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**MATERIALS AND MECHANICAL DESIGN ANALYSIS
OF BORON CARBIDE REACTOR SAFETY RODS —
FINAL REPORT (U)**

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EXECUTIVE SUMMARY

The purpose of this task was to analyze the materials and mechanical design bases for the new boron carbide safety rod. These analyses included examination of the irradiation response of the materials, chemical compatibility of component materials, moisture considerations for the boron carbide pellets and susceptibility of the rod to corrosion under reactor environmental conditions. A number of issues concerning the mechanical behavior were also addressed. These included: safety rod dynamic response in scram scenarios, flexibility and mishandling behavior, and response to thermal excursions associated with gamma heating. A surveillance program aimed at evaluating the integrity of the safety rods following actual operating conditions and justifying life extension for the rods was also proposed.

Based on the experimental testing and analyses associated with this task, it is concluded that the boron carbide safety rod design meets the materials and mechanical criteria for successful operational performance.

INTRODUCTION

During the Emergency Cooling System (ECS) phase of a postulated Loss of Coolant Accident (LOCA), the core coolant level would drop potentially causing elevated temperatures in the safety rods from gamma heating. It is possible that these high temperatures could cause failure in the cladding of the present cadmium-based safety rods in SRS reactors [1]. As an alternative, a boron carbide safety rod was designed to meet the requirements of the functional design criteria. The design of the new safety rods is based on an alternating stack of boron carbide pellets and stainless steel spacers contained in a 304L stainless steel tube. The alternating design provides a weight distribution similar to the Cd safety rod while maintaining adequate negative reactivity for the rod. To assure the materials and mechanical design basis of the boron carbide safety rods, a number of experimental and analytical activities were undertaken at the request of the Reactor Restart Division (RRD) as detailed in the "Materials and Mechanical Design Analysis of Boron Carbide Safety Rods" task plan [2].

To effectively assess the design basis of the safety rods, the task was divided into a Materials Design Package and a Mechanical Design Package. The materials design package included four subtasks: irradiation response of component materials, chemical compatibility of materials, corrosion susceptibility and moisture considerations for B_4C . The mechanical design package was comprised of safety rod dynamic behavior (scram response), flexibility and mishandling behavior, thermal response of safety rods and

mechanical design certification. In a number of these analyses both normal operating conditions as well as hypothesized accident scenarios were considered to thoroughly evaluate the safety rod response and materials integrity. An additional subtask involved assisting in the definition of a surveillance program for the new safety rods. This surveillance is intended to provide the data necessary to form a technical basis for service life extension of the safety rods.

The results of these analyses are presented in this report. The "Safety Rod Characteristics" section discusses important aspects of the safety rod design. This section precedes the results section and highlights important design features considered in the analysis efforts.

SAFETY ROD CHARACTERISTICS

A number of factors were considered in the design of the alternative safety rod in an attempt to duplicate the functionality and physical characteristics of the cadmium safety rod. These considerations included: overall rod weight, weight distribution, external dimensions, flexibility, and nuclear reactivity (ie. neutron rod worth). A relatively short term development, analysis and fabrication program was also necessary to coincide with the reactor schedule.

A boron carbide based rod was determined to be the best alternative to satisfy the conditions specified above [3]. Boron carbide (B_4C) is a ceramic material which has many advantages for neutron absorber applications. These include: a high neutron worth, proven use over many years to median exposures in control rods, relatively low cost, broad vendor base for high quality powder and pellets and good chemical compatibility with clad materials [4]. The material was, thus, a proven candidate for the SRS safety rod application.

The design is based on an alternating stack of boron carbide pellets and stainless steel spacers (Figure 1). This configuration provides the desired rod weight and weight distribution as well as a neutron worth similar (~30% greater) to the Cd rod. The pellet stack and internal structural components are contained within a 304L stainless steel tube. The stainless steel tube is fabricated from one inch 304L stainless steel tubing that is swaged to a 0.940" OD with a wall thickness of 0.049". The minimum clad tube ID is specified as 0.826". The pellet vendor (Advanced Refractories Technology, Inc., Buffalo, NY) cold pressed and sintered the B_4C pellets to approximately 72% of theoretical density. These cylindrical pellets are nominally 1" long by 0.795" in diameter. The stainless steel spacers are nominally 0.6" long by

0.795" in diameter with a 0.15" x 15° bevel. The active core stack consists of 104 B₄C pellet/stainless steel spacer pairs in each rod. Moisture in the stainless steel tube is minimized during assembly by careful drying and "flushing" of the rod with helium. An axial loading spring provides 45 to 55 lbs-force to stabilize the pellet stack. The bottom and top safety rod fittings are connected with mechanical roll joints. The rod is backfilled with helium and a non-load bearing weld is used to atmospherically seal the rod. The rod was designed to meet the design life criteria of five years as specified by the Reactor Restart Division (FRD).

RESULTS AND DISCUSSION

Materials Design Package

Irradiation Response of Materials

The irradiation response of materials was primarily assessed using the existing literature. A memorandum (SRL-MTS-920037) discussing the results of this analysis was issued [5]. The "Summary and Conclusions" section of that report follows. A more detailed discussion of this subtask can be found in the referenced memorandum.

The anticipated levels of irradiation for the new boron carbide safety rods are not expected to limit the performance of the safety rods during the five year design life either under normal conditions or during an inadvertent two week exposure in an operating reactor. In either case, the safety rods are projected to survive a postulated LOCA event and associated gamma heating. Temperatures from gamma heating as high as 800° C can be tolerated by the safety rods under normal operating conditions and after an inadvertent irradiation of two weeks (longest historical inadvertent irradiation at SRS).

Internal pressurization of the safety rod by helium release and pellet swelling, and irradiation effects to the stainless steel sheaths were evaluated. Internal pressures, pellet swelling, and pellet fragmentation are negligible after five years service under normal operating conditions with the rods withdrawn during reactor operation. For an extremely conservative case of a rod exposed for two weeks at 35% reactor power (0.3 percent burnup of boron-10), the highest calculated internal pressure is 165 psi. This assumes a uniform boron-10 burnup throughout all pellets, a gas temperature of 400° C, a proportionate distribution of the helium between the internal void space in each pellet and the free space within the rod, and includes the helium introduced during fabrication. A seemingly more precise expression, based on empirical work by Homan

on actually irradiated B_4C , accounted for the accommodation of He in the boron carbide crystal lattice [6]. Calculated helium pressure values using this expression and corresponding hoop stresses are found in Table I. The axial and hoop stresses are significantly less than the yield strength of the swaged sheath ($\sim 40,000$ psi) in contact with water at $\sim 100^\circ C$. Estimated pellet swelling is less than 0.13 percent volume increase; therefore there is no deformation of the sheath due to contact between the pellet and sheath. Based on experience with boron carbide control rods, pellet fragmentation is not expected for a B-10 burnup < 1 atom percent.

A temperature as high as $800^\circ C$ could be tolerated for up to ten hours without creep rupture of the sheath (Table II and Figure 2). Figure 3 shows a composite of the mechanical properties and pressure values following an inadvertent irradiation for two weeks at 35% historical power. The safety rods would survive the postulated LOCA event and subsequent gamma heating even if the accident were to occur immediately after exposure of a rod in a reactor that had operated at 35% power for two weeks.

Additional analyses and experiments were performed to examine the response of the boron carbide pellets to thermal gradients postulated to occur with irradiation. Conservative calculations at 50% historical power showed that a thermal gradient across the pellet would not exceed $250^\circ C$ [7]. Worst case calculations, excluding contributions of pellet thermal conductivity, heat transfer of medium in contact with the pellets, and geometry and microstructure of the pellets showed that a thermal gradient as low as $\sim 70^\circ C$ could cause fracture of the pellets [8]. In contrast, including the aforementioned factors, calculations showed a thermal gradient in excess of $700^\circ C$ would be necessary for pellet fracture [9]. It is clearly evident, especially considering the nature of fracture of ceramic materials, that a thermal gradient intermediate of these values would likely cause pellet fracture.

Thermal shock experiments were performed to examine the effects of gradients between these extreme values [10,11]. It is important to note that a thermal shock test is a very conservative test for examining thermal gradient effects on materials as very sharp thermal gradients at the surface are created. Actual thermal gradients for irradiated pellets would be much less severe. In general, the pellets exhibited good thermal shock resistance and no detrimental effects to safety performance are anticipated.

A $475^\circ C$ quench from a heated furnace to water resulted in no fracture or fragmentation of the pellets [10]. Examination of pellet sections revealed some "microcracking" resulting from the shock (Figures 4 and 5). It is apparent that the extensive porosity in the pellets (approaching 30%) effectively blunted the cracks

causing extensive crack branching thus inhibiting fracture or fragmentation of the pellets.

Thermal shock tests, examining the effects of a 250° C gradient, were performed in accordance with ASTM Standard D116-86 [11]. Heated glycerine served as the "hot" medium while ice water was the "cold" medium for this test. Ten cycles from one medium to the other were completed for 3 pellets in each of three separate tests. In each of the three tests, one pellet fractured in half on the tenth (last) cycle. The fracture was in the axial direction apparently originating at a pre-existing corner chip or flaw and resulting in essentially two identical halves (Figure 6). No small pellet fragments resulted from this fracture and no compromise to safety rod performance would occur. Additionally, calculations showed that the pellet stack height would not decrease in the extremely unlikely event of gross pellet fracture. As was evident in the previous thermal shock tests, microcracking occurred to relieve stresses associated with the sharp thermal gradients. In successive shocks, these microcracks undoubtedly coalesced resulting in the axial fracture. However, the extensive porosity again significantly inhibited crack propagation.

Chemical Compatibility of Boron Carbide

The compatibility of boron carbide with the stainless steel cladding and the moderator was evaluated using the existing literature database. A number of studies have evaluated the interaction of B₄C and 300 series stainless steels [12-17]. The general findings show that there is no interaction below 700° C. Dahl [12,13,14] and Thorley [17] developed empirical equations to determine the penetration depth of boron carbide into type 316 stainless steel at elevated temperatures. The most conservative equation [13] was based on experiments involving a 3000 psi induced pressure to assure intimate contact between the B₄C powder and stainless steel. Using this expression, an exposure of almost 500 hours at 1000° C would be required to cause a penetration depth equivalent to the cladding thickness of the safety rod (Figure 7). These time-temperature conditions for clad penetration far exceed the postulated accident conditions.

The potential for interaction of the boron carbide pellets with the moderator was also examined using the existing database [4,18-20]. The literature indicates that there is generally good resistance of boron carbide to water at light water operating temperatures. A study involving 70% dense pellets exposed to flowing water at 307° C introduced by intentional clad defects showed no corrosion of the boron carbide for exposures up to 880 hours [18].

Despite the relatively good compatibility between moderator and boron carbide, concern arose due to the potential loss of soluble boron from this interaction. The proposed mechanism for boron dissolution is by the reaction of boron carbide with water to form B_2O_3 , followed by further reaction of B_2O_3 with water to dissolve as boric acid (H_3BO_3) [4]. This reaction is said to be directly dependent on temperature and B-10 depletion, thus results have shown in-reactor boron corrosion to be significantly greater than ex-reactor corrosion. Two studies discussed the loss of boron following interaction with light water. Experience at St. Lucie showed an approximate 10% loss of boron from Al_2O_3 - B_4C pellets after approximately 1 month irradiation in light water operating conditions [4]. Work by Gray and Lynam found significant loss of boron in thin boron carbide plates under high B-10 burn-up (28-72%) and water temperatures between 232 and 266° C [21]. In the extreme burn-up and temperature cases of this study, essentially a complete loss of the boron occurred from the plates. The corrosion was thought to be enhanced by reaction with the radiolytic decomposition products of the water in the high irradiation cases.

In typical operating conditions, with safety rods withdrawn from the reactor, boron loss due to aqueous corrosion is essentially impossible. When rods are inserted in the reactor, the possibility of a leaking rod is also considered extremely unlikely. Even though pellet/moderator interaction is doubtful, the relatively moderate operating conditions of SRS reactors indicate boron dissolution would be insignificant in a leaking rod. The moderator temperature of SRS reactors (100° C) is significantly lower than the temperatures in the studies above describing boron loss. For a two week inadvertent irradiation (longest historical inadvertent rod insertion at SRS), B-10 depletion is expected to be less than 1 percent. This depletion amount is well below the depletion in the two cases where significant loss of soluble boron occurred. A study by Hart, et al. found that heavy water was more stable than light water with regard to radiolytic conversion [22]. This coupled with the hypothesis of Gray and Lynam [21] that the radiolytic conversion of water caused enhanced B-10 dissolution in light water implies dissolution in heavy water is not as likely because less radiolytic conversion occurs. In addition, chemical analysis of the boron carbide pellets for the SRS safety rods was also below the ASTM specification of 0.2% for water soluble boron (Table III) [23].

Moisture Considerations for Boron Carbide

The moisture criteria for the B_4C pellets in the assembled rods was less than 750 ppm per ASTM C751-86a (Table III) [23]. Initial attempts were made to determine the

moisture contents of the pellets exposed to various humidity environments using thermogravimetric analysis (TGA). This technique, however, did not provide the sensitivity necessary for the moisture analysis. Constant-voltage coulometry was found to be a more suitable analytical method. This technique was employed by the pellet vendor (Advanced Refractory Technologies, Inc.) in accordance with ASTM C791-83 [24] to measure the moisture contents of the as-received pellets and pellets exposed to various humidity conditions. Coulometry is a very accurate technique to measure moisture content in ceramics. In this method, water is driven from the sample by heating it to 400° C. A carrier gas sweeps the water into an electrolytic cell where the water is absorbed by phosphorous pentoxide. A potential applied across the cell hydrolyzes the absorbed water. The total current is integrated resulting in a displayed value of micrograms of water. A non-conformance report (NCR) was closed assuring that the coulometric analysis performed by ART, Inc. was consistent with Level 1 Procurement requirements [25].

Moisture contents of less than 150 ppm were measured for the as-received pellets. Measurements on pellets exposed to a high humidity environment (~100 % rh) for 4 hours resulted in moisture contents of less than 250 ppm. Additional measurements on pellets exposed to actual assembly conditions (ie. removed from an assembled rod) showed an average moisture content of approximately 120 ppm with no measurement exceeding 130 ppm. Based on these results, an exposure limit of 4 hours to the assembly environment was recommended. This limit will insure that the moisture contents of the pellets in the assembled rods will be well below the ASTM specification of 750 ppm.

Corrosion Susceptibility

Significant corrosion of the safety rods is not anticipated due to the low moisture and chloride specifications for the boron carbide pellets and the exposure limits specified for assembly. The chloride and fluoride contents of the pellets were chemically determined to be less than 5 ppm [26]. These values were well below the ASTM C 751 - 86a specification for nuclear-grade boron carbide pellets of 75 ppm and 25 ppm for chloride and fluoride ions, respectively (Table III) [23].

The concern of chloride content in materials in contact with austenitic stainless steel centers on the potential for chloride stress corrosion cracking (SCC). Stress corrosion cracking is a failure mechanism where typically ductile, corrosion resistant materials undergo brittle fracture with little associated corrosion or loss of thickness. This failure can occur at stress levels which are routinely thought to be too low to promote purely

mechanical fracture. The SRS standard and specification for stress corrosion cracking states that no material with an excess of 250 ppm chloride content should be in contact with austenitic stainless steel [27]. The boron carbide pellets for the safety rod clearly satisfy this criterion. The potential of chloride leaching from the pellets, in the unlikely event of a leaking safety rod, was also considered for the potential effect of concentrating the chloride ions. Calculations, where water was hypothesized to fill the entire pellet column height and leach out all the chloride present in the pellets, revealed a chloride concentration of about 50 ppm in the water [28]. An even more conservative scenario where only one B_4C pellet was exposed to the moderator was also considered [28]. In this case, a concentration of less than 90 ppm resulted. It is extremely unlikely that these low chloride concentrations would cause stress corrosion cracking. A parallel assessment involving stress corrosion crack growth rate calculations if SCC did occur indicated safety rod failure would not likely occur [28]. In this analysis, crack growth due to SCC following a two week interaction with the moderator was calculated to be less than 0.2 inches. A total crack length was found by adding the growth amount to the "pinhole crack" length necessary for leaking (ie. the depth of the cladding). This value (~0.22") was significantly less than the critical flaw size (essentially the cladding circumference of ~3") necessary for unstable fracture of the safety rod.

Leaching experiments were performed in an attempt to quantify relative chloride leach amounts and rates from the boron carbide pellets. These tests were performed in 95° C distilled/deionized water for a two week interval using condenser loops to prevent evaporation of the water [29]. Five experiments were run simultaneously involving a water standard, sample pellet, and pellet fragments and powder in -3000 μm , + 841 μm ; -841 μm , +297 μm ; and -297 μm sizes. Constant pellet mass to water volume ratios were used for all particle sizes to standardize the tests. Chloride concentrations of the water samples were measured to part per billion levels using ion chromatography (IC).

Although the relative chloride leach amounts trended as anticipated (ie. higher chloride leaching for finer particle sizes), there was significant uncertainty and error in the concentration values. The relatively small concentrations, chloride contamination from pellet "handling" and crushing, and IC analysis limitations and interferences contributed to this error. The results from the pellet sample, which did not require excessive "handling", indicated approximately 2 ppm chloride ion leached from the pellet sample. The remaining fragment samples showed relative leach amounts equivalent to or in excess of the measured chloride amounts in the pellets. Since the aforementioned analyses indicated essentially no susceptibility of the rods to corrosion both under normal operating conditions and in the unlikely event of a leaking rod, no additional leach testing was performed.

The presence of free iron in the boron carbide was also apparent following the leach testing. A reddish, brown residue appeared following exposure to the water. This residue was identified using energy dispersive spectroscopy (EDS) as containing elemental iron. The presence of iron in boron carbide powder has occurred elsewhere and is speculated to be an artifact of powder milling [30]. The iron content in the pellets (0.12 wt %) was well within the ASTM C751-86a specification of 0.6 weight percent so no deleterious effects from the iron were anticipated [23]. However, additional experiments were performed to ensure that the free iron would not cause a breakdown in the passive oxide film on the stainless steel. In the presence of a corrosive medium (ie. moisture) iron and stainless steel can potentially form a galvanic couple leading to a breakdown of the protective oxide film on the stainless steel and the possible onset of corrosion. This effect was of primary concern in the assembly environment when the boron carbide pellets were in contact with the stainless steel cladding in the presence of atmospheric moisture. To examine the potential for this effect, pellets were placed in intimate contact with 304L stainless steel coupons for 4 hours in a laboratory environment and in high humidity conditions for 16 hours [31]. Electrochemical impedance spectroscopy (EIS) was used to examine for any breakdown or deterioration of the stainless steel oxide film following this exposure. No changes in the oxide film on the 304L coupons were evident when compared with unexposed samples (Figures 8 and 9).

Mechanical Design Package

Dynamic Analysis of Safety Rods

To examine the response of the safety rod to scram scenarios, two dynamic analyses were performed [32,33]. Both analyses considered the response of the safety rod to a scram scenario involving mechanical snubbing with no electrical snubbing. The first, more conservative analysis did not account for effects from fluid resistance or response of the reactor superstructure [32]. The primary findings from this analysis included: the safety rod was considered resistant to impact loading conditions, the maximum anticipated elongation of the 304L stainless steel clad tube is 0.42", the spring will not lose compression, no separation between pellets and spacers is expected, and a maximum of 6000 lbs force will be exerted on the safety rod bottom fitting. At the request of the Defense Nuclear Facility Safety Board (DNFSB) staffers, a second analyses was performed which included effects attributed to water resistance and the reactor superstructure [33]. This was performed primarily to assess the degree of conservatism in maximum force calculations. Accounting for these two factors, a maximum force of approximately 5460 pounds was estimated to be exerted on the

bottom fitting resulting in a decrease of about 9% or 540 pounds from the previous estimation.

In response to the maximum load findings, tensile tests were performed on the bottom fitting to ensure the mechanical integrity of this joint under scram conditions [34]. Load to failure values ranged from approximately 7000 pounds to 11,000 pounds. It should be noted that the 7000 lbs measurement was for an "initial production joint" while tensile test measurements on rods fabricated following optimization of the joining process were generally about 10,000 lbs. The rod integrity in this scram scenario was further substantiated by the mechanical snubbing tests performed by the Equipment Engineering Section (EES) of SRL [35]. In these tests, no end fitting failure or degradation was evident in safety rods subjected up to and including 5 drops on a mechanical snubber.

Flexibility and Mishandling

Flexibility testing was performed to compare the stiffness of the boron carbide safety rods to the cadmium rods and to examine for damage to safety rod components following extensive bowing [36]. Following testing, radiography was performed to examine and assess any damage to safety rod internal components [37]. Midpoint deflection tests indicated the B₄C safety rod has roughly 60% of the stiffness of the cadmium rod. This result correlated well with the findings of the Equipment Engineering Section (EES) in their assessment [35]. Drop tests performed by EES also showed that this greater flexibility had no effect on the drop times of the safety rods [35]. Bowing tests, involving an induced 12" midpoint deflection, resulted in no damage to the stainless steel cladding or pellets.

Although various precautions exist, including rigid support of the safety rods from assembly to implementation in the reactor, abuse tests were performed to assess safety rod damage in postulated mishandling scenarios [36]. Radiography was again used to examine the internal components of the rods, in particular the B₄C pellets. Dropping of a safety rod from approximately 3-1/2 feet in the horizontal position, as might result from rolling off a platform or workbench, resulted in minor chipping to B₄C pellets at the ends of the pellet stack. The damage to one pellet was attributed to a machining tolerance error which was remedied before the final safety rod production run (Figure 10). Handling and movement of a rod by a person at each end of the rod resulted in no apparent damage to the stainless steel cladding or rod internals. The handling and transport of a rod in the middle by a single person, however, resulted in permanent bowing of the steel cladding at the point of handling.

Thermal Response Tests

The potential for failure of the cadmium safety rods in certain postulated accident scenarios necessitated a safety rod design capable of withstanding exposure to elevated temperatures. It was expected that a boron carbide based safety rod would provide the refractory properties needed to withstand these conditions. To ensure this, a number of tests were performed to gauge the thermal response of the new safety rods [38,39]. This was accomplished by using a specially designed 4 foot test rod with an internal solid to void volume ratio equivalent to a full length rod (Figure 11). The test configuration included the exposure of a 2 foot pellet column to the hot zone of a furnace. The rod was backfilled with He to a pressure of 1 atm to correspond to normal operating conditions. The internal rod pressure was monitored continuously by a pressure gauge on the end of the rod. Temperature throughout the tests was monitored by a series of 7 thermocouples equally spaced and welded along the length of the rod. Tests were run under "fast" ramp-rate conditions with maximum temperatures of 600, 800 and 1000° C. The "fast" ramp tests involved a 15 minute ramp from ambient to the maximum temperature, followed by a 1 hour hold, and a 50° C/hr cooling ramp-rate (Figure 12). "Slow" ramp-rate tests involved a 4° C/min ramp-rate to maximum test temperatures of 800 and 1000° C (Figure 13). Visual inspection, dimensional measurements and radiography were used to assess changes to the test rods following exposure to the elevated temperatures.

A test rod exposed to a 600° C maximum temperature using the fast ramp profile showed no dimensional changes or degradation. A "darkening" of the surface was observed as anticipated due to oxide film formation on the stainless steel. This same test rod was then exposed to 800° C under fast ramp conditions. Radiographic analysis, visual inspection and dimensional measurements indicated no changes to the rod occurred. A final fast ramp profile to 1000° C with this same test rod, followed by a complete inspection, again revealed no safety rod degradation. Confirmatory testing was also performed on a second test rod. This testing similarly indicated that successive exposure of the safety rods to 600, 800 and 1000° C caused no apparent damage to the exterior cladding or internal components of the safety rods. These findings confirmed that clad rupture and degradation is unlikely under thermal cycle conditions expected under the gamma heating scenarios.

The slow ramp-rate tests were performed using an additional test rod. The visual, dimensional and radiographic inspection following exposures to 800 and 1000° C also showed no safety rod degradation.

Mechanical Design of Safety Rods

The "Materials and Mechanical Design Analysis of Boron Carbide Reactor Safety Rods" task (Task No. 91-047-1) was originally subtask 12 (Investigation of Alternative Designs and Mitigating Operational Procedures) of the "Reactor Components Response to Gamma Heating Accident" task (Task No. 91-001-1) [40]. The boron carbide safety rod design evolved from subtask 12 of Task No. 91-001-1 and was incorporated into a new, separate task (Task No. 91-047-1) [2]. The QA plan and task technical plan were not formalized for the new task until after design activities were initiated. The final approved design, however, was completed as part of Task No. 91-047-1. A nonconformance report (NCR) was written indicating that only preliminary design work was performed under Task No. 91-001-1 and that the final design was completed and approved under Task No. 91-047-1 [41]. The NCR was closed on February 18, 1992.

Surveillance Program

A memorandum "Safety Rod Surveillance For Life Extension" was issued [42]. Since the analyses indicate that the safety rods can be expected to perform satisfactorily long beyond the prescribed five year design life, it was deemed important to develop a surveillance program to justify life extension of the safety rods. While the fabrication cost of replacing the rods every five years would not be prohibitive, the benefits from waste minimization and limitation of employee radiation exposure are attractive. The recommended surveillance program consisted of radiography and photography of a fraction of safety rods before installation to provide a "baseline" database. Periodically, at scheduled shutdowns, a portion of these baselined rods would be visually reinspected and radiographed. An assessment of mechanical rod degradation would be made by these nondestructive techniques. Spatially resolved activity measurements of the lower end of the safety rod could be made to calculate pellet swelling.

Due to the uncertainty in K-Reactor status and the investment necessary in implementing the surveillance program, the program will not be implemented at the present time. The task team will provide consultation on any future implementation of a surveillance program if it is deemed prudent.

CONCLUSIONS

The boron carbide safety rod design is sound from a materials and mechanical standpoint based on the conclusions of the subtasks:

1. Anticipated levels of irradiation for the boron carbide safety rods will not limit the performance of the rods during the five year design life either under normal conditions or during an inadvertent two week exposure in an operating reactor.
2. Boron carbide is compatible with the stainless steel cladding. In the unlikely event of a leaking safety rod, the interaction of the boron carbide pellets and the moderator will not affect the safety performance of the rods.
3. Moisture absorption by the boron carbide pellets is not sufficient to cause deleterious effects to safety rod performance.
4. Significant corrosion of the safety rods is not anticipated due to the low moisture and chloride specifications for the boron carbide pellets and the exposure limits specified for assembly.
5. The safety rod is resistant to impact loading conditions associated with mechanical snubbing under scram conditions.
6. The boron carbide rod has approximately 60% the stiffness of the cadmium rod. The new safety rod was found to adequately withstand bowing and mishandling.
7. The safety rods can withstand temperatures up to 800° C with no apparent degradation. A test rod exposed to 1000° C with an ambient internal He pressure of 1 atm experienced no apparent degradation.
8. A surveillance program is considered an important aspect for life extension of the safety rods. A program consisting of "baselining" a fraction of the safety rods and inspection of a portion of these rods periodically was proposed.

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TABLE I
HOMAN MODEL FOR HELIUM RELEASE FOR
SAFETY ROD IRRADIATED FOR TWO WEEKS [5]

<u>Temperature (C)</u>	<u>Fraction He Release</u>	<u>Helium Pressure (psi)</u>	<u>Hoop Stress (psi)</u>
100	0.0161	19	155
200	0.0453	27	213
300	0.0886	36	288
400	0.1421	47	381
500	0.2016	61	492
600	0.2640	77	621
700	0.3270	95	765
800	0.3893	115	923
900	0.4499	136	1093
1000	0.5082	159	1275

TABLE II
YIELD STRENGTH AND
EXPECTED STRESS TO RUPTURE VALUES FOR
TYPE 304 STAINLESS STEEL BAR STOCK [5]

<u>Temp. (C)</u>	<u>Yield Strength (ksi)</u>	<u>1 Hour Stress to Rupture (ksi)</u>	<u>10 Hour Stress to Rupture (ksi)</u>
427	16.8	57.0	57.0
482	16.2	55.5	55.5
538	15.6	52.5	50.0
593	14.7	45.0	35.2
650	14.1	32.0	24.7
704	13.2	23.0	17.4
760	11.6	16.5	12.1
815	9.5	12.0	8.6
870*		8.5	6.0
982*	6.0	4.3	3.0

* Values were extrapolated from lower temperature values.

TABLE III
PELLET SPECIFICATIONS (ASTM C751-86a) AND
CHEMICAL ANALYSIS RESULTS

<u>Constituent</u>	<u>ASTM Limits</u>	<u>Chemical Analysis (avg)</u>
Total boron	81.0 wt %, max. 73.0 wt %, min.	75.77 wt %
Nitric acid-soluble boron	0.5 wt %, max.	0.096 wt %
Water-soluble boron	0.2 wt %, max.	0.148 wt %
Calcium	0.3 wt %, max.	0.019 wt %
Iron	1.0 wt %, max.	0.080 wt %
Total boron + total carbon	98.0 wt %, min.	98.71 wt %
Fluoride	25 ppm max.	< 5 ppm
Chloride	75 ppm max.	< 5 ppm
Water	750 ppm, max.	< 250 ppm

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Figure 1. Engineering drawing of boron carbide safety rod (Drawing ST-MDX5-10104).

Figure 2. Temperature dependence of 10-hour creep rupture strengths for type 304 stainless steel. Irradiated specimens contain ~4 appm helium [5].

Figure 3. Temperature dependence of mechanical properties of type 304 stainless steel and hoop stress in safety rod sheath. Internal pressure from helium generated by two weeks exposure at 35% power [5].

Figure 4. Boron carbide pellet sections showing significant crack branching following 450°C thermal shock. Micrograph b) is a magnified section of the lower left hand portion of micrograph a).

Figure 5. Photomicrographs of sections of boron carbide pellets following a 450° C thermal shock. The extensive porosity in the pellets significantly inhibited crack growth.

Figure 6. Micrographs of boron carbide pellet following 10 thermal shock cycles of 250° C. Pellet fractured (a)) axially after the tenth cycle. Micrograph b) shows apparent fracture origination from chip on corner of pellet.

Figure 7. Penetration depth vs. time for elevated exposure temperatures using empirical expression developed by Dahl [13].

Figure 8. Bode plot of 304L control sample and 304L sample exposed to boron carbide for 4 hours in building atmospheric conditions. Virtually no difference in sample behavior indicating no breakdown of oxide film on stainless steel from exposure to boron carbide.

Figure 9. Bode plot of 304L control sample and 304L sample exposed to boron carbide for 16 hours in a high humidity environment. Virtually no difference in sample behavior indicating no breakdown of oxide film on stainless steel from exposure to boron carbide.

Figure 10. Radiograph of safety rod used in bow and abuse tests and "radiograph standard rod". Note the differences in counter-bore depth and apparent debris from machining. These fabrication problems were remedied before the safety rod production run.

Figure 12. Prototypical fast ramp profiles for elevated temperature exposure tests.

Figure 13. Prototypical slow ramp profiles for elevated temperature exposure tests.

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