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In-Reactor Studies of Low Cycle Fatigue Properties of a Nuclear Pressure Vessel Steel

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ABSTRACT

An experimental irradiation assembly and associated instrumentation which have been developed and successfully utilized for the performance of dynamic in-reactor low cycle fatigue tests of reactor pressure vessel steels are described. The equipment provides for the simultaneous reverse bend testing of as many as fifteen sheet type specimens representing a range of strain amplitudes at controlled temperatures in the range 300° to 700°F.

The results of an exploratory investigation on the fatigue resistance of ASTM type A302-B steel during irradiation at 500°F are presented and compared with data from out-of-reactor control tests. These preliminary data do not indicate any pronounced difference in the fatigue strength of irradiated versus unirradiated steel. Exploratory investigations are continuing.

PROBLEM STATUS

This is a final report on the design and operational phase of this problem; work on other phases is continuing.

AUTHORIZATION

NRL Problem M01-14
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Task 0858; AT(49-5)-2110; ERG-5-64

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IN-REACTOR STUDIES OF LOW CYCLE FATIGUE PROPERTIES
OF A NUCLEAR PRESSURE VESSEL STEEL

INTRODUCTION

The operators of nuclear power reactors often consider the notch ductility characteristics and fatigue resistance of the pressure vessel to be their two major concerns in regard to the integrity of the reactor primary system. The resolution of problems associated with notch ductility characteristics has been greatly assisted by the rapid growth of a body of experimental data on the notch ductility of irradiated steels (1), and by the recent development of a fracture analysis diagram (2) which interrelates the conditions of flaw size, temperature, and stress necessary for the incipience and propagation of brittle fractures. By contrast, very little is known on the effects of nuclear radiation on the fatigue resistance of reactor structural materials (3). In recognition of the lack of knowledge in this area, a program has been undertaken to investigate the low cycle fatigue behavior of carbon steels commonly used for pressure vessel construction.

This report describes the experimental assembly and associated equipment developed for dynamic in-reactor reverse bend fatigue tests and presents the results of an initial exploratory investigation performed with this equipment.

SPECIMEN DESIGN AND STRAIN MEASUREMENT TECHNIQUE

Details of the specimen design selected for initial investigations are shown in Fig. 1. The cross section of the active test area is 0.1875 in. by 0.0625 in. which provides a width to thickness ratio of three. As will be pointed out in a later section, the experimental assembly developed for in-reactor tests can readily accommodate larger specimens with only minor modification. Prior to testing, each specimen is radiographed to insure that the test section does not contain cracks, fissures, or impurity concentrations which may influence experimental results.

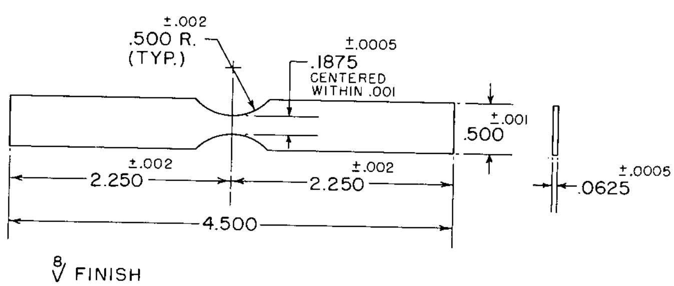

Fig. 1 – The reverse bend fatigue test specimen used for determinations of fatigue resistance under irradiation. The specimen is clamped in the test assembly to a point 0.030 in. below the beginning of the reduced section.

The strain amplitudes developed in specimens are measured by strain gage techniques, using bonded etched foil gages, before irradiation. The strain gage normally utilized has a nominal gage length and width of 0.12 and 0.09 in. respectively. A slightly larger gage (0.25 in. long by 0.17 in. wide) has also been employed on some occasions but is not considered as accurate for defining test section strains.

Strain amplitude is regulated with most reverse bend fatigue test machines by the adjustment of specimen deflection. However, the alternate method of constant deflection with varying specimen moment arm (bending length) was adopted to permit the simultaneous testing of several specimens representing a range of strain conditions with one fatigue machine. As shown in Fig. 2, the level of strain developed in individual specimens can be varied considerably by this method. Thus, an S-N curve (strain vs number of cycles to failure) can be established from the results of one reactor experiment rather than several duplicate experiments.

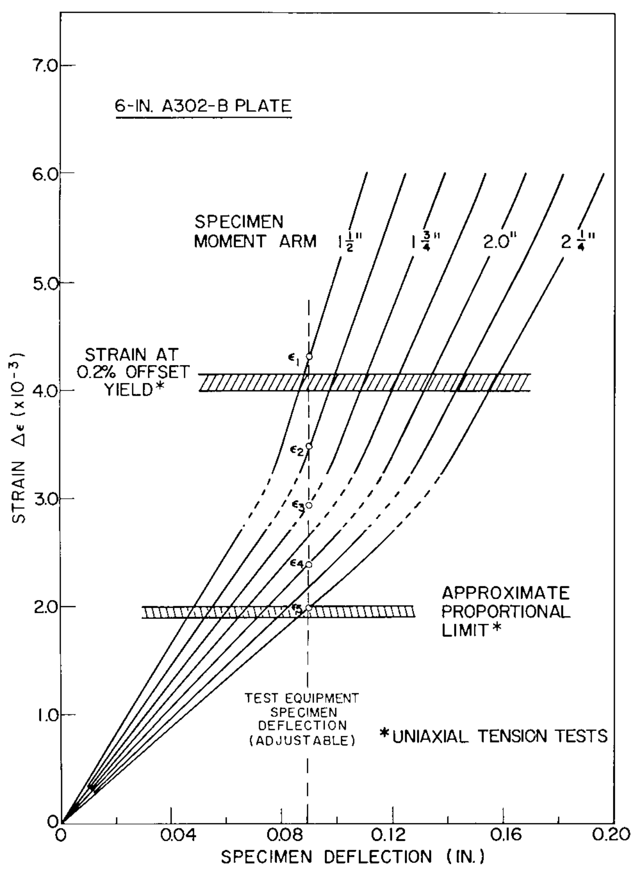

Fig. 2 - A plot of the 1/4 cycle strain developed in specimens of a low alloy steel as a function of deflection for fixed specimen bending lengths

IN-REACTOR FATIGUE TEST APPARATUS

The general experimental objectives considered in the design and development of the in-reactor fatigue test apparatus included the following:

1. The in-reactor assembly was to be designed for insertion in a fuel core position of the Low Intensity Test Reactor (LITR), which has a relatively high fast neutron flux.

2. A maximum number of specimens developing various strain amplitudes were to be tested simultaneously to permit the development of a representative S-N curve with each in-reactor experiment.

3. Test equipment was to provide controlled specimen temperatures in the range 300° to 700°F to encompass conditions of current pressurized water reactor systems.

4. The test equipment and specimen performance were to be continuously monitored during the course of each test.

5. Fully automatic control and recording systems for the monitoring of the experiment were to be incorporated to permit long-term, unattended operations.

Performance tests of the experimental in-reactor assembly and associated control instrumentation developed have shown these design objectives to be fully satisfied.

Fatigue Motor

The basic components of the in-reactor fatigue test assembly are shown in Fig. 3. The final assembly is 17 in. long and has a 2.5-in.-sq. cross section, which permits insertion in a dummy core piece of MTR-type fuel element dimensions. The sequence of component operation which develops fatigue cycling of the specimens begins with the alternate pressurization of the two in-line air cylinders located at the top and bottom of the assembly. The action of these cylinders produces a reciprocating motion of the trunnion which is transmitted to the specimen actuator by means of a roller (ball bearing) and cam arrangement. Finally, a series of uniformly spaced pins on the actuator contact the specimens to produce cyclic deflection of fixed amplitude. The specimen actuator, although free floating, is closely retained within the actuator guide (Fig. 4) for control of directional movement.

As anticipated in the preliminary conception of this equipment, the major problem proved to be the design of a dynamic assembly which "runs dry," that is, without lubrication as normally required by moving parts. At best, various commercially available "radiation resistant" oils and greases would retain their characteristics for only a few days of operation in the nuclear environment contemplated. This problem was solved by incorporating nuclear grade graphite at moving interfaces. As shown in Fig. 3, graphite is used as the air cylinder pistons and for the bearing surfaces of the trunnion. Experience has proved graphite particularly adaptable to this use, since only negligible wear is experienced over long-term service and, with the section sizes employed, dimensional changes of the graphite resulting from radiation-induced growth are not large enough to be troublesome.

Figures 5 through 8 show the in-reactor fatigue test assembly at various stages of fabrication. As indicated with Fig. 3, the fifteen test specimens are arranged in groups of three to provide five average strain amplitudes. The sizes of clamp blocks required to produce specific strain amplitudes are determined from strain measurements on duplicate specimens cycled in a special bench model of the reactor assembly. To verify these results, measurements of individual specimen strain amplitudes are made on completion

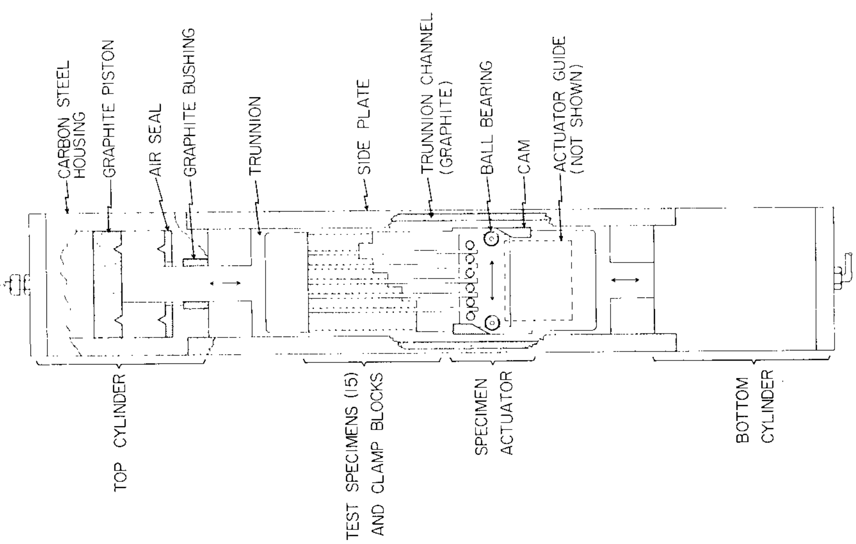

Fig. 3 - Key components of the experimental irradiation assembly developed for in-reactor fatigue testing

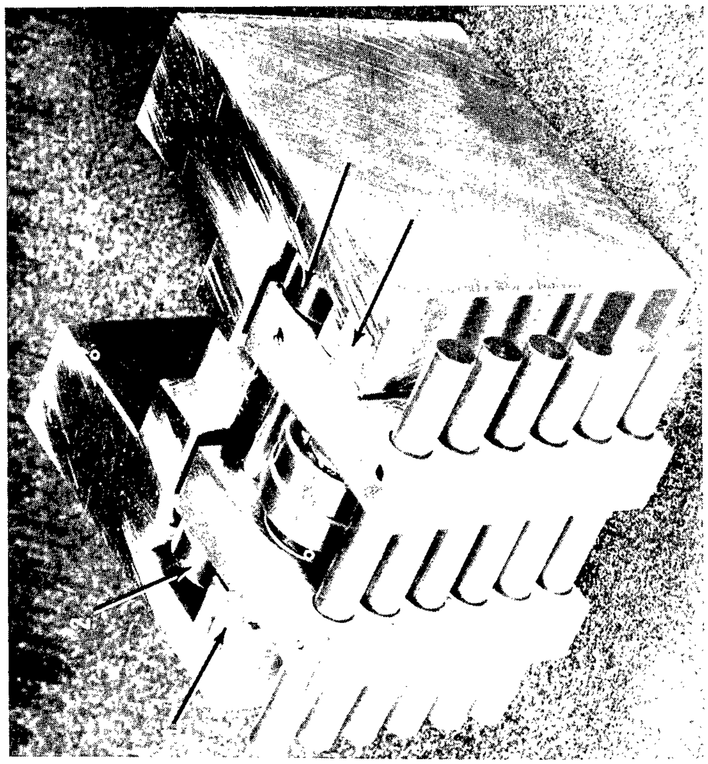

Fig. 4 - The assembly actuator as retained in the actuator guide assembly. The directional movement of the actuator is controlled by the graphite inserts (1) and by the ball bearing rollers (2).

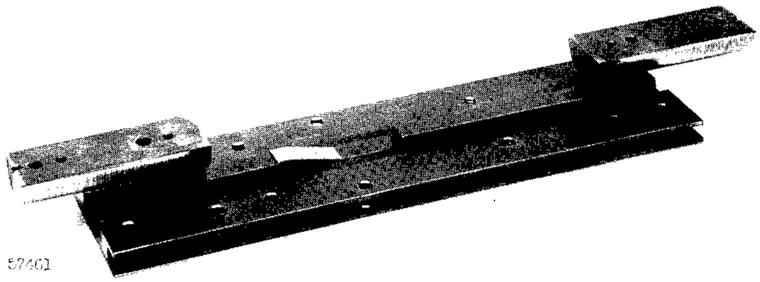

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Fig. 5 – A view of an assembly side plate and one half of the trunnion showing the graphite trunnion channel and the trunnion cam

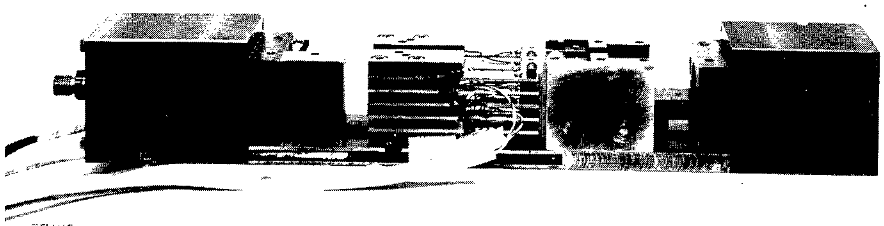

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Fig. 6 – The experimental irradiation assembly partially fabricated. One half of the trunnion assembly was removed for this photograph.

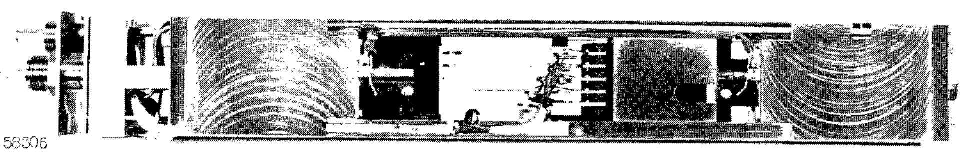

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Fig. 7 – The experimental irradiation assembly in late stages of fabrication

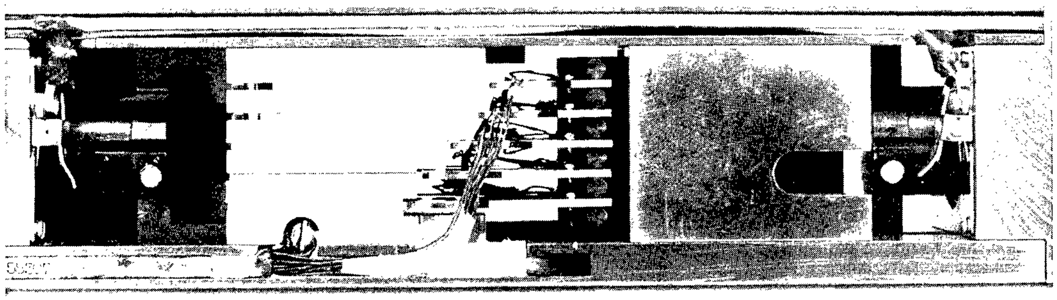

Fig. 8 – A view of the irradiation assembly showing placement of the thermocouples and switches which monitor specimen and assembly performance

of the reactor assembly. Prior to these measurements, the specimens are subjected to a minimum of fifteen complete cycles to establish hysteresis characteristics.

Figure 8 indicates the number of thermocouples employed for temperature control and recording. Each specimen is served by at least one thermocouple. Also shown in this figure are the two switches which continuously monitor the action of the air cylinders and trunnion motion, indicating whether or not full specimen deflection is attained. Determination of the neutron flux dosage ($n/cm^2 > 1$ Mev) received during the course of the experiment is made from iron and titanium threshold detector wires placed at selected locations in the clamp blocks and in the channels provided for thermocouple and switch leads.

Fatigue Motor Control System

The cycling controls for the fatigue test assembly, shown schematically in Fig. 9, perform three basic functions. These include: (a) the automatic interruption of experiment cycling when the reactor power drops below a set minimum, (b) the intermittent operation of the experiment for specific periods at selected time intervals, and (c) the actuation of the experiment air cylinders in accordance with the desired rate of fatigue cycling. The first function is performed by the reactor power sensor, which is simply a potentiometer which monitors the body temperature of one of the air cylinders. By design, the temperature of the cylinders is governed primarily by the level of ambient gamma heating, which varies with reactor power. The second function of the control system is achieved by the timer circuit. Although incorporated for possible future detailed investigations of the influence of cycling pattern on fatigue behavior, this circuit may also serve to periodically check the performance of the experiment and the instrumentation systems. The third system function, the control of experiment air cylinder pressurization sequence, is accomplished with two switches activated by a motor driven cam (Fig. 10). Electromagnetic counters (SV1C and SV2C) in series with the individual solenoid valves (SV1 and SV2) are employed to monitor the switch action. By selecting motors of different rpm and various cam configurations, a range of specimen cycling rates can be obtained.

A schematic diagram of the air system for the fatigue test experiment has also been included in Fig. 9. The combination filter and dryer removes small foreign particles and moisture from the air stream which might affect the operation of the solenoid valves or the in-reactor equipment. The discharge of the assembly air cylinders is vented directly to the reactor off-gas system by the three-way solenoid valves.

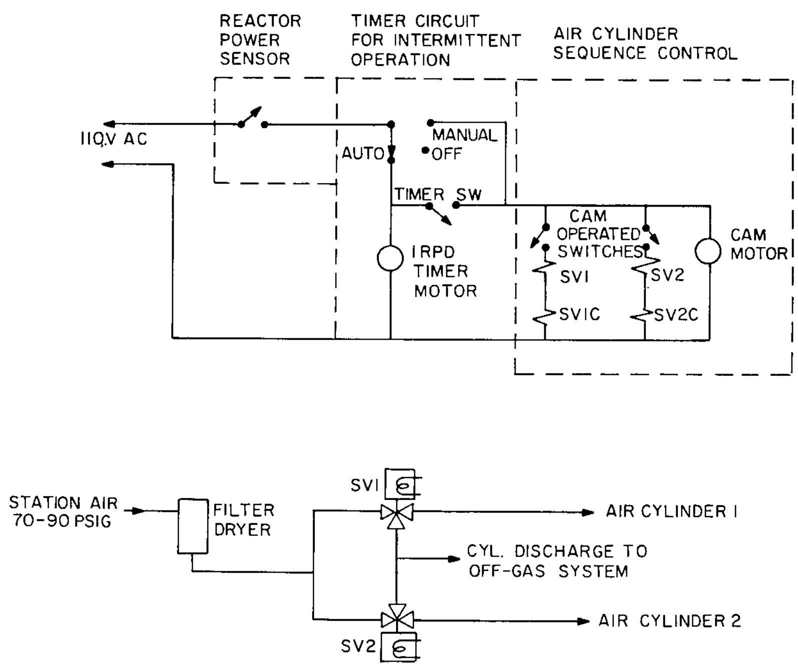

Fig. 9 - Schematic diagram of the assembly cycling control system and components of the external air system

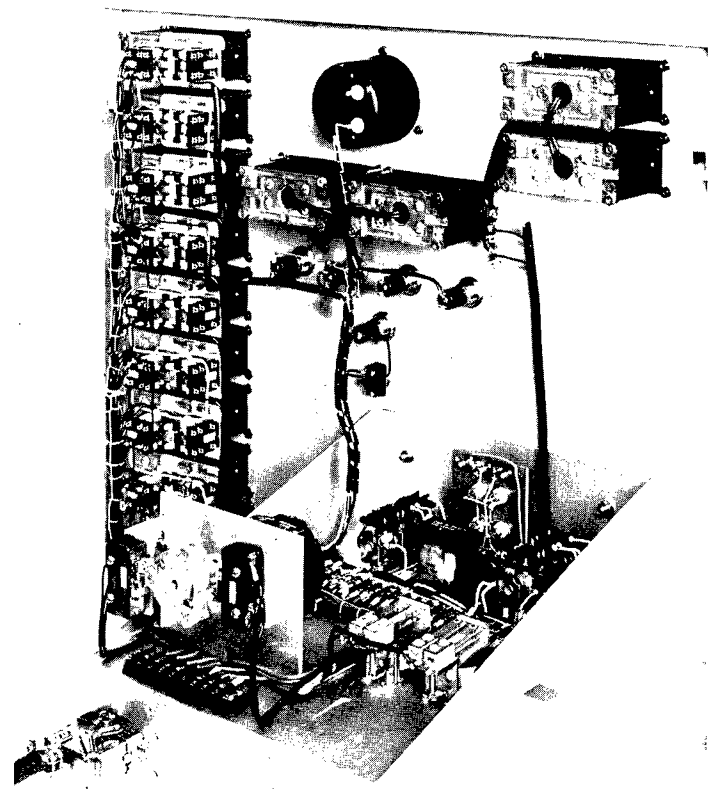

Fig. 10 - Rear view of the air cylinder sequence control panel showing the cam motor and cam-actuated switches which determine the fatigue cycling rate

Specimen Break Detection Systems

One system developed for monitoring the fatigue resistance of individual specimens takes full advantage of the specimen temperature control and recording system. By employing a high sensitivity temperature recorder equipped with upscale thermocouple burnout protection, it was found that both crack initiation and specimen failure could be detected. As illustrated in Fig. 11(a), the development of a small fatigue crack produces an oscillation of the recorder pen corresponding to the opening and closing of the crack in the specimen. Examination of specimens cycled in control tests indicate that cracks in the order of 3/32 in. long can be detected by this method.

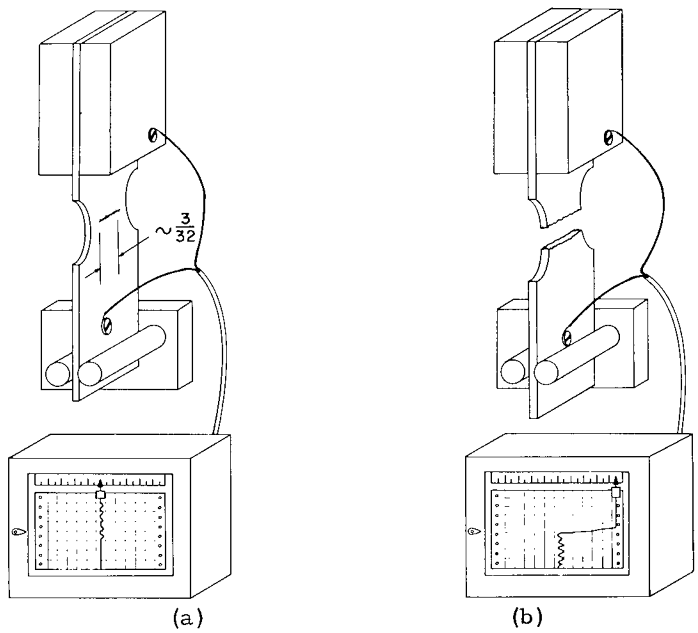

Fig. 11 – Monitoring of specimen fatigue resistance with a temperature recorder having thermocouple burnout protection; (a) recorder response to the development of a small fatigue crack, (b) recorder response to complete failure of the test specimen

Figure 11(b) illustrates the recorder response to the complete failure and separation of the test specimen. The recorder, sensing an open thermocouple circuit, drives the print wheel fully upscale. This action, however, is dependent upon the complete electrical isolation of each test specimen. To achieve this isolation, aluminum oxide coated specimen actuator pins and spacers are used.

Since console space is at a premium at the test reactor installation, a multipoint recorder was modified to enable the monitoring of the fatigue behavior of all specimens with one instrument. A second thermocouple selector switch and fifteen back-set switches were added to the recorder to provide individual circuits for relays which operate the specimen cycle counters. The relays are of the latching type and prevent the re-energization of the counters once failure of the specimen has been detected.

An alternate system for the detection of crack initiation and specimen failure has also been developed and is designed to monitor electrical resistance changes of the specimen. With this system a transistor controlled relay stops a clock when a predetermined resistance, corresponding to either crack initiation or specimen failure, is exceeded. This equipment requires separate leads to each specimen, however, and thus adds to the complexity of the in-reactor assembly.

Temperature Control System

The system used for the control of specimen temperatures is simple, yet very accurate and dependable. Basically, the system controls the specimen temperatures by varying the heat transfer characteristics of the experimental assembly. The functioning of this system depends upon: (a) the attainment of a stagnant temperature (no external control), developed by gamma heating, below that desired for irradiation, (b) a flexible containment sheath which can be moved away from the enclosed assembly by internal pressurization, and (c) instrumentation to automatically increase the internal pressure on the sheath when temperatures are too low and to decrease this pressure when temperatures are too high.

As shown in Fig. 12, the in-reactor assembly is double encapsulated with two welded stainless steel containment sheaths. The annulus between the inner sheath and the flexible outer sheath is pressurized with air by a pneumatic controller in accordance with a thermocouple signal. The action of the outer sheath during heating and cooling situations is depicted in Fig. 13. Although some fatigue action on the outer sheath is developed, this action is minimal since small increments of pressure change result in large temperature changes. Experience has shown that, with a controller having proportional band features for selective pressurization of the assembly according to need, individual specimen temperatures can easily be maintained within ±5°F at operating temperatures in the range 300° to 700°F. Temperature gradients developed between individual specimens are held to less than 15°F by the clamp block arrangement, which acts as a heat sink.

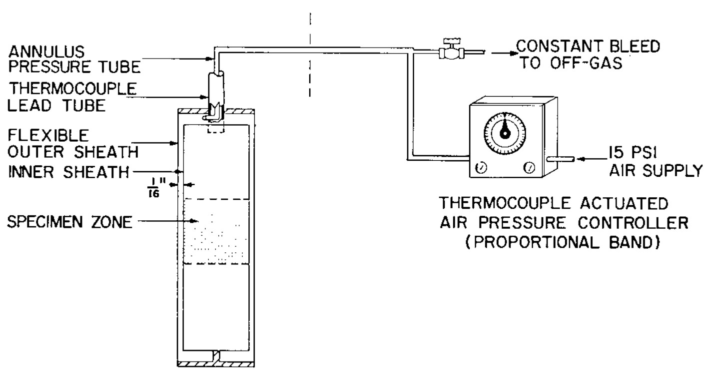

Fig. 12 – Design of the assembly double containment and air pressurization system for specimen temperature control

PRELIMINARY EXPERIMENTAL RESULTS

Material and Specimen Location

An ASTM type A302-B steel was selected for the initial in-reactor fatigue test experiment. The chemical composition, heat treatment, and mechanical properties of this plate are given in Appendixes A and B. For representation of average 1/4-thickness properties, a specimen distribution shown in Fig. 14 was used.

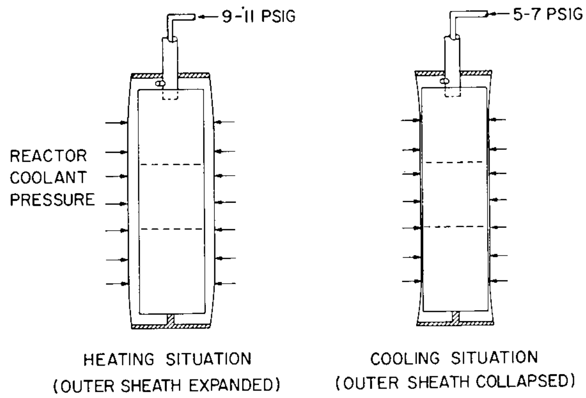

Fig. 13 - Action of the assembly outer containment sheath during specimen heating and cooling situations

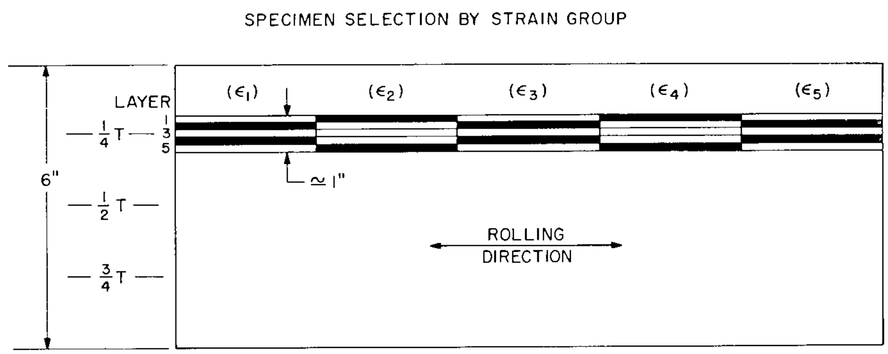

Fig. 14 - Distribution of specimens in the test plate for representation of 1/4-thickness properties. Each average strain group consisted of three test specimens as shown.

Experiment Irradiation and Cycling Procedures

The irradiation tests were conducted in a fuel core position of the Low Intensity Test Reactor (LITR) at a controlled temperature of 500°F. The average instantaneous fast neutron flux in the core facility utilized was $\sim 3.2 \times 10^{12}$ n/cm^2-sec (>1 Mev) as determined from iron and titanium dosimeters included in the experiment. During irradiation a partial helium atmosphere was maintained in the assembly to minimize corrosion.

Cycling of the specimens was performed in two phases. Cycling operations were limited to no more than 20 cycles per day during the first phase of operation. This phase of approximately four weeks duration permitted the accumulation of a significant neutron dosage before initial specimen failures. After an exposure of $\sim 8 \times 10^{18}$ n/cm^2 (>1 Mev) and 465 cycles, continuous cycling at 18 cycles per minute was initiated. The experiment was terminated after 250,000 cycles.

Summary of Test Results

The results of the in-reactor fatigue test and 500°F control data obtained with duplicate test equipment are compared in Fig. 15. The data points represent the cycles to failure of individual specimens. The approximate neutron dosage accumulated at various times during the in-reactor test has been included in the figure for reference. A summary of information pertaining to each specimen test is given in Table 1.

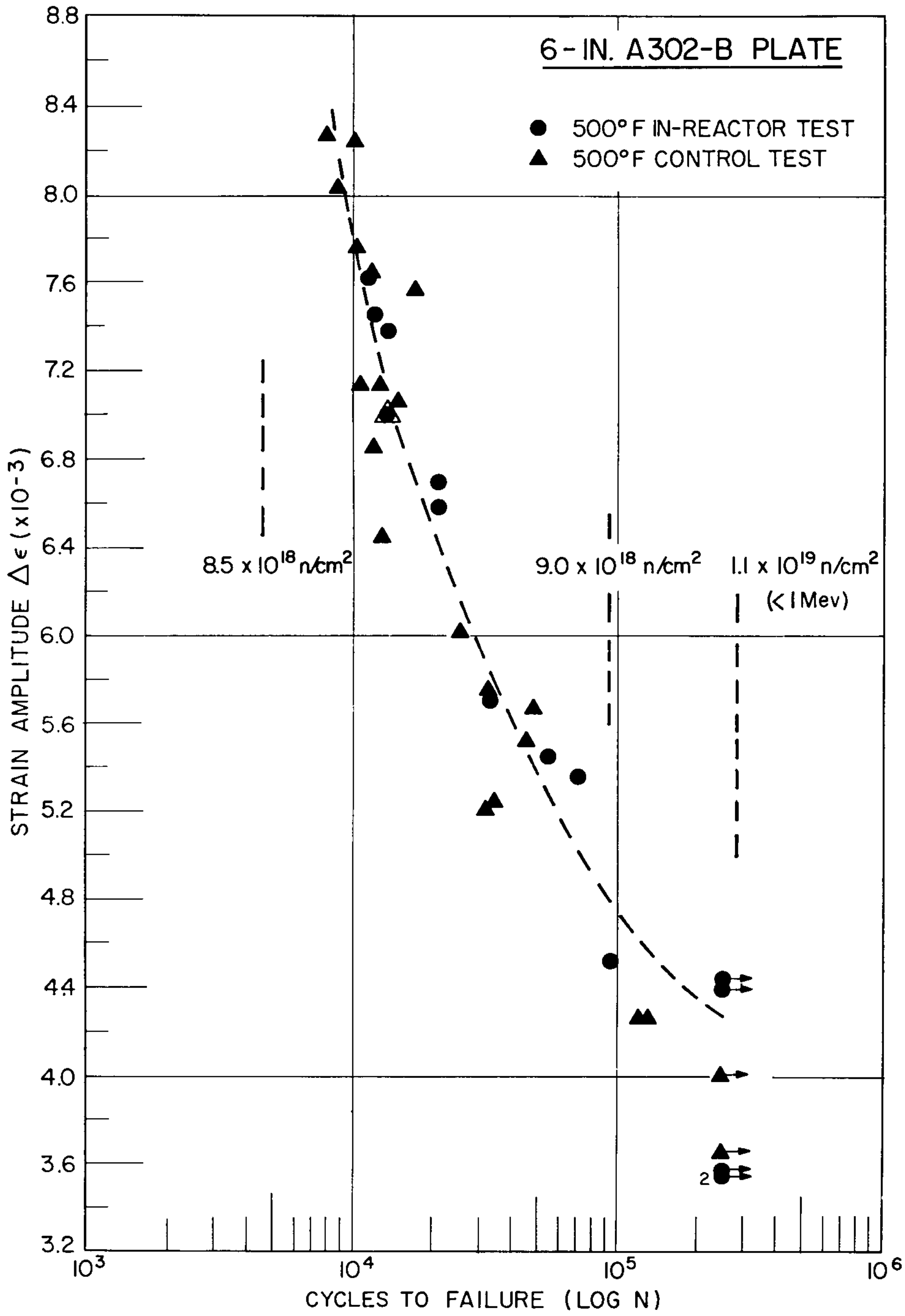

Fig. 15 - Comparison of fatigue data for ASTM type A302-B steel developed by in-reactor tests at 500°F with the results of out-of-reactor control tests. The measurements of strain amplitude were performed at room temperature.

Table 1
Summary of Experimental Data
on the Fatigue Resistance
of ASTM Type A302-B Steel at 500°F

Strain Amplitude $\Delta\epsilon$ ($\times 10^{-3}$)	Cycles to Failure
A. In-Reactor Tests	
7.62	11,168
7.44	12,040
7.38	13,356
7.02	13,143
6.70	20,865
6.58	20,649
5.70	32,309
5.46	54,490
5.36	70,262
4.52	93,393
4.44	*
4.40	*
3.58	*
3.54	*
3.54	*
B1. Out-of-Reactor Control Test 1	
8.26	8,128
8.24	10,170
8.04	8,788
7.62	11,899
7.14	10,815
7.14	12,710
6.14	25,629
5.74	32,556
5.66	47,092
B2. Out-of-Reactor Control Test 2	
7.74	10,266
7.56	17,204
6.84	11,806
7.00	14,040
6.44	12,886
5.52	44,336
5.24	33,565
5.22	30,866
4.28	121,108
4.26	130,212
4.00	†
3.64	†

*Specimen intact after 250,000 cycles.
†Specimen intact after 245,000 cycles.

The data from the two test series do not indicate any pronounced difference in the fatigue strength of the material in the irradiated versus the unirradiated condition, although there is perhaps a suggestion that the strength of irradiated specimens is higher at the high (>50,000) cycle end of the curve. This effect would be consistent with the findings of other investigators (4,5) of strain controlled fatigue regarding the correlation of fatigue life with ultimate strength for high numbers of cycles. However, conclusions concerning material response to irradiation must await the development of more extensive data.

The fatigue life observed for this experimental steel in unirradiated control tests, when compared to that determined by other investigators (6,7) for other heats of this steel at 400° to 675°F, may appear somewhat low. This difference may be attributed to a number of experimental variables, including specimen design, method of strain measurement, cycling rate, definition of failure, and the properties of the test materials. However, the data presented here were obtained with duplicate test equipment using identical experimental techniques, which permitted a direct comparison of fatigue life under the environmental conditions investigated.

DISCUSSION

The effects of neutron irradiation on the fatigue resistance of reactor structural materials is not readily predicted from the response of tensile properties to irradiation. On one hand a possible increase in fatigue strength is suggested by the known increase in yield strength during irradiation. Conversely, because of the concurrent loss in ductility, a reduction in fatigue life might be projected. Thus, experimental tests to evaluate the combined influence of an increase in strength and a loss in ductility on fatigue resistance are necessary.

One approach to this problem has been outlined. The experimental in-reactor fatigue test equipment developed has proven highly successful and capable of performing tests simulating a wide range of reactor service conditions. Although the initial exploratory study using this equipment indicated a material insensitivity to neutron irradiation, additional experiments investigating other materials as well as the influence of known exposure variables are necessary.

FUTURE PLANS

Exploratory investigations of the effects of nuclear radiation on the fatigue properties of pressure vessel steels are continuing and will assess the general effects of irradiation temperature, neutron dosage accumulation, and fatigue cycling pattern. In addition to ASTM type A302-B steel, other select steels and alloys are being considered for study.

ACKNOWLEDGMENTS

The NRL studies on the effects of irradiation on the fatigue properties of reactor structural materials is supported by the Office of Naval Research and the Navy Bureau of Ships, Code 341-A. The steel for the foregoing investigation was supplied by the U.S. Steel Corporation.

The authors wish to thank Messrs. R. F. Bryner, Jr., F. F. Newman, and J. R. Reed for their assistance in the development and fabrication of the test equipment and associated electronic instrumentation.

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APPENDIX A

CHEMISTRY AND HEAT TREATMENT OF 6-IN. ASTM TYPE A302-B PLATE

Chemistry (Weight Percent)

C	Mn	Si	P	S	Ni	Cr	Mo
0.20	1.31	0.25	0.012	0.023	0.20	0.17	0.47

Heat Treatment

A. Austenitized at 1650°F for 2 hours; water quenched.

B. Tempered at 1200°F for 6 hours; furnace cooled to below 600°F.

APPENDIX B

MECHANICAL PROPERTY TEST DATA FOR 6-IN. ASTM TYPE A302-B PLATE

A. Tension Test Data for 1/4 Thickness

Test No.*	Yield Strength (0.2% Offset) (psi)	Tensile Strength (psi)	Elongation in 2 in. (%)	Reduction of Area (%)
1	69,500	91,600	21.0	66.6
2	72,200	93,500	20.0	67.3

*Tests performed at room temperature using 0.180-in.-diam. 1-in.-gage-length specimens.

B. Notch Impact Test Data for 1/4 Thickness

1. Drop Weight Test Results:

   Nil-Ductility Transition (NDT) Temperature was +10°F

2. Charpy V-Notch Test Results:

Temperature (°F)	Energy (ft-lb)	Shear (%)
100	86, 90	99, 100
90	77, 84	88, 98
80	66, 82	81, 98
70	56, 82	67, 85
60	44, 50	56, 60
50	37, 50	48, 58
40	41, 42	48, 52
30	33, 42	44, 46
20	31, 34	42, 42
10	24, 29	28, 42
0	23, 25	30, 34
-20	16, 18	15, 17
-40	10, 14	6, 10
-60	5, 8	2, 2