ORNL

Design and Safety Report for Hobbie Capsules

A Series of Zircaloy Creepdown Irradiation Tests

K. R. Thoms
Theo van der Kaa

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Under Interagency Agreements DOE 40-551-75 and DOE 40-551-76

OAK RIDGE NATIONAL
OPERATED BY UNION CARBIDE CORPORATION

Contract No. W-7405-eng-26

ENGINEERING TECHNOLOGY DIVISION

DESIGN AND SAFETY REPORT FOR HOBBIE CAPSULES

A Series of Zircaloy Creepdown Irradiation Tests

K. R. Thoms
Oak Ridge National Laboratory
Oak Ridge, Tennessee, USA

Theo van der Kaa
Energieonderzoek Centrum Nederland (ECN)
(Netherlands Energy Research Foundation)
Petten (NH), The Netherlands

Date Published: December 1977

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights.

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Under Interagency Agreements DOE 40-551-75 and DOE 40-552-75

NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

CONTENTS

	<u>Page</u>
ABSTRACT	v
1. INTRODUCTION	1
2. EXPERIMENTAL REQUIREMENTS	3
2.1 Neutron Flux	3
2.2 Thermal Conditions	4
2.3 Process Conditions	4
2.4 On-line Measurements	4
2.5 Off-line Measurements	4
3. IRRADIATION TARGET	4
4. IRRADIATION FACILITY	7
4.1 In-core Section	7
4.1.1 Irradiation capsule	7
4.1.2 Thimble	10
4.2 External Installations	10
4.2.1 Cooling system	12
4.2.2 Gas supply system	12
4.2.2.1 Specimen gas system	12
4.2.2.2 High-pressure gas system	12
4.2.2.3 Gas service system	15
4.2.2.4 Nitrogen system	16
4.2.2.5 Compressed air system	16
4.3 Instrumentation	16
4.3.1 Measuring systems	19
4.3.1.1 Scientific instrumentation	19
4.3.1.2 Process instrumentation	20
4.3.1.3 Neutron dosimetry provisions	23
4.3.2 Data registration and control	24
4.3.2.1 Scientific data registration	24
4.3.2.2 Process data registration	24
4.3.3 Safety instrumentation	24
4.4 Utilities	25
4.4.1 Electrical power supply system	25
4.4.2 Gas supply system	25
4.5 Waste System	25
4.6 Shielding	26
5. CHARACTERISTICS	26
5.1 Nuclear Characteristics	26
5.2 Thermal Characteristics	28
5.3 Stress Characteristics	29
5.4 Reactivity Characteristics	35
5.5 Radioactivity Characteristics	37

BLANK PAGE

	<u>Page</u>
6. FABRICATION, INSPECTION, AND TESTING	40
6.1 Irradiation Capsule	40
6.2 Thimble	40
7. OPERATION	41
8. HANDLING, TRANSPORT, DISMANTLING, AND DISPOSAL	42
9. POSTIRRADIATION MEASUREMENTS AND REPORTS	44
9.1 Measurements	44
9.2 Reports	44
10. SAFETY ANALYSIS	44
10.1 Failure Analysis	44
10.1.1 Target failure	44
10.1.2 Failure of containment	45
10.1.3 Cooling system failure	46
10.1.4 Electrical power failure	46
10.1.5 Instrumentation failures	46
10.1.6 Utility failures	47
10.1.7 Operating errors	47
10.1.8 Handling errors	47
10.2 Maximum Credible Accident Analysis	48
ACKNOWLEDGMENTS	49
REFERENCES	51
Appendix A. REMOVED HEAT ANALYSIS	53
Appendix B. MATERIALS USED IN THE PRESSURIZATION SYSTEM	55
Appendix C. PRESSURE EQUALIZATION CALCULATIONS	57
Appendix D. CALCULATION OF SURFACE HEAT TRANSFER COEFFICIENT	59
Appendix E. MEMO - D. A. CANONICO TO D. O. HOBSON, <i>FRACTURE</i> <i>TOUGHNESS</i> , May 12, 1977	61
Appendix F. ACTIVATION OF GAS INSIDE CAPSULE	71

ABSTRACT

This report presents the design and safety features of a series of irradiation experiments designed to study the in-reactor creepdown behavior of Zircaloy tubing typical of LWR fuel cladding. This is a cooperative program between Oak Ridge National Laboratory (ORNL) and Energieonderzoek Centrum Nederland (ECN) at Petten, The Netherlands. The experiments were designed and will be fabricated by ORNL and will be irradiated in the High Flux Reactor (HFR) at Petten. This report is to serve as a basis of technical and safety assessment leading to approval for installation and operation in the HFR.

These experiments will use eddy-current monitoring devices to monitor continuously the creepdown behavior of a test specimen. The test specimens will be irradiated at temperatures between 316-427°C (600-800°F) with a pressure differential across the specimen wall of up to 20.7 MPa (3000 psi).

DESIGN AND SAFETY REPORT FOR HOBBIE CAPSULES

A Series of Zircaloy Creepdown Irradiation Experiments

1. INTRODUCTION

The purpose of the Zircaloy creepdown program is to provide data on the deformation characteristics of Zircaloy tubes, typical of LWR fuel elements, under conditions of axial and tangential compressive stress. These data will be used to verify and improve the material behavior codes that are used for the description of fuel pin behavior in nuclear reactors. Knowledge of clad deformation under biaxial compressive stresses is important because of the following fuel behavior aspects: (1) the thermal characteristics of operating fuel pins, which are dependent upon the effective fuel-cladding gap and thus on the clad diameter; and (2) pellet-clad mechanical interaction, which is also directly dependent upon the clad dimensions.

The creepdown characteristics, both in- and out-of-reactor, are presently somewhat unknown quantities. This is especially true with regard to creep rate. Insufficient information presently exists on this subject. In fact, unless prior specimen history is available, any mechanical properties data are of limited value for the verification and development of correlations and codes. It is apparent that tubing deformation is both a geometrical and a mechanical property phenomenon. Geometrically, the tubing dimensions and, very importantly, the unsupported length of the tubing significantly affect the magnitude and distribution of the stresses induced in the tubing. The strength properties of the material determine the behavior in response to these stresses. If these properties are anisotropic, the behavior of the tubing is further complicated by the need to consider stress states of the tubing as a function of its geometry.

It is reasonable to assume that the directions of the stresses in the tube wall will not change during creepdown - at least not in the early stages of the process. The magnitudes of those stresses will change, however, as will the associated stress ratios.

Although it is pointless to speculate on the specific forms that the deformation would take (this will be determined experimentally), it is important to note that a single tube will have a number of different stress states present during collapse or creepdown. Each of these states will interact with the tube material to cause strain, either by yielding and plastic flow or by creep. In an isotropic material, these strains should be fairly predictable and, indeed, a large volume of literature exists in the multiaxial stress-strain behavior of the more isotropic materials. The anisotropic behavior of Zircaloy, however, adds a further complication to the problem. In this metallurgical variable, the mechanical properties are functions of the degree of

BLANK PAGE

preferred orientation (texture) and the direction and sense of the applied stress, relative to the particular texture. Thus this variable will modify and may even negate many of the observations and conclusions concerning tubing creepdown behavior described by various investigators and based on tests of more isotropic materials.

As far as data are concerned, numerous experiments, both in- and out-of-reactor, have contributed to the knowledge of the effects of both uniaxial and biaxial tensile stresses on annealed and lightly cold-worked Zircaloy tubing. Virtually no data are available, at the temperatures of interest (300-400°C), for the conditions that are considered important for LWR clad creepdown behavior: (1) uniaxial and biaxial compressive loading; and (2) the 70% cold-worked plus stress-relieved condition. Consequently, little insight exists into the phenomena that, qualitatively and quantitatively, control the creepdown behavior.

The US-NRC creepdown program is carried out to provide such data. Experiments will be conducted both in- and out-of-reactor. The former program is necessary because neutron irradiation is expected to influence substantially the creepdown rate.

A number of out-of-reactor creepdown tests have already been carried out at ORNL. Apart from having produced useful data, the experiments have been very helpful in proving the feasibility of the clad deformation sensing instrumentation which was specially developed for this project.

The in-reactor tests, which have been designated HOBBIIE capsules, will determine the rate of cladding creepdown under essentially the same ranges of temperature, external pressure, and pellet-to-cladding gap used in the out-of-reactor tests. These tests will take place in the H2 position of the HFR in an independently water-cooled thimble provided by ECN. Each test will irradiate a Zircaloy tube test specimen of 10.92-mm (0.430-in.) OD by 0.64-mm (0.025-in.) wall thickness and 152.4-mm (6.0-in.) length. The test tubes are selected from a batch of Zircaloy-4 tubing that was manufactured by Sandvik for the US-NRC reactor safety program. Ninety tube sections from this delivery were carefully characterized, as regards:

- geometrical shape, i.e., inner and outer diameter, wall thickness, straightness, etc. by NDT procedures. Tubes with the most uniform dimensions will be chosen for the initial experiments, but later tubes with varying amounts of ovality may be used to investigate the influence of ovality itself on the deformation characteristics.
- texture, by investigating x-ray texture specimens taken from representative tubes and by determination of basal pole figures on tube sections which have been machined and chemically thinned to a number of decreasing wall thicknesses. The investigation revealed that the textures showed very little variance through the wall and that they were typical of those expected from present-day commercial tubing.

2. EXPERIMENTAL REQUIREMENTS

The planned test matrix for the first four creepdown experiments is given below.

Test	Temperature (°C)	Overpressure (MPa)
1	371	17.5
2	371	16
3	371	14.6
4	371	13.3

It is expected that the test time for these experimental conditions will range from 600-1200 hr. The information from these tests will provide a basis for the choice of test matrixes for the long range tests.

The experimental variables for the creepdown tests series are given below.

- I. Independent variables
 - A. Temperature - 316 to 427°C (600 to 800°F)
 - B. Pressure differential across the wall of up to 20.7 MPa (3000 psig)
- II. Dependent variables
 - A. Time to first contact of mandrel
 - B. Radial strain
 - C. Radial strain rate
 - D. Strain geometry
- III. Covariate variables
 - A. Tubing dimensions
 1. Diameter
 2. Wall thickness
 3. Wall thickness variation
 4. Ovality
 - B. Texture
 - C. Fast neutron flux ($E_n > 1.0$ MeV)

The specific requirements for this test series are as follows.

2.1 Neutron Flux

It is anticipated that only the fast neutrons ($E_n > 1.0$ MeV) will have a significant effect on the creepdown rate of the Zircaloy tubing. It is desired to irradiate the test specimens at fluxes typical of a large LWR, i.e., on the order of 6.0×10^{17} neutrons/m²·sec ($E_n > 1.0$ MeV).

2.2 Thermal Conditions

The test specimens will be irradiated at temperatures between 316–427°C (600–800°F). The irradiation experiment is designed so that gamma heat alone should provide a test specimen temperature of 232°C (450°F). The temperature control above 232°C (450°F) will be obtained from an electrical heater located in the center of the test specimen. With about 360 W of heat from this electrical heater, a test specimen temperature of 427°C (800°F) should be obtained.

2.3 Process Conditions

The test specimen will be irradiated in a helium environment at the temperatures stated in Sect. 2.2. The pressure inside the test specimen will be maintained at about 0.1 MPa (15 psi), while the helium pressure external to the test specimen will be maintained at pressures of up to 20.7 MPa (3000 psi).

2.4 On-line Measurements

Continuous monitoring of the temperature and pressure within the irradiation experiment will be necessary. Each capsule will be provided with four Chromel/Alumel thermocouples to monitor the test specimen temperature and up to four additional Chromel/Alumel thermocouples to monitor temperatures at various regions within the capsule.

Each capsule will also be provided with up to 24 eddy-current coils to monitor the deformation of the test specimen.

2.5 Off-line Measurements

Each capsule will be provided with flux monitors that must be recovered and measured after the irradiation test is completed. In addition, it may be desirable to examine visually and possibly make measurements of a collapsed test specimen.

3. IRRADIATION TARGET

The principal target of these irradiation experiments is of course the Zircaloy test specimen. This test specimen is shown in Fig. 1 along with associated parts necessary to complete the simulation of a LWR fuel rod and to monitor the temperature and deformation of the test specimen. As shown in the figure, an electrical heater is located in the center of the test specimen to supply the heat necessary to obtain

Fig. 1. Irradiation target and associated components used to mock up LWR fuel rod and to monitor creepdown of Zircaloy tube.

the desired operating temperatures. Beryllium filler pieces surround the heater both to minimize the operating temperature of the heater and to simulate fuel pellets within the test specimen. The OD of the beryllium is 9.45 mm (0.372 in.) to provide the desired 0.1-mm (0.004-in.) radial fuel-to-cladding gap. In the region used to simulate the pellet-to-pellet gap, if one should be desired, a nickel sleeve surrounds the electrical heater. This sleeve is 1.24 mm (0.049 in.) thick, providing the same mass per lineal inch and therefore producing about the same amount of gamma heat per lineal inch as the beryllium filler pieces. The nickel sleeve will vary in length depending on the pellet-to-pellet gap that is to be simulated in a particular test. In addition, it is insulated from the beryllium sleeves with graphoil wafers; four 0.13-mm (0.005-in.) thick wafers are placed at both ends of the nickel sleeve.

There are four Chromel/Alumel thermocouples welded to the ID of the test specimen just above the end of the top set of beryllium filler pieces. These are Iconel-600 sheathed thermocouples of 0.71-mm (0.028-in.) OD that are welded in at 90° intervals at the same elevation. The test specimen itself, as stated previously, is Zircaloy tubing of 10.92-mm (0.430-in.) OD and 0.64-mm (0.025-in.) wall thickness. Surrounding the test specimen is a 0.76-mm (0.030-in.) radial gas gap between the test specimen and the type 6061 aluminum transducer block. The transducer block has a 44.88-mm (1.767-in.) OD and a 12.45-mm (0.490-in.) ID. There are holes drilled into the block to house the eddy-current probes in the orientation shown in Fig. 2. The OD of the transducer block is grooved to provide space for the lead wires to the probes.

ORNL-DWG 76-21003

Fig. 2. Orientation of eddy-current probes in creepdown test.

4. IRRADIATION FACILITY

The irradiation of the HOBBIE capsule will be performed in the High Flux Reactor (HFR) at Petten. This reactor, which is of the same type as the Oak Ridge Reactor (ORR), is operated under contract by The Netherlands Energy Research Foundation for the Commission of the European Communities. The HFR is a materials testing reactor (Fig. 3) and normally operates at 45 MW. Characteristics of facilities and standard irradiation devices have been described in ref. 1.

4.1 In-core Section

The in-core section of the irradiation facility can be divided in two parts: (1) the irradiation capsule and (2) the thimble. The capsule is designed, manufactured, and assembled at ORNL; the thimble, being a standard irradiation facility, has been made by ECN.

4.1.1 Irradiation capsule

The configuration of the in-core portion of the irradiation capsule is shown in Fig. 4. As can be seen in the figure, the test specimen is welded to a Zircaloy reducer tube which in turn is welded to a Zircaloy-to-stainless steel transition joint. This is a commercially obtained tandem-extruded joint, allowing the use of the inner stainless steel bulkhead through which the heater and internal thermocouples can be brazed.

The pressure vessel of the capsule is type 316 stainless steel of 55.12-mm (2.170-in.) OD, 44.95-mm (1.770-in.) ID, and 560-mm (22-in.) length. In order to reduce mechanical and thermal stresses, the bottom part of the pressure vessel is provided with a hemispherical head. Because of the penetration of instrumentation lead wires and gas lines, an identical geometry at the top part of the pressure vessel was impossible. Therefore, a flat plate bulkhead has been designed in which the lead wires and gas lines are furnace brazed with Nicro-Braze 50 at a temperature of 1010°C (1850°F). The width of the brazing annulus between lead wires and gas lines in the holes of the bulkhead is 0.038 mm (0.0015 in.).

Along with the aluminum deformation monitoring device and the specimen with transition tube, there are two aluminum blocks, one above and one below the deformation monitoring device. In this way the correct thermal characteristics of the capsule are obtained while the volume of high pressurized gas is kept as small as possible.

Detailed assembly drawings including parts lists and material specifications are given in ORNL drawings X8E11762-100 et. seq.

Fig. 3. General arrangement of the High Flux Reactor (HFR) at Petten, The Netherlands. (Reproduced with permission of ECN.)

Fig. 4. General configuration of the in-core portion of the HOBBIIE irradiation capsules.

4.1.2 Thimble

Because the end of the irradiation is defined as the time when a certain displacement of the cladding has been reached, the irradiation time is unknown. In order to make the irradiation of the HOBBIE capsules independent of the reactor operation cycles - which are 26 days - the irradiation is conducted in the thimble.

The thimble is a universal irradiation device for straight rigs that makes it possible to load and unload the experiment during reactor operation. This thimble consists of a closed bottom tube whose top side is equipped with two pads that fit in the standard cone clamp assembly of the central reactor top lid.

Made of type 316 stainless steel, the thimble's in-core dimensions are a 68-mm (2.678-in.) ID and 72-mm (2.835-in.) OD so that a normal filler element can be used for positioning the thimble into the reactor core. The irradiation rig, which is provided with centering pads at the top side, is inserted in the thimble and fixed by means of one cone clamp construction that is mounted on the thimble. In this way the same handling tool can be used for thimble fixation in the reactor top lid as well as experiment fixation in the thimble.

Cooling water is injected into the vessel entry part of the thimble. It flows downward through the annulus between thimble and capsule shroud, turns at the bottom of the thimble, and flows upward through the annulus between capsule shroud and pressure vessel. It then flows through the inner cooling channel of the extension tube. The cooling water ultimately leaves the extension tube through downward oriented holes near the connection box into the reactor pool. The overall design is such that cooling of the in-core part of the capsule is guaranteed either when the capsule is inserted or removed from the thimble. A dummy plug will be placed in the thimble in order to guarantee a forced coolant flow when the thimble does not contain an experimental capsule. A schematic drawing of the irradiation rig positioned in the thimble is given in Fig. 5.

Detailed assembly drawings including parts lists and material specifications have been given in ECN drawings 1.085-11-TEW 2 and 1.085-11-TEW 3.

4.2 External Installations

For the performance of the irradiation experiment, several external installations are necessary. Some of these, such as the cooling water system and the gas service system, are standard devices in the HFR. The high-pressure helium system is an installation specially designed for these Zircaloy-cladding creepdown studies.

Fig. 5. Schematic drawing of HOBBIIE capsule installed in thimble.
(Reproduced with permission of ECN.)

4.2.1 Cooling system

The cooling water system consists of three circulation pumps. These pumps are installed in parallel and have a total capacity of 40 m³/hr (176 gal/min) and a pressure head of 60 m (197 ft) of water. The suction sides of the pumps are connected with the reactor pool; the pressure sides of the pumps are connected with a header provided with 12 connections for the cooling water supply for the experiments. The cooling water of each experiment is individually controlled and flows from the experiment directly into the reactor pool.

A flow sheet of the cooling water system being used for the HOBBIE experiments is given in Fig. 6. A detailed flow pattern of the coolant inside the irradiation rig is shown in Fig. 5. With a cooling water flow of 2 m³/hr (8.8 gal/min), the temperature rise will not exceed 7.5°C (13.5°F) (see Appendix A).

4.2.2 Gas supply system

Figure 6 shows a flow sheet of the complete gas system for the HOBBIE irradiation. This flow sheet can be divided into five different parts: (1) specimen system, (2) high-pressure system, (3) gas service system, (4) nitrogen system, and (5) compressed air system. The first number in the code used in the flow sheet refers to the above-mentioned systems. Along with these systems, Fig. 6 shows the cooling water system that is considered as the sixth external installation (see Sect. 4.2.1). Valves which are normally closed during steady-state conditions of the irradiation test are black. A survey of the materials used in the gas supply system is given in Appendix B.

In the following sections each system will be discussed separately.

4.2.2.1 Specimen gas system. The specimen system is a low-pressure system under normal operating conditions. However, this system has been designed to withstand up to 3 MPa (425 psig) because of the risk of a leak from the high-pressure system through the specimen into the specimen system. Because the specimen must be checked for leakage, the pressure inside the specimen must be monitored. In order to reduce the pressure range of contact manometer 1PIA1, a buffer tank (No. 1) has been installed in the specimen system. In this way the contact manometer is normally operating at about 10% full scale and can withstand the equalized pressure should a leak develop. For pressure calculations see Appendix C.

4.2.2.2 High-pressure gas system. In order to achieve the high pressure for the creepdown experiment, a separate pressurization gas system has been installed. Briefly, this system consists of two helium cylinders (one serves as a spare), a compressor, an accumulator tank, a pressure regulator, and a pressure equalizer.

The gas cylinders are connected with the gas section of an air-driven gas booster compressor (trademark HASKEL, type AC-30) by means of valve 2V8. This valve is part of an automatic pressure control system, together with valve 2V7 and the valve in the air section of the compressor 5V1. The valves 2V7 and 2V8 are high-pressure ball valves that are remotely controlled whereas 5V1 is a magnetic valve that is open only when activated. The valves are actuated by means of pressure controller 2PIC5 as described in Sect. 4.3.1.2.

The HASKEL gas compressor is an automatic cycling device cycled by a non-detented, unbalance spool that is alternately pressurized and vented by a pilot system. The gas compressor is designed to run without lubrication to supply gas free of hydrocarbon contamination. The exhaust air from the driving system is used to cool the gas barrel of the compressor. The model number 30 identifies the ratio of the air drive area to the gas piston area, indicating the approximate ratio of maximum gas discharge to air drive pressure.

The compressed helium is accumulated in a high-pressure cylinder. This accumulator tank (No. 2) is also provided with two parallel relief valves, 2V20 and 2V21, as a failsafe system, while valves 2V22 and 2V23 have been installed for venting and evacuating this part of the pressurization system.

Fine pressure regulation is achieved by regulator 2V6. The feed and bleed lines of this regulator are provided with filters 2F1 and 2F2 to protect the apparatus from dust, etc.

In order to separate the gas volume in the capsule pressure vessel from the pressure supply system, a pressure equalizer has been designed and built in the gas supply line after the pressure regulator. This pressure equalizer forms a mechanical separation between the compressor and the capsule parts of the pressure supply system. The separation, achieved by a spring-loaded piston in a cylinder, can be bypassed for initial pressurization with the help of metering valve 2V3 and ball valve 2V4. In this way the pressure in the capsule is kept constant but the gas volume that could leak out due to a failure is kept to a minimum. (Note: The high-pressure helium system will always leak somewhat because of gas diffusion through materials. It will also be difficult to get fully tight valves and fittings.)

If the pressure in the capsule becomes higher than the pressure in the accumulator tank, check valve 2V5 protects the pressure supply system from any return of contaminated gas. Further, the capsule can be disconnected from the pressure supply system by means of ball valve 2V2. Thus the pressure supply system can be kept under required helium pressure.

Ball valve 2V1 connects the inlet and outlet lines and is normally closed; it is only used under certain service circumstances. In case of overpressurization of the capsule, the pressure is relieved

automatically by valves 2V11 and 2V12, and the amount of contaminated gas is then dumped into buffer/decay tank No. 3. (See Appendix C for pressure equalization calculations.)

The two relief valves, 2V18 and 2V19, have been installed as a failsafe system for the automatic dumping system. The pressure in the capsule can be slowly relieved by means of metering valve 2V3 and ball valve 2V30. Because it is required that a set of double valves be installed between a high- and a low-pressure system or atmospheric surroundings (see ref. 2), valves 2V15, 2V16, and 2V31 have been built in as a second barrier after valves 2V11, 2V12, and 2V30. Check valve 2V17 is mounted in the vent line to the buffer tank to prevent the return of contaminated gas should a dump occur due to leakage in the capsule.

The capsule will be connected with the gas service system before and after high-pressure conditions. During this time valves 2V13 and 2V14 are closed to achieve gas circulation for capsule flushing.

4.2.2.3 Gas service system. As described in the previous section, the capsule is connected with a gas service system when there is no requirement for high pressure. For this purpose a standard gas service system in the HFR is used for the following:

- to supply pure helium, nitrogen, or neon (valves 3V17, 3V18, and 3V19) to a maximum gas pressure of 1 MPa (142 psig) by means of regulator 3V16,
- to permit composing a gas mixture in a mix tank (valve 3V20),
- to connect with the HFR off-gas system for dumping of contaminated gas (valve 3V23),
- to connect with a vacuum pump (valve 3V22),
- to connect with a dump/decay tank for highly active or contaminated gas (valve 3V21).

In order to avoid high-pressure pulses in the general off-gas system or dump tank, the gas service system of HOBBIIE is provided with a private buffer/decay tank (No. 3). Therefore valve 3V21 is not used. Because the gas service system is a low-pressure system with a maximum permissible pressure of 1 MPa (142 psig), both vent lines to and from the service panel are provided with a double set of relief valves (3V3, 3V4, 3V5, and 3V6). Also the buffer/decay tank is provided with relief valves (3V7 and 3V8) at 1 MPa (142 psig) relief pressure as a failsafe system for gas dumping in case the tank is already filled. The buffer tank can be drained to off-gas by means of ball valve 3V2. Valve 3V1 has been built in the system as a high-pressure barrier in case of specimen failure. For allowable pressures in the buffer tank see Appendix C.

4.2.2.4 Nitrogen system. The instrumentation hose of the HOBBIE rig is kept under 0.2 MPa (28 psig) of nitrogen pressure in order to keep it free of moisture. This nitrogen is obtained from a common manifold and enters the hose in the connection box at the bottom side by means of pressure regulator 4V1 and membrane valve 4V2. The exhaust of the hose is at the top side through valve 4V3. In this way the hose can be flushed through to the off-gas system.

4.2.2.5 Compressed air system. The compressed air necessary for driving the gas booster pump is also obtained from a common manifold in the HFR by means of remotely controlled valve 5V1. The maximum working pressure is 0.8 MPa (114 psig) and the pumping ratio is 30 to 1. Hence a maximum gas discharge of 24 MPa (3400 psig) can be derived when the compressor is in use.

4.3 Instrumentation

The main instrumentation components of the HOBBIE irradiation capsule are also shown in the flow sheet of the gas and cooling water system (Fig. 6). As mentioned, the first number in the code used in the flow sheet refers to the external installation as described in the previous sections.

The first letter in the code refers to the measured unit within the system:

D - displacement	(mm)
F - flow	(m ³ /hr)
P - pressure	(MPa)
T - temperature	(°C)
V - valve	(-)
Q - power	(watts)

The second and following letters refer to the action of the measuring equipment:

A - annunciator
C - controller
I - indicator
R - recorder

The second number is a sequence number within the measured unit. The complete instrumentation of the irradiation experiment can be divided into three parts: (1) measuring systems, (2) data registration and control, and (3) safety instrumentation.

A block diagram of all instrumentation components is given in Fig. 7. The complete alarm and interlock system, which is presented in Fig. 8, will be discussed in the next sections where the different measuring systems are described.

Fig. 8. Tabulation of alarm and interlock system for HOBBIE irradiation capsules.
(Reproduced with permission of ECN.)

4.3.1 Measuring systems

For the operation of the Zircaloy fuel-cladding irradiation capsules, three types of measuring systems are in use: (1) scientific instrumentation, (2) process instrumentation, and (3) neutron dosimetry provisions.

4.3.1.1 Scientific instrumentation. In order to define the strain behavior of the cladding as a function of external pressure, temperature, and neutron fluence, the strain must be monitored continuously as the cladding deforms. The scientific instrumentation includes a device for monitoring deformations based on an eddy-current system that measures the distance (lift-off) between a coil and a flat-plate follower that is lightly spring-loaded against the specimen surface.

Twenty of these probes will be applied simultaneously in order to obtain a good definition of the shape of the deformation zone (1DR1-24). Reproducible measurements from 0 to 1.5 mm (0.06 in.) of lift-off have been made with a resolution of approximately 0.013 mm (0.0005 in.). The probe orientation is shown in Fig. 2. The probes extend over a 50.8-mm (2-in.) length and are spaced 6.35 mm (0.25 in.) apart. The coil and probe arrangement within the aluminum cylinder in the pressure vessel of the capsule is shown in Fig. 9.

Fig. 9. Arrangement of coil and probe with respect to test specimen.

Figure 10 shows a block diagram of the eddy-current instrument. The oscillator and power amplifier supply a 2-MHz sinusoidal signal to 24 coils including a reference coil. The impedance of each coil (and therefore the voltage developed across the coil) is affected by the spacing between coil and probe. The analog multiplexer, controlled by a micro-computer, passes the 2-MHz signal that is developed across the coil to a differential amplifier. This module amplifies the difference between that signal and a signal from the reference coil. The difference signal is converted to a direct-current voltage and, by command of the micro-computer, is converted to a digital signal. This signal is then read into the micro-computer through a parallel interface integrated circuit and is converted into actual displacement by the micro-computer. The displacement and the channel number are recorded and the multiplexer is switched to the next channel.

The instrumentation discussed above has been developed by ORNL (refs. 3 and 4).

The independent irradiation variables such as temperature and pressure are measured by conventional methods. The temperature is measured by means of eight Chromel/Alumel thermocouples (1TRA1-2 and 1TR3-8) that are located in the mandrel of the Zircaloy specimen. The cold junctions of the couples are electronically compensated with the help of an Acromag reference box. The pressure in the capsule is measured by means of two pressure transducers, 2PCRA1 and 2PCRA2 (trademark Bell & Howell), type 4-801), with an accuracy of $\pm 0.5\%$. The transducers are mounted out-of-pile in the two gas lines directly after the end of the instrumentation hose.

All millivolt signals from thermocouples and pressure transducers are connected with a matrix pinboard from which the signals are available for data retrieval and/or process control.

4.3.1.2 Process instrumentation. The process instrumentation can be divided into (1) the temperature and pressure control system, and (2) the experiment safety system.

The required irradiation temperature is achieved primarily by the nuclear heat production in the capsule. In order to reach the required temperature, the remaining heat is supplied by an electric heater element inside the mandrel of the specimen (see Sect. 4.1.1). The maximum heat discharge of 750 W is manually controlled by means of a variable transformer (1QC1), which is provided with an ampere- and a voltmeter.

Pressure control of accumulator tank No. 2 is achieved automatically by means of controller 2PIC5. The control system consists of a contact manometer, a switch unit, and valves 2V7, 2V8, and 5V1. The design is such that the contact manometer activates the switch unit at a preset low- and high-pressure limit. This switch unit, which operates the valves, has two modes of operation: one for initial pressurization and one for pressure control.

Fig. 10. Block diagram of computer-controlled eddy-current instrument for measuring creep for an array of coils.

In case of initial pressurization, valves 2V7 and 2V8 are opened and the compressor is started by means of a push button that activates valve 5V1. When the complete system is under pressure and the high-pressure contact is actuated, the switch closes 5V1 and 2V8. Thereafter, one low-pressure contact has no effect and the switch unit has to be set in the pressure control position to maintain operating pressure. In this position, a low-pressure contact closes 2V7, opens 2V8, and starts the compressor while a high-pressure contact stops the compressor, closes 2V8, and opens 2V7.

The experiment safety system protects the capsule from excessive temperatures and pressures by means of the millivolt Alarm Unit (mVAU). One mVAU consists of a double annunciation unit with two identical electronic circuits. In each circuit the measured (millivolt) signal is conditioned and compared with a preadjusted high and low annunciation level. If the measured signal exceeds one of these levels, a high or low alarm signal is generated. The alarm signals can be used for experiment control purposes and for reactor safety interaction. In the last case, all the alarm signals of the mVAUs in use for the experiment are guided to a coding unit. In this coding unit three different safety signals are generated, namely "Alarm" (AL), "Automatic Power Decrease" (APD), and "Scram" (SC), according to the function of the alarm signals and the way they are connected (one out of one or two out of three systems, etc.).

For normal temperature control one thermocouple (1TRA1) is used and connected with a mVAU. A high and low temperature alarm indication at the local mVAU panel and in the control room warns the experiment operator of a need for heater power adjustment. A second thermocouple (1TRA2) is also connected with a mVAU, but its high annunciation level is adjusted higher than that of 1TRA1. The low level is not in use. Besides the normal indication this alarm signal switches off the heater power.

In order to protect the capsule from overpressurization, the pressure is relieved by automatically opening valves 2V11 and 2V12 if the pressure exceeds a preset value. In this instance, the signal of pressure transducers 1PCRA1 and 1PCRA2 are connected, by way of the pinboard, with a mVAU. Only the high alarm signal is used for actuation of the corresponding valve and for indication in the control room. Both valves can also be operated manually.

For pressures that are not registered (nonscientific instrumentation) and that are manually controlled, protection is achieved by means of contact manometers (trademark Econ-Wika, type Tourdon, accuracy, 1% full-scale). These manometers are provided with minimum and maximum cutoff slide contacts. When current obtained from a Contact Alarm Unit (CAU) is cut off, an alarm signal is generated in the same way as it is in the mVAU. All alarm signals from the mVAUs and CAUs are connected with the same coding unit either for experiment alarm in the control room or for APD.

The manometers of 1PIA1, 2PIA3, 2PIA6, 3PIA1, 3PIA2, and 6PIA1 are connected with the CAUs in the aforementioned way and will be discussed below, excluding 6PIA1, which is discussed in Sect. 4.3.3.

- 1PIA1 annunciates at high alarm for check on leakage from the pressure vessel to the specimen,
- 2PIA3 annunciates at high and low alarm for manual pressure control. It will be clear that this high alarm is adjusted a bit lower than the high alarm of 2PCRA1 and 2PCRA2.
- 2PIA6 annunciates at low alarm to warn of the need to change gas cylinders,
- 3PIA1 annunciates at high alarm to warn of a full buffer/decay tank,
- 3PIA2 annunciates at low alarm when the capsule is connected with the gas service system (before and after irradiation).

All actions of annunciators with the adjusted - maximum - limiting values are given in Fig. 8, which illustrates the alarm and interlock system. The remaining pressure indicators are installed for service purposes:

- 2PI4 warns against equalizing the pressure when the pressure at the primary side of the equalizer is lower than the secondary side,
- 3PI3 is used during vacuum check for capsule helium purge,
- 4PI1 monitors pressure in instrumentation hose.

4.3.1.3 Neutron dosimetry provisions. The neutron fluence of the specimen must be measured after the irradiation has been finished. Therefore activation detectors are placed in aluminum tubes [1.0-mm (0.039-in.) OD, 0.6-mm (0.024-in.) ID, 180-mm (7.087-in.) length], which are mounted in the annulus between the pressure vessel and the capsule shroud. The minitubes are diametrically mounted on the outside of the pressure vessel at the location of the test specimen. They are attached by means of stainless steel wires that are spot welded on the surface of the pressure vessel. Each minitube contains four iron detectors with a mass of about 2 mg and four nickel/1% cobalt detectors with a mass of about 0.2 mg. The detectors are separated with the aid of aluminum filler pieces [0.5-mm (0.020-in.) OD].

Thus the neutron fluence outside the pressure vessel is measured. To obtain the fluence values of the specimen the measured fluences must be extrapolated with the help of the measured values that will be obtained from the irradiation of the mockup capsule of HOBBIE.

4.3.2 Data registration and control

The data registration and control system consists of two different parts.

4.3.2.1 Scientific data registration. The first part consists of the registration of the displacement values obtained from the deformation monitoring devices. Operating fully independent of the rest of the instrumentation, the data registration system is controlled by the eddy-current instrument. The micro-computer in this instrument provides all of the needed control signals to operate the multiplexer, the analog-to-digital convertor, the printer, the teletype, and the paper tape punch. The system can operate in several different run and calibration modes, controlled by the front panel switches or by the clock. A block diagram of this system is incorporated in Fig. 10.

4.3.2.2 Process data registration. The second part is the registration of temperature and pressure signals that are available at the matrix pin-board. Registration and control is achieved by the data logging system shown in Fig. 7. The scanner, controlled by a programmable scanner controller, passes the analog signal to a digital voltmeter. This module converts the analog signal to a digital signal. Before the data scan starts, the date and time are printed and punched. Thereafter the measured signals are printed sequentially and punched, together with the corresponding channel number.

Direct data acquisition of the punched paper tape is possible with the help of a standard packet of programs available on the CDC 6600 computer (ref. 5). The result of this acquisition will be a statistical survey of the irradiation performance and a plot of the measured units versus irradiation time. As backup to the data registration, an analog recorder (Phillips 8235) is also connected with the matrix pinboard.

4.3.3 Safety instrumentation

This instrumentation has a direct interaction with reactor operation. As can be seen in the failure analysis (see Sect. 10), failure of the coolant flow requires an automatic power decrease from 45 to 1 MW of reactor power. The coolant flow is monitored by means of a pressure measurement between the secondary side of the flow controller and the cooling water inlet to the thimble. In case of a cooling water blockage the pressure will increase, but a rupture anywhere between circulation pumps and thimble will cause the pressure to decrease. An Econ-Wika manometer with a minimum and maximum current cutoff slide contact is used as the pressure sensor. With the help of a CAU, one alarm signal is generated for either high or low pressure. In the coding unit this signal is transferred to a safety signal that produces an APD.

The pressures by which the contact manometer 6PIA1 annunciates are found experimentally. At normal flow operation of 2 m³/hr (8.8 gal/min), the pressure will have a certain value. The minimum annunciation contact

is adjusted at the pressure corresponding with a flow of $1 \text{ m}^3/\text{hr}$ (4.4 gal/min), whereas the maximum annunciation contact is adjusted 50% above the normal operating pressure.

The APD on cooling water failure is the only safety instrumentation used for the HOBBIE irradiation (see also Fig. 8).

4.4 Utilities

As mentioned in Sect. 4.2 several external installations are provided for the operation of the HOBBIE irradiation. This section discusses the utilities necessary to run the external installations.

4.4.1 Electrical power supply system

Though most of the electronic equipment is connected with the 220-V 50-Hz supply system, a few exceptions have been made for the instrumentation provided by ORNL that are based on 110-V 60-Hz supply system. For the heater controller and the eddy-current equipment, a 2-kVA transformer is available. The change in frequency has no influence on the behavior of this instrumentation.

4.4.2 Gas supply system

The gases available at the gas service system (panel K on the second experiment balcony) are obtained from the manifold that is connected to a group of accumulator tanks in the high maintenance hall of the HFR. Since the experiment is only connected with the gas service system for conditioning the capsule before and after irradiation, the consumption rate of gas is negligible. This negligible rate also holds true for the nitrogen consumption for the instrumentation hose because of the steady-state conditions in the hose. The nitrogen is obtained from the same group of accumulator tanks whose condition is monitored from the reactor control room.

4.5 Waste System

The HOBBIE irradiation tests are connected with the off-gas system for waste retrieval. In order to avoid high-pressure pulses in the off-gas system for other permanently connected experiments, HOBBIE vents its wasted helium gas into a buffer tank (No. 3 in the flowsheet of Fig. 6). Under normal operating conditions, no off-gas has to be vented. In case of finishing the irradiation or under special circumstances during irradiation, the high pressure is always reduced to at least 1 MPa, which is the relief pressure of the buffer tank before final waste retrieval takes place.

4.6 Shielding

Because the HOBBIE capsule does not contain any fissile material or objects that can disappear out of the capsule in a highly active condition, no special shielding provisions are required for the performance of the experiment.

5. CHARACTERISTICS

5.1 Nuclear Characteristics

As mentioned before, the irradiation of the HOBBIE capsule will be performed in core position H2. Since the instrumentation proposal was written in March 1977, the neutron fluxes and nuclear heating have changed somewhat because of small changes in core configuration.

The relative axial distribution of the fast and thermal neutron flux density is presented in Figs. 11 and 12. These curves have been obtained from ref. 1. The relative nuclear heat distribution has been derived from recent measurements in graphite probes and is given in Fig. 13 (see ref. 6).

The best estimated average values of these fluxes at the center line of core position H2 are

Average fast neutron flux density: 0.5×10^{18} neutrons/m²·sec
($E_n > 1.0$ MeV)

Average thermal neutron flux density: 0.9×10^{18} neutrons/m²·sec

Average nuclear heating in graphite: 2.0 W/g

Because of self-shielding of the capsule and the fact that it will be irradiated in a reflector position in the core, a certain horizontal flux depression has to be expected across the capsule. The value of this flux depression will be measured with the help of the mockup capsule, which is provided with enough activation detectors to measure the local flux depression at different heights in the capsule. Nuclear calculations have also been executed on the CDC 6600 computer (refs. 7-9). For these calculations a capsule model was developed utilizing the actual construction materials with appropriate physical properties at working temperature. Out of these calculations the fast flux depression across the specimen amounts to about 11.5% on the average values mentioned above. The thermal flux depression is less than 1%.

Fig. 11. Relative axial distribution of the fast neutron ($E_n > 1.0$ MeV) flux density in HFR core position H2. (Reproduced with permission of ECN.)

Fig. 12. Relative axial distribution of the thermal neutron flux density in HFR core position H2. (Reproduced with permission of ECN.)

Fig. 13. Relative axial distribution of nuclear heating in graphite in HFR core position H2. (Reproduced with permission of ECN.)

5.2 Thermal Characteristics

The HOBBIE capsules contain no fuel, and therefore heat generation is due only to gamma heating and electrical heat. An axial profile of gamma heating in graphite in core position H2 is presented in Fig. 13. The relative gamma heating rates in the materials used in the HOBBIE capsules are presented in Table 1.

Table 1. Relative gamma heating rates for materials used in HOBBIE capsules

Material	Relative gamma heating rate
Graphite	1.0
Aluminum	1.2
Beryllium	1.0
Nickel	1.4
Stainless steel	1.3
Zircaloy	1.25

The radial temperature profile was calculated by incorporating the data from Fig. 13 and Table 1 into the one-dimensional heat transfer code, GENGTC (ref. 10). The resulting test specimen horizontal midplane temperature due to gamma heating only (no electrical heat) is 223°C (433°F). The maximum desired test specimen temperature is 427°C (800°F). By adding electrical heat at the rate of 24.6 W/cm (0.75 kW/ft) the GENGTC code predicts a test specimen temperature of 427°C (800°F). The radial temperature profile for cases both with no electrical heat and with 360 W electric heat is presented in Fig. 14. Both of these temperature distribution plots are for the elevation equal to the horizontal midplane of the test specimen which is located 50 mm (1.97 in.) below the horizontal midplane of the reactor. Figure 13 shows the maximum gamma heat to exist at that elevation; therefore, the temperature distributions in this figure are for the hottest part of the capsule.

For all heat transfer calculations, a surface heat transfer coefficient of 0.85 W/cm²·°C (1500 Btu/hr·ft²·°F) was used. This coefficient is considered conservative when compared to the calculated heat transfer coefficient of 0.99 W/cm²·°C (1741 Btu/hr·ft²·°F). For the calculation used to obtain this heat transfer coefficient, see Appendix D.

An axial distribution of heat flux at the outside of the pressure vessel is shown in Fig. 15.

The maximum coolant temperature rise will be 7.5°C (45.5°F) as shown in Appendix A.

The only means of temperature control in the HOBBIE capsules is the electrical heater. The expected maximum heater power requirement will be the 370 W needed to obtain the highest desired specimen temperature of 427°C (800°F). To produce 370 W of power the heater will require an electric current of 18.2 A. The heater controller is capable of delivering a maximum of 30 A; at this rate the heater will produce about 980 W. Figure 16 shows that the heater is capable of controlling the test specimen temperature at the rate of about 45°C/100 W. Therefore the specimen temperature control margin over the maximum output range of the heater is 400°C (824°F).

5.3 Stress Characteristics

The stress analysis necessary for the HOBBIE irradiation can be divided into three parts: (1) the main pressure vessel of the capsule, (2) the pressure equalizer in the pressurization system, and (3) other components.

The pressure vessel analysis will be discussed in detail below. Calculations concerning the pressure equalizer are described in ref. 11. Other components are those items in the high-pressure system that are bought commercially, such as valves and tubing. Stress information for these components is contained in Appendix B.

ORNL - DWG 77-17484

Fig. 14. Calculated radial temperature profiles in the HOBBIE capsule at the test specimen horizontal midplane.

Fig. 15. Axial distribution of heat flux at the outside surface of the pressure vessel.

Fig. 16. Test specimen temperature at specimen horizontal midplane vs total electrical heater power.

The stress analysis of the pressure vessel has been performed to determine the adequacy of the vessel under the applicable rules of the ASME Boiler and Pressure Vessel Code, Sect. III, for Class I components (ref. 12). In accordance with the requirements of the code, the vessel was analyzed at the design conditions listed below.

Material	316 SS
Operating pressure	20.7 MPa (3000 psi)
Design pressure	24.1 MPa (3500 psi)
Heat transfer coefficient	0.85 W/cm ² ·°C (1500 Btu/hr·ft ² ·°F)
Water temperature at half coolant flow	
inlet (bottom of vessel)	50°C (122°F)
outlet (top of vessel)	65°C (150°F)

In addition to these design conditions, an axial distribution of the heat flux at the inner wall of the pressure vessel is given in Fig. 17. The distribution of this flux is also calculated with the help of the GENGTC computer code (see Sect. 5.2) using the inside capsule geometry, the materials, the nuclear heat, and the specific heat rates as input data.

The stress analysis was performed using the Finite Element Analysis - Temperature and Stress (FEATS) computer code. This program can determine, by finite element analysis, the stress and temperature distribution for an axisymmetric body. A detailed description of the procedure used by the program can be found in ref. 13. The accuracy and reliability of the FEATS program have been demonstrated, and the program is widely used at ORNL for similar problems.

Fig. 17. Axial distribution of heat flux at the inside surface of the pressure vessel.

Because there are no mechanical loads imposed on the vessel, the only stresses under consideration are those originating from the internal design pressure of 24.1 MPa. In order to get an overall view of the stresses appearing in the pressure vessel, preliminary calculations were determined by the FEATS computer code. A breakdown of the vessel geometry into finite elements was set up in the R-Z plane of the complete capsule. From these calculations, which are described in ref. 11, it was determined that the maximum stress is located in the junction of the flat bulkhead and the shell. Consequently, this part of the capsule has been calculated in detail, and the breakdown of this vessel part into finite elements is shown in Fig. 18.

Fig. 18. Finite element breakdown used by the FEATS program in the region where the flat bulkhead joins the shell of the HOBBIE capsule pressure vessel.

To assist in interpreting the results of the FEATS analysis, a brief summary of the nomenclature and calculation procedures is helpful. The vessel geometry is aligned along a chosen set of orthogonal components [tangential (t), longitudinal (l in Code, z in deck), and radial (r)]. Stress components in these directions are calculated and identified on the computer output as follows (with the corresponding code nomenclature in parenthesis):

$$\left. \begin{array}{ll} \text{R STRESS } (\sigma_r) \\ \text{Z STRESS } (\sigma_z) \\ \text{T STRESS } (\sigma_t) \end{array} \right\} \text{Direct Stresses}$$

$$\text{RZ STRESS } (\tau_{lr}) \quad \text{Shear Stress}$$

Since the body is axisymmetric, τ_{lt} and τ_{rt} are zero. Thus the T STRESS (σ_t) is equivalent to one of the principal stresses (σ_1). The other two principal stresses (σ_2 and σ_3) can be found by using the R STRESS, Z STRESS, RZ STRESS, and Mohr's Circle technique. This operation is done internally by the deck and the resulting values are printed out as MAX STRESS and MIN STRESS (σ_2 and σ_3). Thus, one can calculate:

$$S_{12} = \text{T STRESS } (\sigma_1) - \text{MAX STRESS } (\sigma_2)$$

$$S_{23} = \text{MAX STRESS } (\sigma_2) - \text{MIN STRESS } (\sigma_3)$$

$$S_{31} = \text{MIN STRESS } (\sigma_3) - \text{T STRESS } (\sigma_1)$$

The stress intensity (S) is the largest absolute value of S_{12} , S_{23} , and S_{31} .

At design conditions, the stress intensity (S) includes both the membrane and bending components, and thus is equivalent to $P_L + P_b$ as defined by the code. At any section the bending stresses can be separated by the following procedure. Recognize that the stress due to pure bending at the node on the inside wall of the vessel will be equal to that at the node on the outside wall at the same section. The total stress at both nodes (which is known from the computer output) is the sum of the membrane plus bending components, where the membrane component at the inside wall will be higher than that at the outside wall by an amount equal to the internal pressure.

For normal operating conditions, secondary (Q) and peak (F) stresses are caused by temperature gradients, discontinuities, etc. The stress calculated from the FEATS output is equivalent to $P_L + P_b + Q$, and because it is this sum which is limited by the code, no breakdown of this sum into its component stresses is required. Peak stresses can be calculated using stress concentration factors, and the sum $P_L + P_b + Q + F$ can be found and compared to its allowable limit.

Highlights of the stress analysis for both design and normal operating conditions are detailed in Table 2. As indicated in the tabulation for design conditions, which considers only pressure loads, the highest stresses are in the cylindrical shell. However, there is a considerable margin between the maximum calculated stress intensity, 141 MPa (20,400 psi), and the allowable value, 186 MPa (27,000 psi). At normal operating conditions, which include thermal effects as well, the most severe stresses are found in the region near the corner formed by the junction of the flat head with the cylindrical shell. Again, the highest calculated stress intensity, 250 MPa (36,700 psi), is well below the allowable value of 300 MPa (54,000 psi).

In addition to the conventional stress analysis, a fatigue analysis was conducted using an applicable code procedure (ref. 12). Twenty start-up cycles [simultaneous pressure and temperature cycles from 0 MPa (0 psi), 21°C (70°F), to 24.1 MPa (3500 psi), operating temperature] were considered and these were assumed to be the only significant cycles traversed. Results of the fatigue analysis, also presented in ref. 11, indicate no problems due to the specified cyclic history.

The sensitivity to stress corrosion cracking has also been taken into account. Although the austenitic stainless steel used is extremely sensitive to stress corrosion cracking in an enriched environment of oxygen and/or chloride, the chance of this type of failure is negligible because the capsule is not subjected to these operating conditions. If the stress corrosion failure does occur, however, it will undoubtedly lead to a penetration of the capsule wall and a leakdown of internal pressure without rupture.

The failure modes of particular interest to the reliability and safety of the pressure vessel are those wherein a fracture occurs rapidly and leads to fragmentation or major deformation of the capsule. By reference to the theory of fracture mechanics, it was determined that with a calculated stress intensity factor K_{Ic} of 174 MPa $\sqrt{m}$ (174 ksi $\sqrt{in.}$) and a flaw size of 4 mm (0.16 in.) the fracture stress amounts to 1725 MPa (245 ksi). More detailed analysis concerning this fracture analysis can be found in Appendix E.

5.4 Reactivity Characteristics

As a consequence of the reloadable character of the experiment during reactor operation, reactivity effects will appear when inserting the capsule in the core and when removing it from the core.

Using the same nuclear calculations mentioned in Sect. 5.1, the reactivity of the experiment has been calculated in the following configurations:

Thimble loaded with dummy plug	74 pcm (10.5 cents)
Thimble empty (i.e., filled with water)	66 pcm (9.2 cents)
Thimble loaded with HOBBIE capsule	117 pcm (16.2 cents)

Table 2. Highlights of stress analysis results with comparisons to allowable stress intensity

Location	Node point	Calculated stress (psi) ^a				
		Design conditions			Normal operating conditions	
		P_m (Allowable ^b) $S_m = 18,000$	P_L (Allowable) $1.5S_m = 27,000$	P_b	$P_L + P_b$ (Allowable) $1.5S_m = 27,000$	$P_L + P_b + Q$ (Allowable) $1S_m = 54,000$
Shell	233-236 ^c	16,330	16,330	850	17,180	20,900
Flat head	184-446	1,530	1,530	1,650	3,180	16,900
Spherical head	17-20 ^c	8,010	8,010	560	8,570	9,200
Shell at flat head junction, inside wall	172		15,400	3,600	19,000	35,600
Shell at flat head junction, outside wall	179		4,700	3,600	8,300	16,700
Shell, 1/4 in. down from junction, inside wall	108		12,370	130	12,500	25,300
Shell, 1/4 in. down from junction, outside wall	115		9,070	130	9,200	27,600
Shell, 1/2 in. down from junction, inside wall	53		17,410	390	17,800	23,600
Shell, 1/2 in. down from junction, outside wall	60		13,110	390	13,500	21,600
Shell, 1 in. down from junction, inside wall	13		19,620	780	20,400	15,200
Shell, 1 in. down from junction, outside wall	16		14,620	780	15,400	24,800
Flat head at shell junction, inside wall	192		12,980	3,220	16,200	11,800
Flat head at shell junction, outside wall	450		3,030	3,220	6,250	14,700
Flat head 1/2 in. from vessel G_L , inside wall	184		1,630	1,650	3,280	27,500
Flat head 1/2 in. from vessel G_L , outside wall	446		1,430	1,650	3,080	6,300

^a100 psi = 0.6895 MPa.

^bThe allowable stress is based on an S_m evaluated at 500°F. Since much of the vessel operates at a considerably lower temperature, the allowable stresses are conservative.

^cThe shell and spherical head data are based on a different model and the node point numbers do not correspond to Fig. 18.

From these calculations it can be concluded that the largest reactivity addition is given during removal of the capsule from the thimble, that being 51 pcm (7.1 cents). With a control rod worth of 55 pcm/cm (19.6 cents/in.) at a 60-cm setting, the displacement of the control rod will be 0.93 cm. If no control rod movement took place at all, the reactor period would amount to 150 sec according to Fig. 19. The control rod velocity is ~ 0.2 cm/sec, which is equivalent to 11 pcm/sec (1.55 cents/sec). In case the control rod is controlled automatically the experiment can be removed in 5 sec. If the effective length of the capsule is 60 cm, then the maximum permissible velocity of capsule removal is 12 cm/sec.

Since the maximum hoisting speed of the bridge winch can be limited to 2.2 cm/sec, it may be concluded that loading or unloading the capsule in the core can be performed at 45 MW reactor power.

Because the calculation model is a simplified version of the capsule, the real reactivity can be measured empirically with the help of the mockup capsule. In that case the control rod movement can be monitored directly during loading or unloading of the thimble.

Since the gas volume in the capsule is very small (about 50 cm³), the reactivity effect can be neglected were this volume to be filled completely with water in case of leakage.

5.5 Radioactivity Characteristics

The HOBBIE capsules contain no fissionable material and therefore there will be no radioactive fission products. The only radioactivity that will be present will be the induced activity of the capsule structural components that are mostly stainless steel and aluminum, and from the impurities in the helium used to pressurize the capsule. The activity to be expected from the structural components will be similar to that experienced from previous irradiation experiments and will not be calculated here.

The inventory of gaseous activity will be the result of activation of impurities in the helium and the production of tritium through a neutron/proton reaction with the small amount of ³He that is present in helium. Listed in Table 3 are the estimated impurity concentrations, and in Table 4 is the calculated inventory of possible resultant radioactive activation products. The calculations used to obtain these values are presented in Appendix F.

Fig. 19. Reactor period vs reactivity change for the HFR. (Reproduced with permission of ECN.)

Table 3. Estimated impurity concentrations
in helium used to pressurize
HOBBIE capsules

Impurity	Concentration (ppm)
^3He	1.37
H_2O	10
N	20
O_2	5
H_2	5
Ne	10

Table 4. Calculated inventory of activation products
in high pressure helium system (see Appendix F)

Radioactive isotope	Total activity in capsule (Ci)
^3H	4.0
^{14}C	9.4×10^{-5}
^{16}N	2.9×10^{-4}

6. FABRICATION, INSPECTION, AND TESTING

The irradiation facility comprises two major assemblies, i.e., the irradiation capsule and the thimble. The irradiation capsule will be manufactured by ORNL; the thimble will be manufactured by ECN. Because of the distinct separation of manufacturing responsibilities, this section will be divided into two subsections.

6.1 Irradiation Capsule

A detailed quality assurance program has been prepared for the ORNL portion of this project (see ref. 14). Some of the detailed inspection and testing procedures that will be performed during assembly of the irradiation capsule include:

- Dimensional inspection of all mechanical parts.
- Material traceability of all mechanical components.
- Dye-penetrant and hydrostatic testing of pressure vessel.
- Dye-penetrant, x-ray (where applicable), and helium leak check of all containment welds.
- Helium leak check of all braze joints.
- Calibration of eddy-current probes.

After arrival of the capsule at the HFR the program will be continued as follows:

- Check straightness of the capsule after transportation.
- Try out the complete rig in the spare thimble with cone clamp lock.
- Test the instrumentation and alarm and interlock system.
- Pressure check entire high-pressure system.

6.2 Thimble

The thimble necessary for the irradiation has been designed at ECN. Its design has been approved by the Experiment Assessment Committee while the operation procedures are discussed in the Technical Review Meeting. The thimble will be manufactured in the Central Workshop of ECN.

The quality assurance program includes:

- Measuring report from independent measuring room. In this report the designed and measured values of critical data are enumerated.
- Check on fixation in dummy passage block of the reactor vessel top lid.
- Measurement of the pressure drop of the cooling water flow with dummy plug. If necessary the flow resistance will be adapted in the cooling water outlet at the top end of the plug to achieve a reasonable pressure in the cooling water system.
- Measurement of the pressure drop of the cooling water flow with the mockup capsule (which is geometrically identical to the pressurized capsule) for adjustment of the annunciation levels of 6PIA1.

7. OPERATION

Although the operation instructions will be written in Dutch in a separate report, a brief outline of the checkout, start-up, and operation is given below:

- After the acceptance tests as described in Sect. 6.1, the instrumentation hose is connected with the irradiation rig. The first step of the checkout is a test to determine if all connections are properly made and if the instrumentation is functioning well (heater control, thermocouple signals, and connections with eddy-current coils).
- A 24-hr pressure test at operation pressure is performed when the capsule is stored in a storage rack in the reactor pool. Possible leakages can be repaired before the capsule is radioactive.
- During the pressure test the eddy-current probes are calibrated by the experiment instrumentation group. The alarm and interlock system is also checked (except 6PIA1, which is checked in the next stage).
- The dummy plug is removed and the capsule positioned in the thimble at 45 MW reactor power. After locking the cone clamp the cooling water flow is adjusted. The annunciation levels of 6PIA1 are then checked and the interlock system is switched from position "test" to "operation" by means of the experiment annunciator unit in the control room.
- The operation temperature is achieved by adjustment of the electrical heater power. The operating pressure in the capsule should be revalued and eventually controlled.

- If the instrumentation is working satisfactorily, the experiment is monitored manually every four hours. If necessary the temperature and/or pressure are controlled and instrumentation is checked for proper functioning. The most important experiment data and data which are not recorded are written on a logsheet, which will contain at least the following:

• Specimen temperature	1TRA1
• Number of the eddy-current probe with the largest deformation	1DRI-24
• Value of the displacement	
• Pressure in the specimen	1PIA1
• Pressure in the capsule	2PCRA1
• Pressure in the buffer tank No. 3	3PIA1
• Pressure in instrumentation hose	4PI1
• Cooling water flow	6FI2

- In case of specimen collapse, or when a certain displacement has been reached, the ending of the irradiation is discussed with the project engineer.
- When the experiment is finished, the heater element is switched off and the pressure relieved. The APD on cooling water flow is bypassed and, in cooperation with the control room, the capsule is removed from the thimble and stored in a storage rack for irradiated experiments. Then the dummy plug is replaced in the thimble.
- Possible leakages in the specimen or high-pressure system will be intensively checked according to the guidelines mentioned in the operation instructions.
- The project engineer is contacted in cases of leakage or failure of the heater element. Also, problems with the eddy-current instrumentation are directly reported.

As mentioned in the introduction of this section, the above guidelines are a summary of the operation report. An intensive discussion of all handling operations is, however, beyond the purpose of this report.

8. HANDLING, TRANSPORT, DISMANTLING, AND DISPOSAL

The handling of the capsule actually starts after the assembling of all capsule parts. Because the capsule will be manufactured at ORNL and irradiated at ECN, it must be shipped over a long distance. Quite clearly special requirements will be necessary to carry out this shipment. Hence the capsule will be packed in a transportation box, specially designed for this shipment, which should be provided with rubber buffers to protect the rig from shocks.

The top end of the capsule is provided with a so-called blind box to protect the instrumentation connections during transportation. A hoisting eye is fixed on the top end of the blind box for vertical transport of the capsule. Either the blind box or the instrumentation box is mounted on the top side of the rig. In the latter case the instrumentation leads are connected and handling the rig is always performed with the complete instrumentation hose. Hoisting the rig in this way is possible with a U-shaped handling tool with a 72-mm jaw size. This tool, with which one is able to push, pull, and rotate, is provided with a feature for changing the load from one crane to another.

Under normal circumstances the thimble will be handled only with an inserted dummy plug. The dummy plug is provided with a hoisting eye and is easy to handle with general handling tools. During installation of the thimble in the passage block of the reactor vessel, one should notice that the cooling water inlet is positioned in good order with respect to other experiments. The capsule is loaded and unloaded in the thimble by means of the U-shaped handling tool. Experiences with this type of tool have proved satisfactory. The bolting of the capsule into the thimble is achieved by a standard cone clamp assembly. Locking and unlocking this cone clamp is done with the help of the standard handling tool for cone clamps.

The capsule will be removed from the thimble when a certain specimen deformation has been reached. After this time, the top end of the capsule will be hoisted above the pool water surface, and the connection box will be replaced by the blind box. This blind box is provided with a quick joint for gas supply. When stored in a rack for finished experiments, it is connected with a nitrogen service system to keep the capsule dry until dismantling.

If the capsule has to be stored temporarily, it must be placed, with connected instrumentation hose, in a storage rack for experiments. In this case the capsule is filled with helium from the gas service system shown in Fig. 6 and is also monitored by means of pressure indicator/annunciator 3PIA2.

The capsule will be transported to the Laboratory of Strong Active Objects (LSO) in the so-called "Tonolli" coffin. Because the capsule is too long for this coffin, the extension tube must be cut to the proper length. This procedure is performed above the lead floor in the corner of the storage pool. The upper end is direct waste disposal and is not activated; therefore, it can be transported in a plastic bag to waste disposal. The capsule can be loaded in the Tonolli under water according to established loading procedures, and then the coffin can be transported to the LSO.

The capsule is dismantled in the LSO hot cell. First the shroud tube is sawed through at the location of the cooling water return holes; then the shroud tube can be removed and the minitubes with fluence detectors are peeled off of the pressure vessel. Next the pressure vessel is

sawed through at a location above and below the Zircaloy specimen. The deformed specimen is then retrieved and stored for further examination.

9. POSTIRRADIATION MEASUREMENTS AND REPORTS

9.1 Measurements

The main purpose of the postirradiation examination is the determination of the fast neutron fluence ($E_n > 1.0$ MeV). To make this determination the retrieved fluence detectors are sent from the LSO to the Neutron Metrology Department of the Physics Division. In this department the activities of the iron, nickel, and cobalt wires are measured and the fluences determined.

Although the primary information of this Zircaloy creepdown irradiation test is obtained during the irradiation itself, the specimen is stored because postirradiation examination might be necessary in the future.

9.2 Reports

Reporting the experiment will be done in two different parts:

- (1) Irradiation report, written under the authorship of the ECN Reactor Division. This report contains (after a brief description of the experiment) the results of the acceptance tests, the performance of the experiment, the results of the experiment, a chronological survey of the irradiation and, if required, a discussion with remarks concerning the irradiation.
- (2) Neutron Metrology Report, written under the authorship of the ECN Physics Division. The contents of this report should include the experimental data, the counting procedure, and the data treatment procedure and results.

10. SAFETY ANALYSIS

10.1 Failure Analysis

10.1.1 Target failure

The primary purpose of the HOBBIIE irradiation tests is to determine the creepdown behavior of Zircaloy tubing test specimens. It is possible, although not likely, that the test specimen will fail during the test. Should this occur, all of the high-pressure helium will be vented to buffer tank No. 1 (as shown in Fig. 6), which will cause an alarm signal

to be initiated by contact manometer 1PIA1. The high-pressure helium system has a limited volume, and buffer tank No. 1 has been designed to contain the equalization pressure between the high-pressure system and the low-pressure specimen gas system. The calculations presented in Appendix C show that the equalization pressure of the two gas systems is 2.3 MPa (334 psig) while the maximum allowable pressure for buffer tank No. 1 is 2.9 MPa (406 psig).

If the test specimen does fail, the test will be considered completed and the irradiation capsule will be removed from the reactor.

10.1.2 Failure of containment

The HOBBIE capsules have a single containment. The in-reactor containment consists of a pressure vessel designed to operate at 24.1 MPa (3500 psig). The failure of this pressure vessel, considered to be part of the Maximum Credible Accident, is discussed in detail in Sect. 10.2.

The out-of-reactor containment consists of the gas lines that are connected to the gas supply system. If one of these lines fails, contaminated helium would be released to the HFR building, and the activation products presented in Table 4 would be dispersed into the atmosphere within the building. Assuming uniform dilution within the building and a building volume of 10,000 m³, the resultant concentrations of activation products would be as shown in Table 5. Also presented in Table 5 are the occupational Maximum Permissible Concentration (MPC) levels for ³H and ¹⁴C as stated in Chap. 10 of the *U.S. Code of Federal Regulations*, Part 20.

Table 5. Uniform concentration of activation products in HFR building resulting from failure of HOBBIE capsule containment compared with MPC values

Radioactive isotope	Concentration in air (μCi/ml)	MPC (μCi/ml)
³ H	4.0×10^{-4}	5×10^{-6}
¹⁴ C	9.4×10^{-9}	4×10^{-6}
¹⁶ N	2.9×10^{-8}	Not applicable

The MPC values given in Table 5 are for 40 hr per week for 13 weeks or a total time of 520 hr. If we assume that local concentration can be 10 times that obtained by considering uniform dilution in the free volume of the building, then the ³H concentration could be 4.0×10^{-3} μCi/ml or a factor of 800 above the MPC value. Therefore, if personnel

can evacuate the building in 39 min, their exposure would not exceed the maximum allowable dose for a quarter. The ^{14}C activity, even if the concentration is assumed to be 10 times the uniform distribution value, is at most a factor of .025 times the MPC value. Since ^{16}N has a half-life of 7.1 sec, it will not be considered a radiation hazard.

10.1.3 Cooling system failure

The condition of the cooling system is continuously monitored by contact manometer 6PIA1. If the cooling system develops a blockage, the pressure monitored by 6PIA1 would increase and an APD would be initiated. If the cooling system develops a leak prior to the point where it enters the thimble, the pressure monitored by 6PIA1 will decrease and an APD will also be initiated.

10.1.4 Electrical power failure

If the main power supply is switched off, the instrumentation is completely out of operation. As a consequence an APD is obtained from 6PIA1, the cooling water pressure sensor. This APD signal is obtained from a self-resetting contact alarm unit (CAU). Hence if the power supply is switched on directly, the power decrease will stop and the reactor can be manually brought to 45 MW as soon as possible. (Note: As mentioned in Sect. 4.2.2.2, valve 5V1 is open only if the coil of the magnetic valve is powered. Therefore, during a power failure, valve 5V1 would close, stopping the compressor immediately and making overpressurization impossible.)

10.1.5 Instrumentation failures

The success of the experiment depends on the reliability of the scientific instrumentation. For temperature data enough thermocouples have been placed in the capsule. In case of failure, spare thermocouples can be connected with the analogue recorder or millivolt Alarm Unit (mVAU) by means of the matrix pinboard. For pressure data a double set of pressure transducers has been built into the pressure system. If both fail, the integrity of the system is not affected because relief valves 2V18 and 2V19 are installed as a failsafe system. Pressure data is obtained from manometer 2PIA3, which can be replaced during experiment operation.

For displacement data, one eddy-current instrument is available. Failure of this instrument leads to interruption of the pressurized irradiation until this instrument is repaired. All electronic drawings of the instrument are present and also essential electronic printed circuits are available as spare parts.

For the process instrumentation enough knowledge and spare parts are available to safeguard the continuation of the irradiation. A failure of the gas booster pump will also lead to an interruption of the irradiation until the pump is repaired. Documentation of this compressor is available, but there are no spare parts and hence interruption could occur.

10.1.6 Utility failures

The alarm and interlock system reveals that failure of the cooling water system leads to an automatic power decrease of the reactor. Most of the time this failure will be due to human errors and can be reset immediately. If a real utility failure occurs, all poolside facility experiments would run without coolant; hence a reactor scram would be inevitable.

Failure of the gas- and air-supply systems would not affect the integrity of the experiment. Failure of the nitrogen system makes it possible for water to enter the instrumentation hose with unfortunate consequences for the experiment only. A defect in the air supply system will only lead to a pressure decrease in the capsule.

10.1.7 Operating errors

The overall design of the experiment is such that the integrity of the reactor is never affected by human operating errors. The gas system has been equipped with enough relief valves to avoid overpressurization of the system and can, therefore, prevent a rupture or explosion of the capsule high-pressure vessel.

The most serious operating problem would be a succession of two human errors. The first one would be an error in following the checkout list when the operator forgets to reset the bypassed APD signal of the cooling water flow after loading the thimble with the capsule. The second mistake would be to close valve 6V2 after which the cooling water flow is blocked. In this case the pressurized capsule would be without coolant, which may be considered the Maximum Credible Accident. This accident is intensively discussed in Sect. 10.2.

10.1.8 Handling errors

Only mechanical damages can occur because of improper manipulations with handling tools. Two possible handling errors can be considered that might lead to incorrect reactor operation. The first one is a sudden insertion or removal of the capsule in or out of the thimble. In this case it is possible that the automatic control rod system might not be able to compensate for the rapid change in reactivity. However, theoretically it is never possible to reach the critical reactor period of 3 sec with a reactivity change of 51 pcm (7.1 cents) [see Fig. 19]. The other,

even worse handling error, is leaving the thimble empty. Although the thimble is cooled by the primary cooling water flow, the power generated in the dead water in the thimble may cause boiling, which may cause small, rapid changes in reactivity and consequential oscillating of the control rods. Because of the negative temperature coefficient of the reactor, however, dangerous conditions should not occur.

10.2 Maximum Credible Accident (MCA) Analysis

The MCA is assumed to be the result of the following two conditions:

- (1) Following installation of the capsule, the operator neglects to reset the APD bypass switch on low cooling water flow (6PIA1);
- (2) A blockage develops in the cooling water inlet line.

As a result of these two conditions, the capsule will be without cooling water flow and the reactor will remain at 45 MW. Without cooling water flow, the water in the annulus between the pressure vessel and the capsule shroud will boil, which will probably result in the entire annulus being filled with steam. Because the thermal conductivity of steam is quite poor when compared to the heat removal capacity of flowing water, the temperature of the pressure vessel will rise above the point where any strength remains in the stainless steel. At this point the vessel will probably fail. A failure of the pressure vessel will be a ductile one (Appendix E) and will allow about 0.05 m³ of contaminated helium to leak into the capsule extension tube and then into the reactor pool. The activation products that could be released and the consequences of that release have been discussed in Sect. 10.1.2. No radiation hazard exists provided that the building is evacuated in 39 min.

There is a possibility of release of radioactive oxides of materials within the capsule; however, these oxides should be released as particulates and should remain in the pool water.

ACKNOWLEDGMENTS

The authors wish to thank the following persons for their contributions to the program:

From ORNL — D. O. Hobson, Metals and Ceramics Division, serving as Program Manager for providing the design criteria, planning, and coordination for the entire project; C. V. Dodd and L. D. Chitwood, Metals and Ceramics Division, for the design and fabrication of the eddy-current instrumentation; J. A. Conlin, Engineering Technology Division, for assisting in the design of the experiments; and R. W. Clark of Engineering for providing the stress analysis.

From ECN Petten — A. van den Dool, R. J. M. van de Pol, and N. Lindemans for preparation of the irradiation facility; H. J. Nolthenius for preparation of the flux monitors; and N. van der Kleij for the help, cooperation, and technical knowledge made available to us.

REFERENCES

1. H. Röttger et al., *High Flux Materials Testing Reactor*, EUR 5700 (ed. 1976-77).
2. Th. van der Kaa, *Instrumentation Proposal HOBBIIE Irradiation*, HFR memo 77-03 (March 14, 1977).
3. D. O. Hobson and C. V. Dodd, *Interim Report on the Creepdown of Zircaloy Fuel Cladding*, ORNL/NUREG/TM-103 (May 1977).
4. C. V. Dodd and G. D. Connell, *The NDT-COMP8 Micro-computer*, ORNL/TM-5773 (March 1977).
5. Th. van der Kaa, *DACON, Description and Use of a DATA CONVersion Program for HFR Irradiation Experiments*, HFR memo 76-09 (May 12, 1976).
6. G. J. A. Teunissen, *Compilation of Nuclear Heating Data*, HFR memo in preparation.
7. A. Tas, D. van Ligten, and A. W. den Houting, *MICROFLUX-2, A Program for Calculation of Thermal Group Constants*, RCN-94 (October 1968).
8. J. Adir and K. D. Lathrop, *Theory of Methods Used in the GGC-4 Multigroup Cross-Section Code*, GA-9021 (October 1968).
9. A. Tas and G. J. A. Teunissen, *HIP-TEDDI, A Two-Dimensional, Four-Group Code for the Calculation of the Neutron Fluence and Operating Data of the Material Testing Reactor HFR at Petten*, ECN-77-058 (May 12, 1977).
10. H. C. Roland, *GENGTC: A One-Dimensional Computer Program for Capsule Temperature Calculation in Cylindrical Geometry*, ORNL/TM-1942 (December 1967).
11. W. R. Clark, *Response to Stress Analysis Review*, memo to K. R. Thoms (September 14, 1977).
12. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, sect. III, div. I, subsects. NA and NB.
13. J. A. Swanson, *FEATS User Manual*, WANL-TME-1888 (December 1968).
14. K. R. Thoms, *Quality Assurance Program Plan for Zircaloy Creep Collapse Irradiation Experiments*, Doc. Q-11762-ET-001-5-0 (May 9, 1977).
15. E. Lybrink, personal communication June 6, 1977.
16. H. Etherington (ed.), *Nuclear Engineering Handbook*, 1st ed., McGraw-Hill, 1958.

Appendix A

REMOVED HEAT ANALYSIS

For a conservative calculation of the heat production in the HOBBIE capsule, it will be assumed that the capsule consists of a pressure vessel of 60-cm length that is completely filled with aluminum and surrounded by a shroud.

The total nuclear heat production Q_n amounts in that case to

$$Q_n = \bar{q}_Y''' * \frac{\pi}{4} l [(D_{sh}^2 - d_{sh}^2 + D_{pv}^2 - d_{pv}^2) * \rho_{ss} + d_{pv}^2 * \rho_{Al}]$$

$$= 16.55 \text{ kW} ,$$

where

$$\begin{aligned} \bar{q}_Y''' &= \text{average nuclear heat generation in position H8} = 2 \text{ W/g} , \\ l &= \text{length of capsule} = 60 \text{ cm} , \\ D_{sh} &= \text{outer diameter of shroud} = 6.35 \text{ cm} , \\ d_{sh} &= \text{inner diameter of shroud} = 5.93 \text{ cm} , \\ D_{pv} &= \text{outer diameter of pressure vessel} = 5.512 \text{ cm} , \\ d_{pv} &= \text{inner diameter of pressure vessel} = 4.495 \text{ cm} , \\ \rho_{ss} &= \text{specific weight of stainless steel} = 7.9 \text{ g/cm}^3 , \\ \rho_{Al} &= \text{specific weight of aluminum} = 2.7 \text{ g/cm}^3 . \end{aligned}$$

The total heat production Q_t in the capsule is the nuclear heat production Q_n increased with the maximum electrical power Q_c of the heater element, which was 0.75 kW.

If the cooling water flow m amounts to 0.55 kg/sec ($\cong 2 \text{ m}^3/\text{hr}$), then the temperature rise of the coolant is

$$\Delta T = \frac{Q_t}{m C_p} = 7.5^\circ\text{C} ,$$

where

$$\begin{aligned} Q_t &= \text{total heat production} = 17.3 \text{ kW} , \\ m &= \text{mass flow of coolant} = 0.55 \text{ kg/sec} , \\ C_p &= \text{heat capacity} = 4.2 \text{ kJ/kg}\cdot^\circ\text{C} . \end{aligned}$$

Appendix B

MATERIALS USED IN THE PRESSURIZATION SYSTEM

With the exception of the irradiation capsule, the complete pressure supply system has been designed for an operating pressure of 30 MPa (4260 psi). The design pressure is 1.1 times the operating pressure whereas pressure tests on the pressure equalizer have been performed at 1.3 times the operating pressure.

All valves used in the system, such as ball valves, check valves, needle valves, and metering valves, are high-quality stainless steel valves for operating pressures up to 35 MPa (5000 psi) at 20°C (trademark HOKE Inc.).

The tubing used is annealed seamless stainless steel minitube AISI-316 with 4-mm ID/6-mm OD for the pressurization part up to the pressure equalizer, and 1-mm ID/3-mm OD for the capsule part. The allowable stress in the minitube is 1320 MPa (187 ksi), and the Rockwell hardness is less than B90 in order to obtain a proper connection with flareless tube fittings (trademark GYROLOK).

The buffer tank No. 1 and accumulator tank No. 2 are sampling cylinders of the HOKE Corporation. The buffer tank is a low-pressure cylinder good for 2.8 MPa (400 psi); the accumulator tank is a high-pressure cylinder good for 35 MPa (5000 psi). Minitubes and cylinders are delivered with a material certificate. The pressure equalizer is provided with a material analysis performed at the Metals Division of ECN (see ref. 15).

The accumulator tank is provided with two relief valves (trademark Dr. Thiedig) adjusted at 35 MPa (5000 psi). Therefore an operating pressure of up to 30 MPa (4260 psi) can be obtained downstream of pressure regulator 2V6, a HOKE-precision high-pressure controller of type 5562.

Since the operating pressure of the capsule is 21 MPa (3000 psi), the capsule side of the pressure vessel is provided with two Dr. Thiedig relief valves adjusted to 23 MPa (3270 psi).

Appendix C

PRESSURE EQUALIZATION CALCULATIONS

As mentioned in Sect. 4.2.2.2 the capsule part and the compressor part of the pressure supply system are protected against overpressurization by means of two pairs of relief valves installed in parallel. In this appendix the equalized pressures are calculated after failure of the specimen and after dumping of the gas system into buffer tank No. 3.

The volumes used for these calculations are given in Table C-1.

Table C-1. Volumes used for pressure equalization calculations

Volume compressor part V_{com}	800 cm ³
Volume accumulator tank No. 2	500 cm ³
Volume minitube (21 m with 13 cm ³ /m)	273 cm ³
Volume pressure equalizer piston right positioned	27 cm ³
Volume capsule part V_{cap}	247 cm ³
Volume pressure equalizer piston left positioned	74 cm ³
Piston displacement	69 cm ³
Volume minitube (2 × 10 m @ 2.4 cm ³ /m; 2 × 4 m @ 0.8 cm ³ /m)	54 cm ³
High-pressure vessel (estimated) is V_{pv}	50 cm ³
Volume specimen part V_{spec}	2286 cm ³
Volume specimen (estimated)	9 cm ³
Volume minitube (14 m with 10 m @ 2.4 cm ³ /m; 4 m @ 0.8 cm ³ /m)	27 cm ³
Buffer tank No. 1	2250 cm ³

Specimen failure

Because of the very small volume in the specimen, the temperature influence on the equalized pressure can be neglected in the calculations.

This equalized pressure P_{eq} is

$$P_{eq} = \frac{P_{spec} * V_{spec} + P_{cap} * V_{cap}}{V_{spec} + V_{cap}} = 2.3 \text{ MPa} ,$$

where

P_{spec} = operation pressure inside the specimen = 0.3 MPa ,

P_{cap} = maximum operation pressure in capsule = 21.0 MPa .

Note: maximum allowable pressure, buffer tank No. 1 = 2.8 MPa ,
annunciation pressure, 1PIA1 = 0.5 MPa .

Capsule Gas Dumping in Buffer Tank No. 3

Before calculating the equalized pressure between the capsule and buffer tank No. 3, it is necessary to split the capsule part into two parts, namely (1) the pressure vessel part, V_{pv} (50 cm³), at an average gas temperature T_{pv} of 500°K, and (2) the remaining part, $(V_{cap} - V_{pv})$, at room temperature T_{ref} of 300°K.

The equalized pressure is the sum of the partial pressures in the total volume.

$$P_{eq} = \frac{P_{cap} * V_{pv} * T_{ref}}{(V_{cap} + V_{bdt}) * T_{pv}} + \frac{P_{cap} * (V_{cap} - V_{pv}) + P_{bdt} * V_{bdt}}{V_{cap} + V_{bdt}} = 0.6 \text{ MPa} ,$$

where

P_{bdt} = pressure, buffer/decay tank No. 3 = 0.1 MPa ,

V_{bdt} = volume, buffer/decay tank No. 3 = 10,000 cm³ .

Note: maximum allowable pressure, buffer tank No. 3 = 1.0 MPa ,
annunciation pressure, 3PIA1 = 0.3 MPa .

Appendix D

CALCULATION OF SURFACE HEAT TRANSFER COEFFICIENT

The surface heat transfer coefficient is calculated for a water flow rate of 0.5 kg/sec (1.10 lb/sec). The first step is to determine the type of flow, i.e., laminar^m or turbulent, in the annulus between the pressure vessel and the shroud. This is done by calculating the Reynold's Number according to the equation:

$$RE = \frac{D_h G}{\mu} = 11,437 ,$$

where

$$\begin{aligned} D_h &= \text{hydraulic diameter of flow annulus} \\ &= 4 \times \frac{\pi/4(OD^2 - ID^2)}{\pi(OD + ID)} = 0.418 \text{ cm (0.137 ft)} , \\ OD &= 5.928 \text{ cm (2.334 in.)} , \\ ID &= 5.118 \text{ cm (2.170 in.)} , \end{aligned}$$

$$\begin{aligned} G &= \text{mass flow rate/unit area} \\ &= 0.138 \text{ kg/cm}^2 \cdot \text{sec (273 lb}_m\text{/ft}^2 \cdot \text{sec)} , \\ \mu &= \text{viscosity. For water at } 60^\circ\text{C} \\ \mu &= 0.49 \text{ centipoises} \\ &= 4.89 \times 10^{-6} \text{ kg/cm} \cdot \text{sec (3.29} \times 10^{-4} \text{ lb}_m\text{/ft} \cdot \text{sec)} . \end{aligned}$$

The water flow in the annulus between the pressure vessel and the shroud is therefore turbulent. To calculate the surface heat transfer coefficient we may then use the relationship:

$$\frac{hD_h}{k} = 0.023 (Re)^{0.8} (Pr)^{0.4} ,$$

where

$$\begin{aligned} k @ 60^\circ\text{C} &= 6.60 \times 10^{-3} \text{ W/cm} \cdot ^\circ\text{C (0.381 Btu/hr} \cdot \text{ft} \cdot ^\circ\text{F)} \\ Pr @ 60^\circ\text{C} &= 2.98 ; \end{aligned}$$

therefore

$$h = 0.98 \text{ W/cm}^2 \cdot ^\circ\text{C (1731 Btu/hr} \cdot \text{ft}^2 \cdot ^\circ\text{F)} .$$

Appendix E

**MEMO — D. A. CANONICO TO D. O. HOBSON,
FRACTURE TOUGHNESS, May 12, 1977**

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 12, 1977

To: D. O. Hobson
From: D. A. Canonico
Subject: Fracture Toughness
Reference: D. O. Hobson to G. G. Fee, "Visit of T. van der Kaa,
ECN-Petten to ORNL", January 18, 1977

I have prepared the attached review of the toughness of the austenitic stainless steel containment for the specimen which you plan to irradiate in the Dutch HFR at ECN-Petten. Your choice of an austenitic stainless steel containment material eliminates the possibility of a rapidly propagating brittle fracture. One must, however, contend with the possibility of stress corrosion cracking and plastic instability due to overpressurization. I have assumed that these latter two failure modes have been considered in your design and their probability of occurring minimized through environmental control (chlorides, oxygen, etc.) or design (sufficient wall thickness to withstand any internal pressurization).

D. A. Canonico

DAC:wmb

Attachment

cc: G. G. Fee
G. M. Slaughter
DAC/RC

FAILURE ANALYSIS OF CONTAINMENT CAPSULE

D. A. Canonico

I have assessed the probability of fracture of the pressure containment capsule for the experiment planned for the Dutch HFR at ECN-Petten. The material of construction¹ for the capsule is austenitic stainless steel and its nominal operating temperature range¹ will be limited to 38 to 100°C (100 to 212°F). My assessment of the reliability of the capsule is centered about the sensitivity of the austenitic stainless steel to a rapid catastrophic mode of failure. Of course, austenitic stainless steel is extremely sensitive to stress corrosion cracking (SCC) when highly stressed in an oxygen and/or chloride enriched environment. The source of the stress can be designed load or residual stresses due to welding. If the capsule is subjected to these operating conditions then capsule failure is highly probable. The SCC failure, however, if it does occur, will undoubtedly lead to a breach of the capsule wall and a loss of internal pressure, i.e. "leak before break". I assume that the conditions leading to a SCC will be avoided during the radiation experiment.

The failure modes of particular interest to the reliability (and perhaps safety) of the irradiation experiment are those wherein a fracture occurs rapidly and leads to fragmentation or major deformation of the capsule. These modes of failure can damage the reactor core and/or fuel plates and therefore are of most concern. Both types of failure, classic brittle fracture and low energy ductile tearing, are related to the toughness of the material from which the capsule is constructed.

¹Private discussion, D. O. Hobson, January 1977.

BLANK PAGE

The austenitic stainless steels do not undergo a ductile to brittle transition. This transition is common in body centered cubic materials and is particularly evident in the low alloy, high strength structural steels. Figure 1 represents the affect of temperature on the ability of a notched material to absorb energy when impact loaded. The upper curve represents austenitic stainless steel, a material which does not undergo a ductile to brittle transition, even down to liquid nitrogen temperatures and below. The lower curve, which shows a transition, is representative of ferritic materials. The temperature at which the transition occurs is affected by the chemical composition of the steel. Further, the transition temperature for ferritic steel of a given chemical composition can be varied through heat treatment.

Fig. 1. Graphic Representation of the Affect of Temperature on the Charpy V-Notch Toughness of Austenitic and Ferritic Structural Materials

In the remainder of this assessment I will concentrate on some toughness data available for austenitic stainless steels. Charpy V-notch (C_V) studies have been conducted on 25.4 mm (1 in.) thick type 304 stainless steel base plate at ORNL. We use an instrumented Baldwin impact tester which has a maximum stored energy of 325 joules (240 ft-lbs). Our C_V tests were conducted at -150, -50 and 250°C (-238, -58 and 482°F). At the two higher temperatures the full size C_V specimens (10 mm by 10 mm) stopped the hammer in the impact tester, i. e. the specimens absorbed, more than 325 joules. At -150°C the C_V specimens absorbed 267 joules. The ability of an austenitic stainless steel to absorb this level of energy at the temperatures investigated in the ORNL studies indicate that neither brittle fracture nor low toughness ductile tearing is of much concern.

A more quantitative assessment of the toughness of an austenitic stainless steel can be made from the data presented by C. G. Chipperfield.² Chipperfield obtained J_{IC} values for type 316 stainless steel with various degrees of cold work. He reported a valid J_{IC} value at 20°C (68°F) for 0% cold worked type 316 stainless steel of 966 lb/in. His value for 5% cold worked material is 481 lb/in. A stress intensity value can be obtained from the J integral through the equation³

$$J = G = \frac{K^2}{E} (1 - \nu^2) \quad (1)$$

or

$$K_{IC} = \sqrt{\frac{J_{IC} E}{1 - \nu^2}} \quad (2)$$

²C. G. Chipperfield, "Detection and Toughness Characterization of Ductile Crack Initiation in 316 Stainless Steel", *International Journal of Fracture*, Volume 12, 1976, pp. 873-886.

³J. F. Knott, *Fundamentals of Fracture Mechanics*, John Wiley and Sons, New York, 1973, p. 168.

Employing Eq. (2) and assuming Poisson's ratio, ν , to be 0.29 and the modulus of elasticity, E , to be 29.0×10^6 at room temperature⁴ the K_{IC} values for the 0% and 5% cold worked type 316 stainless steel are 174 ksi $\sqrt{\text{in.}}$ and 123 ksi $\sqrt{\text{in.}}$, respectively.

If we assume that small amounts of local plastic flow precede crack extension, referred to as "quasi-brittle behavior" by Knott,⁵ then the critical value for crack extension can be related to the failure stress by linear elastic methods. Applying the methods of linear elastic stress analysis permits the following relationship between fracture stress, δ_F , flaw size, a , and the critical stress intensity factor K_{CRIT} .

$$K_{CRIT} = \delta_F \sqrt{\pi a} \quad (3)$$

We can assume that the K_{IC} values that were obtained for the austenitic stainless steel through Eq. (2) represent the critical stress intensity value K_{CRIT} . Using these values and assuming various capsule capsule wall thicknesses, we can calculate the stress value that corresponds to a flaw size that is identical to the wall thickness. Table 1 provides data for wall thickness of 1, 2, 4 and 6 mm (0.04, 0.08, 0.16 and 0.24 in.). It is evident from the results provided in Table 1 that the fracture stress values are considerably in excess of the ultimate tensile strength of austenitic stainless steels.

⁴J. P. Hammond et al., "Effects of Materials and Test Variables on Elastic Constants of 2 1/4 Cr-1 Mo Steel and Type 304 Stainless Steel", Proceedings of the Second International Conference on Mechanical Behavior of Materials, Aug. 16-20, 1976, Boston, Mass., p. 1037.

⁵J. F. Knott, *Fundamentals of Fracture Mechanics*, John Wiley and Sons, New York, 1973, p. 111.

Table 1. Relationship Between Fracture Stress
and Flaw Size for Type 316 Stainless
Steel with Various Critical
Stress Intensity Values.

K_{CRIT} ksi $\sqrt{\text{in.}}$	Flaw Size in.	Fracture Stress ksi
174	0.04	490
123		347
174	0.08	347
123		245
174	0.16	245
123		173
174	0.24	200
123		142

A second assessment can be made by assuming a fracture stress equal to the ultimate tensile strength of the austenitic stainless steel and calculating the critical flaw size. The ultimate tensile strength of the 0% cold worked material used in Chepperfield's experiments is 91 ksi. This calculation suggests a critical flaw size of over one inch (1.08 inches) a value considerably larger than the wall of the containment capsule.

In summary, this review indicates that the critical flaw size and/or fracture stress values for a rapid catastrophic failure of the austenitic stainless steel containment capsule are extremely large. Therefore, I must conclude that the probability of a "brittle" failure of the containment capsule is nil. This does not, however, relieve the designers and operators of the responsibility for providing an assurance that the capsule will not burst as a consequence of an overpressurization during operation.

Appendix F

ACTIVATION OF GAS INSIDE CAPSULE

Total void volume in capsule = 55 cc .

Conditions of gas in capsule: P = 3000 psi (20.7 MPa) ;

Average: T = 150°C (423°K) .

Calculate volume at standard temperature and pressure:

$$\frac{V_1 P_1}{T_1} = \frac{V_2 P_2}{T_2} ,$$

$$V_2 = \frac{V_1 P_1}{T_1} \times \frac{T_2}{P_2} = \frac{55(3015)}{423} \times \frac{293}{14.7} ,$$

$$V_2 = 7814 \text{ cc} .$$

Table F-1. Impurities in helium and equivalent volumes

Impurity	Impurity concentration (ppm)	Total volume, STP ^a (cc)
³ He	1.3	1.02
H ₂ O	10	7.8
N	20	15.6
O ₂	5	3.91
H ₂	5	3.91
Ne	10	7.8

^aStandard temperature and pressure.

The following equation will be used to calculate the amount of activation products.

$$N_2 = \frac{\phi \sigma_1 N_1}{\phi \sigma_2 + \lambda_2} \left\{ 1 - \exp [-(\phi \sigma_2 + \lambda_2) t] \right\} .$$

BLANK PAGE

Transmutation of ^3He to ^3H

Calculate number of atoms of ^3He .

$$n = \frac{PV}{RT} = \frac{1 \text{ atm } (1.02 \text{ cc})}{82.057(293^\circ\text{K})} = 4.24 \times 10^{-5} \text{ mole .}$$

$$N_1 = 4.24 \times 10^{-5} (6.023 \times 10^{23}) = 2.55 \times 10^{19} \text{ atoms .}$$

$$\sigma_1 = 5327 \text{ b ,}$$

$$\sigma_2 \approx 0 ,$$

$$\phi = 0.9 \times 10^{14} \text{ neutrons/cm}^2 \cdot \text{sec ,}$$

$$\lambda_2 = \frac{0.693}{12.26 \text{ year} \times \frac{365 \text{ day}}{\text{year}} \times \frac{24 \text{ hr}}{\text{day}} \times \frac{3600 \text{ sec}}{\text{hr}}} = 1.7 \times 10^{-9} \text{ sec}^{-1} ,$$

$$t = 7.2 \times 10^6 \text{ sec .}$$

$$N_2 = \frac{0.9 \times 10^{14} (5327 \times 10^{-24}) (2.55 \times 10^{19})}{0 + 1.7 \times 10^{-9}}$$

$$\left\{ 1 - \exp[-(1.7 \times 10^{-9}) (7.2 \times 10^6)] \right\} = 8.75 \times 10^{19} \text{ atoms .}$$

$$A = \frac{N\lambda}{3.7 \times 10^{10} \frac{\text{dis/sec}}{\text{Ci}}} = \frac{8.75 \times 10^{19} (1.7 \times 10^{-9})}{3.7 \times 10^{10}} = 4.0 \text{ Ci .}$$

Transmutation of ^{14}N to ^{14}C

Calculate number of atoms of ^{14}N .

$$n = \frac{PV}{RT} = \frac{1 \text{ atm } (15.6 \text{ cc})}{82.057(293^\circ\text{K})}$$

$$= 6.49 \times 10^{-4} \text{ mole .}$$

Since nitrogen exists as a diatomic gas, the number of atoms,

$$N_1 = 2 \frac{\text{atoms}}{\text{molecule}} (6.49 \times 10^{-4} \text{ moles}) (6.023 \times 10^{23} \frac{\text{molecules}}{\text{mole}}) \\ = 7.82 \times 10^{20} \text{ atoms .}$$

$$\lambda_1 = 1.8 \text{ b ,}$$

$$\lambda_2 = 0 ,$$

$$\phi = 0.9 \times 10^{14} \text{ neutrons/cm}^2 \cdot \text{sec ,}$$

$$\lambda_2 = \frac{0.693}{T_{1/2}} = \frac{0.693}{5730 \text{ year} \times \frac{365 \text{ day}}{\text{year}} \times \frac{24 \text{ hr}}{\text{day}} \times \frac{3600 \text{ sec}}{\text{hr}}} \\ = 3.84 \times 10^{-12} \text{ sec}^{-1} ,$$

$$t = 2000 \text{ hr} = 7.2 \times 10^6 \text{ sec .}$$

$$N_2 = \frac{0.9 \times 10^{14} (1.8 \times 10^{-24}) (7.82 \times 10^{20})}{0.9 \times 10^{14} (0) + 3.84 \times 10^{-12}}$$

$$\left\{ 1 - \exp[-(3.84 \times 10^{-12}) (7.2 \times 10^6)] \right\} = 9.12 \times 10^{17} \text{ atoms .}$$

$$A = \frac{N\lambda}{3.7 \times 10^{10} \frac{\text{dis/sec}}{\text{Ci}}} = \frac{9.12 \times 10^{17} (3.84 \times 10^{-12})}{3.7 \times 10^{10}} = 94 \text{ } \mu\text{Ci .}$$

Transmutation of ^{16}O to ^{16}N

Quoting from pp. 7-18 of ref. 16:

The ^{16}N isotope decays with a half-life of 7.4 sec, emitting very energetic (6- to 7-MeV) gamma rays. Neutrons must have an energy greater than 9 MeV to produce the reaction $^{16}\text{O}(n,p)^{16}\text{N}$. Only one in several thousand fission neutrons has an energy in excess of 9 MeV. Rather than use the actual fission neutron spectrum and the variation of the ^{16}O cross section with energy it is convenient to use the measured cross section value of $4 \times 10^{-29} \text{ cm}^2$ for virgin fission

neutrons as a whole. Since a single scattering in H_2O is likely to reduce the energy of virgin neutrons below the 9-MeV threshold, only the virgin flux need be considered. This virgin flux is calculated from the relation

$$\phi_v = \frac{n}{\Sigma_{in}} = \frac{(3.1 \times 10^{10})(2.5)(P)}{N_H \sigma_H},$$

where

n = number of virgin neutrons released per second in 1 cm^3 of the reactor core,

Σ_{in} = total scattering cross section of hydrogen for fission neutrons above 9 MeV/ cm^3 of the core,

P = power level of core, watts/ cm^3 ,

3.1×10^{10} = fissions/sec-watt,

2.5 = number of neutrons per fission,

N_H = number of hydrogen atoms/ cm^3 of core,

σ_H = scattering cross section of hydrogen for neutrons above 9 MeV $\approx 1 \text{ b}$.

Assume average power density of 340 W/cc .

Fraction of water volume in core = 0.62 (ORR).

Therefore power density per unit volume of water = $\frac{340}{0.62} = 550 \text{ W/cc}$.

$$\phi_v = \frac{3.1 \times 10^{10}(2.5)(550)}{6.6 \times 10^{22}(1 \times 10^{-24})}$$

$$= 6.5 \times 10^{14} \text{ neutrons/cm}^2 \cdot \text{sec} .$$

Calculate Total Oxygen Atoms in Capsule

From H_2O ,

$$n = \frac{1 \text{ atm } (7.8 \text{ cc})}{82.057(273)}$$

$$= 3.48 \times 10^{-4} \text{ mole .}$$

$$N_0 = 3.48 \times 10^{-4} (6.023 \times 10^{23})$$

$$= 2.10 \times 10^{20} \text{ atoms .}$$

From O_2 ,

$$n = \frac{1 \text{ atm } (3.91 \text{ cc})}{82.067(273)}$$

$$= 1.75 \times 10^{-4} \text{ mole .}$$

$$N_0 = 2 \frac{\text{atoms}}{\text{molecule}} (1.75 \times 10^{-4} \text{ moles}) (6.023 \times 10^{23} \frac{\text{molecules}}{\text{mole}})$$

$$= 2.10 \times 10^{20} \text{ atoms .}$$

Total oxygen atoms = 4.2×10^{20} atoms .

Saturated ^{16}N activity is therefore

$$\phi_v N_0 \sigma = 6.5 \times 10^{14} (4.2 \times 10^{20}) (4 \times 10^{-29})$$

$$= 1.09 \times 10^7 \text{ dis/sec .}$$

$$A = \frac{1.09 \times 10^7}{3.7 \times 10^{10}} = 2.9 \times 10^{-4} \text{ Ci .}$$