{11} watts

16 each - In-core energy probes, ranged from 0 MW-sec to 20 MW-sec

8 each - Fuel temperature thermocouples, ranged from 70°F to 1800°F

8 each - Fuel clad strain gages, ranged from 0% to 2% elongation

1 each - Vessel strain gage, ranged from 0% to 2% elongation

4 each - Fuel rod pressure transducers, ranged from 0 psig to 3000 psig

2 each - Fuel rod pressure transducers, ranged from 0 psig to 300 psig

2 each - Fuel rod pressure transducers, ranged from 0 psig to 150 psig

1 each - Fuel rod pressure transducer, ranged from 0 psig to 250 psig

1 each - Fuel rod pressure transducer, ranged from 0 psig to 50 psig

In the case of many of these detectors, the range of signal output will be expanded so as to indicate trends more clearly in the range of interest. In addition, a 15 channel sampling recorder with a 30 second cycling time will be used to monitor radial and axial arrays of energy probes and fuel thermocouples following the nuclear heatup to indicate the spatial temperature distribution prior to performing a subsequent test.

IV. Expected Results

Based on the extrapolation of results of the previous step tests, it is anticipated that no reactor damage will occur throughout this test series. The isothermal data presented in NAA-SR-5483 indicates that at a fuel temperature of 1400°F a pressure of 310 psig would result. However, a pressure of only 5 psig was actually observed to occur in the center fuel rod as a result of the 1400°F hotspot fuel temperature observed in the previous step test series. This lower pressure is the result of axial heat flow away from the hotspot toward the cooler ends of the fuel rod. The 5 psig correspondes to a predicted fuel temperature of less than 1100°F and appears to be related to the equilibrated center fuel temperature rather than the hotspot of the center fuel rod. It is therefore estimated that an equilibrated center fuel rod temperature of 1400°F to 1500°F would be encountered before the 800 psig pressure limit is reached. However, since in some isolated cases of oven heating of the fuel at Atomics International, a temperature of as low as 1385°F was found to cause clad rupture, a 1350°F upper limit will be imposed on the average temperature of the hottest fuel rod (i.e., the center fuel rod). Under these conditions, the limits for damage as defined in TSP No. 4 are not expected to be exceeded during this test series.

V. Test Sequence

The testing sequence to be used for those step insertions starting at ambient temperature (~ 70°F) will be the same as those presented in TSP No. 3. The operating limits to be used for termination of this portion of the test series will be the same as those specified in TSP No. 4 except for the 1700°F hotspot temperature limit and the 1350°F average hot fuel rod temperature limit as discussed above.

The transient tests which are to be initiated from elevated fuel temperatures will be performed in a manner similar to those initiated from ambient fuel temperature, except that the spatial temperature distribution will be examined to determine the temperature rise at the hotspot and in the hottest fuel rod which can safely be accommodated by the ensuing step transient. Cognizance of this distribution will be exercised by means of the 15 channel display recorder as it records the output signals of pre-calibrated energy probes and thermocouples.

The drum position setting for the intended step insertion will be selected on the basis of cold or ambient drum worth curves and a pre-transient critical performed to compensate for the overall temperature defect. This method will result in a slightly smaller reactivity insertion than predicted since the hot beryllium is less reflective than the cold, however, the projected energy release will be extrapolated on the basis of what was previously determined for a given reactivity insertion at temperature.

The sequence of testing for these tests will consist of performing a transient from ambient which results in about 800°F average core temperature rise. Following the relaxation of the temperature to an acceptable distribution and level, a step reactivity insertion yielding about 2 MW-sec total energy release will be imposed. Since the thermal decay for an 800°F average core temperature rise is about 80°F/hr approximately 2 hours will be available for evaluation of the previous test results and preparation for the next test. This duration is expected to be more than adequate to perform mechanical checks on the reactor for insuring freedom of drum movement. Subsequent tests using successively larger reactivity steps will be performed only after the initial elevated temperature conditions prior to each test return to the conditions comparable to those for the first elevated temperature step test. The limiting criteria will be, as before, strain, pressure, and the temperature of the hotspot and the average fuel temperature of the hottest fuel rod.

VI. Safety Considerations

Radiological monitoring will be provided in and around the test cell in the same manner as for previous transient tests. Based on present understanding of the available data on hydrogen diffusion, no reactor damage is expected in this test series. However, the available data concerning hydrogen release levels and rates is subject to considerable uncertainty and therefore makes an accurate prediction of test results difficult. Hence, it is conceivable that some partial fuel damage may result for tests involving insertions greater than 1.5 dollars or those tests approaching the hotspot temperatures associated with the 1350°F hot fuel rod limit. However, such fuel damage is not expected to cause undue radiological hazard to operating or other site personnel since the tests will be conducted under meteorological control. Under the conditions imposed for this test series, a maximum energy release of 20 MW-sec could result. Should the fission product cloud (assuming a 100% release) associated with a 20 MW-sec energy release be directed toward the Technical Support Facilities (TSF) area, less than a quarter of the yearly allowable radiological dose would be incurred by NRTS site personnel. To minimize the undue radiological dose incurred by site personnel, evacuation of personnel in the TSF area (the location nearest to the test area) will be undertaken. To allow sufficient time for this evacuation, a maximum wind speed when in the direction of the TSF area will be imposed as necessary conditions for conduct of the test. Specifically, the meteorological restrictions for these tests will be as follows:

(1) No wind speed restrictions will be imposed when the wind is blowing into the unrestricted sectors encompassed by an arc extending from 270° through 0° to 150° (0° representing due north).

(2) The wind speed will not exceed 5 mph when blowing into the restricted sector encompassed by 150° through 180° to 270°.

VII. Schedule

These tests will be started during the latter part of April and are expected to require two weeks for performance.

VIII. Approvals

Reviewed by STEP Senior Staff: T. R. Wilson Date: 4/14/65

Approved by Engr. & Test Branch Mgr: T. R. Wilson Date: 4/14/65

SPERT-STEP Safeguard Committee: G. O. Bright Date: 4/16/65

Approval by Asst. Mgr. Nuclear Safety Technology
Wm. Thomas for GEN Date: 4/16/65

Approval Received from ID: ______________________ Date: ______

Attachment

cc: W. E. Nyer
W. E. Nyer /r/ J. P. Lyon
W. E. Nyer /r/ J. F. Kaufmann (2)
T. R. Wilson
M. E. Thomas
F. L. Bentzen
G. F. Brockett (3)
W. J. Neal (3)
O. M. Hauge
R. P. Rose
W. E. Kessler
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Committee Files (S. Ward)

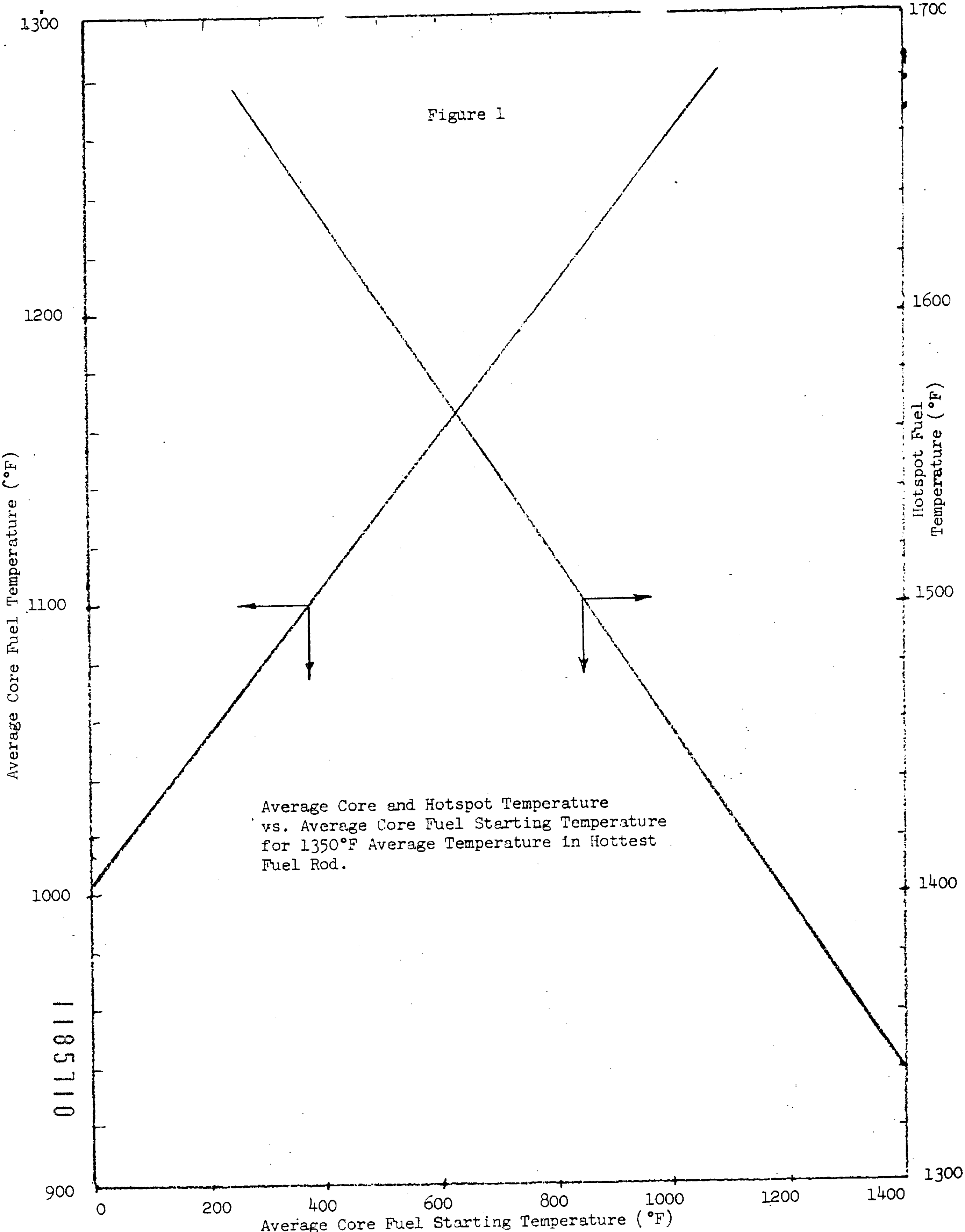

Figure 1
Average Core and Hotspot Temperature
vs. Average Core Fuel Starting Temperature
for 1350°F Average Temperature in Hottest
Fuel Rod.
Average Core Fuel Temperature (°F)
Hotspot Fuel Temperature (°F)
Average Core Fuel Starting Temperature (°F)
1300
1200
1100
1000
900
1700
1600
1500
1400
1300
0
200
400
600
800
1000
1200
1400
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