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Title: ADVANCED TEST REACTOR (ATR)
REFLECTOR SAFETY ANALYSIS

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ADVANCED TEST REACTOR (ATR) REFLECTOR SAFETY ANALYSIS

1. INTRODUCTION

The Advanced Test Reactor (ATR) reflector, composed primarily of beryllium, experiences significant material changes as a result of its prolonged exposure to the intense neutron and gamma ray environment during reactor operation. Radiation-induced lattice vacancies and interstitials, as well as deleterious transmutation products, eventually lead to significant internal stresses in beryllium. The main reactions producing nuclear transformations of concern in beryllium are those which produce helium and tritium.[1] These products result from (n,2n) and (n,alpha) reactions. Gradual buildup of gaseous products, such as helium, contribute to the internal stresses. Diffusion of the gaseous atoms to dislocation centers can introduce stresses which lead to deformation and failure.[2] The internal stresses are largely due to bulk swelling of the material. The internal stresses, in the case of the ATR reflector, are largely due to fast neutron flux gradients from the beryllium/fuel interface to the outside diameter of the reflector and from the center of the core to the top and bottom of the core.

Historically, the aging of beryllium in the ATR has resulted in bowing, cracks, small holes through thin ligaments, and spalling. An understanding of when and where these phenomena are expected to occur allows continued operation of the reflector after some aging effects have occurred. Economic and programmatic considerations make it desirable to continue operation of a reflector which has experienced some aging effects. This report provides the basis for continued reactor operation with an aged reflector. Aging effects beyond those analyzed to be acceptable constitute reflector failures. This report applies to Reflectors IV and V and also to subsequent reflectors if they are designed, manufactured, and operated under the same conditions.

2. REFLECTOR DESCRIPTION

The Advanced Test Reactor (ATR) reflector is composed of machined beryllium components surrounding a serpentine-shaped core. The major reflector components are eight, 51-inch high by 45-degree segments machined from hot-pressed beryllium block. The reflector blocks extend 2 inches above and 1 inch below the four-foot active core. A plan view of the ATR core and reflector is shown in Figure 1. Reflector blocks are numbered in a clockwise manner beginning with No. 1 at the one o'clock position in Figure 1. Near the inner tip of each beryllium block is an "inner B-hole" which extends through the length of the block. The number of each inner B-hole in Figure 1 corresponds to the number assigned to the block. The analysis contained in this report was based on reflector block design drawings 419609-Rev. N/C and 419610-Rev. N/C.

Two outer shim control cylinders (7.24 inches O.D.) are contained within each reflector block. Outer shim control cylinders (OSCC) are primarily composed of beryllium (machined, hot-pressed block) but include curved hafnium plate segments (122 degrees) on the periphery of each cylinder. OSCCs are rotated to adjust the power distribution across the core. Each OSCC is uniquely identified as indicated in Figure 1. OSCC assembly drawing numbers are 401894-Rev. C, 401895-Rev. C, 401896-Rev. C, and 401897-Rev. C.

I-holes near the outside diameter of the reflector may be filled with beryllium filler pieces or used for irradiating various materials. The I-holes are 1.5, 3.25 and 5.0 inches in diameter.

The beryllium reflector assembly has an outside diameter of 50.5 inches and is contained in an aluminum core reflector tank. Each reflector block assembly includes two aluminum pads which rest on an internal flange at the bottom of the core reflector tank. These aluminum pads, located on the outer corners of each block, provide the vertical (axial) support for the block. Radial location of each reflector block is achieved via upper and lower alignment cups associated with the outer shim control cylinders.

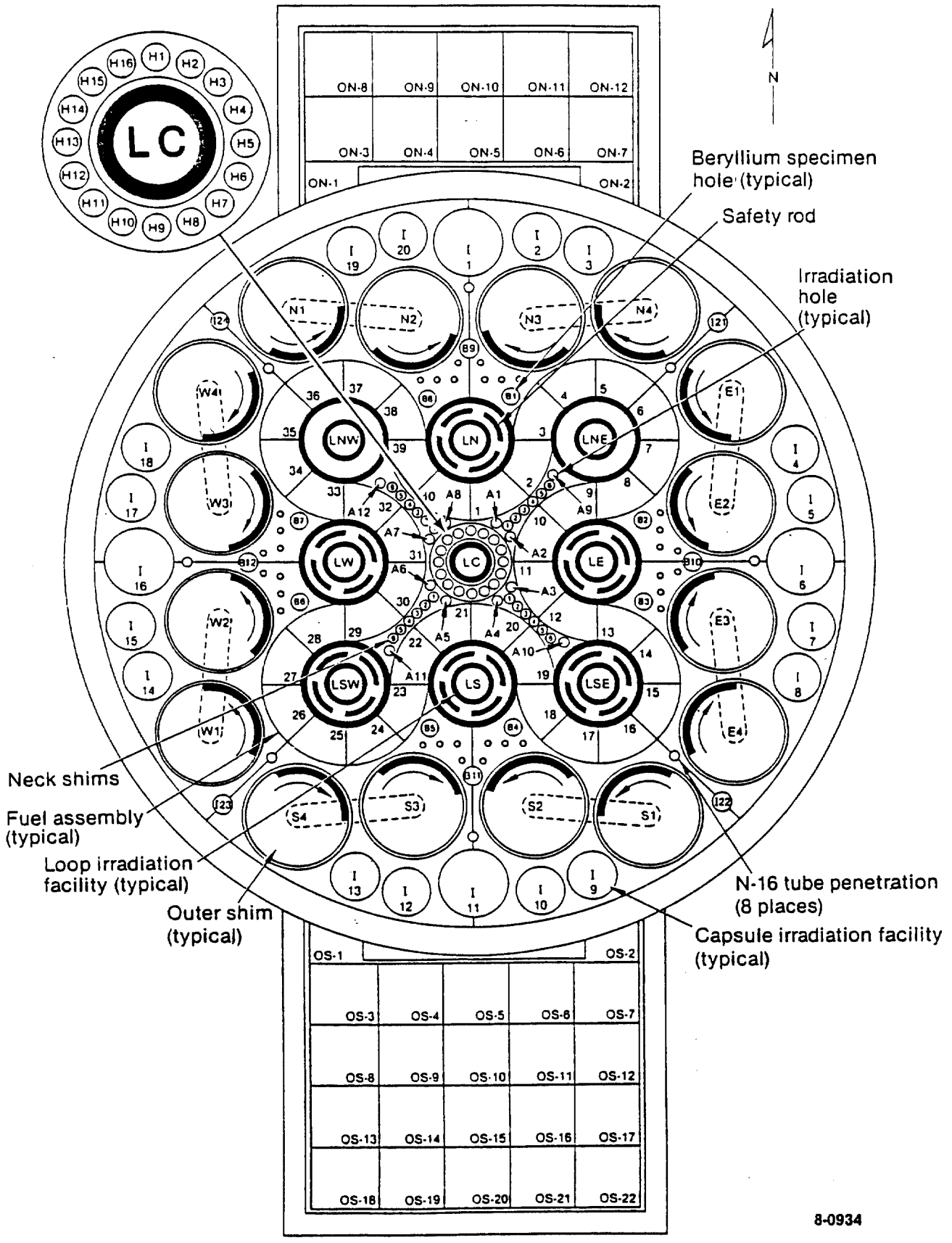

Figure 1. Plan View of ATR Core.

The beryllium utilized in present ATR reflectors is a high grade material manufactured per Specification ANC 80005 entitled, "Material Specification - Beryllium Pressings and Components for Nuclear Reactors and Reactor Systems." This specification addresses requirements for the beryllium powder, average grain size, chemical composition, trace impurity limits, physical properties, mechanical properties, quality assurance provisions, and preparation for shipment.

3. REFLECTOR AGING EFFECTS CAUSED BY RADIATION DAMAGE

ATR reflector aging can potentially affect the reactor in different ways. Aging effects can be generally categorized as follows:

(a) Cracks which can produce beryllium pieces which can move against the outer plate (No. 19) of a fuel element.

(b) Cracks which can produce beryllium pieces which can restrict or prevent OSCC rotation.

(c) OSCC bowing which can cause interference between the OSCC and the reflector block during rotation.

(d) Spalling or cracks which can produce free beryllium pieces which can circulate through the primary coolant system (PCS).

(e) Cracks which produce beryllium pieces which can fall into and remain in the reflector support tank.

(f) Cracks which produce beryllium pieces which can move against other core components.

(g) Cracks which result in small holes in the reflector face adjacent to fuel.

Radiation-induced stresses in the reflector blocks are expected to be greatest in the thin ligaments adjacent to the fuel elements. The radial fast neutron flux gradients are generally highest in these areas; typical flux contour lines are shown in Figure 2. A combination of total fast neutron flux, fast neutron flux gradient, and reflector geometry largely determine the magnitude of radiation-induced stresses. The axial fast neutron flux gradient is generally greatest near the top and bottom of the reflector. A typical inner B-hole axial fast neutron flux distribution is shown in Figure 3.

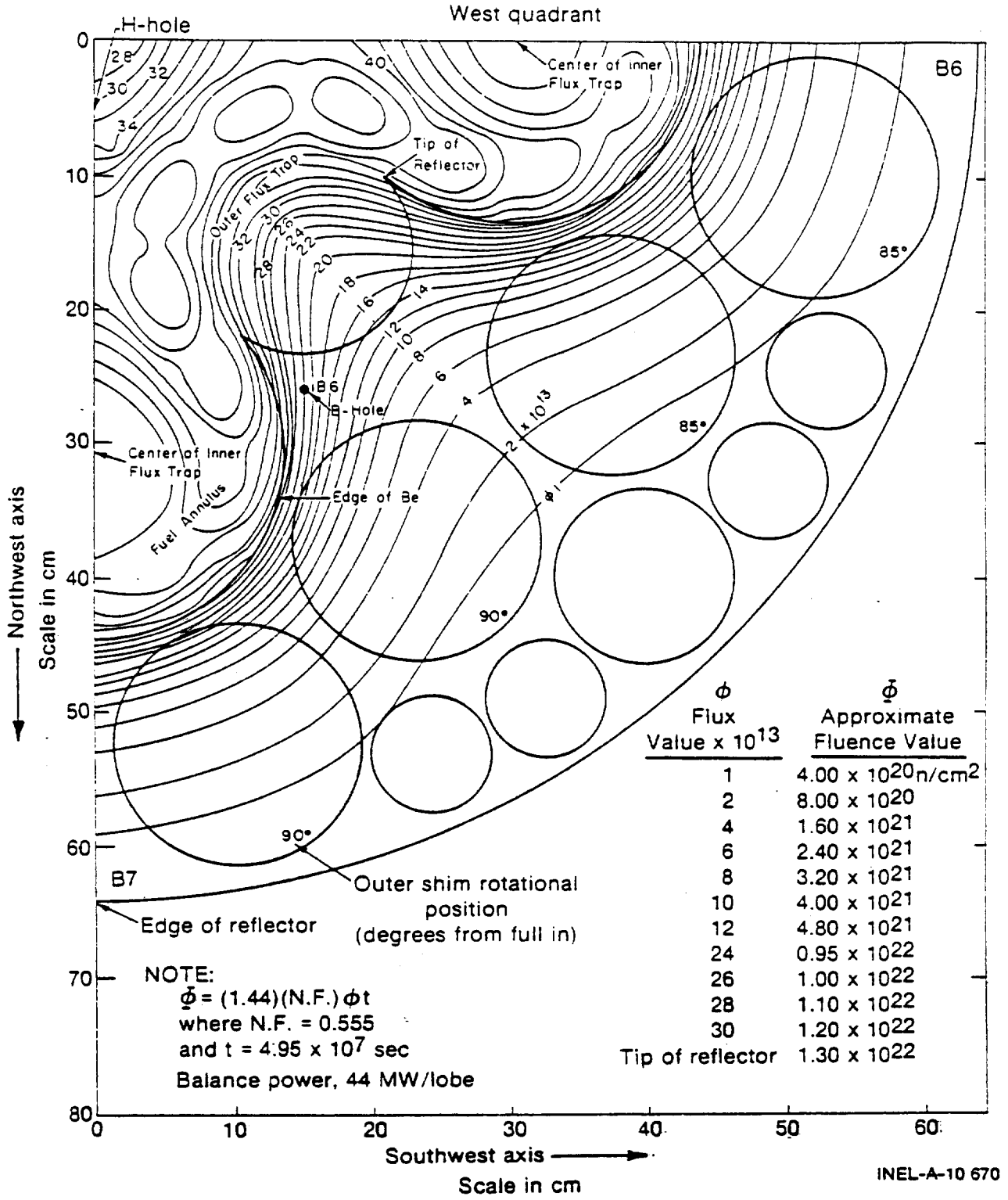

Figure 2. ATR Core Midplane Fast (>1 MeV) Neutron Flux Contours for Cycle 10A.

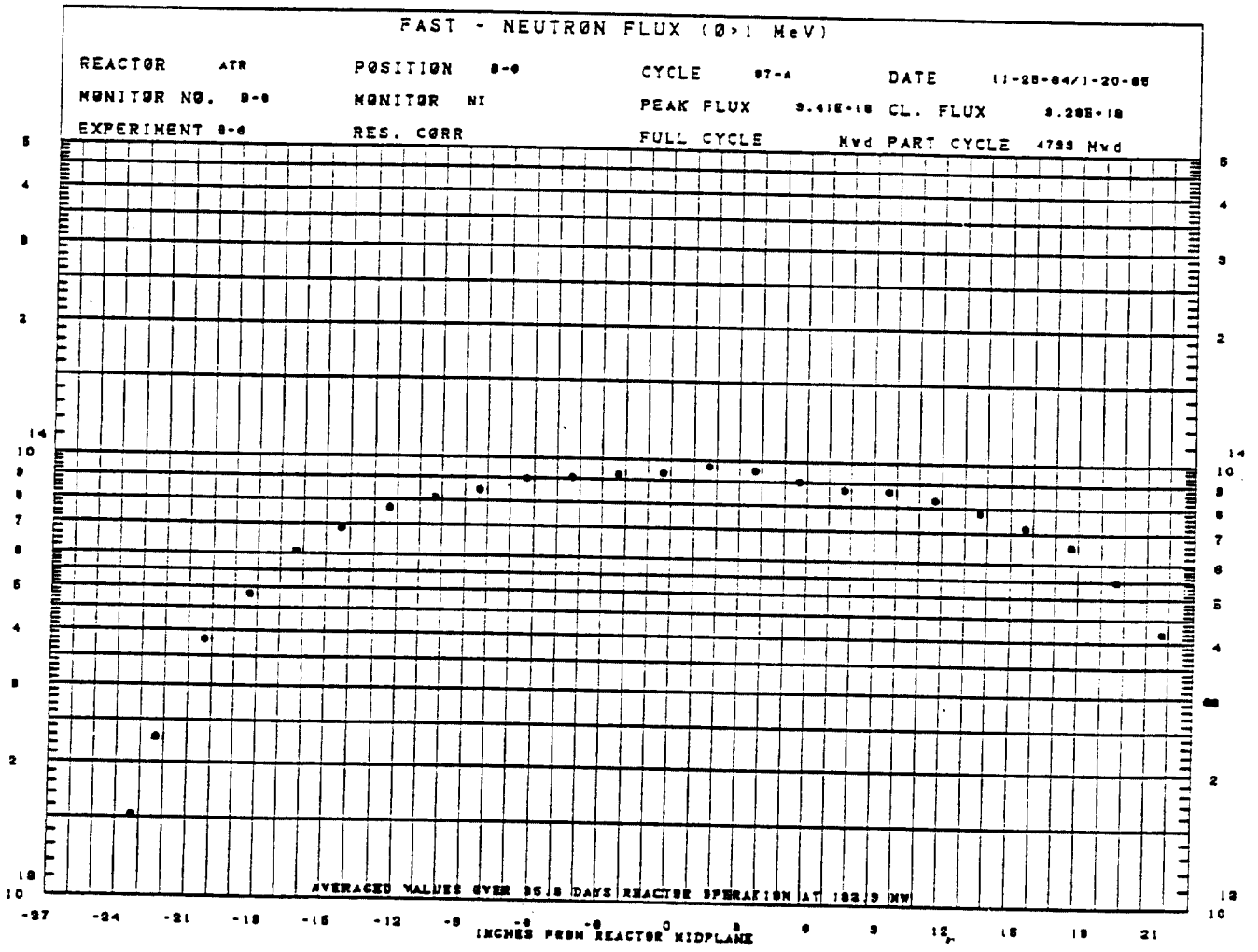

Figure 3. Inner B-Hole Axial Fast Neutron Flux Distribution.

Several thin ligaments in the reflector blocks have been designated for stress analysis and surveillance purposes. Ligament designations are shown in Figure 4. Previous reflector experience has indicated that the highest ligament stresses initially occurred in the A and E ligaments near the top and bottom of the reflector. Cracks have occurred in the A and E ligaments as well as in the ligaments associated with the inner B-hole.

Operating experience with ATR reflectors I and II led to incorporation of horizontal saw cuts in reflector III (and subsequent reflectors) blocks to delay buildup of internal stresses. Axial locations of these saw cuts are shown in Figure 5. Figure 4 indicates the radial location of the saw cuts in relation to the ligaments. The saw cuts limit initial cracking in the A and E ligaments to the saw cut segment in which the crack starts. Eventually, after prolonged reflector use, the A and E ligament cracks can extend over all the saw cut segments from the top to the bottom of a reflector block. When this occurs the operation of the reflector is allowed to continue until cracking of a B ligament is approached. Cracking of a B ligament following complete cracking in the A ligament could result in a large free piece of beryllium (hereafter referred to as the nosepiece). Since the B ligament is not segmented by the horizontal saw cuts, an initial crack in the ligament would be expected to quickly propagate across the full length of the block. Therefore, measures are taken to assure that the B ligament will not experience cracking.

Two ligaments are associated with each inner B-hole. Complete cracking of both of these ligaments could allow a free "tip" piece of beryllium. Note that the inner B-holes are within the saw cut regions.

The nominal gap between an OSCC and a reflector block is 0.130 inch. The OSCCs operate for extended periods of time in a relatively narrow range of rotational positions. Therefore, radiation-induced bowing, in association with adverse tolerance conditions, could cause interference between an OSCC and its surrounding reflector block.

A LIGAMENT THK. = 0.249
B LIGAMENT THK. = 0.220
C LIGAMENT THK. = 0.332
D LIGAMENT THK. = 0.211
E LIGAMENT THK. = 0.249
F LIGAMENT THK. = 0.411

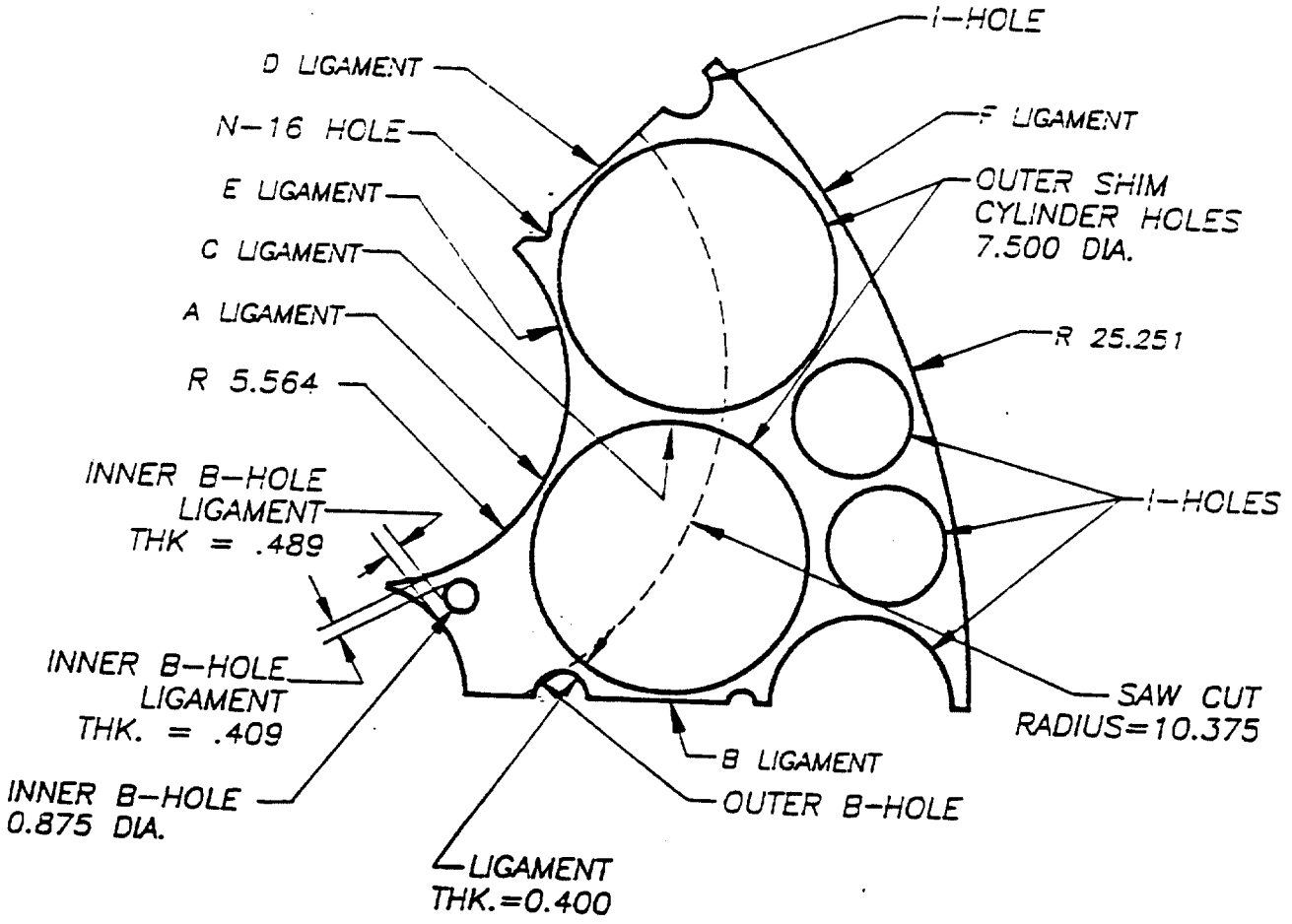

Figure 4. ATR Beryllium Reflector Block Ligaments (Ref. Dwg. 419609, Rev. N/C) (all dimensions are in inches).

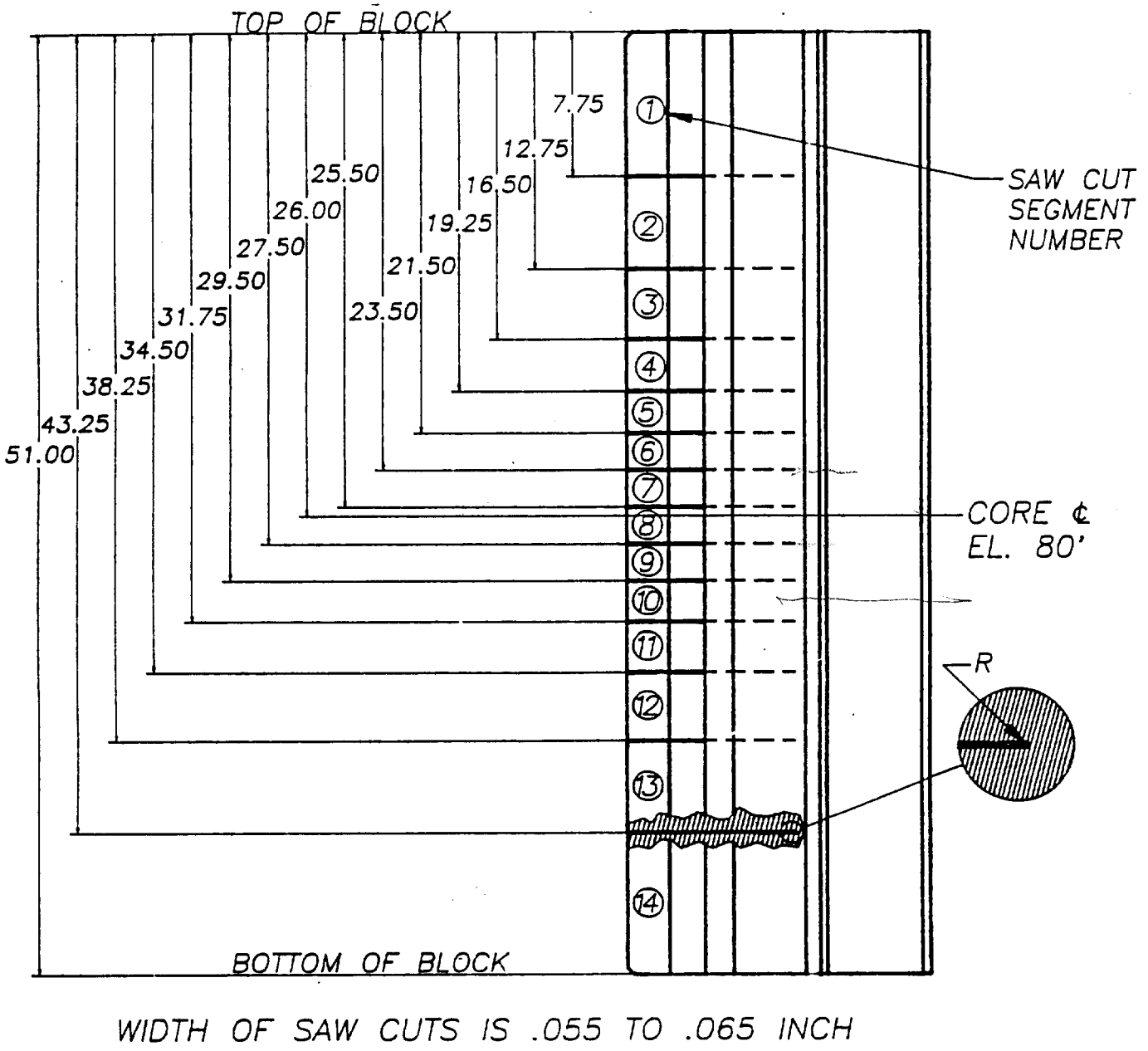

Figure 5. ATR Beryllium Reflector Block Saw Cuts (Ref. Dwg. 419609, Rev. N/C) (all dimensions are in inches).

The initial evidence of reflector aging is expected to be progressive cracking leading to a full length crack of the A ligament. However, the potential for other aging effects exists. In order to envelop the consequences of such occurrences, several other specific effects have been evaluated. These effects include, but are not limited to, a 1-in. by 2-in. hole in the reflector face adjacent to fuel elements, a 1-in. by 2-in. piece of beryllium against fuel plate 19, a 3.8-in. by 2.25-in. piece of beryllium against fuel plate 19 (complete flow blockage), and complete cracking of the A and B ligaments resulting in a free "nosepiece" of beryllium.

4. REFLECTOR EXPERIENCE

4.1 Reflector I

The operating period for the first ATR reflector, Reflector I, was from February of 1968 (Cycle I) to October of 1972 (Cycle 15A). Cracks were discovered in three of the eight reflector blocks in March of 1972 after an accumulated reactor operation of 113,593 megawatt days (MWd) with a balanced lobe power distribution.[3] All cracks were in the A ligaments of the blocks and were determined to have initiated at the top of the beryllium.[3] An extensive analytical program following discovery of the cracks verified that they were the result of excessive tensile bending stresses due to irradiation swelling and material embrittlement.[3] It should be noted that Reflector I did not include horizontal saw cuts which were incorporated in the reflector design beginning with Reflector III. A thorough technical evaluation following discovery of the Reflector I cracks proved that the reactor could safely continue operating (maximum lobe power of 44 MW) as long as certain restrictions were met.[3] Reactor operation was continued, without incident, for another 27,647 MWd; the reactor was then shut down to replace the reflector.[3] Reflector I was therefore operated for a total of 141,240 MWd. No malfunctions of the moving reactor internal components were experienced, no loose beryllium pieces were reported, and radiation levels in the process systems remained at their normal levels.[3]

4.2 Reflector II

The operating period for the second ATR reflector, Reflector II, was from October of 1972 (Cycle 15A) to April of 1977 (Cycle 35A-1). Reflector II was similar in composition and geometry to Reflector I but was subjected to reactor power tilts.[3] The greatest power tilt was produced during Cycle 16E when the southwest lobe was operated at 55.0 MW and the northeast lobe was operated at 26.2 MW.[3] Outer shim control cylinders (OSCC) from Reflector I were used as part of Reflector II.[4] This was possible because of the longer life predicted for the OSCCs. The OSCC life was analytically predicted to be 340,000 MWd.[4] Reflector II

was operated for 153,734 MWd. Some control over cracking in the A ligaments was obtained by scribing certain A ligaments to induce cracking.

Reflector III

The operating period for the third ATR reflector, Reflector III, was from April of 1977 (Cycle 35A-1) to February of 1986 (Cycle 73A-1). The major difference between Reflector III and previous reflectors was the incorporation of horizontal saw cuts in Reflector III blocks to delay buildup of radiation-induced stresses. Saw cut locations are shown in Figure 5. The saw cuts required other design changes in the upper and lower alignment cups for the OSCCs and in the design details of the OSCCs. Inspection of Reflector III during Shutdown 64A-1 revealed the onset of cracking in saw cut segments 12, 13, and 14 of ligament A in beryllium block 4.[5] Saw cut and ligament identifications are shown in Figures 4 and 5 . Shutdown 64A-1 (start date was 3-4-84) occurred after Reflector III had operated for 228,438 MWd. The saw cuts proved to be effective in that Reflector III was operated for 305,246 MWd, a significant extension beyond the lifetimes of Reflectors I and II.

5. DIRECT CONSEQUENCES OF BERYLLIUM AGING

5.1 Basis for the Consequence Study

Several operational variables for the ATR can affect the direct consequences of beryllium aging. Consequences can be different depending on whether the reactor is in a power operational mode (neutron power greater than 3 MW and vessel inlet pressure greater than 350 psig) or in a low power testing mode (neutron power less than 2.5 MW and vessel open to the atmosphere). Consequences can vary depending on the type or types of fuel elements involved (7F, 7NB, or YA). Consequences can vary depending on the primary coolant flow rate - 3 primary coolant pump (PCP) operation or 2 PCP operation during the power operational mode. These variables have been considered in the following consequence study. Any beryllium aging changes during reactor shutdown periods or during reactor startup are not expected to cause any consequences outside those which could occur during power operation or low power testing. The ATR is infrequently operated in the low power testing mode. In fact, low power testing is usually only performed with a new reflector. Therefore, consequences of beryllium aging for the low power testing mode are not specifically addressed in this report.

The following consequence study uses some results from previous analyses supporting reactor operation during use of Reflectors I, II, and III. These analyses are primarily documented in References 3, 6, and 7.

5.2 Cracked Beryllium Contacting the Outer Plate of a Fuel Element

The relationship between the 40 ATR fuel elements and the 8 reflector blocks is indicated in Figure 1. Each fuel element consists of a top and bottom end box and 19 parallel curved fuel plates attached to two side plates to form a 45-degree sector of a right circular cylinder. The side plates are vented by localized slots to provide interconnection between all channels and thereby reduce lateral pressure differentials. The outside fuel plate (plate 19) of each element is 0.100 inch thick with 0.020 inch of fuel and 0.040 inch of cladding on each side of the fuel. A

fuel element is shown in Figure 6. The cooling channel between plate 19 and the reflector is designated as channel 20 (note that fuel element side plates extend 0.073 in. beyond the outer surface of plate 19). The fuel element end boxes are designed for support and positioning on the flux trap baffles.

Several potential reflector cracking modes could result in beryllium pieces which could move across cooling channel 20 and come in contact with plate 19 of a fuel element. Cracking across the full length of the A and B ligaments would allow the largest possible piece (nosepiece) of beryllium to move against a fuel plate (Note that the horizontal saw cuts do not intersect with the B ligament). Cracking across both ligaments of an inner B-hole could allow a "tip" piece of beryllium to move against a fuel plate. Cracking across an E and D ligament in a given saw cut segment could allow a beryllium piece to move toward a fuel element (N-16 tube would tend to limit movement to approximately 0.092 inch, possibly preventing contact with plate 19). Cracking across an A, C, and E ligament in a given saw cut segment could also allow beryllium movement against a fuel element. It should be noted that cracking across one ligament tends to relieve stresses in the adjacent ligament and therefore helps to prevent cracking in the adjacent ligament.

5.2.1 Consequences of Mechanical Loading on a Fuel Element

The maximum mechanical loading of plate 19 due to broken beryllium would be expected to occur during operation with three primary coolant pumps (PCP). All three types of fuel elements can be treated the same in terms of their structural ability to sustain the resulting mechanical loads since they all have the same external geometry and are fabricated in the same fashion. The only major structural difference in Type YA elements is that plate 19 is solid aluminum and contains no fuel; plate 19 of a YA element would operate at lower temperatures than plate 19 of 7F or 7NB elements and therefore would be able to sustain mechanical loads at least as great as those which could be sustained by 7F or 7NB elements.

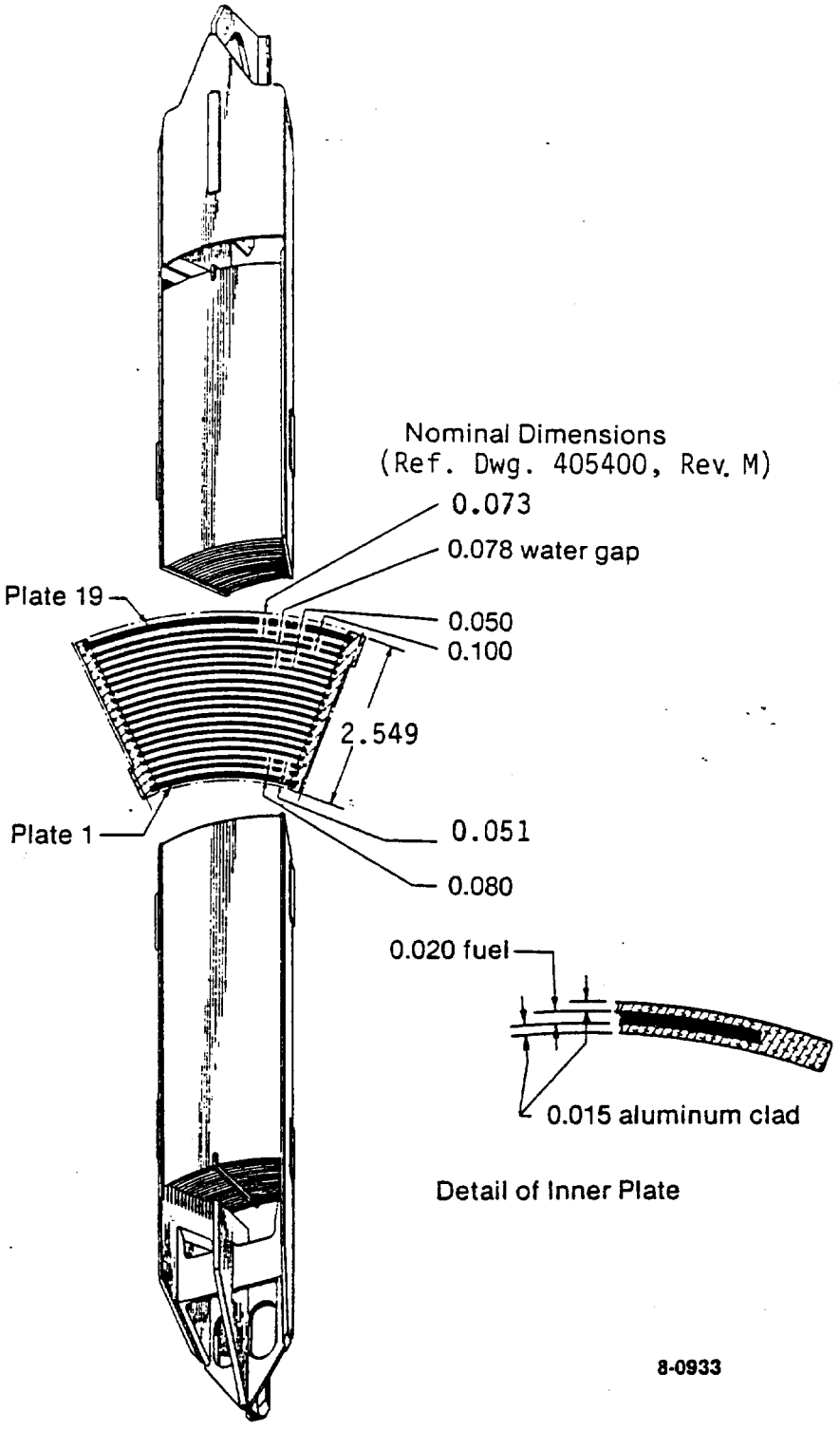

Figure 6. ATR Mark VII Fuel Element.

Present ATR reflectors are designed to minimize the hydraulic gradient between the OSCC annuli and the fuel annulus to limit the cross flow through the reflector block saw cuts. Figure 7 indicates the differential pressure between the OSCC annulus and the fuel annulus for 3 PCP operation. The maximum hydraulic gradient (approx. 20 psid) tending to force beryllium toward the fuel occurs near the bottom of the reflector. The maximum load on fuel plate 19 would require a beryllium piece with an arc length slightly less than that of plate 19; any larger piece would result in at least one side of the beryllium piece contacting the fuel side plate. The side plate would then bear part of the load.

Beryllium cracking and displacement toward a fuel element could mechanically load plate 19 by point, line, or area contact. Numerous measurements of the gap between plate 19 and adjacent beryllium were made for the original ATR core. The maximum measured gap was approximately 0.200 inch.[8] The A and E ligaments of beryllium separating the OSCC annulus and the fuel annulus are 0.249 in. thick. The aspect ratio of a maximum gap of 0.200 inch and a beryllium ligament thickness of 0.249 inch clearly prevents any significant piece of beryllium from breaking loose and rotating sufficiently to present a point loading condition on plate 19. Deflection tests in which plate 19 was loaded with a 30 degree conical point with a 0.020-in. flat end indicated that 330 lb. would force plate 19 into plate 18.[7] Failure of fuel plate 19 due to mechanical loading is considered to occur if the load is sufficient to deflect plate 19 into plate 18.

The differential pressure distribution in Figure 7 indicates that the only significant differential pressure tending to force beryllium toward the fuel occurs near the bottom of the reflector. One of the saw cuts exists at a distance of 16.5 inches above the bottom of the beryllium. A break in the beryllium at this elevation and rotation of the broken piece about the break point has been assumed to estimate the width of the broken piece which would be necessary to produce a point load of 330 lbs. Using the differential pressure profile of Figure 7, the calculated width would need to be 4.0 inches. The arc length of fuel plate 19 (between the two side plates of an element) is only 3.8 inches. Therefore, it can be

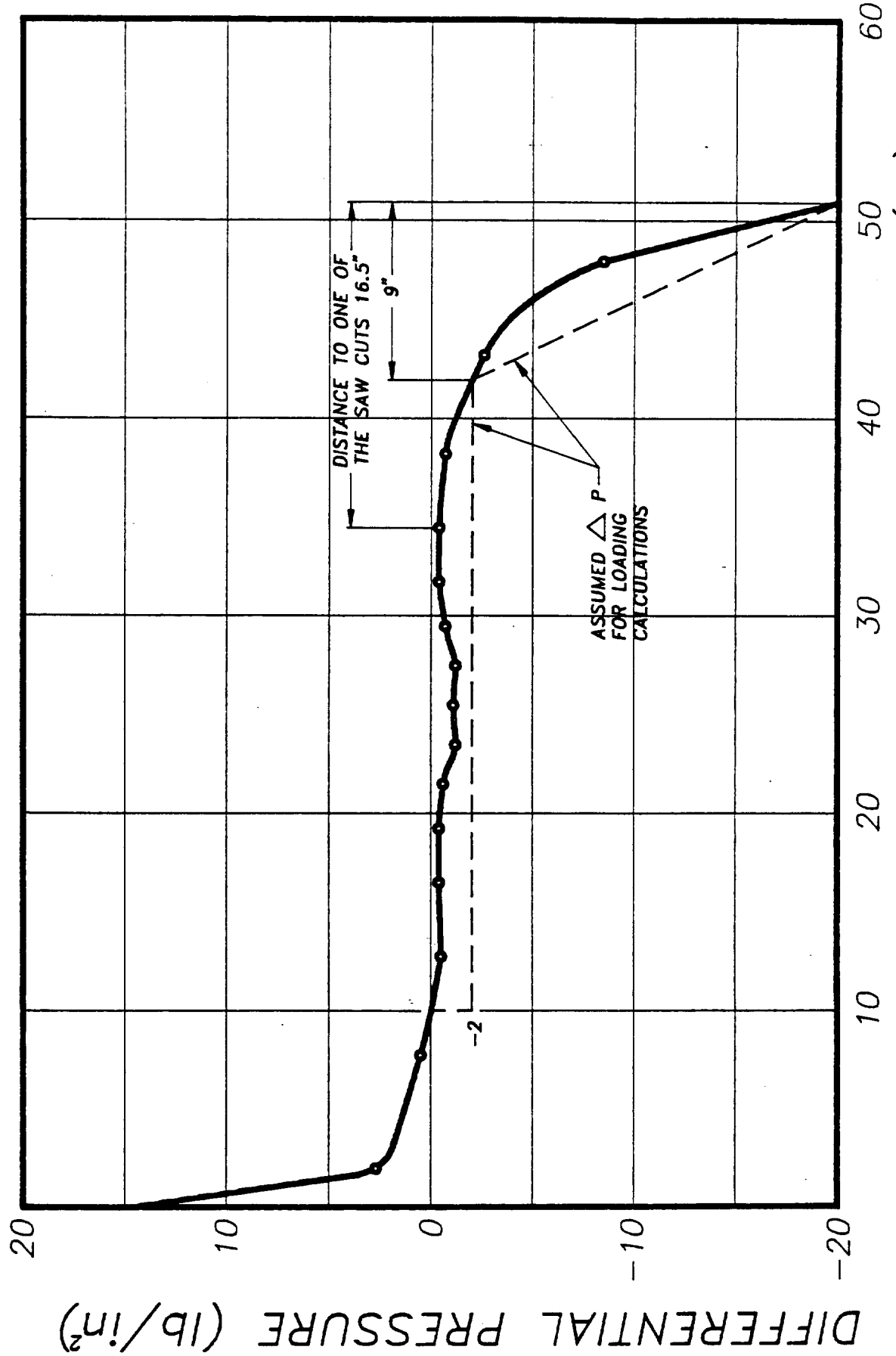

Figure 7. Pressure Difference Between an OSCC Annulus and Channel 20 as a function of elevation. Negative values indicate a force toward the fuel.

concluded that a broken piece of beryllium of sufficient size to produce a 330 lb. point loading would be larger than that which would fit between fuel element side plates. A large enough piece of beryllium to produce a 330 lb. load would always bear partially on a side plate and thus it would be incredible to produce a 330 lb. point load on plate 19. Therefore, plate failure from point loading is considered incredible.

Deflection tests in which plate 19 was loaded with a blunt 90 degree edge, aligned with the plate length, indicated that a minimum load of 200 lb./in. (at plate midspan) is required to force plate 19 into contact with plate 18.[7] The maximum conceivable line load on plate 19 can be determined by assuming cracks in the A, C, and E ligaments in the bottom saw cut segment of a reflector block. An axial line load on plate 19 could then occur by rotation of the broken piece of beryllium about the crack at the A ligament such that the edge of the E ligament could contact plate 19 at approximately midspan. The distance between the A and E ligaments is 4.56 inches. If an initial gap of 0.200 in. (See above discussion of results of gap measurements) is assumed between the beryllium and plate 19, the broken piece of beryllium could rotate about ligament A and result in contact of the edge of the E ligament with plate 19. This contact could occur before the beryllium would come to rest on the fuel element side plates between the A and E ligaments (if no restraint of movement at the C ligament crack is assumed). Assuming the maximum driving force (from Figure 7) of 20 psid over the span of 4.56 inches results in a maximum line load on plate 19 of 46 lb./in. Since this is significantly less than the 200 lb./in. limit it is concluded that no fuel failure from postulated axial line loads can occur.

A maximum azimuthal line load can be postulated if a section of the beryllium block would break at an elevation approximately 10 inches below the top of the reflector, i.e. at an elevation corresponding to that at which the differential pressure begins to be in the direction which would force the beryllium toward the fuel (as shown in Figure 7). In this case the broken piece of beryllium is assumed to rotate about the break elevation such that the bottom of the piece would swing into plate 19. Based on the pressure profile of Figure 7 the driving force is assumed

(conservatively) to be 2 psid from the top of the broken piece to the bottom, plus a varying load from 2 psid (at 32 inches from the top) up to 20 psid (at 41 inches from the top). The resulting azimuthal line load is 116 lb./in. Previously reported calculations have indicated that even an azimuthal line load of 152 lb./in. would not crush plate 19.[9] It is therefore concluded that no fuel failure from postulated azimuthal line loads can occur.

Since the above maximum point and line loads associated with broken beryllium do not result in fuel failures, it can also be concluded that area contact loads will not produce structural failures. This conclusion is possible since the maximum point and line loads were based upon the differential pressure across large beryllium areas; the loads resulting from area contact would be less severe, structurally, than the point and line loads. Heat transfer considerations for loss of cooling due to area contact are addressed in another section of this report.

The above mechanical loading analyses do not consider the increased hydraulic communication between the OSCC annulus and channel 20 as a result of cracks and movement of beryllium toward the fuel. This communication would reduce the hydraulic load on the beryllium and reduce the consequences on the fuel.

5.2.2 Hydraulic Loading From Obstructed Channel 20

Complete obstruction of channel 20 would result in the pressure differential across plate 19 as shown in Figure 8.[7] This differential pressure would result in a maximum load directed toward the center of curvature of 8 psi. The arc length of plate 19 is 3.8 inches. Application of 8 psi over an arc length of 3.8 inches would result in a total axial load of 30 lb./in. Since this is less than the 200 lb./in. axial line loading which is required to deflect plate 19 into plate 18, it is concluded that channel 19 could not be blocked by a complete obstruction of channel 20.

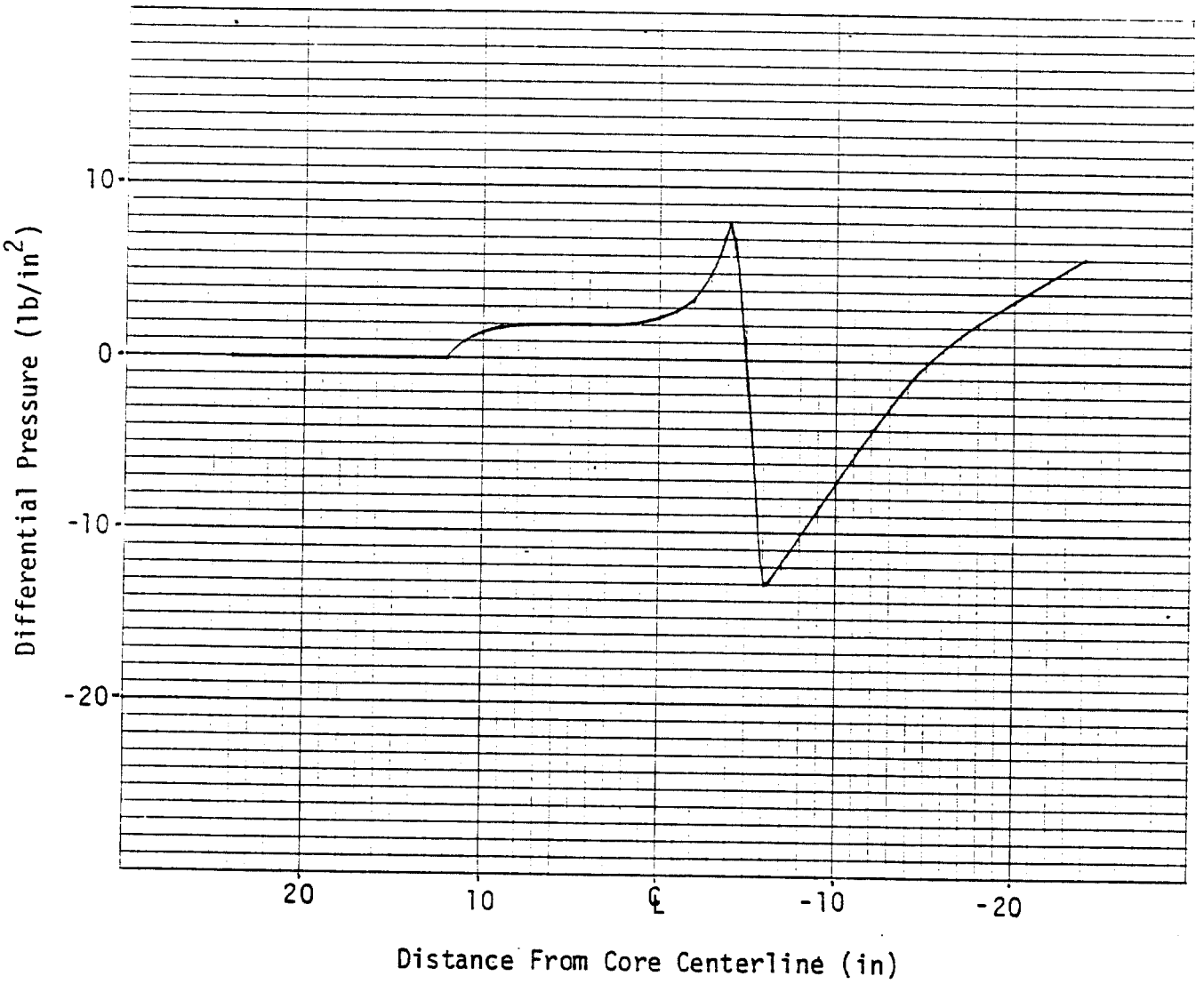

Figure 8. Pressure Difference Between Channel 19 and Channel 20 with Channel 20 blocked. The positive pressure results in a force toward the fuel element.

Figure 8 indicates that the maximum pressure directed away from the center of curvature of the fuel is 13 psi. This type of loading allows analysis of plate 19 as a thin membrane, in tension, supported by the fuel element side plates. The inside radius of curvature of plate 19 is 5.340 in. The corresponding tension load on the swage joint at each side plate would therefore be 69 lb./in. Since the fuel element fabrication specification[10] requires the swage joint to be capable of withstanding 150 lb./in. of swage joint, no element failure from hydraulic loading due to obstruction of channel 20 is expected.

5.2.3 Consequences of Inadequate Fuel Element Cooling

Complete blockage of channel 20 by a piece of beryllium is highly improbable but has been evaluated. This event is not expected to occur due, in part, to the fact that no two reflector ligaments exactly coincide with a full span across plate 19. Fuel element side plates exist between every reflector ligament and therefore free beryllium pieces resulting from cracks in two adjacent ligaments would always be prevented from total area contact with plate 19 by the side plates. Side plates, as shown in Figure 6, extend 0.073 in. beyond the outer surface of plate 19. The consequences of a partial blockage of channel 20 by a piece of beryllium, which is feasible, are addressed in Section 6.12.0 of this report.

Complete blockage of channel 20 can obviously compromise the cooling of plate 19 and cause the temperature of the plate to increase significantly. Alternate flow paths provided by vents in the side plates and saw cuts would, however, allow partial cooling. The most severe conditions would naturally result from blockage in the high-power fuel element positions (typically position numbers ending with 3,4,7, or 8). Blockage at the beginning of the cycle would result in the maximum temperature for plate 19. Type 7F and 7NB fuel elements would be of concern in this situation whereas Type YA elements, without any fission heat source in plate 19, would not experience inadequate cooling.

The temperature and resulting stresses for plate 19 were evaluated using the following assumptions: (1) complete blockage of channel 20 just

below the bottom set of vents (top of blockage at 3.5 in. below core centerline, at top of saw cut segment number 10), (2) two PCP operation, (3) 48 MW lobe power, (4) no slippage at the swage joint, and (5) fuel plate oxide thickness of one mil.[11] Both BOL and EOL (OSCC bowing) beryllium configurations were considered and the worst effect occurred for the EOL case.

The maximum calculated plate temperature for complete blockage was 648 degrees F (two sigma temperature was 736 degrees F).[11] This maximum temperature was predicted to occur at an elevation 28.75 in. below the top of the fuel plate. The calculated stress at this elevation was greater than the yield strength over the entire width of the plate, but the stresses at 24.75 and 37.75 inches below the top of the plate were found to be below the yield strength. Therefore, not more than 13 in. of the plate could be above the yield strength. Plates 1 through 18 would be expected to operate in the elastic range since the blockage would not significantly affect their heat loads.

The above results are considered conservative since Engineering Test Reactor (ETR) experience has shown that the swage joints at the side plates allow axial slippage and thereby relieve the thermal stress. ETR experience has also shown that fuel plate rippling is a progressive phenomenon in that stresses above yield strength do not produce large initial deflections. Small amplitude deflections (less than 0.010 in.) relieve the stresses and the fuel plates reach a stable configuration.[3] With prolonged operation, however, the thermal differences increase and the deflection amplitude increases.[3] Deflections due to beryllium blockage of channel 20, however, would tend to be self limiting in that localized deflections or rippling would disrupt uniform contact between plate 19 and the beryllium thus reestablishing partial cooling channels.

Channel 20 blockage could result in deflection of plate 19 (over the 13-in. length previously noted) toward plate 18. The nominal gap between plate 19 and plate 18 is 0.078 in. Information presented in Reference 12 (Section V.C.1.a) indicates that critical-hydraulic-collapse velocities cannot be achieved prior to developing deflections that could result in

contact of plates 18 and 19.[7] Contact between plates 18 and 19 would not necessarily occur, however, because localized deflection of plate 19 toward plate 18 would imply movement away from the adjacent surface of the beryllium and this would tend to reestablish a cooling channel. If contact between the two plates were to occur, the heat load generated by plate 18 could be removed in channel 18 without approaching CHF (a single plate can be adequately cooled from one side).[7] Partial melting of plate 19, if any were to occur, would be limited to the 13-in. length. A 13-in. length of plate 19 is about 1.8% of an element and represents considerably less than 1% of the total core. Consequences of a 1% release are stated in Section 5.2.4.

The minimum blister temperature (two standard deviations) for ATR fuel is 835 degrees F for fission densities of 2.7×10^{21} fiss/cm^3.[13] The maximum calculated plate 19 temperature (plus two standard deviations) for complete blockage of channel 20 was 736 degrees F. Therefore, blistering is not expected to be a potential fission product release mechanism resulting from total blockage of channel 20.

5.2.4 Consequences of Fission Product Release to Primary Coolant System

As previously indicated, mechanical loading of plate 19 by cracked, displaced beryllium is not expected to be great enough to result in any plate failure. Thermal stresses caused by flow blockage could, however, result in some melting of plate 19. The maximum possible amount of melting was concluded to be less than 1% of the total core. This was based on a single reflector failure (independent of any previous failure). Multiple simultaneous failures have an extremely low probability because only a few areas of the reflector are highly stressed by aging effects. Also, the reactor is usually operated with an imbalance in the lobe powers and therefore different parts of the reflector age at different rates.

The consequences of cracked beryllium contacting plate 19 are not expected to result in any fuel damage beyond possible melting of a portion of plate 19; nevertheless, in order to demonstrate the consequences of

damage to several plates (i.e., 2.79-in. span of plates 11 through 19 for the maximum distance between side plate vents), a release to the primary coolant system (PCS) of one percent of the core fission products has been considered. An ORIGEN2-calculated fission product inventory for the ATR core after 60 days of operation at 250 MW indicates that a release of 1% of the inventory would be approximately 2.4×10^6 curies of solids (Cs, Rb, Ba, Te), 1.0×10^6 curies of halogens (I, Br), and 1.0×10^6 curies of noble gases (Xe, Kr).[14] All released fission products would be expected to go into ionic solution with the water.[14] Assuming complete mixing with the 73,800 gal. of primary coolant, the resulting activity level would be 1.6×10^4 microcuries per milliliter. This concentration would exceed the ATR Technical Specification limit of 20 microcuries per milliliter and would therefore require shutdown of the reactor. The resulting increase in direct gamma radiation outside the primary coolant system shielding would be less than 1 mr/hr.[3]

Release of 1% of the core fission products into the PCS could result in significant releases from the ATR stack. The ATR stack effluent is monitored, however, with an alarm set to correspond to not more than 2000 Ci/day (ATR Technical Specification limit). A fission break would result in immediate operator actions to control the discharges from the PCS. Operator actions would include minimizing the degassing flow, increasing the bypass demineralizer flow, and increasing the efficiency of the warm waste treatment system in accordance with the abnormal and emergency operating procedure for fission breaks (O&MM No. 7.3.13.2.3.8). If these actions were to prove insufficient then the Radiation Monitoring and Building Seal Subsystem (RMSS), upon detection of high radiation in the ATR building exhaust, would initiate a confinement isolation by tripping the closure of penetration dampers and shutting off ventilation supply fans and exhaust fans.

The onsite and offsite consequences of a fission break stack release have been evaluated. Efforts would be made, upon experiencing a fission break, to control the immediate stack release rate to less than 400 Ci/day. ATR Technical Specifications require that stack effluents be controlled to be as low as practical and to not exceed an annual average

value of 400 Ci/day. A stack release of 10^5 Ci/day has been calculated to result in a maximum daily dose of 120 mR at a distance of 100 meters downwind from the ATR (assuming a constant wind direction).[3] Established TRA emergency procedures provide for evacuation away from the existing downwind direction. Actual tests have shown that TRA can be evacuated in 15 minutes. The calculated dose would only be 36 mR (assuming a constant degassing rate of 200 gpm) even if an evacuating person remained in the plume until departing from TRA.[3] This dose would be well within the limits established in DOE Order 5480.11 (Radiation Protection for Occupational Workers). The maximum offsite (Arco) dose rate from a 10^5 Ci/day stack release rate would be 0.7 mR/day (assuming a constant wind direction).[3] Assuming a dose rate of 0.7 mR/day, no resident of Arco would be expected to even approach the 10 CFR 100 whole body limit of 25 rem. Therefore, if cracked beryllium resulted in channel 20 blockage such that partial melting of plate 19 did occur, the consequences would be acceptable both onsite and offsite.

The PCS demineralizer system could be used to clean the primary coolant following partial melting of plate 19 and therefore impact on reactor operation would be relatively minor. This incident is not expected to be feasible until the reflector has aged significantly and therefore, if it did occur, it would probably initiate the planned replacement of the reflector. The probability of this incident occurring is limited by the requirement to use type YA fuel elements adjacent to A or E ligaments when certain conditions exist. When A or E ligament stresses are calculated to be less than two standard deviations to failure stress then YA elements are required (per Section 6 of this report) for lobe powers greater than 48 MW.

5.3 Cracked Beryllium Falling Onto the Reflector Support Tank

The worst case consequences of beryllium falling onto the reflector support tank have been determined by postulating that a 51-in. long piece of beryllium, created by total cracking of the A and B ligaments, falls onto the reflector support tank and then leans against the flux trap baffle. The location of the reflector support tank is shown in Figure 9.

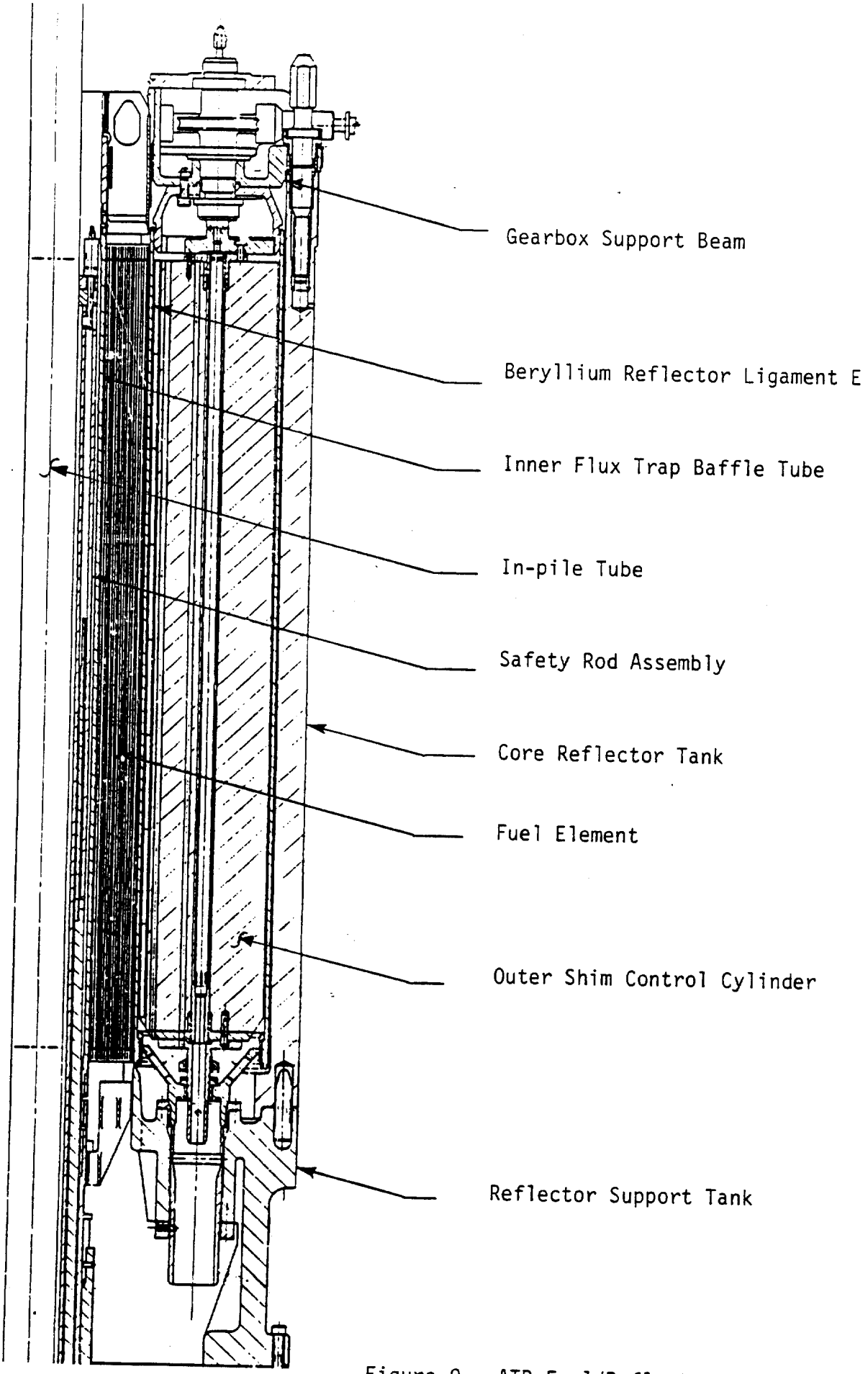

Figure 9. ATR Fuel/Reflector Ligament E/OSCC Cross Section

This piece of beryllium is referred to as the "nosepiece" and is the largest credible piece of beryllium which could potentially break free due to reflector aging. The reflector is normally supported by the core reflector tank. Maximum loading conditions have been applied by assuming 3 PCP operation.

5.3.1 Static Load of Beryllium Nosepiece on Reflector Support Tank

The static load of a nosepiece on the reflector support tank has been evaluated. The cross sectional area of the nosepiece is approximately 12.7 square inches. The design differential pressure across the core for 3 PCP operation is 100 psid. The resulting hydraulic load on the nosepiece would therefore be 1270 lbs. The weight of the 51-in. long nosepiece in water (125 degrees F) would be 20 lbs. Therefore, the total vertical static force on the reflector support tank (aluminum casting) would be 1290 lbs.

The reflector support tank includes an inner ring, as shown in Figure 9, which is adjacent to the fuel elements and the outer flux trap baffles. This inner ring is supported by webs or gussets extending from the inside diameter of the reflector support tank toward the center of the reactor. The vertical load from the nosepiece would be applied to the inner ring and would be transferred to the webs or gussets.

One of the major webs or gussets supporting the inner ring would be located directly below the tip of a beryllium nosepiece. This web is shown in Figure 10. The shear cross section of the web is approximately 1.5 in. by 1 in. (per Dwgs. 120352 and 120358) at a location near the inner ring. Conservatively assuming that this web would carry the entire nosepiece load, the shear stress would be 860 psi (1290 lbs./1.5 sq. in. = 860 psi).

The ASME Boiler and Pressure Vessel Code has been used as a guide to determine an allowable shear stress for the reflector support tank. The very low probability of the nosepiece dropping onto the reflector support tank allows the event to be considered as a Level C condition per the ASME

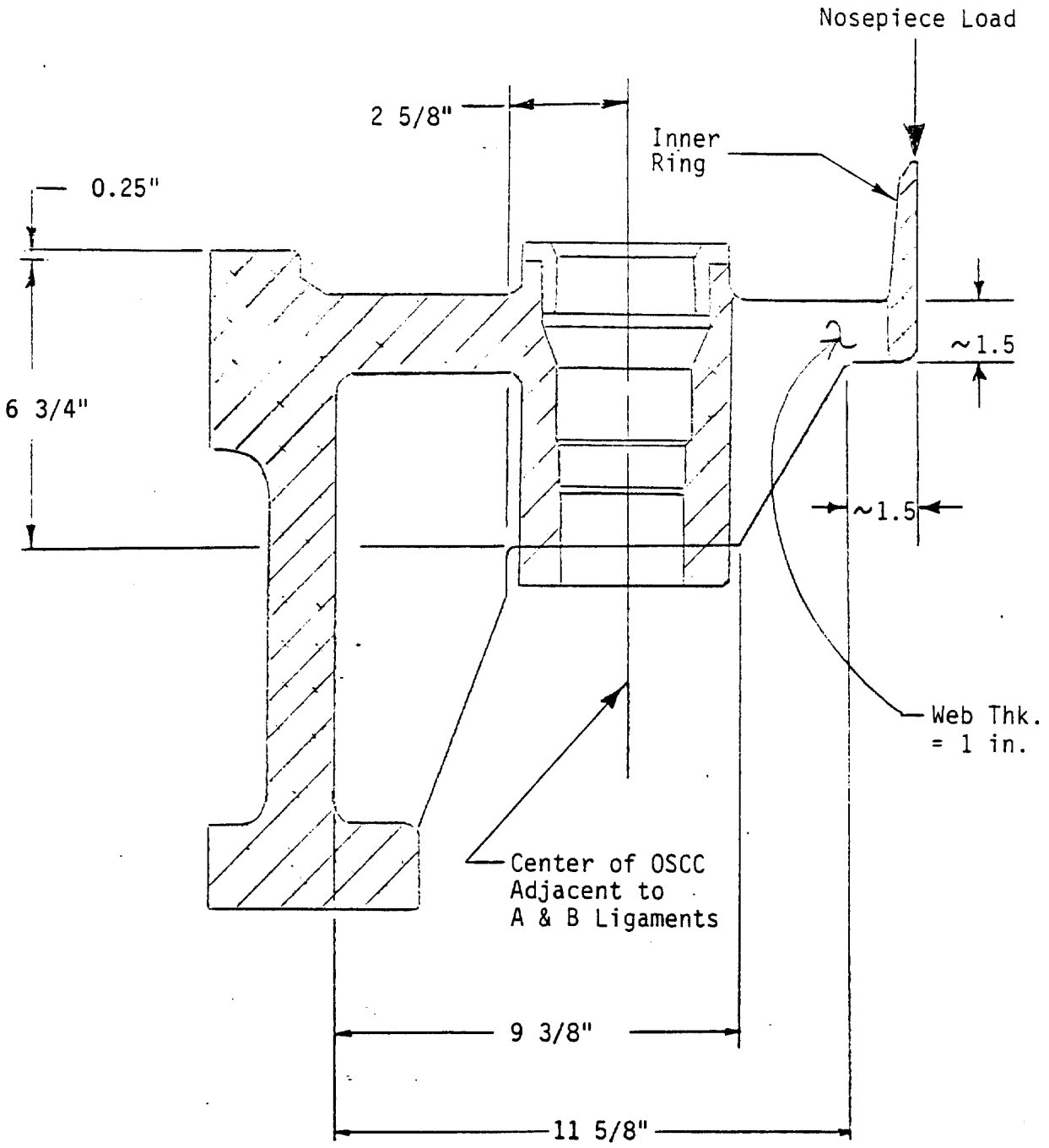

Figure 10. ATR Reflector Support Tank Section.

Boiler and Pressure Vessel Code. The Level C allowable shear stress is the greater of 0.5 S_y or 1.2 x (0.6 S_m). S_m is assumed to be equal to 2/3 S_y. S_y for the reflector support tank aluminum alloy 356-T71, at the ATR vessel design temperature of 240 degrees F, is 15,200 psi.[15] Therefore, the allowable shear stress would be 7,600 psi. Note that aluminum casting alloy 356-T71 is the same material as ASTM Specification B 26-60T, alloy SG 70 A-T71 specified on Dwg. 120352. The calculated shear stress (860 psi) from the static load is very conservative since other webs would share the load. Therefore, since the calculated shear stress is much less than the allowable, it can be concluded that the static load from a fallen reflector nosepiece would not damage the reflector support tank.

5.3.2 Impact Load of Beryllium Nosepiece on Reflector Support Tank

A clearance normally exists between the bottom of a reflector block and the top of the inner ring of the reflector support tank. Therefore, an impact load would occur if a reflector nosepiece were to suddenly break free and fall onto the reflector support tank. The nominal clearance between the nosepiece and the inner ring is 0.035 in. while the maximum tolerance clearance is 0.056 in. (per Dwgs. 120353, 120358, and 419610). The shear stress in the reflector support tank has been conservatively calculated for a 0.06 in. drop assuming the nosepiece is accelerated by a 100 psi pressure difference as well as the force due to gravity (the viscous effect of water was neglected).

As indicated in Section 5.3.1 of this report, the reflector support tank inner ring is supported by webs or gussets, three of which would be located directly below a reflector nosepiece. One web, as described in the static load evaluation of Section 5.3.1, would extend under the tip of the nosepiece. Another web would exist directly under the A ligament of the nosepiece. A third web would be located along the reactor axis directly under the B ligament of the nosepiece.

It has been assumed that the impact of the falling nosepiece would occur along the top of the inner ring from a location directly under the A ligament to the tip adjacent to the outer flux trap baffle. Dwg. 120352

indicates the appropriate details. Impact along the reactor axis (directly under the B ligament) could not be assured due to a large chamfer on the top of the inner ring in that region. The first two webs described above would be located under the defined impact load region. The third web, described above, and another web just outside the defined impact load region would also provide support of the inner ring. Assuming the impact load would be distributed to the two webs directly under the defined impact region and to extensions of the inner ring beyond the impact region, the calculated shear stress was 6,521 psi. This is less than the allowable shear stress given in Section 5.3.1 of this report and therefore, it can be concluded that the impact of a falling nosepiece would not cause failure of the reflector support tank. Also, the probability for a sudden, free-falling nosepiece would be very low because of the binding of the crack and other frictional forces.

5.3.3 Consequences of Reflector Support Tank Deformation

Even though the static and impact load analyses indicated that no failure of the reflector support tank would be expected, it could still be postulated that deformation could occur. The maximum credible deformation situation is as follows.

The top of the inner ring of the reflector support tank (see Figure 10) is at an elevation just below (approx. 0.25 in.) the bottom of the fuel plates. Major lateral deformation of the inner ring could push it into the area below a fuel plate exit thus causing a flow disturbance in the fuel element. Studies of an obstruction below the outlet of an ETR fuel element have shown that obstructions can cause a reduction in the coolant channel velocity. Data in the tested ETR element showed as much as 20% reduction in velocity in the channel corners for an obstruction in the end box 3/4 in. below the channel exit. Since the inner ring on the ATR reflector support tank would act as an air foil angled into the stream, it would produce less velocity reduction than observed in the ETR element. A 20% velocity reduction would produce an increase of almost 20% in fuel plate temperature. The consequences of a 20% velocity reduction, however, would be less than those associated with a total blockage of

channel 20. The consequences of total blockage of channel 20 are addressed in Section 5.2.3 of this report.

The nosepiece impact analysis indicated that shear stresses in the reflector support tank would be less than the allowable values. Therefore, significant deformation would not be expected and no appreciable vertical movement of the reflector would occur.

5.4 Cracked Beryllium Affecting Outer Shim Control Cylinder Gearbox Support Beams

The outer shim control cylinder (OSCC) gearbox support beams provide vertical and partial lateral support for the outer shim control cylinders. In addition, the gearbox support beams provide partial lateral support for the beryllium blocks. Partial lateral support is denoted because the gearbox support beams provide lateral support for the OSCCs and the beryllium blocks only at the top of the reflector. Lateral support at the bottom of the reflector is provided by other means. Cracked beryllium is not expected to make any significant changes to the vertical loads on the OSCCs. The differential pressure between the fuel annulus and an OSCC (as shown in Figure 7) indicates, however, the possibility of cracked beryllium pieces near the top of the reflector being forced against an OSCC. Lateral loads imposed on the OSCCs by broken beryllium near the top of the reflector would be primarily transferred to the gearbox support beam. The lateral load on the gearbox support beam would essentially be the same as for the uncracked reflector since the gearbox support beam normally provides the lateral support for the top of the reflector. Cracked beryllium is not expected to increase the loads on gearbox support beams and could, in fact, reduce the loads if large beryllium pieces such as a nosepiece would fall onto the reflector support tank.

5.5 Cracked Beryllium Affecting Outer Shim Control Cylinders

5.5.1 Effects on Outer Shim Control Cylinder (OSCC) Rotation

Total cracking of the A and B ligaments in a reflector block could allow movement of the resulting free "nosepiece". The potential for a free nosepiece to move into contact with an OSCC has been investigated because of its possible effects on OSCC rotation. The outer shim control cylinders are separated from the surrounding beryllium by a 0.130-in. annulus. This annulus is maintained by alignment cups at the top and bottom of the OSCCs. The alignment cup at the top of each OSCC is attached to a gearbox support beam. The alignment cup at the bottom of each OSCC is connected to the reflector support tank as shown in Figure 9. Any free nosepiece would be adjacent to one of the reflector outer B-holes (B-9 through B-12). Each outer B-hole is currently filled with either an experiment plug (Dwg. 418780) or an experiment basket (Dwgs. 421092 and 421093). Each experiment plug or basket extends from a hole in a gearbox support beam to an elevation within 3 inches from the bottom of the reflector.

An experiment plug or basket in an outer B-hole would tend to restrict lateral movement of a nosepiece in a direction toward the center of the reactor. Significant lateral movement of a nosepiece to a position nearer an OSCC than its original position would not be feasible unless additional pieces of beryllium would break away from the nosepiece at the A or B ligament area. Significant rotation of a nosepiece would be prevented by an Inconel ring which is installed in the bottom of each OSCC hole in each reflector block. Each Inconel ring is match-machined to provide a maximum diametral clearance with each OSCC hole of 0.004 inch (Dwg. 419610). The pinning action of a B-hole experiment plug or basket in combination with the presence of the Inconel ring would prevent significant rotation of the nosepiece and so prevent contact with the OSCC. Therefore, the free nosepiece resulting from total cracking of the A and B ligaments in a reflector block is not expected to affect the rotation of the associated OSCC.

Total cracking of the E and D ligaments across any saw cut segment in a reflector block could allow movement of the resulting free piece of beryllium. The N-16 tubing assembly located between the E and D ligaments would tend to restrict lateral movement of the resulting free piece of beryllium toward the adjacent fuel element but would not prevent rotation of the piece nor its potential movement toward the associated OSCC. Therefore, it can be concluded that contact of the beryllium piece and the OSCC could occur.

Total cracking of the A, E, and C ligaments across any saw cut segment in a reflector block could allow movement of the resulting free piece of beryllium. This free piece of beryllium could conceivably move laterally into contact with one of the adjacent OSCCs.

The worst case consequences of beryllium cracking, relative to OSCC rotation, would be total binding of an OSCC pair. Simultaneous binding of more than one OSCC pair is not considered credible. Even if only half of the OSCCs were operable, the shutdown margin of the reactor with three safety rods, all neck shims, and half the OSCCs inserted has been calculated to be approximately 4.0$.[7] If OSCC binding were to occur during a reactor shutdown period the binding would be quickly detected during startup of the reactor since movement of the OSCCs is required to establish and maintain the prescribed power division. Binding of an OSCC pair during reactor operation would be of no significance until attempts were made to move the pair. ATR Technical Specifications require that the reactor be shut down if a stuck pair of OSCCs cannot be freed within 12 hours. The ATR Technical Specifications also require that the reactor be immediately shut down if more than one OSCC pair becomes stuck. OSCC drives have torque release clutches that limit the amount of torque that can be applied to the cylinders.

The worst case effect of beryllium cracking on OSCC rotation is the potential binding of an OSCC pair. Since adequate shutdown margin would still exist with a bound OSCC pair it can be concluded that the effects of beryllium cracking on OSCC rotation are acceptable.

5.5.2 Effects on Outer Shim Control Cylinder Cooling

Beryllium cracking between the OSCCs and the fuel annulus would allow increased hydraulic communication between these coolant channels. The OSCCs are cooled internally via full-length coolant holes and externally via the 0.130-in. annulus surrounding the entire cylinder. In addition, the curved hafnium plates on the outside diameter of each OSCC are cooled via a 0.110-in. annulus between the hafnium and the beryllium cylinder. The coolant flow through and around each OSCC is limited by restrictive orificing in the upper alignment cup and by roughened walls on the outside diameter of each cylinder. The resulting differential pressure profile is shown in Figure 7. Hydraulic communication between the cylinder annulus and the fuel annulus already exists through the saw cuts. Cracking of the reflector would increase the communication, but no significant change in the coolant flow through the cylinder would result. No cooling problems are anticipated as a result of hydraulic communication through the broken beryllium.

Three major types of beryllium pieces resulting from ligament cracks were described in Section 5.5.1 above. A nosepiece resulting from cracking of the A and B ligaments would encompass the greatest arc length around the adjacent OSCC. Even though it is not considered credible that a nosepiece could contact an OSCC, the consequences of direct contact have been estimated. Contact of a nosepiece with an OSCC could block approximately 115 degrees of the OSCC cooling annulus. The total OSCC cross sectional area for cooling paths is approximately 4.6 in^2. Blocking 115 degrees of the outer cooling annulus would represent a blockage of approximately 21%. This is approximately the same decrease in flow that results in changing from 3 PCP operation to 2 PCP operation. An assessment of the increase in the maximum metal temperature of an OSCC has been made for changing from 3 PCP to 2 PCP.[16] The maximum OSCC metal temperature (at the bottom of the reflector) for 3 PCP is 275 degrees F while that for 2 PCP is estimated to be 295 degrees F.[16] A cracked piece of beryllium against an OSCC during 2 PCP operation would represent the worst case in relation to cooling of the OSCC. The increase in the maximum OSCC metal temperature for a 21% blockage of the flow area should not be any greater than that for changing from 3 PCP operation to 2 PCP

operation. Therefore, it can be concluded that the maximum metal temperature of an OSCC with a nosepiece blocking 115 degrees of the outer annulus would be no greater than 295 + 20 = 315 degrees F. The absolute pressure at the bottom of the reflector during 2 PCP operation would be approximately 300 psia and the corresponding saturation temperature would be 417 degrees F. Since the maximum metal temperature in the OSCC would be significantly less than the saturation temperature it can be concluded that adequate cooling of the OSCC would still exist. Increased temperatures in the cracked piece of beryllium also would not be expected to be excessive due to the increased cooling that would occur on the surfaces facing the wider cooling channels created by the movement of the piece toward the OSCC.

5.6 Cracked Beryllium Moving Against Other Core Components

5.6.1 Static Load of Beryllium Nosepiece on Outer Flux Trap Baffle Tube

Complete cracking of the A and B ligaments of a reflector block would result in a 51-in. long nosepiece of beryllium which could move from its original position. A previous reflector analysis postulated that this nosepiece could fall onto the reflector support tank and lean against the outer flux trap baffle tube with a lateral force caused by the pressure differential between the OSCC annulus and the fuel annulus. This analysis was applicable to reflectors I and II wherein significant lateral forces resulted from the relatively large differential pressures between the OSCC annulus and the fuel annulus. The differential pressures used for the analysis for reflectors I and II varied from 15 psi at the top of the nosepiece to 91 psi at the bottom.[3] The differential pressures for the present reflector design, as shown in Figure 7, are much less than those which applied to reflectors I and II. Therefore, the results of the previous analysis for reflectors I and II envelop operation with the current reflector design.

The results of the previous analysis are repeated in this report and it should be recognized that they are very conservative for the current reflector design. Another element of conservatism of the present design relative to that for reflectors I and II is the potential benefit of the

experiment plugs or baskets which are installed in the outer B-holes of the reflector. The plug or basket in each outer B-hole could act as a pinning device for a nosepiece to restrain its lateral movement toward the outer flux trap baffle tube. The nominal clearance between the intact nosepiece of a reflector block and the outside diameter of an outer flux trap baffle tube is 0.048 in. The nominal difference between the inside diameter of an outer B-hole (per Dwg. 419609) and the outside diameter of an experiment plug (experiment plugs per Dwg. 418780, are normally used rather than experiment baskets) is 0.035 in. These nominal dimensions would indicate that an experiment plug would tend to prevent a free nosepiece from contacting the outer flux trap baffle tube (assuming no rotation of the free nosepiece). Worst case tolerance conditions would, however, allow contact between the free nosepiece and the baffle tube. If contact were to occur it would still appear reasonable to assume that the experiment plug in the outer B-hole would share some of the lateral load with the baffle tube. The following assessment does not take any credit for the current reflector design which includes a significantly reduced differential pressure between the OSCC annulus and the fuel annulus; it also takes no credit for potential pinning effects by the B-hole experiment plugs.

The following results of the analysis for reflectors I and II (conservatively applicable to the current reflector) were obtained from Ref. 3. The model covered the maximum loading condition in that 3 PCP operation was assumed.

The entire length of the baffle tube was modeled for the Piping Flexibility Analysis Program, MEL-40.[17] The support rim on the bottom of the tube was fixed as an anchor with no degrees of freedom. The top part of the tube, where it is latched to the gearbox support beam, was fixed in the horizontal direction but was free to rotate and move vertically. Since there is some slight thermal bowing of the baffle tube away from the reflector, the free nosepiece was assumed to contact the tube at two points approximately 50 in. apart. To obtain the loads to apply to the tube at these points, the nosepiece was considered to be a simply supported beam with a triangular distributed load. The distributed load was calculated from the lateral pressure difference (across the

beryllium piece) which normally varied from 15 psi at the top to 91 psi at the bottom. The reactions at the support points of the nosepiece beam were determined to be 6,000 and 12,000 lbs.

The bending stress in the beryllium was determined to be 35,000 psi without stress concentration from cracks. This value is between the estimated yield strength of 15,000 psi and estimated ultimate strength of 50,000 psi. Due to this magnitude of bending stress it was then postulated that the 51-in. long nosepiece would break into two pieces. These two pieces were superimposed onto the baffle tube and tied fictitiously so that the tube and beryllium acted as a composite beam under the differential pressure loads. A triangular distributed load was applied in steps to the composite beam. The tube and beryllium were tied together with fictitious elements with the following geometry:

Length = 0.1 in.
Cross Sectional Area = 0.1 $in.^2$
Moment of Inertia = 0.0001 $in.^4$.

The cross sectional area of 0.1 $in.^2$ was enough to transmit lateral loads from the beryllium to the tube, and the small moment of inertia was used to prevent moment transfer from the beryllium to the tube. Calculations indicated that the baffle tube would deflect 0.115 in. toward the fuel elements at a point 27 in. below the top of the beryllium.

The analysis also indicated the maximum stress intensity in the aluminum portion of the tube would be 22,700 psi at the reduced thickness (0.156 in.) near the bottom of the tube. The maximum stress intensity for the stainless steel portion (bottom adaptor) would be 39,169 psi, located in a reduced thickness (0.156 in.) section. The room temperature yield strengths for aluminum (Alloy 6061-T6) and the stainless steel (Type 304) are 35,000 psi and 30,000 psi, respectively. Using ASME B&PV Code, Section III criteria, the allowable stress intensities for the aluminum and stainless steel would be 21,000 psi (1.5 S_m) and 30,000 psi (1.5 S_m), respectively, for normal and upset (Level A and B) conditions, but would be 25,200 psi (1.8 S_m) and 45,000 psi (1.5 S_y), respectively, for emergency (Level C) conditions. Therefore, the calculated stresses

would be within allowable values which ASME Section III criteria would give for emergency conditions. Local yielding of the aluminum would not change the deformations given above significantly.[3]

The maximum beryllium stress intensity was determined to be 10,805 psi at a point 21 in. below the top of the reflector. The beryllium yield strength for an irradiation fluence level of 8×10^{21} n/cm^2 is approximately 15,000 psi.

5.6.2 Impact Load of Beryllium Nosepiece on Outer Flux Trap Baffle Tube

Equivalent static loads for impact of a beryllium nosepiece on a flux trap baffle tube would be in the tens of thousands of pounds using the differential pressure distribution for reflectors I and II (see description in Section 5.6.1). Calculations indicated that damage to the flux trap baffle would occur which, in turn, could invalidate the action of the affected outer safety rod.[3] The differential pressure distribution for the current reflector design is much less than that for reflectors I and II and therefore the consequences of a beryllium nosepiece impacting a flux trap baffle tube would be much less severe. The ATR includes six safety rods as shown in Figure 1. Safe reactor shutdown is assured even in the event of a stuck safety rod.[18]

5.6.3 Deflection of Other Components Due to Nosepiece Loading on Flux Trap Baffle Tube

Section 5.6.1 indicated that lateral deflection of an outer flux trap baffle tube (caused by a free beryllium nosepiece) could be significant (0.115 in.). This section addresses the consequences of core component deflections caused by deflection of an outer flux trap baffle tube.

An outer flux trap baffle tube deflection of 0.115 in. toward the neck shim housing would cause the 0.032 in. clearance between the baffle tube and the fuel element to close. Additional baffle tube movement could then move the fuel element 0.083 in. toward the neck shim housing. Clearance between the fuel element side plate and the neck shim housing is nominally 0.063 in. The movement of the fuel element would then deflect

the neck shim housing 0.020 in. along the line of action. The line of action is assumed to be along a line drawn through the centerline of an OSCC (nearest the inner B-hole) and through the centerline of the closest outer baffle tube. This line of action would pass through the wing of the neck shim housing approximately 3.73 in. from reactor centerline.[3] If the neck shim housing were assumed to rotate about the reactor centerline as a result of the 0.020-in. deflection at radius 3.73 in., then the neck shim rod drive upper extension housing (which is rigidly attached to the bottom of the neck shim housing at radius 6.01 in.) would be deflected 0.032 in.[3] The extension housing, however, has 0.080 in. or more clearance on all sides for some 60 in. below the neck shim housing.[3] Therefore, no severe bending or contact with other parts would be expected.

If the neck shim housing were assumed to rotate about the reactor centerline as a result of the 0.020-in. deflection at radius 3.73 in., then the inner flux trap baffle tubes, at a radius of 12.0 in., would deflect approximately 0.065 in. Nominal annulus clearance between a safety rod and associated inner flux trap baffle tube at the bottom of the baffle tube is 0.063 in. or more, depending upon rod position, and is 0.125 in. with the rod withdrawn.[3] It is highly unlikely that such deflections could occur, but if they did, it would only result in the safety rod in an inner position just touching the baffle tube; this would be insufficient to bind the safety rod.

The neck shim housing is keyed to the center flux trap baffle tube by two tightly fitting keys at the top of the neck shim housing. The center baffle tube penetrates the reactor bottom head approximately 18 feet below the two keys and is not keyed to the bottom head. In the event of rotation of the neck shim housing, the top end of the center baffle tube would be rotated which, in turn, would either rotate the center baffle tube in the bottom head or would simply load the center baffle tube in torsion. The latter is probable, but the deflections are so small as to be harmless to the center baffle tube. Therefore, the total calculated deflection of the outer flux trap baffle, if applied to other reactor components with no credit for restraint by the other components, would not bind either neck shim rods or inner safety rods. Also, the calculated

deflections were based on lateral differential pressures for reflectors I and II and would be conservatively high for the current reflector design (due to the significantly lower lateral differential pressures for the current reflector).

5.6.4 Consequences of Outer Flux Trap Baffle Failure

If an outer flux trap baffle were to fail due to a sudden load from a beryllium nosepiece, it would absorb a large part of the dynamic load from deformation and would transmit the load to the neck shim housing, the in-pile tubes, and inner flux trap baffles which are supported above the core from the housing. The neck shim housing is simply supported laterally at the bottom. The outer flux trap baffle movement of 0.032 + 0.063 = 0.095 in. before transmitting loads to the neck shim housing is the amount of movement required to support the neck shim housing on the opposite outer flux trap baffle. The result would be a small movement together of in-pile tubes and outer flux trap baffles at the bottom of the core and about 0.095-in. movement at the top.[3] This type of movement would preserve the safety rod annulus but would cant it. The articulation of the safety rod would preclude binding in this case.

The outer flux trap baffle tube and the associated safety rod are sufficient to protect the associated in-pile tube from crushing forces. The potential lateral deflection of an outer flux trap in-pile tube can reach several inches without exceeding ASME B&PV Code, Section III allowable bending stresses.[3] Therefore, damage to in-pile tubes and resulting flux trap voiding cannot occur in the event of flux trap baffle failure due to beryllium cracking.

5.7 Outer Shim Control Cylinder Bowing

Aging of the outer shim control cylinders (OSCCs) is exhibited in the form of bowing of the cylinders instead of ligament cracking as in the case of the reflector blocks. The OSCCs are normally operated between 80 and 110 degrees withdrawn. The fast neutron flux gradient across the diameter of each OSCC is therefore basically repetitive from reactor cycle to cycle. Extended operation of each OSCC therefore results in some

bowing of the cylinder. Cylinder bowing could potentially lead to binding of the cylinder due to interference with the surrounding reflector block. Cylinder bowing could also affect the thermal hydraulics associated with plate 19 of the adjacent fuel element.

The effects of OSCC bowing have been analyzed and reported in Ref. 19 for 3 PCP operation. In the end-of-life (EOL) case, the OSCC annulus width in the axial and circumferential directions was varied to account for the bowing. The amount of bowing (0.065 in.) used in the analysis was that calculated to occur at 340,000 MWd.[19] The nominal OSCC annulus is 0.130 in. and therefore it can be concluded that OSCC bowing should not cause binding of the cylinder. If, however, OSCC binding were to occur the consequences would be acceptable as indicated in Section 5.5.1 of this report.

The thermal hydraulic effects of OSCC bowing were originally analyzed and reported in Ref. 19 and an extended analysis was reported at a later date in Ref. 20. To represent the eccentricity or circumferential channel width variation, the OSCC annulus was divided into five segments. The maximum bowing of 0.065 in. was assumed to occur at the core midplane elevation. Flow through the saw cuts between the fuel annulus and OSCC annulus was modeled to communicate with two of the five eccentric annular segments. The minimum coolant velocity in Channel 20 (adjacent to fuel plate 19) was determined for the beginning-of-life (BOL) and end-of-life (EOL) conditions. No OSCC bowing was assumed for the BOL condition, while a maximum bowing of 0.065 in. was assumed for the EOL condition. The original analysis indicated that the minimum Channel 20 velocity (for minimum Channel 20 width) was reduced by approximately 4.6% from the BOL to the EOL case. The extended analysis reported in Ref. 20 was performed to incorporate the results of measurements obtained from hydraulic facility testing. The extended analysis indicated that the BOL minimum Channel 20 velocity would be 29.5 ft./sec.[20] The extended analysis did not include a specific bowing evaluation because of its complexity. The Channel 20 configuration affected by the EOL bowing phenomena would be unchanged in the extended analysis from that in the original analysis. Therefore, it can be concluded that the 4.6% reduction in minimum Channel 20 velocity from BOL to EOL would still be applicable to the extended

analysis. Applying a 4.6% reduction to the BOL velocity of 29.5 ft./sec. results in a minimum EOL (bowed) Channel 20 velocity of 28.1 ft./sec. Ref. 21 indicates that this is an acceptable result for an anticipated event. An analysis, reported in Ref. 21, showed that a Channel 20 velocity of 26.5 ft./sec. would still be 3.2 standard deviations to burnout for a 70 MW lobe power.[21] Therefore, it can be concluded that OSCC bowing would not result in unacceptable consequences for 3 PCP operation.

Acceptable consequences for OSCC bowing for 2 PCP operation can be inferred from the above information and the results of the 2 PCP safety analysis (Ref. 16). One of the events analyzed for 2 PCP operation was a flow coastdown (60 MW lobe power) with failure to scram. The nominal primary coolant flow rate of 39,000 gpm decreased by 4.6% to 37,200 gpm in less than 0.3 sec.[16] During the same time period (0.3 sec.) the nominal reactor power only decreased approximately 2% and the margins to DNB and flow instability decreased to approximately 6 standard deviations and 9 standard deviations, respectively.[16] Since only 3 standard deviations above DNB and flow instability are required for an anticipated event, it can be concluded that a conservative margin will be assured for the EOL (bowed) case for 2 PCP operation even after accounting for the predicted 4.6% decrease in the Channel 20 velocity.

5.8 Effects of Circulating Beryllium Pieces on Primary Coolant System Components

5.8.1 Introduction

Cracks in the reflector could lead to pieces of beryllium being washed out of the reflector and into the primary coolant system (PCS). The possibility of damage to reactor or PCS components by these free pieces of beryllium has been assessed. Components for which the assessment was made include the heat exchangers, primary coolant pumps, primary coolant pump check valves, safety rods, neck shim rods, outer shim control cylinders, and fuel elements. A discussion of each of these components is presented in succeeding paragraphs. In addition, a

concluding section summarizes beryllium reflector experience in the Materials Test Reactor (MTR), Engineering Test Reactor (ETR), and Advanced Test Reactor (ATR).

Free pieces of beryllium entering the PCS can be placed into two categories. One is small chips or particles of beryllium resulting primarily from spalling. The second category is relatively large pieces of beryllium resulting from cracks near the bottom of the reflector which lead to pieces falling from the reflector into the reflector support tank. Also in the second category could be pieces of beryllium which fall away from the reflector and into the reflector support tank when fuel elements are removed during refueling operations.

Chips or particles from spalling could enter the primary system by being washed downward through the annulus that exists between the reflector face and outer surface of plate 19 of a fuel element or through the annulus that exists between a reflector block and an OSCC. After exiting the reflector region the chips or particles could flow through the flow distribution tank and into the PCS piping. Numerous measurements of the gap between plate 19 and adjacent beryllium were made for the original ATR core. The maximum measured gap was approximately 0.200 inch.[8] The OSCC annulus is smaller than this. The overall chip size would be limited by the gap between plate 19 and the reflector block and the curvature of the annular surface. Its length and width dimension would not be expected to be more than several times the 0.200-in. maximum annulus width.

Relatively large pieces of beryllium could drop into the flow distribution tank through one of the holes in the reflector support tank casting. The largest piece that could enter the flow distribution tank during reactor operation would have a cross-sectional area of about 11.3 $in.^2$, which corresponds to the largest free opening in the reflector support tank. Thus, a relatively large piece of beryllium could be postulated to wash out of the flow distribution tank and to enter the primary coolant system through one of the quadrant outlet nozzles.

5.8.2 Heat Exchangers

After leaving the reactor vessel, primary coolant flows through the tube side of five parallel U-tube heat exchangers. The inside diameter of the heat exchanger tubes is 0.652 in. Consequently, the heat exchangers would serve to retain beryllium pieces larger than 0.652 in. in diameter. The length of pieces (which could conceivably pass through the heat exchangers) approximating a diameter of 0.652 in. would also be limited because of the 180 degree bend in each tube. Each heat exchanger has approximately 20% excess heat exchange capacity and could tolerate a large number (greater than 300) of blocked tubes without affecting reactor inlet temperature.[3] Radiation hot spots could develop, but experience at MTR and ETR has shown this to not be a serious problem.

5.8.3 Primary Coolant Pumps

After leaving the reactor vessel and flowing through the heat exchangers the primary coolant passes through two or three primary coolant pumps (PCP) during power operation. Each primary coolant pump is a double suction volute centrifugal pump and is driven by a 2000 hp induction motor. Both pump casing and impeller are fabricated of type 304 stainless steel. The only potential area of damage to a pump is for a chip to become lodged or jammed in the nominal 3/8-in. clearance that exists between the pump impeller and a 3/8-in. thick casing web on the bottom of the pump volute. Considering that the chip would be small in size, that its compression fracture strength would likely be reduced because of radiation embrittlement and the large amounts of impact energy imparted by the rotating impeller, it can be concluded that the beryllium piece would likely shatter before deforming pump components. Minor pump damage might result if a chip should retain compressive fracture strengths exceeding the yield strength of the pump material. Under this condition, it can be postulated that local deformation would occur to the casing web or possibly to the tips of the impeller vanes. Cracking would not be expected to occur because of the ductile nature of the pump material. Clearances in all other areas of a primary coolant pump exceed the maximum

chip size. Makeup water flowing into a pump through the pump seals precludes chip or particle damage to the seals. The pump bearings are external to the pump casing.

5.8.4 Primary Coolant Pump Check Valves

A swing check valve is installed on the discharge side of each PCP. Each check valve is equipped with a dashpot to dampen the closing action of the valve disc to eliminate waterhammer effects. The dashpot design and piping arrangement for the check valves have been reviewed to determine if a beryllium particle could interrupt the required operation of the dashpot. The piping arrangement includes a 1/2-in. pipe, located on the upper third of the check valve body, which connects to the dashpot and serves to supply makeup or to remove excess water from the dashpot during operation of the valve. Partial blockage of the 1/2-in. line could be tolerated without any effect to the valve. Even partial blockage, however, is considered unlikely because water in the line is normally stagnant, line routing is vertical and the tap is located on the top third of the valve (remote from areas where particles would tend to settle). Operation of the dashpot is controlled by two bypass lines around the piston on the dashpot cylinder and leakage around the piston. These bypass lines are not connected to the check valve body other than indirectly through a 1/2-in. makeup line. The high velocity (approx. 18.5 ft./sec. for 3 PCP and 22 ft./sec. for 2 PCP) of water flowing through the valve and additional turbulence caused by the check valve disc would further tend to wash any particles flowing in the system through the valve.

5.8.5 Safety Rods

A review of the safety rod drive assembly was performed to determine if it would be possible to lodge a piece of beryllium in any portion of the drive assembly such that the piece of beryllium would bind the safety rod and affect its performance reliability.

There are two places where relatively large beryllium particles could enter a safety rod drive assembly and settle in locations where a

potential problem could develop: (1) in the 0.254-inch clearance between the absorber plate retainer alignment and the in-pile tube insulating jacket, and (2) in the 0.125-inch clearance between the inner flux trap baffle and the absorber plate. If a particle were to enter the 0.125-inch clearance between the inner flux trap baffle and the absorber plate, it would pass through this channel and settle on the cast Stellite No. 3 upper guide bushing located between the 17-4 PH stainless steel outer snubber tube and the 17-4 PH stainless steel rack tube. The largest piece of beryllium which could potentially circulate to the point at which it could affect safety rod movement would be approximately 0.650 in. in diameter (slightly smaller than the inside diameter of the heat exchanger tubes).

The safety rod upper guide bushing has a 10-degree lead-in angle to the minimum clearance section. The total downward force on a safety rod during operation is approximately 850 lbs. from hydraulic loading with two primary coolant pumps operating (1000 lbs. with three PCP) and approximately 400 lbs. from the scram spring force. Any binding of the upper guide bushing sufficient to prevent a scram would have to sustain at least a 1250-lb. load during power operation. Binding of the upper guide bushing would be unlikely since the beryllium particles would enter in from above and this would tend to wedge the safety rod on withdrawal, but not on insertion. A significantly smaller binding force could prevent withdrawal of a safety rod. The withdrawal gear motor has an output torque limit switch which limits the torque to 285-295 foot-pounds. This torque limit switch will shut off the gear motor and is backed up by an electromagnetic clutch, which will slip if excessive binding occurs. This torque limit prevents damage to safety rod components if severe binding occurs.

If beryllium particles were to enter the 0.254-inch clearance between the absorber plate retainer alignment and the in-pile tube insulating jacket, they would pass through this channel and into the space between the in-pile tube insulation jacket and the rack tube. At this point they would settle on the rack tube floating ring (17-4 PH stainless steel), or they would be washed outside the rack tube through one or more of its eighty-four holes (1-1/8-inch diameter) and then settle on the lower guide

bushing. Potential binding problems for the lower guide bushing would be analogous to those of the upper guide tube bushing (discussed above) since the two are identical.

Each safety rod includes a floating seal ring (17-4 PH stainless steel) and a lower bushing retainer ring (17-4 PH stainless steel) just below the lower guide bushing. The floating seal ring and lower bushing retainer ring have diametral clearances of 0.005 (+.0025, -.0000) and 0.016 (+.002, -.000) in. with the inner snubber tube (304 stainless steel), respectively, when the safety rod is seated.[3] These clearances increase gradually as a safety rod is raised approximately 24 in. The diametral clearances increase to approximately 0.270 in. at withdrawals above 24 in. It would be extremely unlikely that a particle would interfere with rod travel in this area due to the acute timing requirement for one of the rings to bind a spherically-shaped particle against the inner snubber tube. If a particle were caught in this position during rod insertion, it would have to sustain a load of at least 1250 lbs. (for 2 PCP operation) to actually create a binding problem during power operation. Since the irradiated beryllium becomes brittle and therefore more easily shattered, it is unlikely the particle of beryllium could withstand the impact load. Particles passing below the rings would land on the cushion plate where they should cause no problems.

The safety rod pinion interfaces with the rack tube in the region between the upper and lower guide bushings. The pinion and bearings in the pinion shaft bearing cage are open. Velocities in this region are very low and the bearings are somewhat away from the main flow paths in the rod. A constant water purge flowing into the reactor from behind the bearings would makes it very unlikely that particles would enter the ball bearings.

In January 1969, the Idaho Nuclear Corporation investigated the effects of metal chips lodged in the articulating joints of the ATR safety rods.[22] The types of chips considered in the investigation were aluminum (similar to alloy 6061) and stainless steel (similar to type 17-4 PH). The chips ranged in size from particles, which require filtration to separate, to 0.15-inch thick by 0.050-inch wide and 0.375-inch long curled

aluminum, to 0.007-inch thick by 0.020-inch wide and 0.050-inch long stainless steel chips. The effects of the metal chips on vertical movement of a safety rod and the hazards to movable components at or below the articulation joint were analyzed in detail. The conclusion of the investigation was that presence of any chips of the stated size and materials in the safety rod system would have no deleterious effects on reactor operation.[22] Beryllium chips, being more brittle than those investigated in 1969, would be less likely to prevent proper safety rod operation.

5.8.6 Neck Shim Rods

A beryllium particle 0.033 in. in diameter, or smaller, could enter the channel between the absorber section of a neck shim rod and the surrounding sleeve. Once in this channel, the particle would settle on the lip of the aluminum follower. If the particle is smaller than 0.023 in. it could pass through a 0.023-in. channel and fall into the wider channel formed by the lower portion of the follower and Inconel X-750 sleeve. Due to the constant water purge through the cable sleeve and into the reactor, the particle would not create any binding in the cable or drive region of the neck shim.

The neck shim rod gear motor has an overload coupling. Keyed to the output shaft of the gear motor and the shim pinion shaft assembly is a torque limiting coupling set at 210 $\pm$ 20 in.-lb. This insures that mechanical damage is prevented by setting the coupling release at a torque below the cable capacity.

The 0.188-in. diameter coolant hole in the center of the shim absorber section could become plugged, but this would not interfere with movement of the shim rod. Complete plugging could result in boiling in the plugged channel, but this would not cause permanent damage to the rod. The possibility of the holes becoming completely plugged is very remote.

5.8.7 Outer Shim Control Cylinders

Two areas of potential damage from a circulating piece of beryllium becoming lodged in the outer shim control cylinder (OSCC) mechanism need to be considered. These are the drive gear and the annular space between the OSCC and the reflector block.

Each OSCC drive mechanism includes a torque release clutch installed between the drive motor and the drive shaft. It is designed to slip if its preset torque is exceeded. This action protects the drive motor in the event an overload condition develops in the horizontal OSCC drive line. It also limits the amount of driving torque that can be applied to the drive line. The torque release clutch is set to release at a torque of 480 in.-lb. (or approximately 300 to 350 in.-lb. for an aged reflector block).

The OSCC drive gear boxes (part of gearbox support beams) are relatively well-shielded from any high velocity flows. Flow through the areas of the worm gears is through a number of holes in the gearbox top plate (0.75-in. dia. holes) and support beam (0.50-in. dia. holes). It should be noted that the current gearbox support beams and associated internals were installed in the ATR in 1986. Pressure differential across a gearbox support beam varies and has a maximum value (3 PCP operation) of 8 psid.[23]

It seems unlikely that a beryllium particle would find its way into a gearbox and become lodged in the gears. If a particle did come to rest in the gears it is expected that the particle could be dislodged without rendering the OSCC inoperable. This would be possible, in part, due to the very slow rotational speed of the OSCCs (0.055 rpm) which would not result in any impact loads during attempts to move them. If an OSCC failed to move in the desired direction upon command, it is likely that the reactor operator would attempt temporary movement in the opposite direction to free the OSCC. Since the OSCC worm drive shafts are horizontal it is likely that the movement in the opposite direction from a stuck position would move the particle up and away from the meshing gear teeth and allow the flow to wash it away from the area.

Applying a modified Lewis formula to the design of the worm drive and gear, it can be shown that the gear and wheel will not sustain damage until a load ten times greater than that allowed by the torque release clutch is applied.[3] This analysis is substantiated by the safety analysis report of the Bendix Corporation in the instruction book for the OSCCs.

The horizontal worm shaft bearings are well protected by their housing and by their retainer rings. It is quite unlikely that particles could be deposited in them. The bearings on the cylinder support shaft are even better protected from contaminants. The upper bearing on a cylinder support shaft is shielded from above by its housing and the lower bearing is shielded by the large gear just above it. However, areas do exist in the bearings where it might be possible for particles to enter the bearing housing. If particles were to enter the bearing housing the bearing retainer would be the most susceptible to damage since it is the softest part of the assembly. The bearing retainer material is 17-4 PH stainless steel. In comparing the compression strength of the bearing retainer to the compression strength of irradiated beryllium it is noted that the strength of the bearing retainer is slightly higher than that for irradiated beryllium ($S_{17\text{-}4}$ = 170,000 psi and S_{Be} = 160,000 psi).[3] The difference is small enough that it cannot be said with any degree of assurance that no damage could occur to the bearing. Gearbox bearings are replaceable, however, it is likely that an entire gearbox support beam would be replaced if a damaged bearing rendered an OSCC inoperable. Adequate reactor shutdown margin still exists with a stuck OSCC pair.

It is quite possible for a circulating piece of beryllium to lodge in the 0.130-in. annulus between the outer shim control cylinder and the reflector block. The lodged particle may be crushed and broken through a brittle fracture mechanism because of the lower brittle fracture strength of the more highly irradiated beryllium comprising the particle. However, if the lodged particle did result in a stuck OSCC pair, the consequences would be acceptable. Adequate shutdown margin would still exist and the reactor could be safely shut down and the necessary repairs made.

It is also possible that a beryllium particle could become lodged between an OSCC and a reflector block with the A ligament cracked. If this were to occur the integrity of ligament C would be of concern. With both ligaments A and C broken, there is a possibility that either ligament D or F would also break. The probability of a broken piece of beryllium having just the right size and shape to become lodged in the annulus exactly at the C ligament is extremely remote. Figure 11 shows a full scale sketch of this postulated condition. It should be noted that in order for this situation to occur, three concurrent conditions would be necessary: (1) The opposite corners of the broken piece of beryllium would need to be sharp, (2) vertical notches would be required on the OSCC and on the thinnest portion of ligament C, and (3) the notches would need to be deep enough and the broken piece of beryllium large enough to prevent shearing or crushing of the piece.

The approximate consequences of the above unlikely situation have been examined by modeling the beryllium at the C ligament as a thin circular ring with a thickness of the C ligament (0.332 in.) and a radius equal to the inside diameter of the OSCC hole in the reflector block (3.75 in.). A radially-applied concentrated load, generated by the OSCC torque acting against the broken piece of beryllium, was assumed to act at the C ligament. The radial load required to produce a bending stress equal to the tensile strength of 52.2 ksi was calculated to be approximately 800 lbs. This calculation was based on an assumed length of 1 in. along the axis of the ring. The assumption of a 1-in. length for the model should be conservative since the distance across the smallest saw cut segment (see Figure 5) is 1.94 in. and a radial load at the C ligament would be likely to at least be spread over this distance.

A scoping calculation has been performed to assess the resultant radial force that could be applied to a C ligament from a broken piece of beryllium wedged in the OSCC annulus. Figure 11 shows a rectangular shaped piece of beryllium (approx. 0.13 in. by 0.5 in.) wedged between an OSCC and the surrounding reflector block. Assuming a smaller dimension than 0.5 in. would result in a larger radial load applied to the C ligament. A small piece of beryllium with one dimension slightly larger

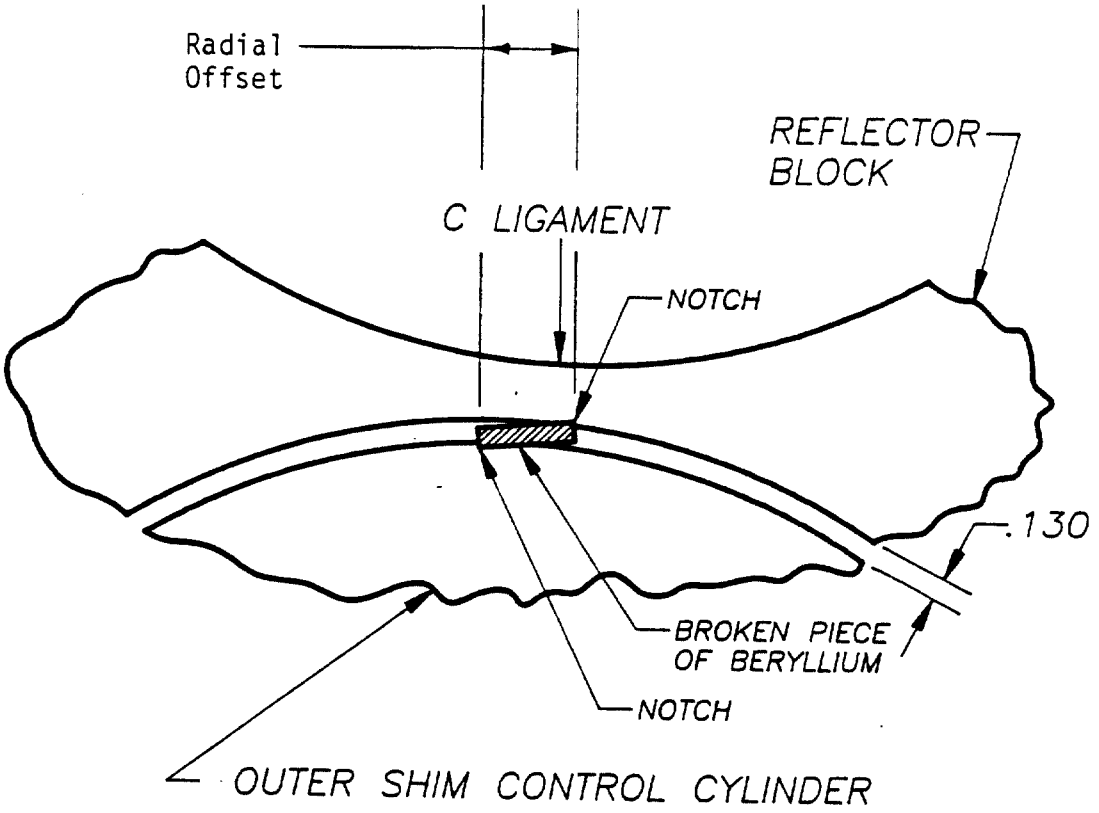

Figure 11. Full Scale Sketch of Ligament C and OSCC Showing a Fragment Trapped in the Annulus.

than the OSCC annulus could be rotated into a binding position by initial movement of the OSCC. A small beryllium piece with a dimension of 0.135in. across the contact corners allows a scoping calculation to be made to predict the resulting radial load that could be applied to the C ligament. A torque on the OSCC worm drive of 325 in.-lbs. (corresponding to use of 300 to 350 in.-lbs. torque release clutch) would produce a corresponding torque on the outer shim cylinder of approximately 1268 in.-lbs. (based on OSCC drive design data by the Bendix Corp.). The radial offset (see Figure 11) between contact points would be approximately 0.036 in. for a nominal annulus width of 0.130 in. and a contact point to contact point distance of 0.135 in. Assuming the piece of beryllium would bind the OSCC, the radial load associated with an offset of 0.036 in. would be approximately 1250 lbs. Therefore, since this is greater than the allowable 800 lbs., it can be concluded that ligament C could be cracked. Even if ligament C cracking were to result, the force producing it would not be envisioned to act through more than a few mils. Therefore, other components such as the flux trap baffle or fuel elements would not be damaged. If a failure allowed subsequent movement of the associated beryllium, the potential consequences would be as discussed previously in this report. Frictional interference resulting from either direct contact or contact through numerous, small foreign particles would be more likely than the postulated chip interference noted above. The resulting loads on the reflector block from a frictional type of interference would, however, be less severe than those determined for a chip type of interference.

5.8.8 Fuel Elements

Movement of beryllium chips through a reactor vessel has, at times, resulted in chips becoming lodged in the fuel element channels or the channels between outer fuel plates and the reflector. Lodging of beryllium pieces in these flow channels was observed during several shutdowns prior to replacement of the beryllium reflectors at the Engineering Test Reactor (ETR). In none of these cases was the flow restriction sufficient to cause a fuel element cladding failure. Potential damage to plates at the ATR from this type of flow blockage is reduced due to the presence of side vents on the fuel elements. These

side vents would reduce the effects of a flow blockage by allowing primary coolant to enter a cooling channel below the point of blockage.

Additional potential damage to the fuel elements exists during their removal if beryllium debris of sufficient size has lodged in channel 20, thus causing increased resistance to upward movement of the elements. This potential damage is reduced for the ATR, as compared to the ETR, since the ATR fuel element plates are of a curved and thus more rigid construction (ETR fuel plates were flat). In addition, the outside plate cladding (plate 19) of the ATR fuel elements is 0.040-inch thick as compared to 0.015-inch for the ETR elements, thus offering a greater measure of protection from failure due to scratches. It has been conservatively estimated that a flow channel area blockage of 75% would be required before a plate cladding failure would occur.[3] It is therefore unlikely that a fuel element failure would occur from one of the above sources. However, if a failure should occur, it would be on the order of a blister failure which would encompass a very small area of the fuel plate.

5.8.9 Test Reactor Operating Experience

Operation of both the Materials Testing Reactor (MTR) and the Engineering Test Reactor (ETR) was continued after observation of initial deterioration of their beryllium reflectors. Operation of the ATR has been successful after initial cracking of Reflectors I, II, and III.

Beryllium damage at the MTR was first evidenced when bowing of a beryllium shim rod was observed during the Cycle 184 shutdown in 1962.[3] Investigation of beryllium lattice pieces, initiated during Cycle 187 in 1963, revealed cracking and spalling of reflector pieces immediately north of the core. A program of damage surveillance was initiated with the aid of photographs taken during each shutdown. The program was continued until the beryllium reflector pieces were replaced during Cycle 295 in 1969.[3] On several occasions during the surveillance period, small pieces of beryllium were found to have been washed from the reactor tank to other locations of the primary system. Beryllium chips of various configurations and approximately 3/4-in. in length, together with other

debris, were found in the seal tank, the evaporator tubes, and in the primary system strainer. These beryllium pieces indicated radiation levels on the order of 5 R/hr.[3] During various shutdowns, some reflector debris was observed and removed from the top of the core. Migration of these beryllium chips had no noticeable effect on the operation of the reactor or the water systems other than to increase the radiation levels at the locations where they came to rest.

Deterioration of the ETR beryllium reflector pieces was first observed during Cycle 94 in February 1968 when cracks were found in the C-13 and C-16 beryllium pieces.[3] This damage, as well as damage at other reflector locations, was kept under photographic inspection surveillance until replacement of the reflector pieces during the Cycle 107 shutdown in the Spring of 1970. During this period, continued spalling of the reflector occurred. Some of these chips were also washed out of the reactor vessel and came to rest at low spots in the primary system piping. Migration of the chips from the vessel presented no operational difficulty.

Beryllium reflector experience at the ATR is described in Section 4 of this report. No operational problems due to circulating beryllium particles or pieces have been identified at the ATR.

The operating experience at the MTR, ETR, and ATR, with aged beryllium reflectors, indicates that migration of small pieces of the beryllium from the reactor vessel to other primary system locations has been tolerated without operating incident.

5.9 Cracked Beryllium Effects on Core Physics

5.9.1 Reflector Support Tank Failure

The reflector support tank includes a support plate structure which is located directly below the reflector. A reflector piece falling onto this plate is not expected to cause failure of the plate (as shown in Section 5.3 of this report). If the plate were to fail, however, allowing

the piece of beryllium to fall totally out of the core region, the reactivity effect would be negative. Replacing beryllium with water causes a negative reactivity effect.

5.9.2 Flux Trap Baffle Failure

Flux trap baffle failure due to the impact from a nosepiece of the reflector is not expected to occur (as shown in Section 5.6.2 of this report). However, in the event of the failure of a flux trap baffle, a safety rod could be rendered inoperable thereby reducing the reactor shutdown margin. The reactivity worth of the safety rods is evaluated for each reactor operating cycle to assure that there is at least 5.8$ reactivity in the safety rods for 3 PCP operation and at least 6.4$ reactivity for 2 PCP operation assuming that one of the rods fails to insert. Sufficient safety rod worth would still be available to shut down the reactor in the event of such a remote failure. Other reactivity effects due to failure of the flux trap baffle are negligible.

5.9.3 Variations in Width of Cooling Channel No. 20

Calculations have been completed to determine the reactivity effects and changes in power generation in fuel plate 19 due to variations in water channel 20. Loss of reflector pieces or reflector movement could cause changes to the width of channel 20. The calculated peak power density in plate 19 increases as the beryllium adjacent to the plate is replaced with water. The peak power density increases as much as 19 percent for the removal of a 25-degree sector of beryllium adjacent to fuel position 27 (a typical high-power position) and as much as 10 percent for the removal of a 25-degree sector of beryllium adjacent to fuel position 28.[24] It may be noted that fuel position 28 (and symmetric positions) is adjacent to the tip of a reflector block and cracking in the tip region has been experienced. The anticipated aging effects of the beryllium adjacent to fuel position 27 are cracking of the A ligament followed by spalling which would not lead directly to large variations in the water channel.

A 0.13-in. increase in the water channel thickness surrounding the fuel annulus was calculated to increase the peak power density in plate 19 by 3.6% to 13.6% (for fuel positions 26, 27, and 28).[24] The peak power density effect of removing the reflector in only a one-inch wide strip at the narrowest portion of the ligament is an increase of approximately 10% near the plate center and only 5% near the edge of the fuel.[25] The effect of a 1/8-inch wide crack in the beryllium is an increase in the peak power density of less than 2%.[26]

The calculated reactivity effect of reducing the thickness of the water annulus between the fuel and the reflector for one reflector block (involves three fuel elements) is 0.13$.[24] Although a reactivity step of this magnitude would not necessarily be fully compensated by the automatic control system, it is not large enough to cause a power transient resulting in damage to the core.

5.9.4 Binding of Outer Shim Control Cylinders

Binding of the outer shim control cylinders (OSCC) due to movement of the beryllium reflector would compromise the reactor shutdown margin. The reactor would, nevertheless, be shut down in excess of one dollar provided five safety rods and all neck shim rods would operate.[7] This shutdown margin would apply to a typical fuel loading near the beginning of the reactor cycle (the worst time for such a failure to occur) and assumes that one safety rod would not operate. This margin would increase as the cycle progressed due to fission product poisoning and fuel burnup. However, the simultaneous damage to all OSCCs such that they could not be inserted is incredible. A more credible failure would involve damage to a safety rod and attendant failure of an OSCC pair. If this failure were to occur with all OSCCs fully withdrawn (e.g., while attempting to override xenon following a scram), the reactor could be easily shut down with the operable safety rods. Insertion of the operable OSCCs would provide an adequate shutdown margin following the complete decay of xenon.

Even if only half of the OSCCs were operable, the shutdown margin of the reactor with three safety rods, all neck shims, and half the OSCCs inserted has been calculated to be approximately 4.0$.[7] This margin would apply to a typical fuel loading (critical at 60 degrees) in an 18-20-24-15-30 power division with the N, W, and SW safety rods assumed to be operable and the E, S, and SE safety rods assumed to be inoperable.[7]

6. ANALYSIS OF BERYLLIUM AGING EFFECTS ON ATR ACCIDENTS

This analysis considers those accidents that result in changes to the fuel element performance and in changes to the margin to CHF (critical heat flux). The limiting accident in each section of the Advanced Test Reactor Plant Protective System and Technical Specifications Design Basis Report (DBR)[27] has been evaluated as described below for conditions peculiar to a reflector after it has aged to the point that calculated stresses reach two standard deviations to failure. In addition, several accidents that are specific to reflector aging effects have been addressed.

As indicated previously in this report, beryllium aging is expected to be exhibited by cracking, spalling, and bowing. Also, some small holes through beryllium ligaments could develop. In order to envelop the many possibilities of the aging effects, a complete blockage of channel 20 just below the bottom set of fuel element side plate vents was evaluated for loss of flow, loss of coolant, and slow power change accidents. This configuration represents the worst condition to be expected for these accidents.

For reactivity addition accidents, the power peaking is more important than the coolant adjacent to fuel plate 19. Therefore, it was assumed that a piece of the reflector was missing resulting in an increase of 10% in the plate power generation.[28]

The analyses reported in this section have established that a lobe power of 48 MW is acceptable with a standard 7F fuel element adjacent to a cracked beryllium ligament. For operation at higher lobe powers, a special fuel element with no fuel in plate 19 (Type YA element) will be required adjacent to a ligament which has been analyzed to have reached a stress level two standard deviations to failure (even if cracking has not initiated).

The description below (for Sections 6.1 through 6.10) follows the DBR format and addresses the limiting accidents in each section of the DBR.

The final digits of section numbers, after the heading number "6", are identical to the major DBR section numbers up to Section 6.11. Several additional accidents are addressed beginning with Section 6.11.

6.1.0 Primary Coolant Flow

6.1.1 High Flow (High Core Differential Pressure)

The operation of a reflector beyond two standard deviations to failure would not change the development or consequences of high core differential pressure accidents. A high differential pressure, however, would change the load on the reflector and perhaps lead to premature or additional cracking. The consequences of such cracking would be enveloped by those reported in Section 5 of this report except for the higher loads on the reflector support tank which may occur if the high differential pressure caused beryllium pieces to break free of the reflector.

Reference 27 indicates that a high core differential pressure event could increase the pressure differential to 130 psid (3 PCP). The impact load of a beryllium nosepiece on the reflector support tank has been evaluated for this increased differential pressure using the same methodology used for Section 5.3.2 of this report. The calculated shear stress in the reflector support tank was 7,401 psi. The allowable shear stress is 7,600 psi (see Section 5.3.1). Therefore, it can be concluded that an aged reflector will not affect the consequences of a high core differential pressure event. The probability of a high differential pressure event is presently less than that during operation with reflectors I and II due to the current predominant operation with 2 PCP instead of 3 PCP.

6.1.2 Low Flow

The limiting anticipated low flow event is the loss of all PCP leading to a flow coastdown. This accident is enveloped by the limiting unlikely event which is a flow coastdown of both the PCPs (2 PCP operation) and the emergency pumps (short-term loss of power to pumps).

Therefore, only the unlikely coastdown was analyzed (at 48 MW lobe power) with the additional assumption of channel 20 blockage by a piece of cracked beryllium.[11] Both BOL and EOL fuel configurations were considered and the worst effect occurred for the EOL case.[11] The calculated margins to DNB (departure from nucleate boiling) and flow instability are shown in Figures 12 and 13.[11] Since these margins are greater than three standard deviations the postulated event is acceptable as an anticipated event.

The limiting extremely unlikely event for low flow is a long term complete loss of flow (CLOFA). The calculations for this event, with cracked beryllium blockage of channel 20, indicated the approach to the aluminum water ignition threshold (AWIT) shown in Figure 14.[11] The closest approach is approximately 5.8 standard deviations for AWIT, and the closest approach to cladding liquidus is 2.9 standard deviations. The CLOFA is acceptable at 48 MW with channel 20 blocked.

6.2.0 Primary Coolant System Pressure

6.2.1 High Pressure

There are no high pressure events that are affected by an aged reflector.

6.2.2 Low Pressure

Low pressure events are enveloped by the consequences of the 3-in. LOCA and the long-term CLOFA (slow depressurization due to leakage). These events, combined with cracked beryllium, will also be enveloped by the 3-in. LOCA and long-term CLOFA with cracked beryllium.

6.3.0 Loss of Primary Coolant

Loss of coolant accidents in the anticipated and unlikely categories are enveloped by a 3-in. pipe break. The calculated margins to DNB and

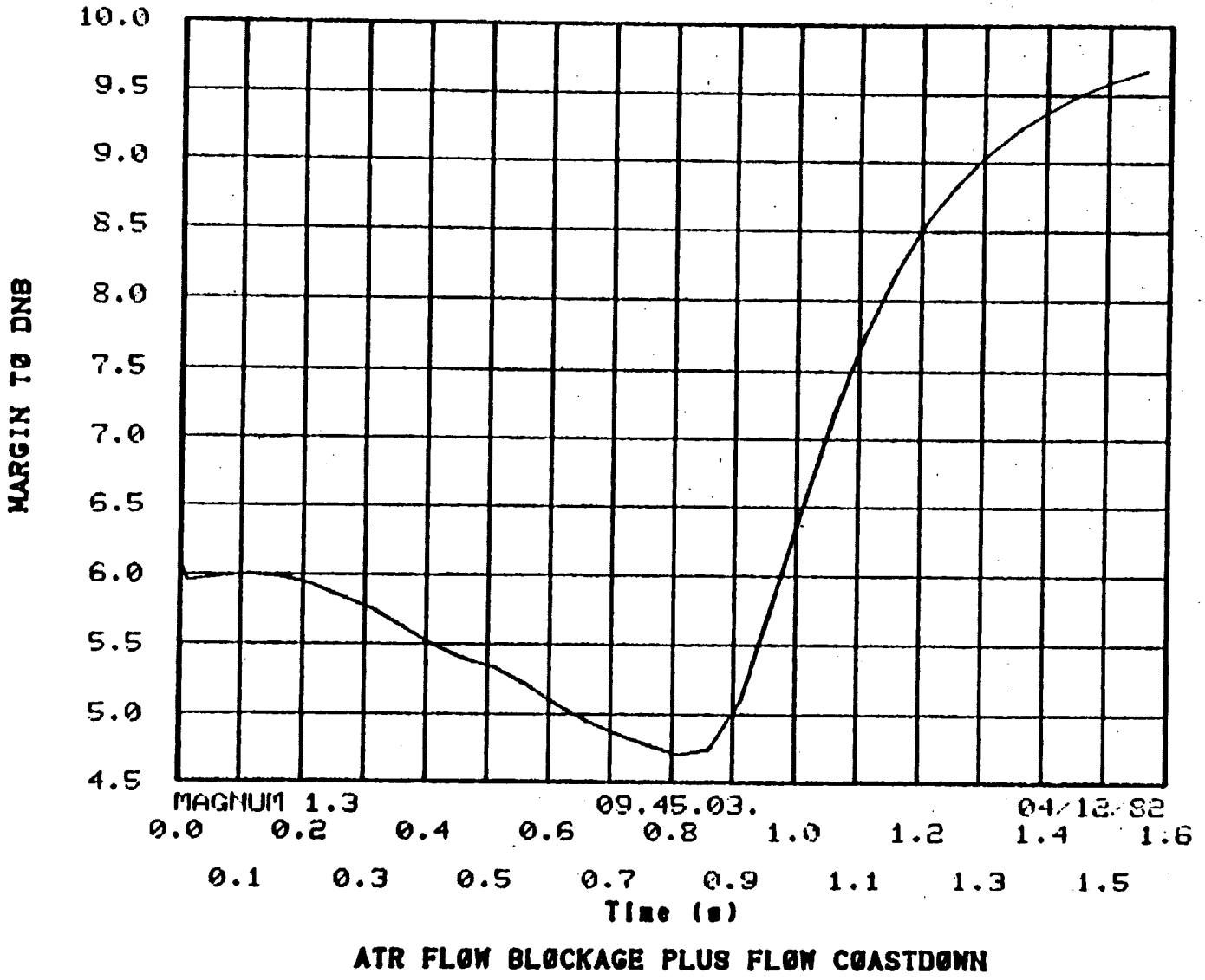

Figure 12. Number of Standard Deviations to DNB for the Coastdown of All Coolant Pumps with Channel 20 Blocked.

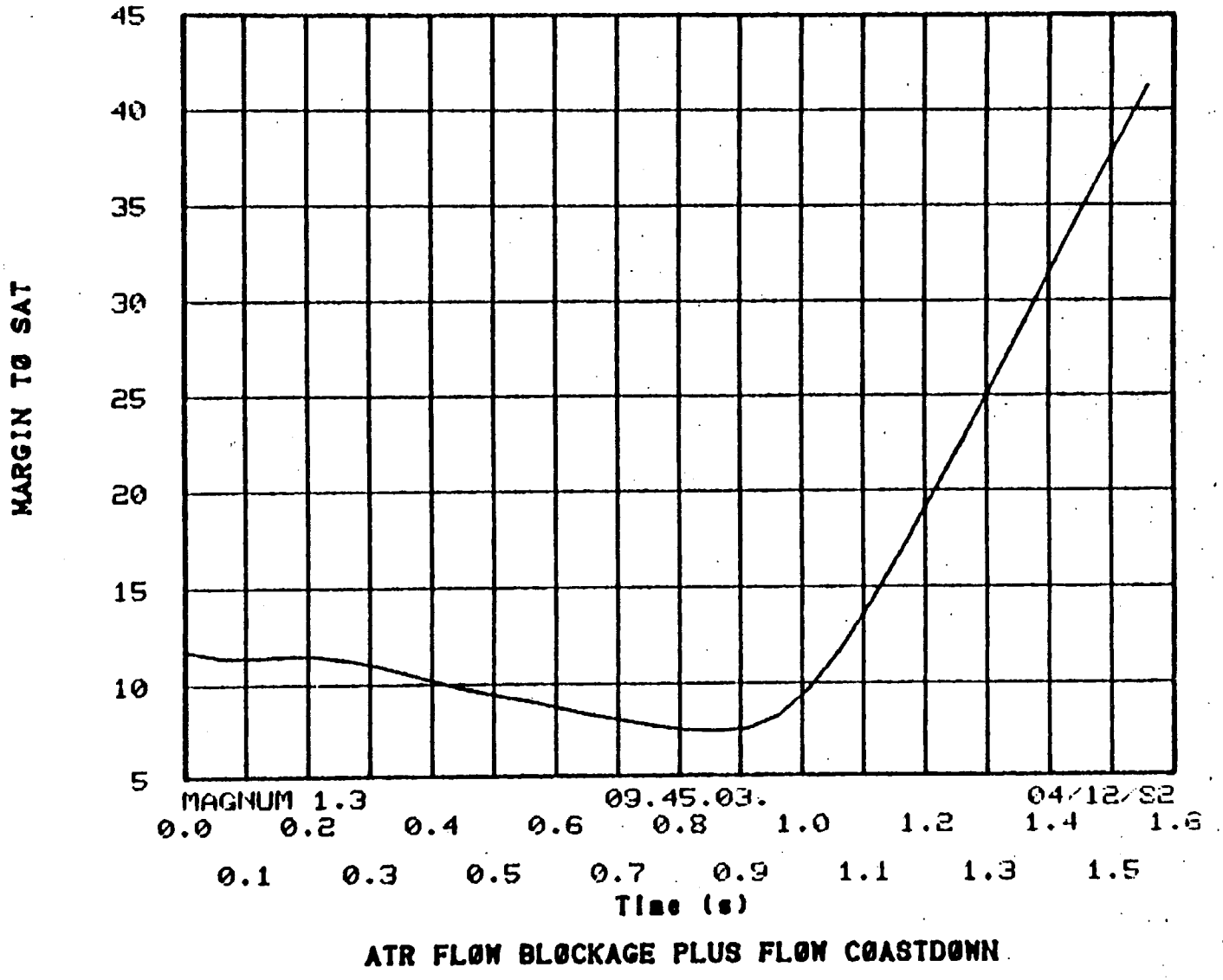

Figure 13. Number of Standard Deviations to Flow Instability for the Coastdown of All Coolant Pumps with Channel 20 Blocked.

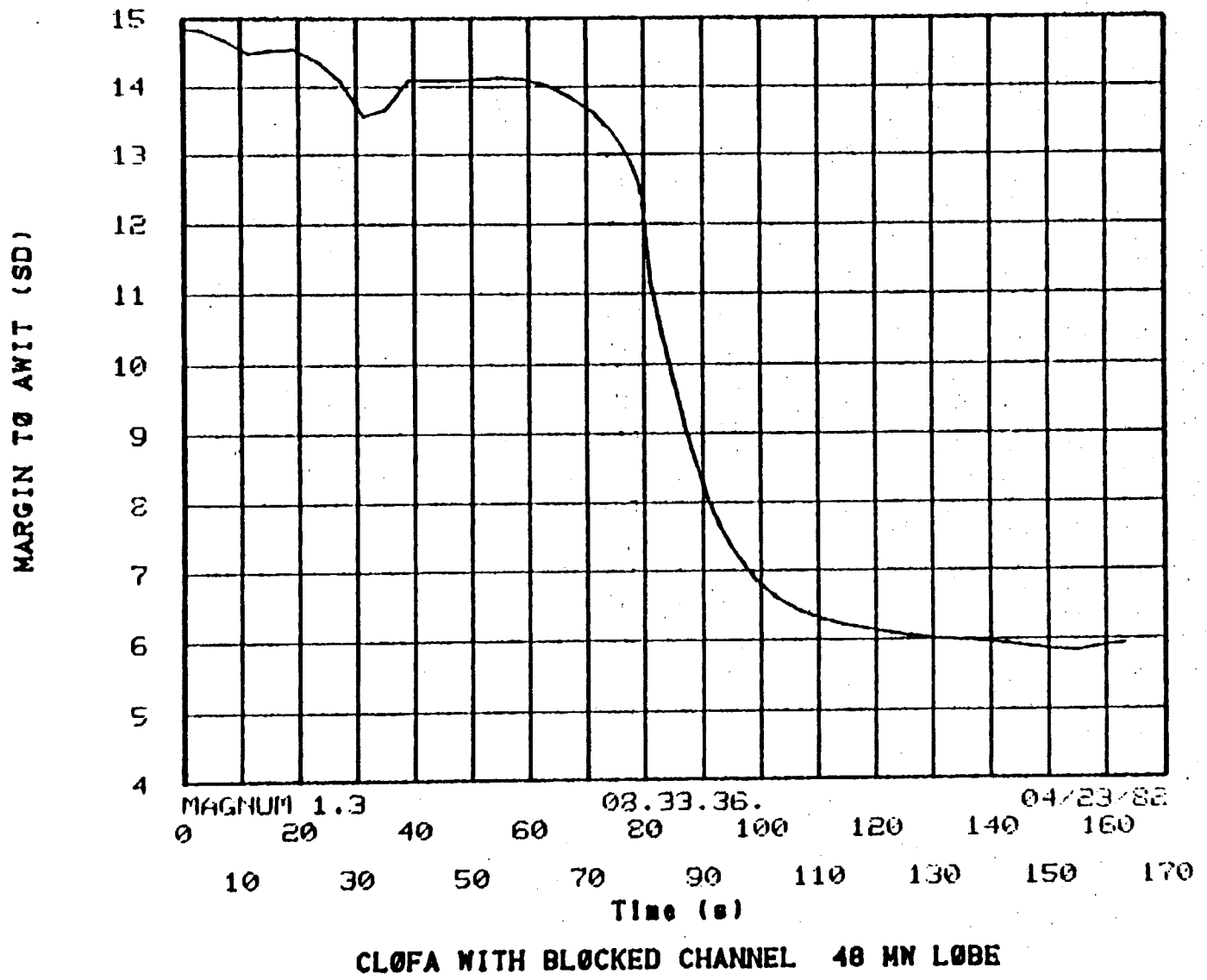

Figure 14. Number of Standard Deviations to AWIT as a Function of Time for CLOFA with Channel 20 Blocked.

flow instability (saturation conditions) for this LOCA with a blocked channel 20 at 48 MW lobe power are shown in Figures 15 and 16. Since the minimum margin is 3.8 standard deviations, the event is acceptable in both the anticipated and unlikely categories.

The limiting extremely unlikely loss of coolant event is enveloped by the 8-in. LOCA. An 8-in. LOCA has been predicted to result in some fuel melting at the hot spot. Combining this event with a blockage of channel 20 by a piece of cracked beryllium could result in some additional melting. The increased melting, if any, would be negligible.

It can be concluded that loss of coolant accidents, with a lobe power of 48 MW, 2 PCP operation, and channel 20 blocked by a piece of beryllium, would result in acceptable consequences.

6.4.0 Insufficient Heat Sink

The abrupt and complete loss of secondary coolant envelopes the accidents in the anticipated and unlikely categories. The abrupt and complete loss of secondary and the long-term CLOFA envelop the extremely unlikely events. The complete and abrupt loss of secondary has been analyzed (48 MW lobe power, 2 PCP) with channel 20 blockage by a piece of beryllium. For an inlet temperature of 150 degrees F, the calculated margin to DNB was 5.6 standard deviations and the margin to flow instability was 9.9 standard deviations.[11] Since the PPS setpoint for the reactor vessel inlet temperature must be no greater than 144 degrees F, it can be concluded that the loss of heat sink event is acceptable with cracked beryllium blocking channel 20. (Note that the PPS setpoint provides protection for the PCS piping against excessive temperatures,and in doing such, it also protects the fuel in this case.)

6.5.0 Reactivity and Power Excursion

6.5.1 Effects of Reflector Aging

The ATR reactivity-initiated accidents are limited by the energy deposition in fuel plate 17 instead of plate 19 which is limiting for

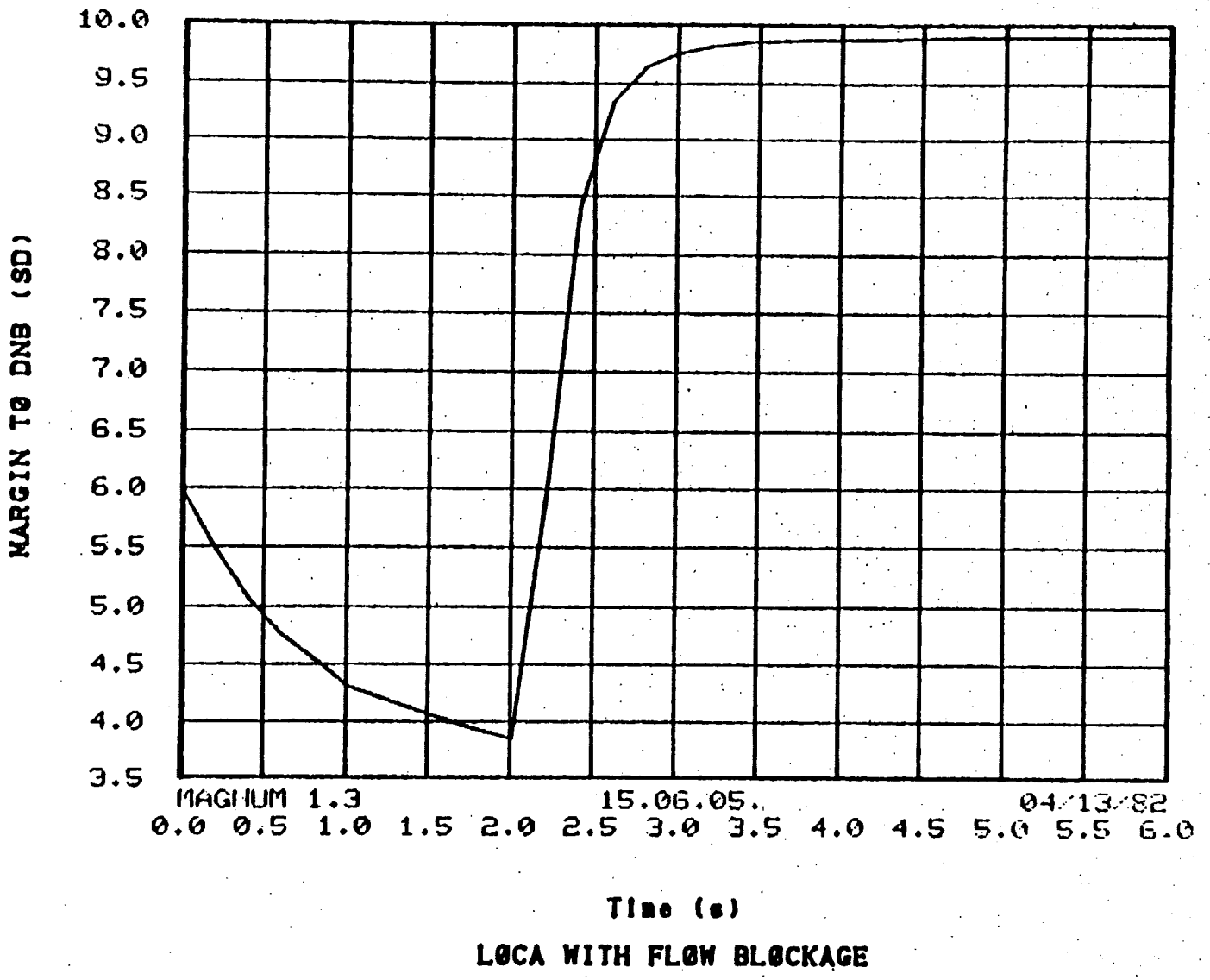

Figure 15. Number of Standard Deviations to DNB as a Function of Time for the 3-in. LOCA with Channel 20 Blocked.

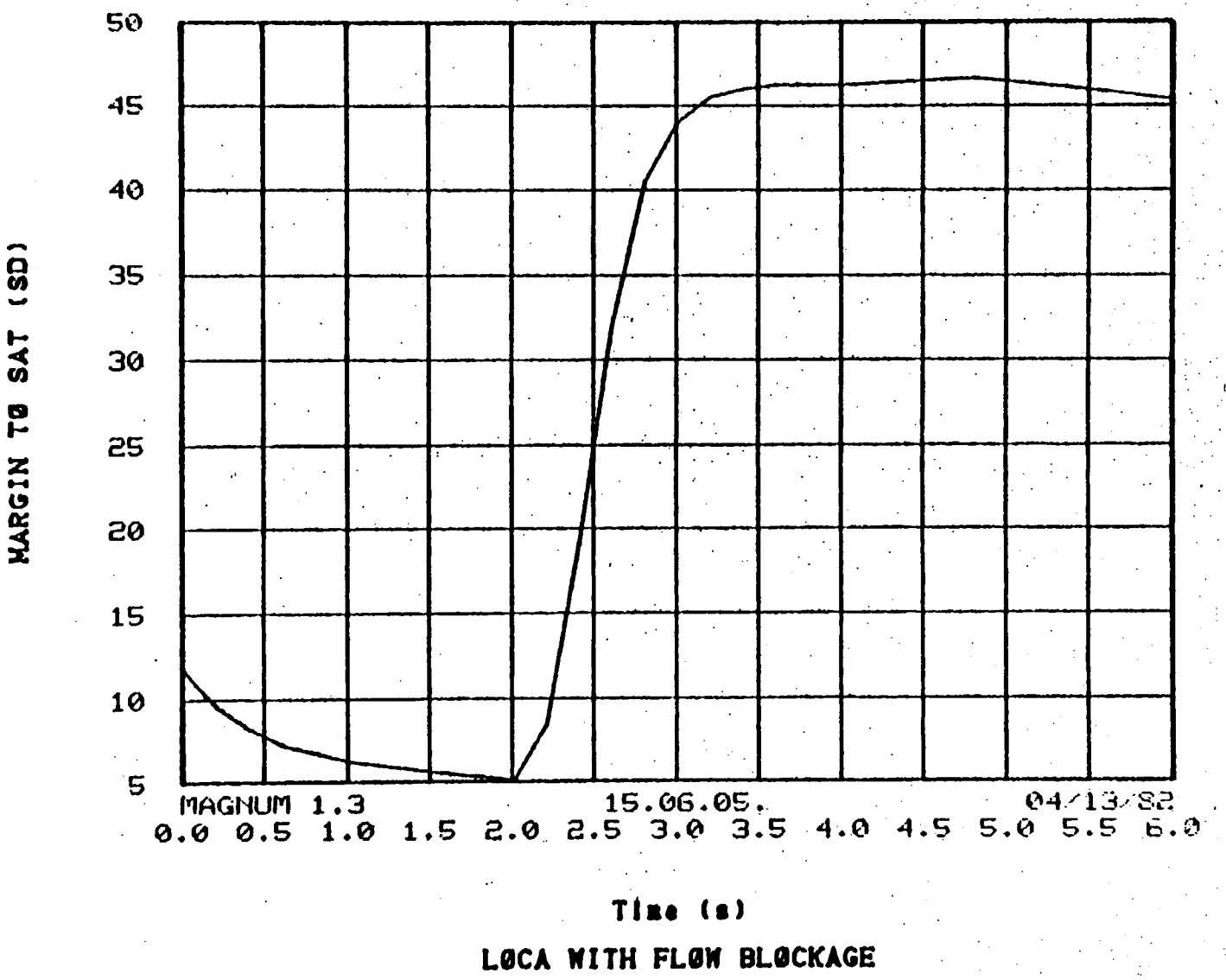

Figure 16. Number of Standard Deviations to Flow Instability as a Function of Time for the 3-in. LOCA with Channel 20 Blocked.

accidents involving coolant perturbations. Also, the reactivity-initiated events occur more rapidly than coolant perturbation events. As a result, insulation of plate 19 by a beryllium blockage in channel 20 is not as severe as a perturbation to the power generation in the fuel plate adjacent to cracked beryllium. Note that replacement of beryllium with water would cause a power increase in the adjacent fuel plate.[7] In order to envelop the various events that could occur, an increase of 10% in plate power for plate 19[28] was used in analyzing the reactivity accidents. The worst-case power peaking considered for the two-PCP analysis contained in Appendix A of Reference 27 was the base distribution. This analysis assures that plate 19 temperatures will not exceed allowable limits for operation up to 48 MW lobe power with a hole in the adjacent reflector block. Above 48 MW lobe power, plate 19 will not be fueled, and the hole will not have any effect on plate 19. Such a hole would, however, cause a perturbation to the plate 17 power distribution. The data of Reference 29 indicate that the power peaking at the hot spot is increased by about 2.5% due to a one-inch wide crack. It is therefore necessary to reduce the allowable peaking for the element adjacent to a ligament that has reached stress levels of two standard deviations to failure.

6.5.2 Limiting Anticipated Events

The limiting anticipated reactivity insertion event is the draining of an in-pile tube (IPT) due to a 1/2-in. pipe break in the experiment loop. The analysis for this event, in combination with a 1-in. by 2-in. hole in the reflector, at a lobe power of 48 MW indicated that the minimum margin to DNB would be 4.8 standard deviations and that the minimum margin to flow instability would be 6.4 standard deviations.[11] These margins, as a function of time, are shown in Figures 17 and 18. Small holes in the reflector face adjacent to the fuel annulus are possible. A small hole approximately 1-3/4 in. by 1/2 in. was observed at the A ligament adjacent to fuel position 34 in Reflector III. Also, reflector experience has included cracks in the tip regions of reflector blocks; it is conceivable that cracked tip pieces could fall away from the reflector during times when fuel is removed from the reactor.

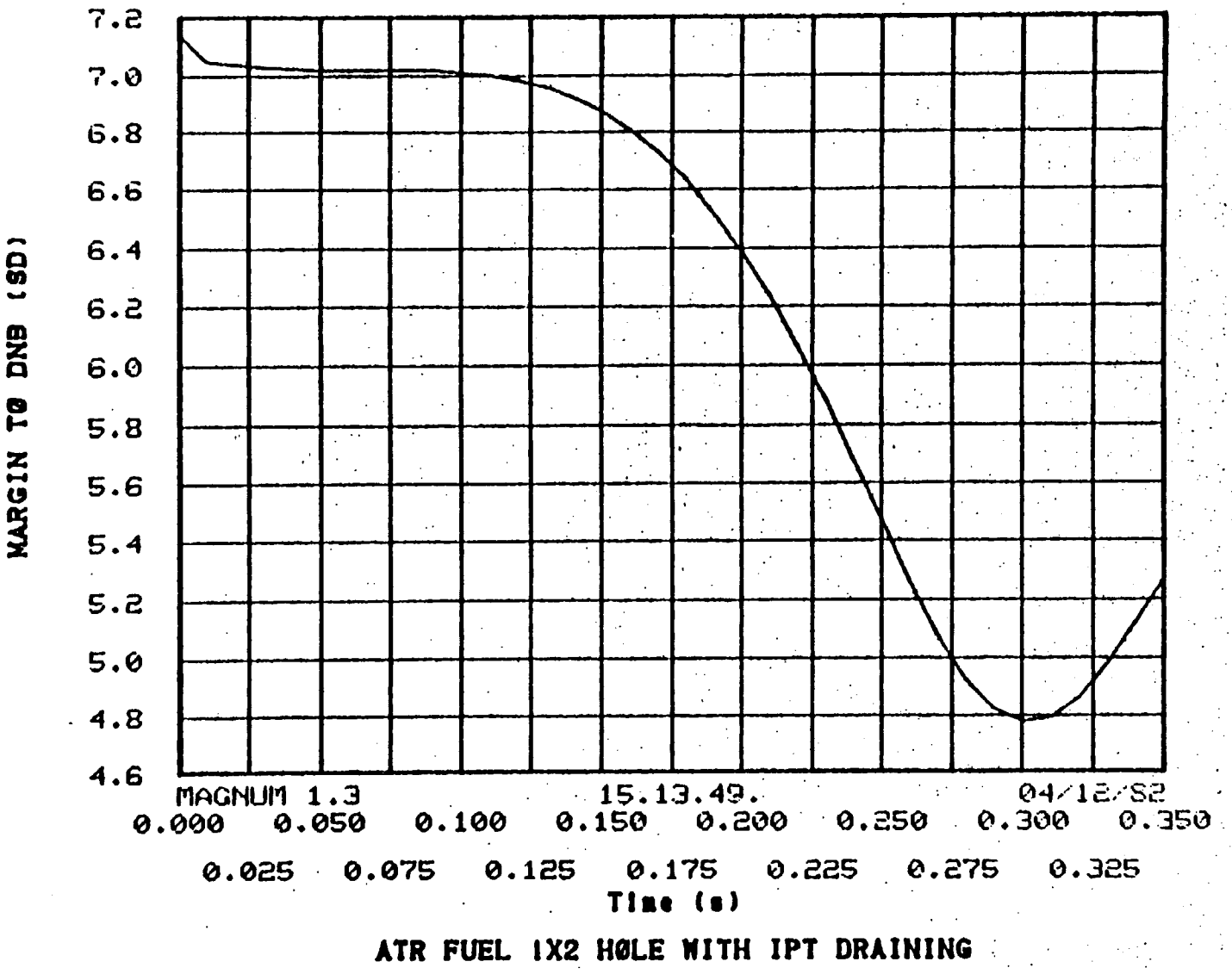

Figure 17. Number of Standard Deviations to DNB as a Function of Time for IPT Draining Due to a 1/2-in. Pipe Break with a 1 in. X 2 in. Hole in the Reflector.

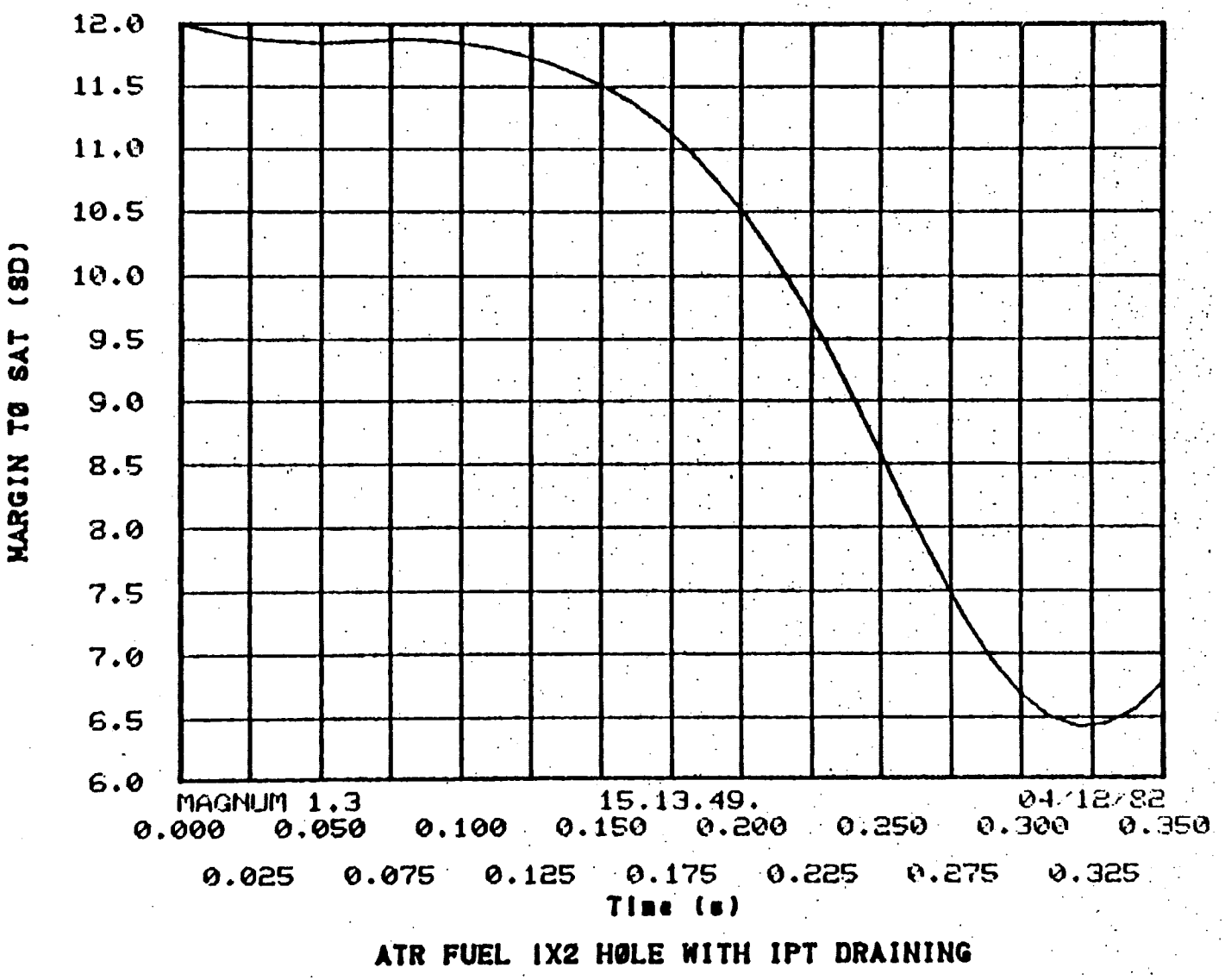

Figure 18. Number of Standard Deviations to Flow Instability as a Function of Time for IPT Draining Due to a 1/2-in. Pipe Break with a 1 in. X 2 in. Hole in the Reflector.

A slow reactivity insertion (slow power increase) must also be considered. This abnormal condition could result in an overpower of one of the lobes.[27] Assuming that this condition would occur in the lobe with a highly stressed beryllium block, an overpower condition greater than 48 MW could occur with channel 20 blocked. Since the reflector condition would not alter the initiating event, the overpower would still be 1.34 N_F.[27] However, as shown in Reference 11, the margin to DNB at 1.37 N_F (where N_F is 48 MW) is 3.8 standard deviations and the margin to flow instability is 6.1 standard deviations. Therefore, the required margins would be maintained for this event as long as N_F for the neutron levels and wide range system (WRS) correspond to 48 MW lobe power.

6.5.3 Limiting Unlikely Events

The limiting unlikely accident is a 0.75$ step reactivity insertion. A 0.75$ step reactivity insertion, in combination with an aged reflector, was not specifically analyzed since the needed conclusion could be drawn from the results of an analysis for a 1.0$ step insertion (with a 1-in. by 2-in. hole in the reflector). The 1.0$ step analysis showed that the plate 19 temperature would not be as severe as the plate 17 temperature for the same power transient without the hole in the reflector. The power transient for both a 0.75$ step and a 1.0$ step would occur in a very short period of time and consequently only retained energy would be important. Therefore, it can be concluded that plate 19 temperatures for a 0.75$ step insertion would also be less than those for plate 17 for the same power transient. The larger mass of plate 19 would contribute to a lower temperature than plate 17 for the same retained energy.

Another unlikely event, the lobe power shift with compensation by shim movement, has been considered to assure that the proper quadrant differential temperature setpoint exists. The analysis for this event showed that a quadrant differential temperature setpoint of 1.2 times the nominal value will limit the lobe power to 137% of the nominal lobe power.[11] Previous analysis had indicated that a setpoint of 1.24 times the nominal differential temperature would limit the lobe power to 145%.[27] The margins to DNB and flow instability for a lobe power of

137% of 48 MW (with channel 20 blocked by beryllium) were determined to be 3.8 and 6.1, respectively.[11] (The power shift is slow enough that blockage is the worst case). Therefore, the lobe power shift event (with shim compensation) is acceptable provided the quadrant differential temperature subsystem setpoint is set at 120% of the normal differential temperature with the adjacent lobe power at 48 MW. (This is the same setting as presently required except the lobe power is limited to 48 MW instead of 60 MW maximum.)

6.5.4 Limiting Extremely Unlikely Events

The limiting extremely unlikely events are the 1.0$ step reactivity insertion event and the IPT draining event due to a shear of the experiment loop piping. The consequences of the 1.0$ step event, however, envelop those of the IPT draining event.[27] Therefore, only the 1.0$ step insertion was analyzed (with a 1-in. by 2-in. hole in the reflector and initial reactor power of 200 MW).[11] The calculated margin to AWIT (aluminum water ignition threshold) is shown in Figure 19 (calculations were terminated at 0.25 seconds). The results indicated that the plate temperature would reach a maximum value at 0.2 sec. and would then begin decreasing.[11] Since the plate temperature would be decreasing at 0.25 sec., it can be concluded that the margin to AWIT would not decrease significantly after 0.25 sec. and that the criterion to maintain at least three standard deviations to AWIT would be met. Reference 27 indicates that a similar analysis (without hole in reflector, 1.0$ step insertion, initial reactor power of 250 MW), resulted in the margin to AWIT decreasing at 0.25 sec. but stabilizing between 0.25 sec. and 0.30 sec.

6.5.5 Design Basis Accident

The limiting design basis event is a 1.3$ step reactivity insertion. This event would be expected to result in a significant amount of core melting. The reflector condition at the time of such an event could cause only a minor, if any, perturbation to the amount of melting. Therefore, it can be concluded that the design basis accident consequences would be unchanged by reflector aging.

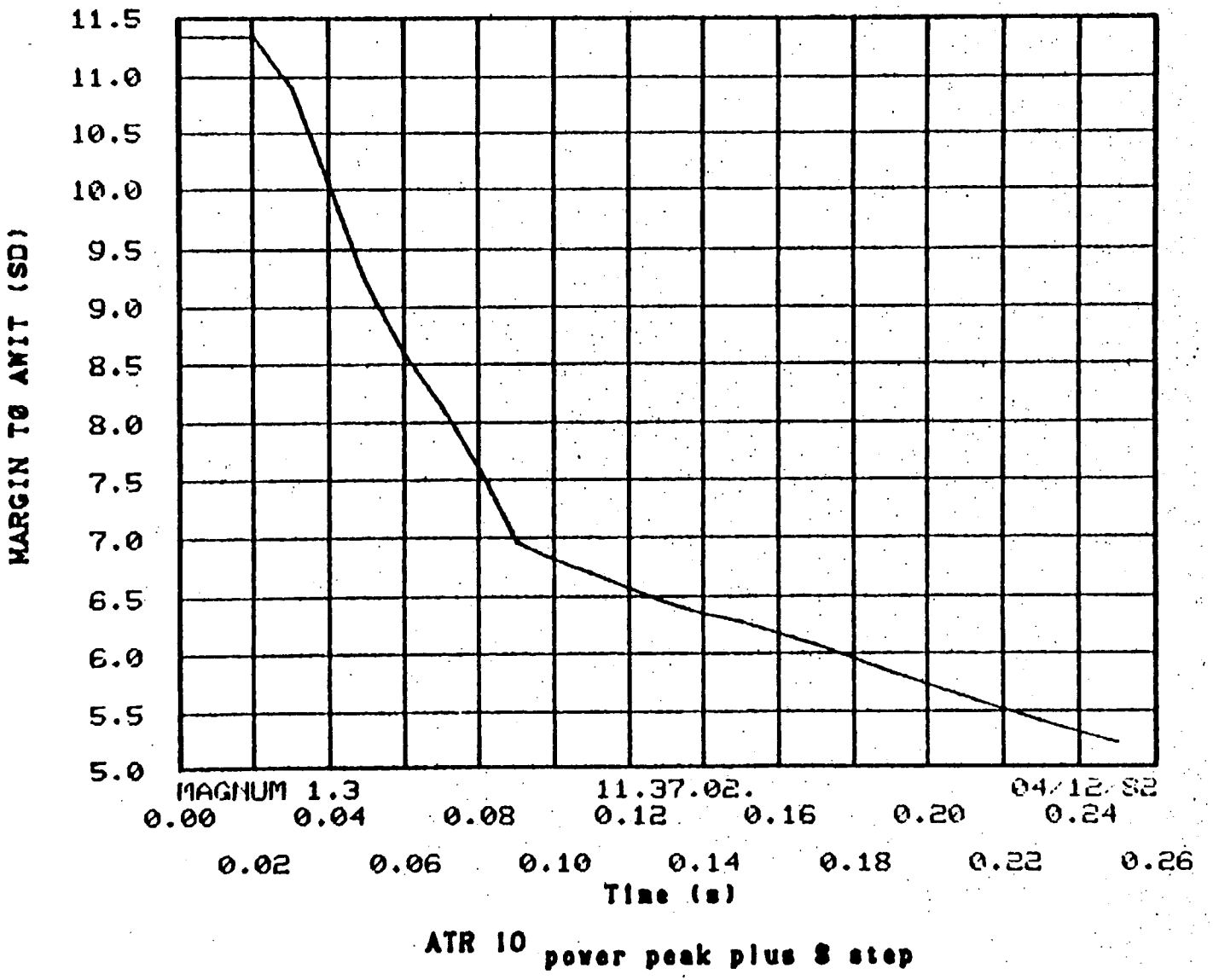

Figure 19. Number of Standard Deviations to AWIT as a Function of Time for a 1$ Step Reactivity Insertion with a 1 in. X 2 in. Hole in the Reflector.

6.6.0 Fuel Element Failure

Anticipated fuel element failures are enveloped by minor channel blockages (not extending completely across a channel). These blockages have a minor effect on channel flow and margins to CHF. The combination of an anticipated channel blockage and beryllium blockage of channel 20 would result in consequences no worse than the two conditions occurring by themselves, that is, unless they would be located on immediately opposite sides of plate 19. This combination of blockages, due to its low probability of occurrence, can be classified as unlikely. The numerous coolant channels in the core would make it unlikely that a minor blockage would occur in channel 19 of an element at a point adjacent to a highly stressed reflector block where a piece could become free and block channel 20.

Unlikely fuel element failures are enveloped by channel blockages extensive enough to result in melting. In order for the combination of an unlikely blockage and a beryllium blockage to be worse than the individual blockages, they would have to occur on opposite sides of plate 19. If these concurrent blockages were to occur on opposite sides of plate 19 the extent of melting would not be expected to be significantly different (the extent of melting for an extremely unlikely blockage is described only in general terms in the DBR and is not quantified). The melting from concurrent blockages could be expected to affect only the lower portion of plate 19 (probably only a portion between two saw cuts).

Extremely unlikely fuel element failures are enveloped by gas leakage into the core thereby resulting in a compromise of the heat transfer from the plates to the coolant. The gas leakage event was evaluated in combination with beryllium blockage of channel 20 for a lobe power of 48 MW.[11] The method used for evaluating this condition did not include a calculation of the standard deviations to CHF. However, consistent with the methods used in the DBR (Ref. 27), the DNB ratio (ratio of heat transfer at DNB to that calculated for the hot channel) and the flow instability ratio (ratio of channel differential temperature to differential temperature required to reach saturation) were calculated for

2 PCP operation. The DNB ratio and flow instability ratio were 2.4 and 0.79, respectively. The corresponding ratios for the gas leakage event at a lobe power of 60 MW, but without channel 20 blockage were 1.9 and 0.91 respectively.[16] Therefore, the margins to DNB and flow instability for gas leakage (48 MW lobe power) with a blocked channel 20 would be greater than those reported in Reference 16. (Note that worst case ratios are compared; axial locations of the worst cases were not necessarily the same.)

6.7.0 Fuel Storage Canal and Cask Handling Accidents

There are no events in this category that would be affected by reflector aging.

6.8.0 Earthquakes

The ATR seismic system is designed to shut the reactor down before physical damage from a seismic event can cause damage sufficient to preclude safety rod insertion. A seismic event could conceivably cause failure of a highly stressed reflector; however, the reactor would be shut down either before or concurrent with such a failure. Since the reflector failure would be expected to be cracking, the consequences of the failure would not be likely to extend beyond those analyzed in this report.

6.9.0 Radioactive Contamination Releases

There are no identified radioactive contamination release events which would be affected by reflector aging. A reflector aging accident involving a fission break is addressed in section 5.2.4 of this report.

6.10.0 Failure of Supporting Systems

The only supporting system event affected by the reflector condition is related to the mispositioning of safety rods and shims. The mispositioning of a safety rod has been reported to result in a power

peaking increase of 27%.[27] However, an overpower event of 37% with channel 20 blocked by beryllium has been shown to meet the minimum margins to DNB and flow instability.[11] Therefore, beryllium aging in combination with a mispositioned safety rod is acceptable for operation with a 48-MW lobe power.

The mispositioning of neck shims (22 neck shims at 50% withdrawn was considered the limiting case) has been reported to result in an increase of 27% in the peak power.[27] This event, in combination with channel 20 blocked by beryllium, would also be acceptable as noted above.

The mispositioning of the OSCCs can cause a significant power tilt across a lobe. Data obtained by measurements in the ATR Critical Facility has shown that the extreme error in positioning (one set fully withdrawn and one set fully inserted) is 24%.[30] Since this is less than 37% (See above discussion of mispositioned safety rod), the error is acceptable for a 48-MW lobe power. The DBR (Ref. 27) also addresses shim mispositioning with compensation by shims adjacent to a different lobe; this is essentially the unlikely event treated in Section 6.5.3 of this report.

6.11.0 Additional Reactivity Effects of Reflector Aging

Several events, in addition to those addressed in preceding sections of this report, have been considered. These can be described as the reactivity effect of reflector aging combined with the following anticipated and unlikely events:

(a) IPT draining from a 1/2-in. experiment loop pipe break.
(b) The limiting unlikely reactivity ramp.
(c) PCP coastdown.
(d) The 3-in. LOCA.

The maximum reactivity effect of a cracked reflector has been reported to be a 0.25$ step insertion associated with movement of a whole beryllium block toward the core.[3] Such a movement could be postulated by assuming complete cracking of the B, C, and D ligaments (without any

restraint to movement of the free piece of beryllium by the experiment plug in the outer B-hole or by the N-16 tubing). Since the maximum reactivity effect of an aged reflector is enveloped by a 0.25$ step insertion, the above events were analyzed in combination with a 0.25$ step. The combination for items (a), (c), and (d) are unlikely events and the combination for item (b) is extremely unlikely. The analysis for these combined events was completed for unperturbed two-PCP flow and unperturbed flux distributions at a lobe power of 48 MW.

The combination of a 0.25$ step with the IPT draining from a 1/2-in. experiment loop pipe break would result in 3.2 standard deviations to DNB and 4.1 standard deviations to flow instability as shown in Figures 20 and 21.[11] This event is clearly acceptable at the anticipated level although such a combination of two anticipated events should be considered at least unlikely.

The limiting unlikely reactivity ramp is enveloped by a step reactivity of 0.75$. The combination of a 0.25$ and 0.75$ step is enveloped by the 1.0$ step. This combination is extremely unlikely (combines an anticipated and an unlikely event), and is clearly acceptable since it is enveloped by the limiting extremely unlikely step.

A PCP coastdown event combined with a 0.25$ step reactivity insertion occurring just before reaching a low flow trip would result in 5.6 standard deviations to DNB and 7.5 standard deviations to flow instability as shown in Figures 22 and 23.[11] This unlikely event would clearly be acceptable since it would not even approach the margin limits for an anticipated event.

The unlikely event of a 0.25$ step insertion from beryllium aging effects in combination with a 3-in. LOCA would result in 4.3 standard deviations to DNB and 5.0 standard deviations to flow stability as shown in Figures 24 and 25.[11] Since this event would not challenge the limits of an anticipated fault, it would clearly be acceptable as an unlikely event.

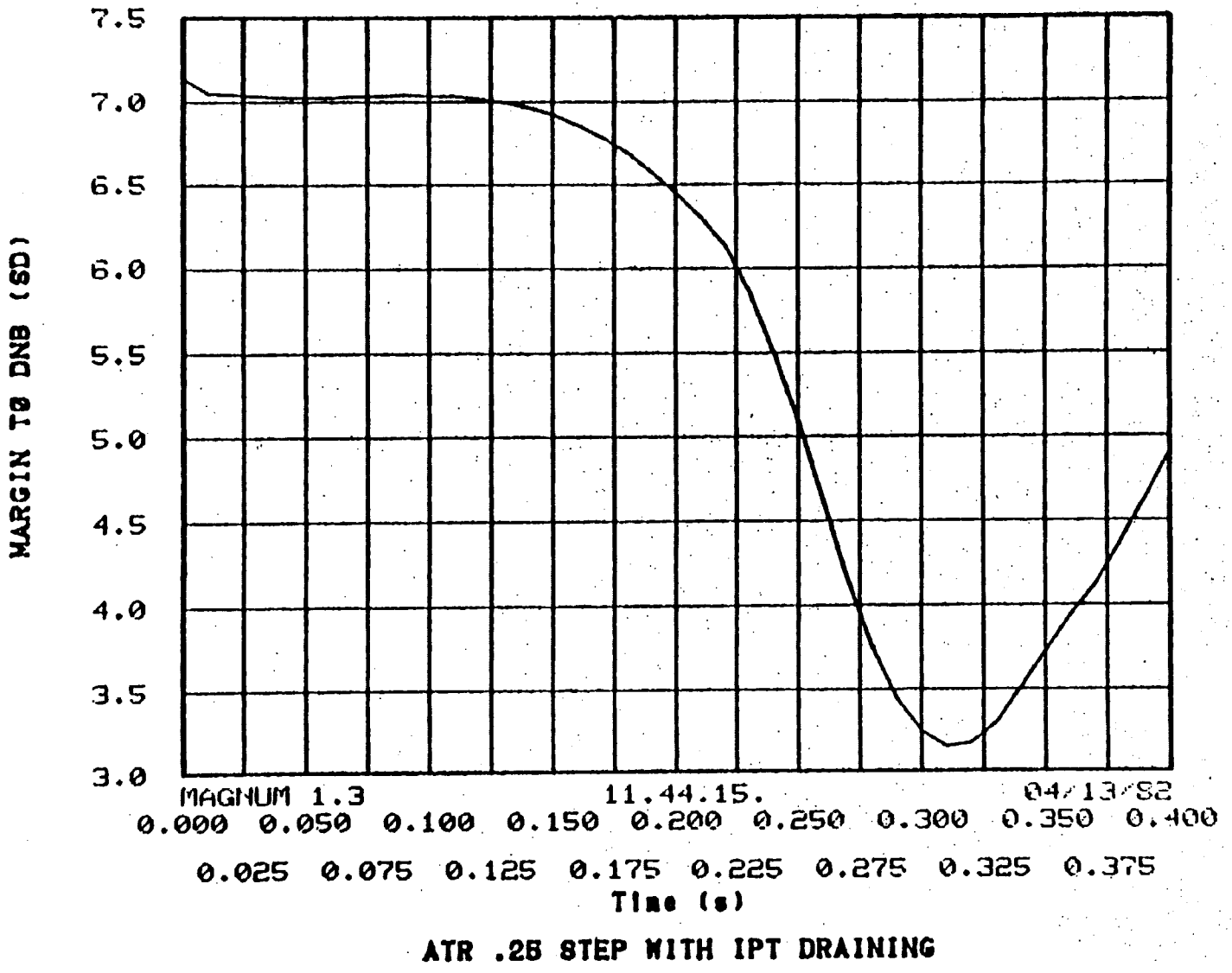

Figure 20. Number of Standard Deviations to DNB as a Function of Time for IPT Draining Due to a 1/2-in. Pipe Break Combined with a 0.25$ Step Reactivity Insertion.

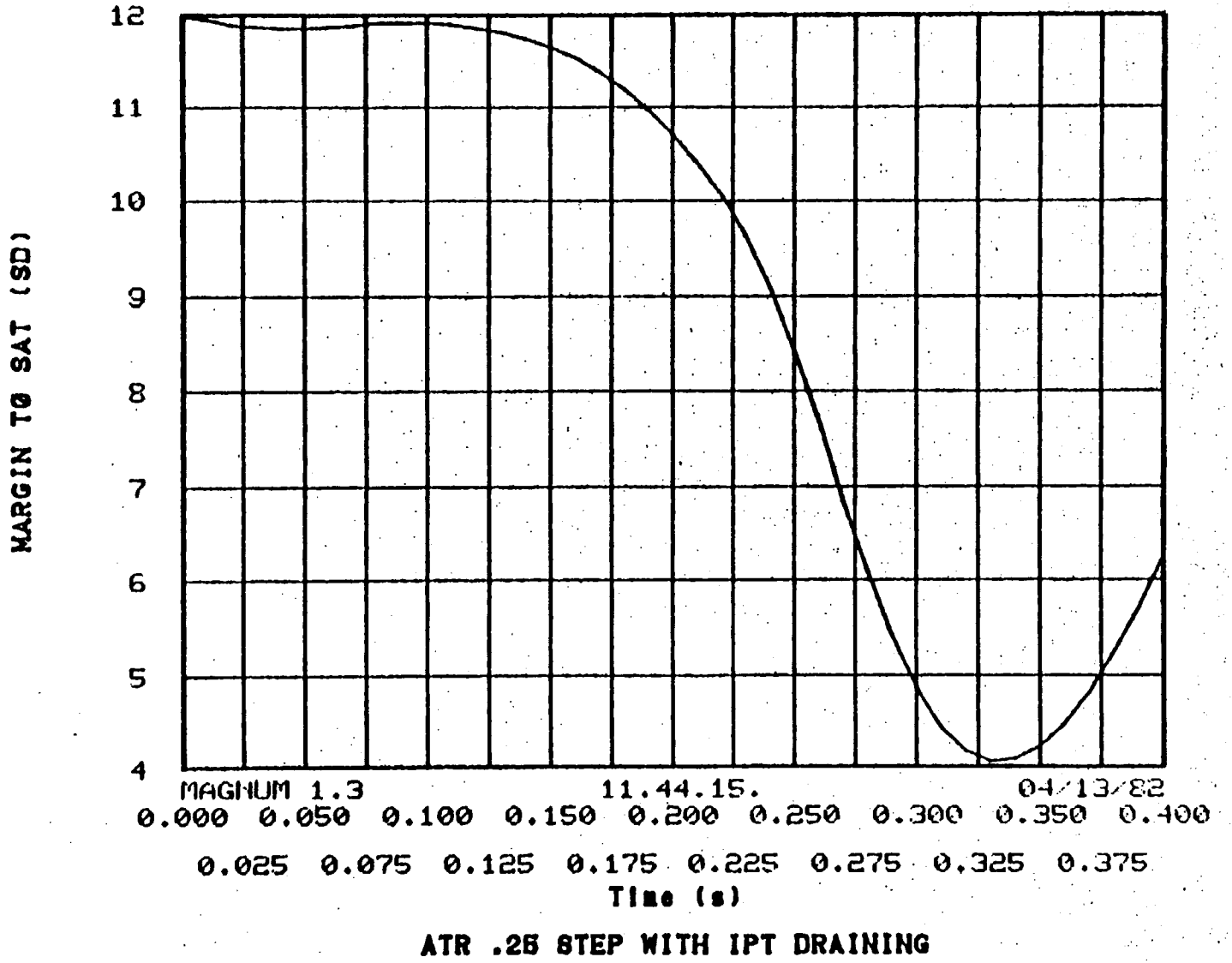

Figure 21. Number of Standard Deviations to Flow Instability as a Function of Time for IPT Draining Due to a 1/2-in. Pipe Break Combined with a 0.25$ Step Reactivity Insertion.

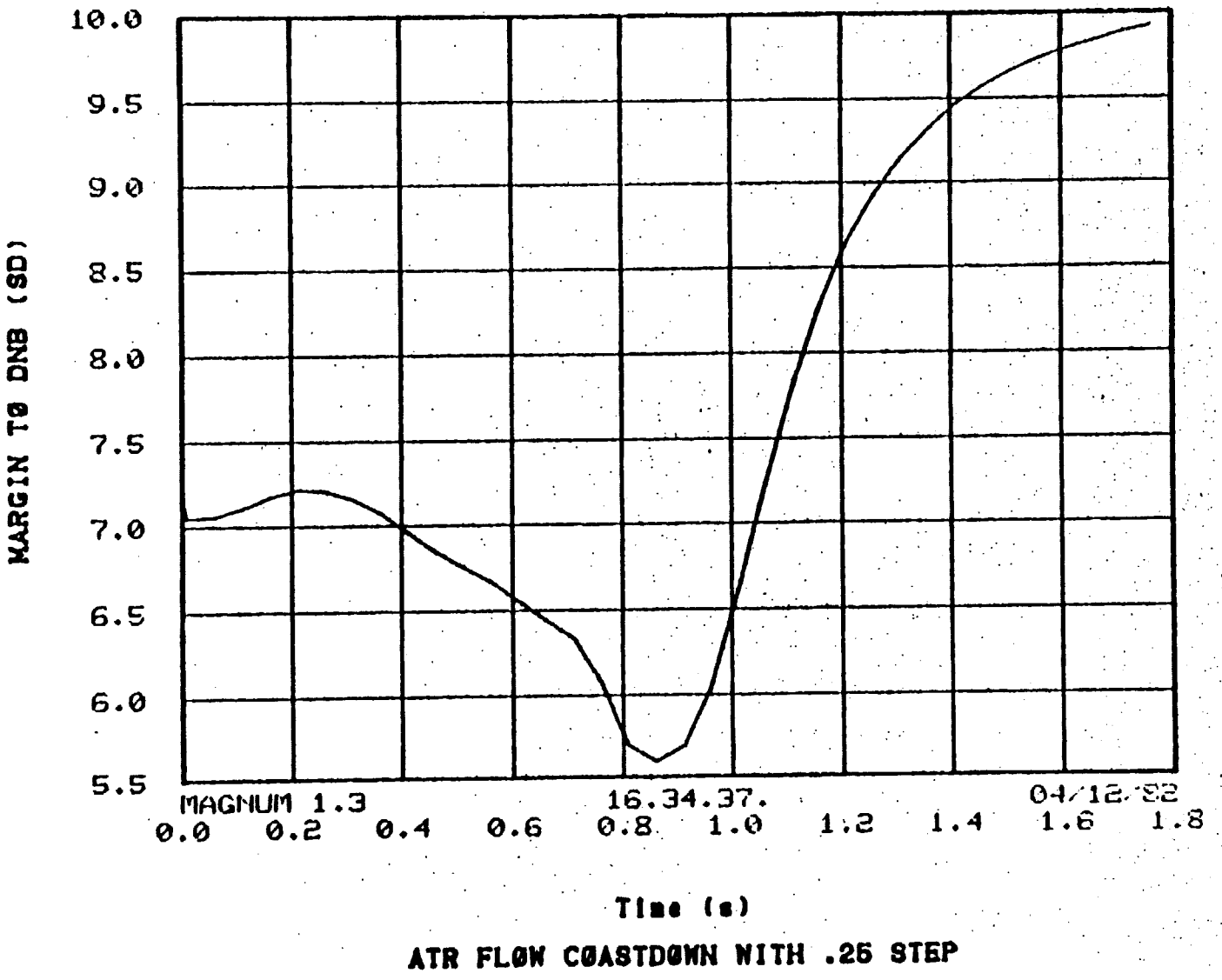

Figure 22. Number of Standard Deviations to DNB as a Function of Time for Flow Coastdown Combined with a 0.25$ Step Reactivity Insertion.

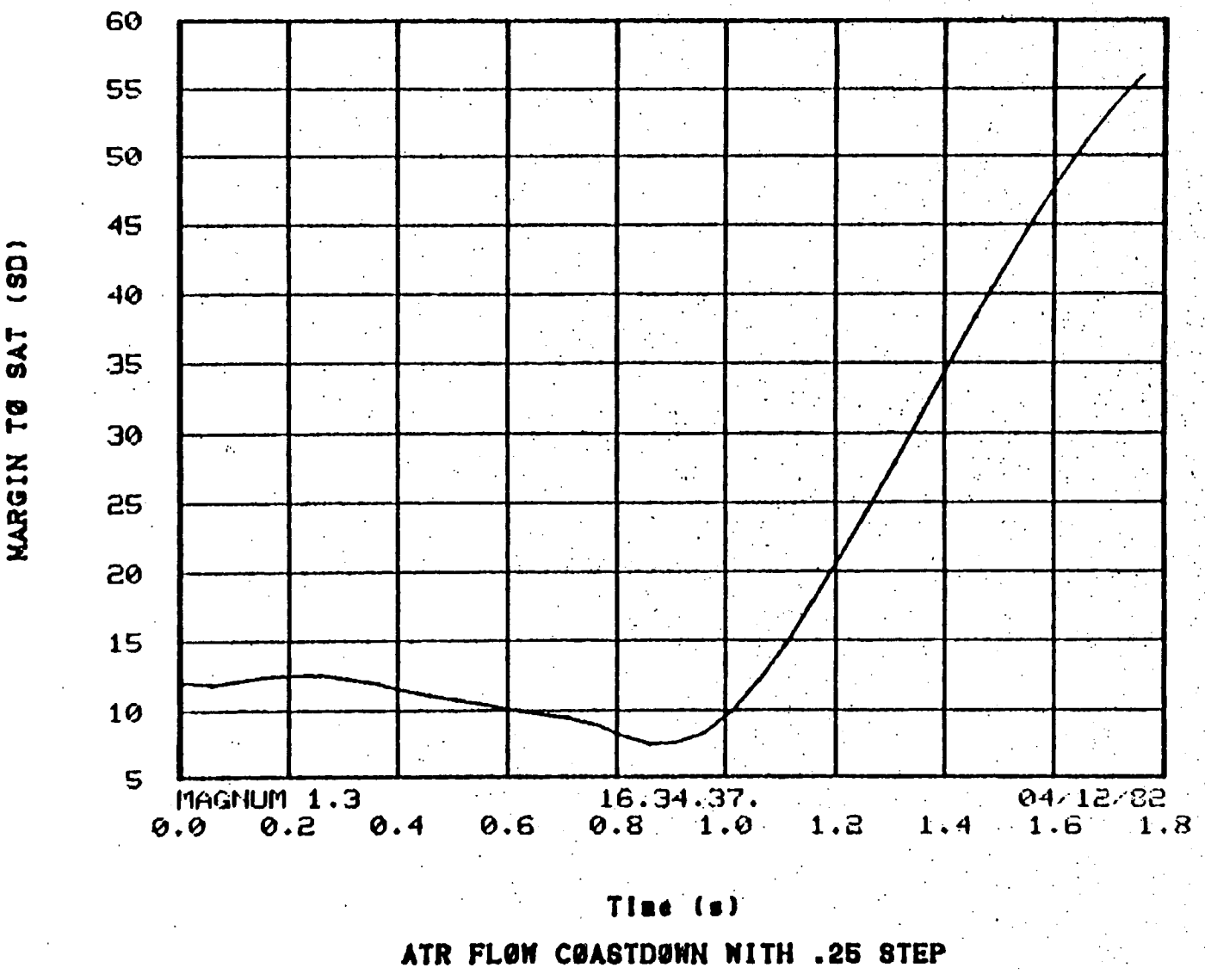

Figure 23. Number of Standard Deviations to Flow Instability as a Function of Time for Flow Coastdown Combined with a 0.25$ Step Reactivity Insertion.

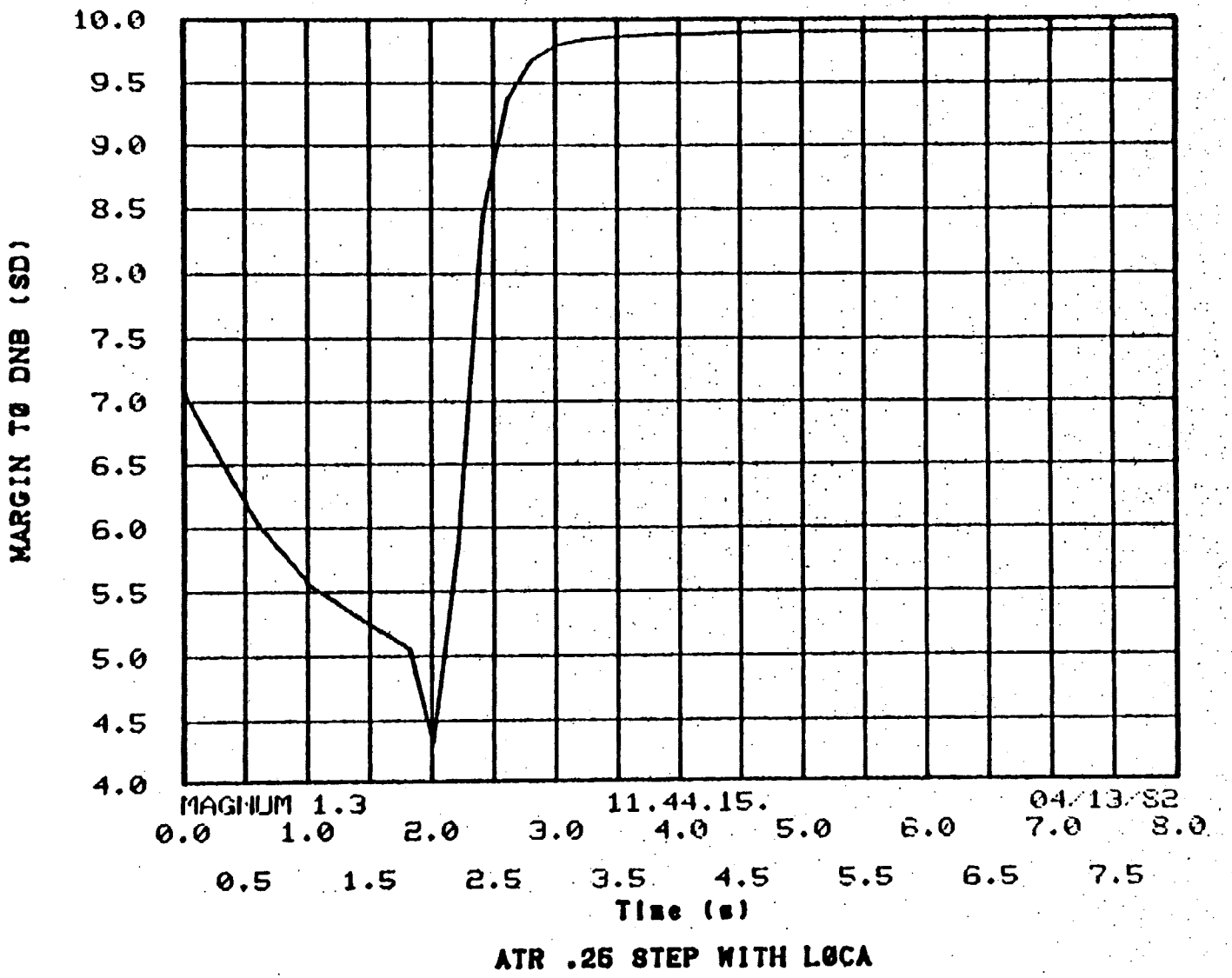

Figure 24. Number of Standard Deviations to DNB as a Function of Time for the 3-in. LOCA Combined with a 0.25$ Step Reactivity Insertion.

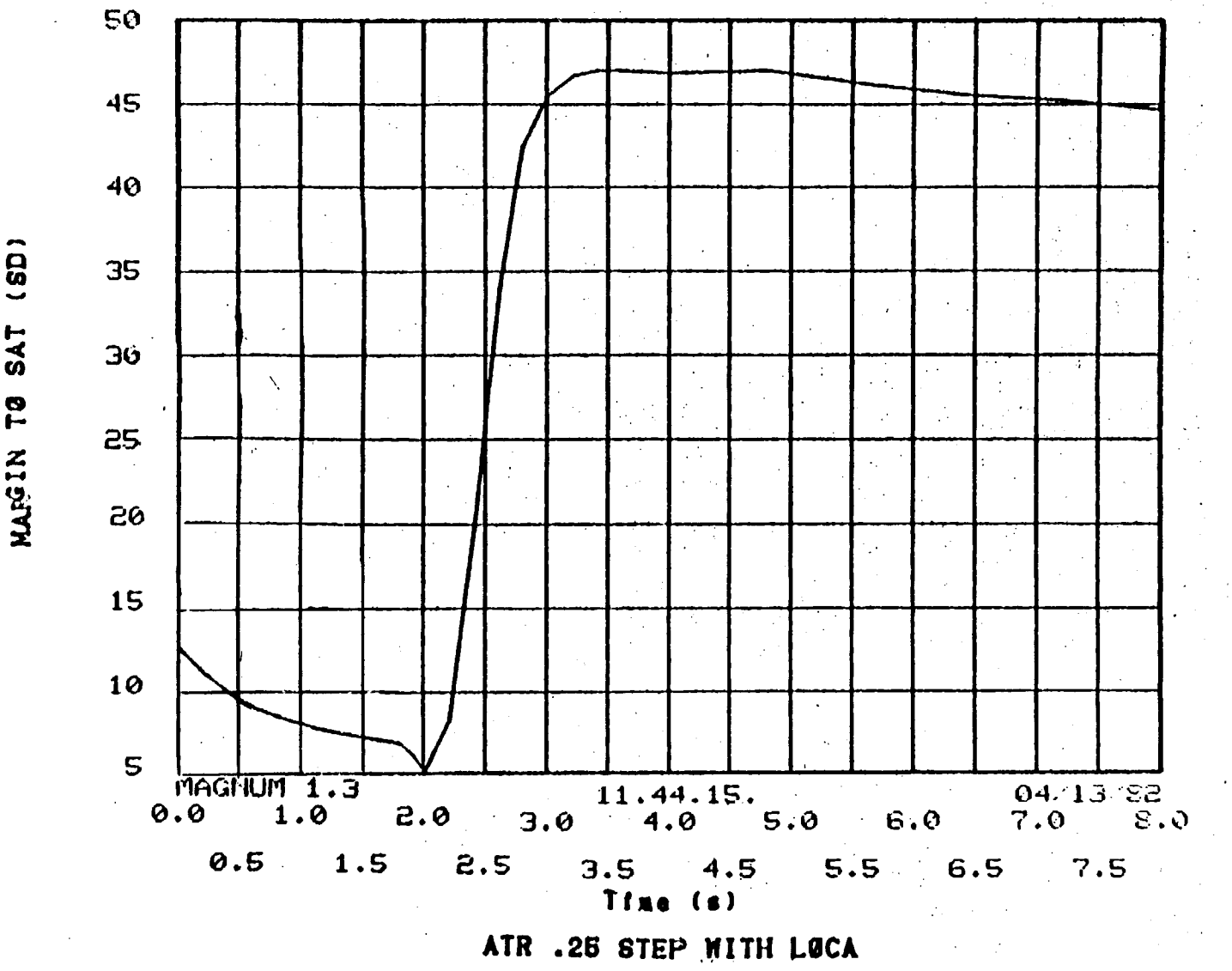

Figure 25. Number of Standard Deviations to Flow Instability as a Function of Time for the 3-in. LOCA Combined with a 0.25$ Step Reactivity Insertion.

6.12.0 Fuel Plate 19 Thermal Stress Analysis

One of the potential effects of reflector aging, as noted in Section 3, includes a 1-in. by 2-in. piece of beryllium lodged against plate 19 of a fuel element. The probability of this occurrence is not established quantitatively, but is estimated to be unlikely based on the following considerations. Operating history has shown that small holes of this size have occurred. Therefore, it is anticipated that such holes would exist, but the associated pieces have not been observed and particularly not lodged against a fuel plate. The cracking of the reflector to produce such pieces would be irregular with rough edges. As a result, any loose piece would tend to be captured. For the hole to develop, the piece would need to deteriorate and be removed in smaller pieces. It is rather improbable that the piece could move to the fuel plate and insulate it (beryllium has a specific heat approximately twice that of aluminum and a thermal conductivity only 19% less than that of aluminum). Additionally, the analysis conservatively assumes that the contact is such that the surface is adiabatic and also that slippage does not occur at the side plate (causing the fuel plate to be completely constrained). The analysis of a small piece of beryllium insulating the plate considers conditions with a probability judged to be equivalent with the unlikely category of occurrence.

A thermal stress analysis of plate 19 for this condition was completed for a lobe power of 48 MW, two-PCP flow, and an unperturbed power distribution.(7) The axial location of the blockage was chosen to be at the left edge of the fuel with the 1-in. dimension extending from about 5% to 31% of the plate width in order to locate the piece consistent with a probable break location and to choose an extreme temperature location. The MITAS code was used to generate the plate temperatures for the analysis. The stress was calculated for the blockage at two locations: 24.75 in. from the top of the plate and at 28.75 in. from the top. The flow perturbation for this blockage was calculated and found to be minor (6.3% decrease).[11]

The results of the thermal stress are shown in Figures 26 and 27 (7). The model divided the plate into 4-in. axial intervals and therefore the results did not indicate the axial distance over which the stress would exceed the yield strength. However, since the flow perturbation was minor (as indicated above) it can be assumed that the extent of axial yielding would not be much different than it would be in the azimuthal direction. The calculated stress about 0.6 in. azimuthally from the edge of the blockage is equal to the yield strength. Using this as an indicator, it is estimated that the area of the plate that would be exceeding the yield strength would be about 1.8 in. by 3.5 in. This is a relatively small portion of the plate, and therefore buckling of the plate would not be expected.

The construction of the element does not bond the fuel plate rigidly to the side plate (a swaging process is used), and therefore axial slippage would be expected under the loading conditions being considered. Experience with ETR fuel elements indicates that slippage will occur.[31] Also, measurements taken on a sampling of ATR elements has shown that the fuel plates have slipped in the side plates and have elongated, thereby reducing compression stresses, whether produced by irradiation growth or thermal expansion.[32] However, if slippage did not occur and the loading resulted in buckling, there could be some limited melting of the fuel plate. The amount involved would be small and well enveloped by the discussion in Section 5.2.4.

The minimum blister temperature (two standard deviations) for ATR fuel elements is 835 degrees F for fission densities of 2.7×10^{21} fiss/cm^3 (13). Therefore, blistering would not be expected for the 1-in. by 2-in. blockage since the calculated temperatures were significantly less than 835 degrees F. Larger blockages are enveloped by the analysis for complete blockage contained in Section 5.2.3. Since the flow is not perturbed by cracks (even large ones), the stress at 48 MW would be acceptable for areas adjacent to cracks.

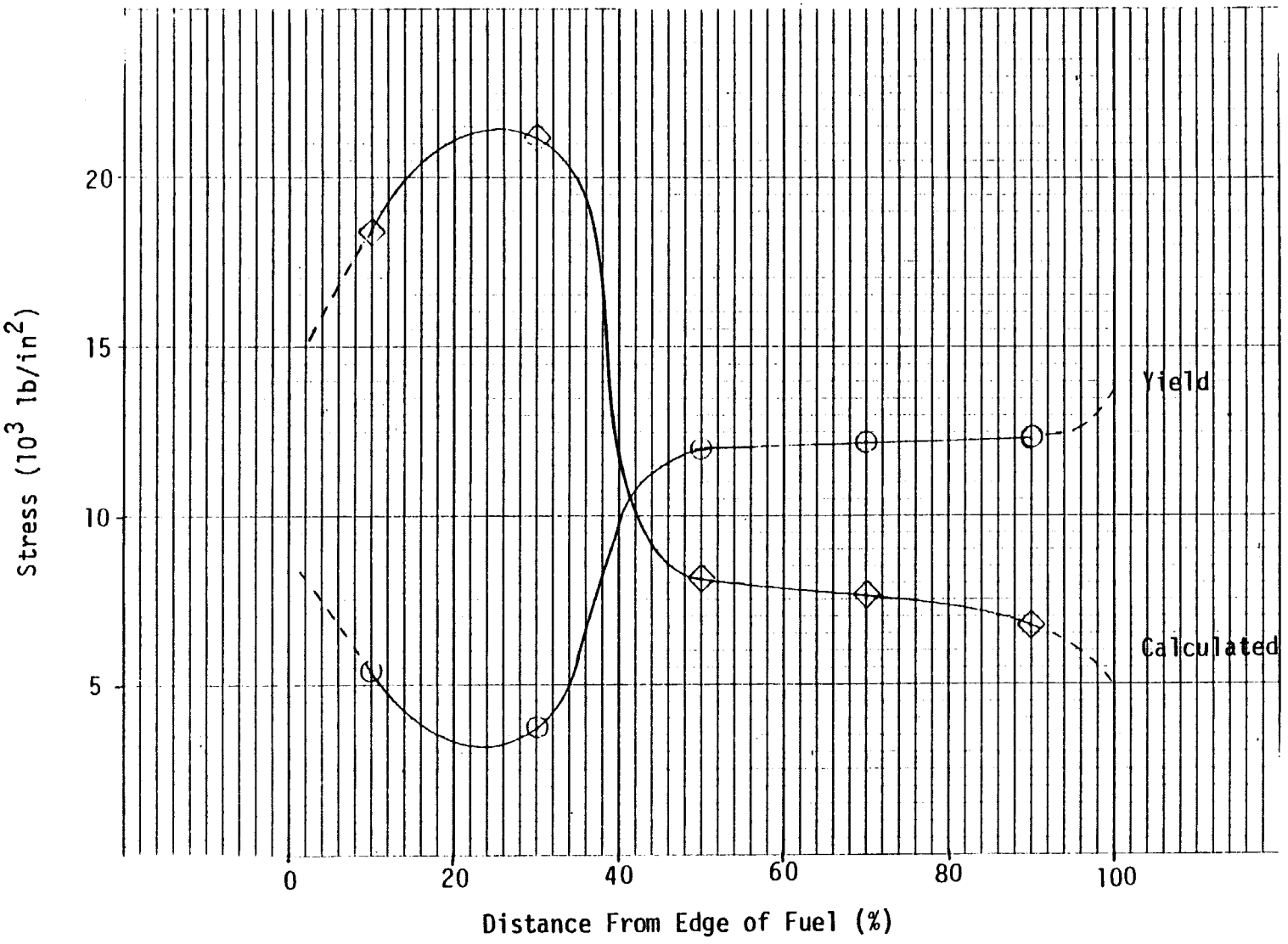

Figure 27. Calculated and yield stresses for plate 19 with a 1 in. X 2 in. piece of beryllium located at 28.75 in.

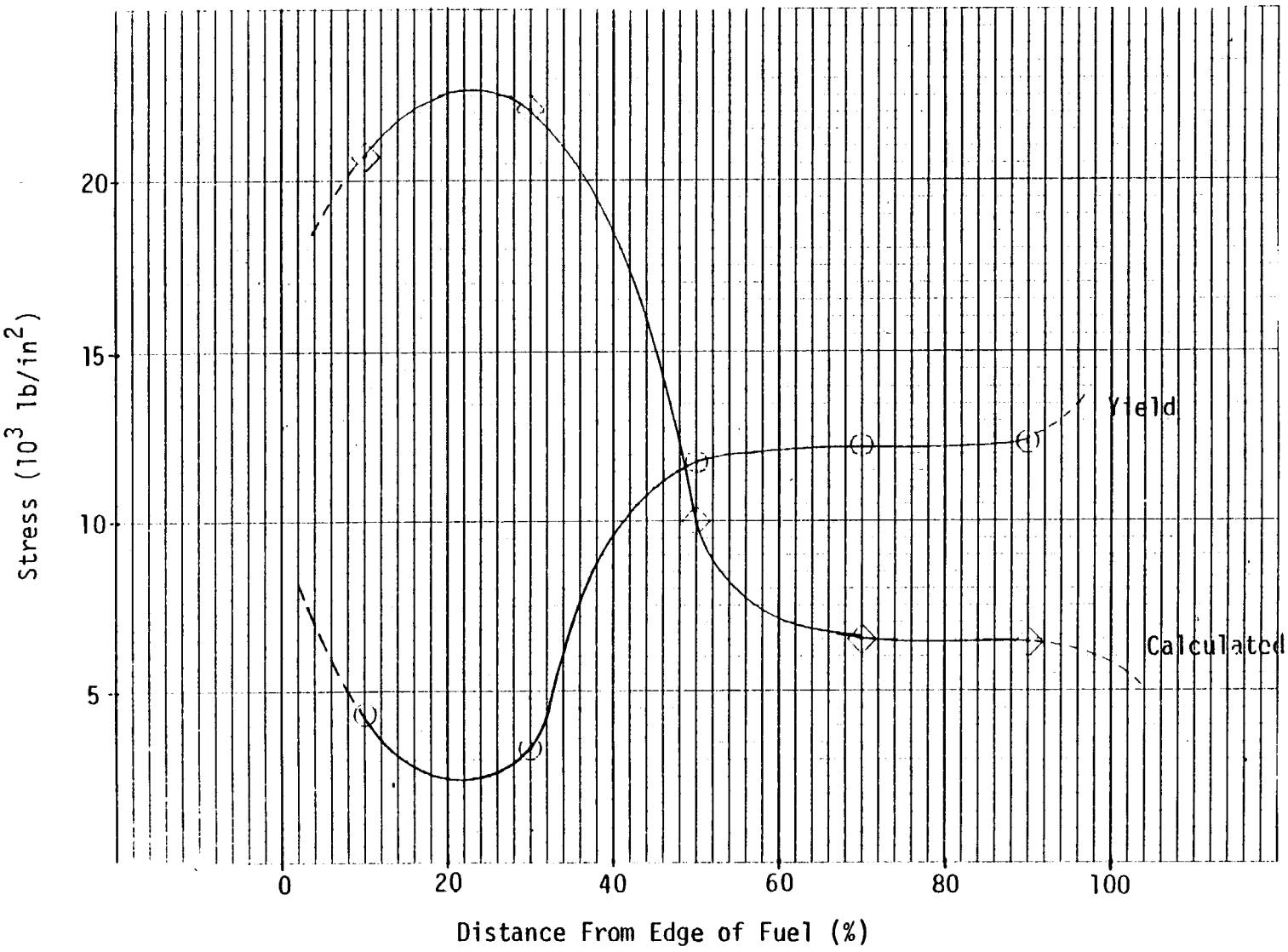

Figure 26. Calculated and yield stresses for plate 19 with a 1 in. X 2 in. piece of beryllium located at 24.75 in.

It can be concluded that the consequences of a 1-in. by 2-in. beryllium blockage would be acceptable for an unlikely event. Blistering would not occur and only a small area of plate 19 could be stressed above the yield strength (expected slippage at the side plate would tend to relieve the thermal stress). If slippage did not occur and fuel plate buckling resulted in some melting, the consequences would be enveloped by the complete blockage case.

7. SURVEILLANCE

7.1 Summary

Surveillance pertaining to reflector safety consists of two parts as follows:

Part I - Surveillance from the time a ligament reaches two standard deviations to failure until cracking of any beryllium block ligament is observed.

Part II - Surveillance from the time cracking of any beryllium block ligament is observed until the reflector is replaced.

Part I surveillance will detect premature failure of the beryllium. This surveillance is initiated in the reactor cycle that the stresses are predicted to reach two standard deviations to failure. This surveillance will continue until cracks are observed in any of the reflector block ligaments.

Part II surveillance is initiated at the time of observed cracking of any ligament and will remain in effect until reflector block replacement. This surveillance will track the condition of the beryllium faces and the respective fuel. Also included in this portion are safety and control rod tests to detect malfunctions which may be the result of beryllium displacement. The inspections and tests included are those used with reflectors I, II, and III (after cracks were discovered) with appropriate deletions and additions.

7.2 Part I Surveillance

7.2.1 Beryllium Surveillance and Inspection

(a) Whenever fuel is discharged from the core, ATR Operations personnel shall note and record any undue binding of the fuel and any possible beryllium block movement associated with fuel element movement.

This information will be recorded in the Reactor Top Log and in the DOP being used for the work.

(b) The full length of the reflector faces which can be seen from the fuel annulus shall be inspected. This includes the faces opposite fuel positions F-4, F-5, F-6, F-7, F-14, F-15, F-16, F-17, F-24, F-25, F-26, F-27, F-34, F-35, F-36, and F-37 and the joints located between beryllium blocks 1 and 2, 3 and 4, 5 and 6, and 7 and 8 (definition of block numbers is given in Section 2. of this report). Beryllium faces should be smooth (except for saw cuts), free of scratches, cracks, and spalling. Clearance at joints should be uniform at initial inspections. Any deviation from these conditions shall be recorded as well as any change observed from previous inspections. All unidentified indications shall also be recorded.

(c) Beryllium surveillance shall be repeated approximately every 40 days (in the refueling shutdown nearest to 40 days from the previous inspection).

(d) Inspection records shall be maintained by ATR Operations as part of the beryllium surveillance documentation. These records shall include condition maps or video recordings of the inspected beryllium faces showing all indications observed. The indications shall be clearly described using such terms as closing of saw cuts, spalling, scratches, marks, holes, missing pieces, or cracks. The locations of indications shall be identified at least by fuel position number and saw cut segment number (See Figures 1 and 5). Joints between the blocks shall also be observed for any changes in width. Records shall be kept showing observed changes from the previous inspection. Photographs or video recordings shall be taken of the beryllium faces, if required, after review of inspection records.

(e) Beryllium inspection data shall be reviewed before reactor startup when any changes from previous inspections are observed.

This review shall be completed by members of ATR Operations and PRP Technical Support, and if necessary, by Reactor Safety Analysis.

7.2.2 Fuel Element Surveillance and Inspection

(a) Fuel elements opposite the beryllium reflector face shall be inspected for abnormal markings or discoloration on the fuel plate and side plate surfaces that interface with the beryllium.

(b) Any evidence (observed during routine fuel oxide and channel measurements) of impact or buckling in the fuel plate adjacent to the beryllium shall be recorded.

(c) Fuel element inspection shall be performed every refueling shutdown.

(d) Fuel element inspection shall be recorded as no observed anomalies for the elements opposite the beryllium faces or as anomalies observed by mapping the indication on a fuel element sketch and recording fuel position and element number. Photographs shall be taken, if required, after review of the inspection records.

(e) Inspection records shall be reviewed by ATR Operations and PRP Technical Support (and, if necessary, Reactor Safety Analysis) personnel prior to reactor startup whenever anomalies or changes from previous inspections are observed.

7.2.3 Outer Shim Control Cylinder Drive Surveillance

(a) Outer shim control cylinder surveillance shall be initiated by replacing the standard 480 in.-lb. torque release clutches with clutches rated for 300 to 350 in.-lb. A torque release clutch with a lower torque rating will provide a means to detect early,

or low level, binding while at the same time minimizing the potential loads that a stuck OSCC could apply to a reflector ligament.

(b) Any OSCC binding problems incurred after installing the 300 to 350 in.-lb. torque release clutches shall be recorded and maintained by ATR Operations as part of the beryllium surveillance documentation.

(c) OSCC binding records shall be reviewed by ATR Operations, PRP Technical Support, and, if necessary, Reactor Safety Analysis prior to reactor startup.

7.2.4 Surveillance During Reactor Operation

Surveillance in addition to the standard mode of operation that requires immediate reactor power reduction for unexplained nuclear instrument abnormality is not required. The importance of directing extra attention to nuclear instrument readout shall be reemphasized to operating personnel by ATR Operations Management.

7.3 Part II Surveillance

Part II surveillance shall be initiated when beryllium block cracking is first observed and shall continue until the reflector is replaced.

7.3.1 Baseline Inspection Data

(a) Photographs or video recordings shall be taken of the ligament cracks and any other observable damage on all beryllium faces which can be seen from the fuel annulus. The beryllium faces involved are those opposite fuel positions F-3, F-4, F-5, F-6, F-7, F-8, F-13, F-14, F-15, F-16, F-17, F-18, F-23, F-24, F-25, F-26, F-27, F-28, F-33, F-34, F-35, F-36, F-37 and F-38. Saw cut segment numbers (See Figure 5) shall be used to designate axial locations of defects.

(b) The joints between beryllium blocks 1 and 2, 3 and 4, 5 and 6, and 7 and 8 shall be inspected and recorded.

(c) Tracings or copies of tracings of all nuclear system recorders from a recent period of normal operation shall be collected.

(d) The baseline information collected above shall be retained by ATR Operations as part of the ATR beryllium surveillance documentation.

(e) Review of the Part II baseline inspection data shall be completed by ATR Operations and PRP Technical Support (and, if necessary, Reactor Safety Analysis) personnel before reactor operation. Additional inspection data shall be obtained as required.

7.3.2 Beryllium Surveillance and Inspection

(a) Initial tank inspections shall be made before in-tank operations begin in each shutdown. These inspections shall include visual inspection and identification of any foreign item found in the tank. Identification of any foreign beryllium item shall require efforts to determine the origin of the beryllium.

(b) Whenever fuel is discharged from the core, ATR Operations personnel shall note and record any undue binding of the fuel and any possible beryllium block movement associated with fuel element movement. This information will be recorded in the Reactor Top Log and in the DOP being used for the work.

(c) The full length of all beryllium faces which can be seen from the fuel annulus shall be inspected using the mirror tool and camera or video equipment. The beryllium faces involved are those opposite fuel positions F-3, F-4, F-5, F-6, F-7, F-8, F-13, F-14, F-15, F-16, F-17, F-18, F-23, F-24, F-25, F-26, F-27, F-28, F-33, F-34, F-35, F-36, F-37 and F-38. Saw cut segment numbers (see Figure 5) shall be used to designate axial locations of defects.

The joints between beryllium blocks 1 and 2, 3 and 4, 5 and 6, and 7 and 8 shall be inspected and any changes shall be recorded.

(d) All changes and additions to the baseline or previous inspection shall be recorded. These records shall include condition maps or video recordings of the inspected beryllium faces showing all indications observed. The indications shall be clearly described using such terms as closing of saw cuts, spalling, scratches, marks, holes, missing pieces, or cracks. The locations of indications shall be identified at least by fuel position number and saw cut segment number (See Figures 1 and 5). Joints between the blocks shall also be observed for any changes in width. Records shall be kept showing observed changes from the previous inspection. Photographs or video recordings shall be taken of the beryllium faces, if required, after review of inspection records.

(e) The beryllium inspection shall be performed every scheduled refueling shutdown occurring after ten days of operation since the previous inspection. The beryllium inspection shall also be performed in every unscheduled shutdown in which fuel is handled unless it can be conclusively argued that the reflector condition was not adversely affected by either the shutdown or the initiating event. The beryllium inspection shall be performed after an unscheduled shutdown if the reflector initiated the shutdown.

(f) Inspection records shall be maintained by ATR Operations as part of the beryllium surveillance documentation.

(g) Beryllium inspection data shall be reviewed before reactor startup when any changes from previous inspections are observed. This review shall be completed by members of the ATR Operations and PRP Technical Support (and, if necessary, Reactor Safety Analysis) groups.

7.3.3 Fuel Element Surveillance and Inspection

(a) Fuel elements opposite the beryllium reflector face shall be inspected for abnormal markings or discoloration on the fuel plate and side plate surfaces that interface with the beryllium.

(b) Any evidence (observed during routine fuel oxide and channel measurements) of impact or buckling in the fuel plate adjacent to the beryllium shall be recorded.

(c) Fuel element inspection shall be performed concurrent with the beryllium inspection. Fuel element inspection shall be recorded as no observed anomalies for the elements opposite the beryllium faces or as anomalies observed by mapping the indication on a fuel element sketch and recording fuel position and element number. Photographs shall be taken, if required, after review of the inspection records.

(e) Inspection records shall be reviewed by ATR Operations and PRP Technical Support (and, if necessary, Reactor Safety Analysis) personnel prior to reactor startup whenever anomalies or changes from previous inspections are observed.

7.3.4 Safety Rod and Control Rod Surveillance

(a) Each safety rod shall be checked prior to each reactor startup for activation of the lower limit switch and seat light. Reactor startup shall not be permitted unless each safety rod has a functioning lower limit switch and seat light.

(b) Each safety rod (except strapped safety rods) must demonstrate rod release and drop times within existing specified limits before reactor startup will be permitted.

(c) Safety rods may be strapped during reactor operation as allowed by the Core Safety Assurance Package (CSAP).

(d) Prior to reactor startup and with normal primary coolant flow, each outer shim control cylinder (OSCC) shall be cycled 60 degrees to test for binding. If OSCC binding is detected, reactor startup shall not be permitted until the cause is identified and reviewed, and the reflector configuration is determined to be stable. If OSCC binding is detected, it may be possible to alleviate the binding by replacing the 300 to 350 in.-lb. torque release clutch with a clutch rated for 480 in.-lb. Installation of a torque release clutch rated for 480 in.-lb. will require a case-by-case review by ATR Operations, PRP Technical Support, and Reactor Safety Analysis and written approval from the ATR Operations Manager.

(e) Outer shim control cylinders shall be rotated 1 degree ($\pm$ 0.5 degree) once per shift during normal operation to detect binding of the cylinders or drive mechanisms. Binding can be detected four ways:

- Torque clutch release
- No shim movement as indicated by the shim drive position indicators on the reactor console
- No observed reactivity changes on the reactor instrumentation
- No observed movement on the active regulating rod.

Any observed binding of an OSCC pair that is not resolved within 12 hours requires reactor shutdown. Binding of more than a single pair of OSCCs requires immediate reactor shutdown.

(f) The neck shim rods should not be affected by beryllium block movement or by beryllium chips in the circulating coolant. No additional checks beyond the routine operability checks performed at each reactor startup are required.

7.3.5 Reactor Instrumentation Surveillance

Operating crews shall be instructed by ATR Operations Management to direct extra attention to nuclear instrument readout. The standard mode of operation requiring an immediate reduction in reactor power for any unexplained nuclear instrument abnormality shall be reemphasized. Specific instructions concerning instrumentation include the following:

(a) The operating regulating rod has a normal oscillation of two inches during reactor operation. An unexplained three-inch deviation from normal regulating rod movement that is abrupt and sustained will require a review of the nuclear instrumentation. If the regulating rod position change is caused by an unexplained reactivity change, the reactor power shall be reduced until the cause is determined.

(b) Unexplained changes in nuclear power instrumentation shall require action as outlined in the ATR Reactor Console and Tank Operating and Maintenance Manual (7.1.13.1.3.4). For example, if there is an unexplained power variation of 3% or greater for longer than 15 minutes, the reactor must be shut down. It may be noted that a 0.015$ reactivity oscillation has been predicted to result in a 1% power variation in the ATR.[33]

7.3.6 Primary Coolant System Sampling

(a) During reactor operation, samples of the primary coolant are counted twice per shift for gross activity and once per day (except Saturday and Sunday) to identify specific nuclides (including I-131, I-133, and I-135) and their concentrations. This frequency shall be maintained with the importance of early fission break detection stressed to the shift managers and the health physicists. Established guidelines concerning fission product activity and continued reactor operation shall be maintained.

(b) Atomic absorption spectroanalysis is performed on a primary coolant sample once per week during reactor operation. This analysis includes identification of beryllium concentration in the primary system and shall be continued at the present or increased frequency.

8. QUALITY ASSURANCE

An appropriate quality assurance program is maintained at the Test Reactor Area. The quality assurance system in effect meets the requirements of ANSI/ASME NQA-1 (Quality Assurance Program Requirements for Nuclear Facilities) and 10 CFR-50 Appendix B.[34] Reflector components are manufactured to quality assurance Level I (or A) requirements.

9. SUMMARY AND CONCLUSIONS

The analyses for operation of a reflector with stress levels less than two standard deviations to A or E ligament failure provide a basis for continued operation until the B ligament stress reaches two standard deviations to failure. The specific limitations are as follows:

1. Reactor lobe powers are limited to 48 MW or less whenever an adjacent ligament has reached a stress level of two standard deviations to failure and the fuel element next to the ligament includes fuel in plate 19.

2. Operation in excess of 48 MW lobe power requires an element without fuel in plate 19 (Type YA) adjacent to any ligament that has reached a stress level of two standard deviations to failure.

3. Power peaking factors in fuel elements adjacent to a ligament that has reached stress levels of two standard deviations to failure must be reduced by 2.5% (assumes 1-in. piece of beryllium missing and plate 17 is limiting). The allowable values are as follows:

   (a) For 3-PCP operation, the peak-to-average fission power density ratio limit is 3.25 (or alternately, equivalent peak power of 727.0 MW) for 7NB elements, 3.41 (or alternately, 764.0 MW) for 7F elements, and 3.27 (or alternately, 731.6 MW) for YA elements.

   (b) For 2-PCP operation, the peak-to-average fission power density ratio limit is 2.97 (or alternately, equivalent peak power of 595.0 MW) for 7NB elements, 3.12 (or alternately, 624.0 MW) for 7F elements, and 2.99 (or alternately, 597.7 MW) for YA elements.

4. The power level setpoints and quadrant differential temperature setpoints must be consistent with a full power (N_F) of 48 MW for any lobe adjacent to a ligament that has reached stress levels of two standard deviations to failure and has a fuel element with fuel in plate 19 adjacent to it.

5. The surveillance described in Section 7 of this report must be followed unless ATR Operations, PRP Technical Support, and Reactor Safety Analysis concurrence is obtained, in writing, for any deviations.

It may be noted that this reflector safety analysis considered both three-PCP operation and two-PCP operation. It may also be observed that there is not a requirement for inner B-hole pins. However, outer B-hole experiment plugs or baskets should always be installed during reactor operation. The pressure distribution in the reflector and fuel annulus is well balanced and therefore lateral forces on any loose piece of beryllium is minimized. Furthermore, the outer B-hole experiment plugs or baskets will provide a pinning action to help restrain movement of any free nosepiece of beryllium toward the fuel or toward an outer flux trap baffle.

This reflector safety analysis does not specifically identify a need for reduced ratings for the OSCC torque release clutches. However, it is recognized that using 300 to 350 in.-lb. torque release clutches will reduce the potential for damage to reactor components if interference should occur between an OSCC and the adjacent reflector block. Also, torque release clutches with lower torque ratings will provide a means to detect early, or low level, binding. The 300 to 350 in.-lb. torque release clutches should be used for the shim pair in any reflector block that has reached stress levels of two standard deviations to failure. If persistent binding problems develop with a lower rated clutch, a specific review of the situation will be necessary before allowing replacement with a standard 480 in.-lb. clutch. Binding could occur as a result of deterioration in the drive system or from interference (directly or indirectly via beryllium chips in the annulus) between an outer shim

cylinder and the surrounding reflector block. It appears that continued operation with a degraded drive system would not lead to damage of other components. Clutch replacement can be accomplished in a few hours.

This analysis did not specifically address low power (depressurized) testing with an aged reflector (A or E ligament stresses within two standard deviations to failure). The ATR is infrequently operated in the low power testing mode and such operation is not planned with the reflector in an aged condition. Low power testing is conducted for relatively short periods of time under low flow/low temperature conditions which would not be expected to cause any reflector configuration changes during operation. It is not apparent that there are any reflector safety issues associated with such short and infrequent operating modes. A specific analysis will be needed, however, to formally address the potential consequences if such operation becomes necessary.

10. REFERENCES

1. G. E. Darwin and J. H. Buddery, Beryllium, Butterworths Scientific Publications, 1960.

2. D. R. DeBoisblanc and S. Cohen, Safety Analysis Report, Advanced Test Reactor, IDO-17021, April 1965.

3. S. A. Atkinson et al, ATR Reflector II Lifetime Extension Analysis, TR-559, November 27, 1974.

4. ATR Operating Envelope Extension, TR-386, May 10, 1973.

5. G. W. Holman ltr. to D. R. Mousseau, Chronology of ATR Beryllium Crack Growth, GWH-11-84, August 6, 1984.

6. E. H. Smith ltr. (with attachment) to R. J. Nertney, Revision to Attachment of Letter EHS-76-72 (Dated March 31, 1972) - Request to Operate the ATR at Power Levels Not to Exceed 240 MW with Cracks in the ATR Beryllium Reflector, EHS-77-72, April 7, 1972.

7. J. L. Durney, Analysis for Extension of Reflector III Lifetime, PR-T-82-010, Rev. 4, January 23, 1986.

8. M. L. Griebenow, Isothermal Hydraulic Measurements in Fuel Elements in ATR Core, IN-1158, September 1968.

9. B. L. Harris, Stress Analysis of ATR Fuel Plates Loaded by Broken Beryllium (Reflector III), TRTSB-ATR-027, December 29, 1981.

10. Specification for Advanced Test Reactor Mark VII Zone Loaded Fuel Elements, IN-F-9-ATR, Rev. 9, November 14, 1986.

11. R. G. Ambrosek, et al, Safety Analysis for Assumed Reflector III Failure Modes, PR-T-83-009, May 1982.

12. M. L. Griebenow, Ed., ATR Extended Burnup Program, ANCR-1015, November 1971.

13. J. M. Beeston, et al, "Development and Irradiation Performance of Uranium Aluminide Fuels in Test Reactors," Nuclear Technology, June 1980.

14. P. D. Bayless, et al, Analysis of a Hypothetical Large LOCA in the ATR, PG-T-89-004, February 1989.

15. Design Manual for the Advanced Test Reactor, Vol. 18, Reactor Reflector Support Tank, March 1964.

16. R. G. Ambrosek, et al, ATR Two-Pump Safety Analyses, PR-T-80-012, February 4, 1980.

17. Users Guide to MEL-40, A Piping Flexibility Analysis Program, Documented by L. M. Kaldor, et al, Program Written by J. A. Olson and R. V. Cramer.

18. J. L. Durney and T. A. Tomberlin, ATR Shutdown Assurance, PG-T-88-012, July 1988.

19. H. H. Nielsen, Final Reflector III/Mark II Outer Shim Cylinder Thermal-Hydraulic Design Report, TR-448, November 8, 1973.

20. H. H. Nielsen ltr. to C. E. Friedrich, ATR Mark II/Reflector III Flow 3 Analysis, Ni-1-77, February 24, 1977.

21. F. O. Haroldsen ltr. to D. R. Mousseau, ATR Reflector III/Mark II Outer Shim Cylinder Thermal-Hydraulic Analysis, Haro-1-75, January 14, 1975.

22. Investigation of the Effects of Metal Chips Lodged in the Articulating Joints of the Advanced Test Reactor Safety Rods, ANC Internal Report CI-1137, January 1969.

23. R. W. Thomas, Design Report for the Modifications to the ATR Beryllium Reflector (Mark IV) and Associated Core Internals, PG-T-84-025, November 1985.

24. A. C. Smith ltr. to M. L. Griebenow, Flux and Reactivity Changes Due to ATR Beryllium Reflector Damage - GWA 15300-320-009, ACS-5-72, July 17, 1972.

25. A. C. Smith ltr. to J. L. Durney, Power Peaking Increase Due to a 1-inch Water Gap Between Core Position F-14 and the Be Reflector, ACS-4-81, January 30, 1981.

26. A. C. Smith ltr. to J. L. Durney, Power Peaking Increase Due to a 1/8-inch Water Gap in the Be Reflector Adjacent to Core Position F-27, ACS-17-75, May 22, 1975.

27. Advanced Test Reactor Plant Protective System and Technical Specifications Design Basis Report, Issue 023, November 7, 1988.

28. J. L. Durney ltr. to M. S. Vargo, Analysis of ATR Reflector Failure, JLD-6-81, June 23, 1981.

29. A. C. Smith ltr. to J. L. Durney, Plate 17 Power Peaking Increase Due to a 1-inch Water Gap Between Core Position F-14 and the Be Reflector, ACS-6-81, February 18, 1981.

30. J. L. Durney ltr. to File, Change In Power Peaking for ATR Shim Unbalance, JLD-11-82, April 22, 1982.

31. M. L. Griebenow, Ed., ATR Extended Burnup Program, ANCR-1015, November 1971.

32. C. M. Rice ltr. to W. L. Ginkel (AEC-ID), ATR Fuel Failure Mode Analysis, CMR-243-71, April 12, 1971.

33. J. A. Close ltr. to N. C. Kaufman, Surveillance of Reactivity Variations in ATR, Close-3-74, July 26, 1974.

34. TRA Quality Program Plan, Issue 019, April 21, 1989.