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ABB Atom Inc.

Report

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Title

Oskarshamn 2 Fuel Failures

Abstract

During the period from December, 1987, through mid-February, 1988, stepwise increases in the Oskarshamn 2 offgas and primary coolant water activities indicated that fuel failures had occurred. Subsequent evaluations performed since that time have established that four fuel rods in separate assemblies failed because these fuel rods were operated for some period of time under dryout conditions. The resulting overheating of the cladding led to local breaching of the rods which resulted in the leakage of radioactive material from the fuel rods to the primary coolant. This document contains an overview describing the event, the investigations to date establishing the failure mechanism, and the causes of the failures.

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The dryout condition occurred for these rods because they were operated at an actual power which was substantially higher than that reported by the Plant Process Computer which monitors the core thermal limits. The four failed fuel rods were located in the corner of fresh SVEA 64 type assemblies adjacent to the control rod gaps containing nonboiling water. These fuel rods were operating at higher powers than expected primarily because the widths of the control rod water gaps were substantially greater for the assemblies containing these four fuel rods than assumed in the Plant Process Computer calculations. The increased neutron moderation associated with the increased control rod gap widths led to the increase in fuel rod powers.

There is no direct relationship between the Oskarshamn 2 fuel failures and the SVEA-64 design. Traditional BWR fuel designs without the watercross are affected in the same way as the watercross design with respect to this type of failure.

The control rod gap widths were underpredicted in the input to the Plant Process Computer primarily for two reasons. The major cause was failure to account for the accelerated excessive channel bow which had occurred for assemblies adjacent to those containing the fuel rods which failed. The assemblies containing the rods which failed were located adjacent to 8X8 assemblies with reused fuel channels which had experienced exceptionally high radiation exposures. These channels had experienced a relatively large channel bow away from the control rod gap which had caused the width of the control rod gap to increase. Following the normal practice in the United States and Europe, the effects of channel bow were not included in the preparation of the input to the Plant Process Computer. A second contributing factor involved an oversimplification of core lattice geometry in the input to the Plant Process Computer which led to a further underestimate of power in the fuel rods adjacent to the control rod gaps in the absence of channel bow.

The failure of these four fuel rods in the Oskarshamn 2 reactor was no more serious from a plant safety standpoint than failures which have occurred in commercial reactors in the past due to such mechanisms as Pellet-Clad interaction and debris fretting. The Oskarshamn 2 failures did not represent a massive failure mechanism, compromise plant safety, or represent a threat to the environment in any way.

The circumstances which led to the failures are unusual and would be expected to occur with a very low frequency. However, observation of this type of failure in an operating commercial reactor has not previously been reported and must be thoroughly understood to avoid future occurrences. Furthermore, the failure mechanism is not specific to a given fuel type.

The failures were not related to the SVEA watercross design. Misrepresentation of the interassembly gaps provides the potential for causing fuel damage independent of the specific assembly design. Therefore, it is considered important to disseminate this information to BWR utilities and appropriate authorities regardless of the fuel design they are currently operating with.

ABB Atom and the OKG utility performed sufficient poolside and hot cell examinations to definitely establish the failure mechanism as dryout during and subsequent to the 1988 shutdown. Furthermore, ABB Atom has performed analytical studies which demonstrate that the observed channel bow and the oversimplification in the preparation of Plant Process Computer input in the absence of channel bow are sufficient to account for the dryout of the failed fuel rods using ABB Atom methods. The ability of the ABB Atom methods to predict the dryout of these fuel rods when applied to the actual assembly geometry in the OKG core when the failures occurred also demonstrates that these methods when used consistently are effective in predicting dryout even in extreme situations.

The overview of the Oskarshamn 2 failures provided in this document represents a summary of information available up to June 1989. ABB Atom and OKG are performing additional examinations of channels and fuel rods still in the core in conjunction with the 1989 refueling of Oskarshamn 2. Analysis efforts are also under way to describe the phenomena associated with the failures in still greater detail.

CHRONOLOGY OF THE EVENT

In August of 1987, a reload of ABB Atom's SVEA-64 assemblies was installed in the Oskarshamn 2 core replacing discharged ABB Atom 8X8-1 fuel. "SVEA-64" is the designation of ABB Atom's 64-rod watercross fuel bundle design. Following this refueling the Oskarshamn 2 core contained 270 ABB Atom 8X8-1 bundles and 174 SVEA-64 bundles. Following start-up in August of 1987, the plant was operated until a brief shutdown from December 28 to December 30, 1987. During the 6 week period immediately following this outage, stepwise increases in activity were measured in the off gas and primary coolant water. These increases occurred in four discrete steps occurring on December 31, January 6, February 3, and February 8, 1988. Operation of the core continued at full power without further indication of failures until shutdown for annual refueling in August, 1988.

DETERMINATION OF THE FAILURE MECHANISM

Investigation and Failure Mechanism

Investigations were performed during and subsequent to the August, 1988, refueling shutdown which established the nature and the cause of the fuel

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failures. The scope of these investigations and their scheduling were established to minimize any impact on the plant refueling and startup schedule.

Sipping was performed to determine the location of the leaking fuel. Leaking fuel was identified in assemblies 14311, 13544, 13543, and 14312 shown in Figure 1. While initial sipping results indicated the possibility of a failure in assembly 13535, subsequent examination of the fuel showed that all rods in that assembly were intact. The minibundles in each of the SVEA-64 assemblies shown in Figure 1 were visually inspected, and the four failed fuel rods were identified as shown in Figure 1.

Assemblies 10223 and 10232 were removed from the core, and the channel bow was measured. A summary of the existing channel bow measurements and exposures as of 9/88 for the six 8X8 assemblies adjacent to the assemblies with failed fuel is provided in Table 1. All of the 8X8 assemblies adjacent to the assemblies with failed corner rods were highly depleted with highly exposed channels in their second bundle lifetime. Since the SVEA-64 assemblies containing the failed fuel rods had been inserted into the reactor as fresh assemblies in August of 1987, their average burnup at the time the rods failed was about 4 MWD/MtU. Additional channel bow measurements on the remaining four 8X8 assemblies identified in Figure 1 are planned for the 1989 shutdown in August, 1989.

Gamma-scan measurements in nine of the fuel rods in assembly 13536 were performed during the shutdown. Power distributions in these rods at end-of-cycle were inferred from these measurements.

In addition to the poolside examinations, four fuel rods were removed from the core for post irradiation examination in a hotcell. These rods are identified in Table 2 which contains a summary of the hotcell results. The convention for numbering the rods is shown in Figure 2. The hot cell examinations provided information on fission gas release, clad microstructure, cladding collapse, and burnup of the rod.

The major results of the visual, pool side, and hot cell examinations for the assemblies shown in Figure 1 can be summarized as follows:

1. The failures occurred in corner rods adjacent to the control rod gap in fresh SVEA-64 assemblies.

2. The assemblies adjacent to those with the failed fuel were relatively highly depleted 8X8 assemblies with reused channels with exceptionally high exposures. Specific measurements on the 8X8 channels adjacent to two of the assemblies containing failed fuel rods confirmed excessive channel bowing (7.3 to 7.4 mm, i.e. 0.29 inches) away from the control rod gap.

3. The failures occurred just below the top spacer on the side of the rod facing the control rod gap. Based on extensive dryout test data, this is the axial location where dryout is most likely to occur.

4. The surfaces of the failed rods facing the control rod gaps were heavily oxided indicating operation at elevated temperatures.

5. Microstructure measurements in the vicinity of a failure revealed a zircaloy phase change indicating temperatures in excess of 860 deg. C.

6. Profilometry measurements showed cladding collapse onto the pellets just below the top spacer in a failed rod as well as in one intact corner rod adjacent to the control rod gap in one SVEA-64 assembly. This collapse implies that the rods operated at very high temperatures in these locations.

7. Cesium migration measurements in an intact SVEA-64 rod adjacent to the control rod gap indicates operation at elevated temperatures.

8. Gamma-scan and fission product release measurements demonstrated that the power generated in the SVEA-64 rods adjacent to the control rod gaps in the assemblies shown in Figure 1 was substantially higher than predicted by the calculations which formed the basis for the Plant Process Computer input. The gamma-scan measurements also demonstrated that the powers in the rods adjacent to the narrow gaps were lower than expected. These measurements demonstrated that the radial power distribution across the assembly was severely skewed toward the control rod gaps for these SVEA-64 assemblies.

9. Examination of the interior of the channels adjacent to the failed rod locations showed no indication of rod damage due to contact with the channel.

10. Secondary failures due to internal (secondary) hydriding were observed at the bottoms of the failed rods.

11. It is estimated that the failed rods were actually in dryout for a total period of time of between two and seven days. It is possible that the dryout periods were not contiguous but occurred in a dryout-rewet cyclic manner.

These examinations conclusively showed that the rods failed because of operation with excessively high cladding temperatures. The gamma-scan and fission gas release measurements demonstrate that there was a severe power

Therefore, the CPR values predicted by the Plant Process Computer were higher than the results predicted by ABB Atom design methods in Table 3. The input to the Plant Process Computer was prepared by OKG using analytical methods which are somewhat different than the ABB Atom design methods. OKG and ABB Atom are currently working together to resolve this discrepancy.

EFFECT OF ASSEMBLY DESIGN ON OSKARSHAMN 2 TYPE FUEL FAILURES

There is no direct relationship between the SVEA watercross design and the Oskarshamn 2 fuel failure mechanism. The Oskarshamn 2 fuel failures occurred because significantly higher powers occurred in relatively undepleted fuel rods than were predicted by the Plant Process Computer used to monitor the plant thermal limits. It occurred in SVEA-64 fuel simply because the SVEA bundles were relatively undepleted and occupied locations in which the control rod gap widths were substantially larger than assumed in the Plant Process Computer input. Fuel failures did not occur in the 8X8 assemblies adjacent to these control rod gaps because the assemblies were relatively highly depleted as shown in Table 1.

The SVEA design is no more susceptible to this type of failure than other designs which do not contain a watercross. In fact, there are some features of the design which can be expected to reduce its susceptibility to this type of failure.

For example, the watercross increases the neutron moderation in the center of the assembly. The improved neutron moderation in the center of the assembly leads to lower fuel rod relative powers than the traditional designs without a watercross. The Oskarshamn 2 fuel failures demonstrate that considerable care must be taken in the selection of the fuel rod enrichments adjacent to the water gaps to assure that their relative power, or local power peaking, does not become excessive. Since relative fuel rod powers are generally somewhat lower in the watercross fuel than other designs with the same number of fuel rods, the constraints required on the selection of enrichments for fuel rods adjacent to the water gaps to accommodate the effects of the water gaps should be more easily accommodated in the watercross design.

The second factor which tends to reduce the susceptibility of the SVEA fuel to the Oskarshamn 2 type failure is the extensive experimental and analytical work which has led to the capability to reliably predict dryout. The division of the assembly into four subchannels reduces the problem to a more tractable two-step process. The first step involves the use of test data to predict the occurrence of dryout in a subchannel. The limited variability in the subchannels relative to the entire assembly makes it more practical to obtain sufficiently extensive test data to assure that the entire operating range is covered. The second step involves the application of the CPR correlation derived from the data to the entire assembly in a manner which ensures that dryout is conservatively predicted considering subbundle power mismatch.

The dryout behavior of the SVEA-64 design has been established by extensive testing of sixteen-rod subbundles at the ABB Atom FRIGG Loop in Sweden and the Westinghouse Canada facility in Hamilton, Ontario. Data were obtained over the entire range of core flow, system pressure, inlet subcooling and local power expected in steady-state or transient operation. The data base for the sixteen-rod subassembly in the watercross design is based on over thirty-five 16-rod local power distributions with corner rod relative powers up to a value of 1.7. The relatively small size of the sub-bundles allowed the thorough understanding of the CPR behavior as a function of fuel rod position and relative power. This extensive CPR data base ensures that the ABB Atom methods can predict the occurrence of dryout in the Oskarshamn 2 case when the appropriate geometry is used in the analysis.

The CPR correlation for the sixteen-rod test data is applied to the SVEA-64 assembly in a manner which conservatively predicts dryout. Dryout is predicted at a slightly lower power than the test data would actually predict. Flow communication holes are provided in the SVEA assembly to equalize pressure between the sub-channels. A conservative estimate of dryout power is accomplished by assuming in the CPR calculations that there is no flow communication between the minibundles. Extensive subchannel analyses using the computer codes COBRA-IV and GOBLIN have confirmed that the assumption of no flow communication between the sub-channels leads to a conservative prediction of dryout power. In addition, ABB Atom has performed full-scale comparative 64-rod and 16-rod watercross assembly tests which confirm the results of the subchannel analyses.

The Oskarshamn 2 failures demonstrate that the uncertainties in the power level of the rods adjacent to the control rod gaps can be somewhat greater than for the interior rods. It is expected that the susceptibility for the Oskarshamn 2 type failures will be greater as the dryout sensitivity to changes in corner rod relative peaking increases. ABB Atom has performed a comparison of SVEA test data with full-scale 8X8 assembly test data using a similar spacer grid, which demonstrates, that the change in critical bundle power as the corner rod local peaking factor is changed is about equal for the SVEA design and the 8X8 design. Thus it has been shown that the SVEA design is not more sensitive to changes in corner rod peaking than fuels without a watercross.

CONCLUSION

The following conclusions regarding the fuel failures at Oskarshamn 2 can be drawn:

1. The Oskarshamn 2 fuel failures occurred due to overheating of the cladding associated with operation under dryout conditions.

2. The dryout of the failed rods occurred because of unanticipated channel bow in assemblies adjacent to those containing the failed fuel rods and an oversimplified treatment of cross sections and fuel rod power distributions in the input to the Plant Process Computer.

3. The fuel failures were not related to any design features specific to the SVEA-64 assemblies. The failures occurred in the SVEA-64 assemblies because the assemblies were fresh and were located adjacent to water gaps which were not accurately represented in the Plant Process Computer used to monitor thermal limits. Traditional BWR fuel designs without the watercross are affected in the same way as the watercross design with respect to this type of failure.

4. Analysis of the Oskarshamn 2 cycle in which the fuel failures occurred with ABB Atom analytical methods confirm that the ABB Atom methods are reliable for predicting this type of failure.

5. The overall conclusion is that current methods can predict this phenomenon without any added uncertainties, provided that the Plant Process Computer input is based on realistic best-estimate type data.

6. The ongoing investigations might provide additional refinements. For example, the analytical methods might predict results in even closer agreement with the actual operating conditions. These refinements are not expected to change the conclusions presented above.

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DETERMINATION OF THE FAILURE MECHANISM

Investigation and Failure Mechanism

Investigations were performed during and subsequent to the August, 1988, refueling shutdown which established the nature and the cause of the fuel failures. The scope of these investigations and their scheduling were established to minimize any impact on the plant refueling and startup schedule.

Sipping was performed to determine the location of the leaking fuel. Leaking fuel was identified in assemblies 14311, 13544, 13543, and 14312 shown in Figure 1. While initial sipping results indicated the possibility of a failure in assembly 13535, subsequent examination of the fuel showed that all rods in that assembly were intact. The minibundles in each of the SVEA-64 assemblies shown in Figure 1 were visually inspected, and the four failed fuel rods were identified as shown in Figure 1.

Assemblies 10223 and 10232 were removed from the core, and the channel bow was measured. A summary of the existing channel bow measurements and exposures as of 9/88 for the six 8X8 assemblies adjacent to the assemblies with failed fuel is provided in Table 1. All of the 8X8 assemblies adjacent to the assemblies with failed corner rods were highly depleted with highly exposed channels in their second bundle lifetime. Since the SVEA-64 assemblies containing the failed fuel rods had been inserted into the reactor as fresh assemblies in August of 1987, their average burnup at the time the rods failed was about 4 MWD/MtU. Additional channel bow measurements on the remaining four 8X8 assemblies identified in Figure 1 are planned for the 1989 shutdown in August, 1989.

Gamma-scan measurements in nine of the fuel rods in assembly 13536 were performed during the shutdown. Power distributions in these rods at end-of-cycle were inferred from these measurements.

In addition to the poolside examinations, four fuel rods were removed from the core for post irradiation examination in a hotcell. These rods are identified in Table 2 which contains a summary of the hotcell results. The convention for numbering the rods is shown in Figure 2. The hot cell examinations provided information on fission gas release, clad microstructure, cladding collapse, and burnup of the rod.

The major results of the visual, pool side, and hot cell examinations for the assemblies shown in Figure 1 can be summarized as follows:

1. The failures occurred in corner rods adjacent to the control rod gap in fresh SVEA-64 assemblies.

2. The assemblies adjacent to those with the failed fuel were relatively highly depleted 8X8 assemblies with reused channels with exceptionally high exposures. Specific measurements on the 8X8 channels adjacent to two of the assemblies containing failed fuel rods confirmed excessive channel bowing (7.3 to 7.4 mm, i.e. 0.29 inches) away from the control rod gap.

3. The failures occurred just below the top spacer on the side of the rod facing the control rod gap. Based on extensive dryout test data, this is the axial location where dryout is most likely to occur.

4. The surfaces of the failed rods facing the control rod gaps were heavily oxided indicating operation at elevated temperatures.

5. Microstructure measurements in the vicinity of a failure revealed a zircaloy phase change indicating temperatures in excess of 860 deg. C.

6. Profilometry measurements showed cladding collapse onto the pellets just below the top spacer in a failed rod as well as in one intact corner rod adjacent to the control rod gap in one SVEA-64 assembly. This collapse implies that the rods operated at very high temperatures in these locations.

7. Cesium migration measurements in an intact SVEA-64 rod adjacent to the control rod gap indicates operation at elevated temperatures.

8. Gamma-scan and fission product release measurements demonstrated that the power generated in the SVEA-64 rods adjacent to the control rod gaps in the assemblies shown in Figure 1 was substantially higher than predicted by the calculations which formed the basis for the Plant Process Computer input. The gamma-scan measurements also demonstrated that the powers in the rods adjacent to the narrow gaps were lower than expected. These measurements demonstrated that the radial power distribution across the assembly was severely skewed toward the control rod gaps for these SVEA-64 assemblies.

9. Examination of the interior of the channels adjacent to the failed rod locations showed no indication of rod damage due to contact with the channel.

10. Secondary failures due to internal (secondary) hydriding were observed at the bottoms of the failed rods.

11. It is estimated that the failed rods were actually in dryout for a total period of time of between two and seven days. It is possible that the dryout periods were not contiguous but occurred in a dryout-rewet cyclic manner.

These examinations conclusively showed that the rods failed because of operation with excessively high cladding temperatures. The gamma-scan and fission gas release measurements demonstrate that there was a severe power

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tilt to the rods nearest the control rod gap. The failures are located precisely where dryout would be expected to occur with the observed power tilt. Therefore, evaluation of the visual, poolside, and hot cell data leads to the conclusion that the four rods failed because of excessively high temperatures associated with operation under dryout conditions. A severe power tilt toward the control rod gap side of the assembly was also established by these measurements.

The examinations also showed that rods near the control rod gap in assemblies 13535 and 13536 which did not fail operated at significantly higher powers than predicted by the core management calculations input to the Plant Process Computer. It is likely that rod H8 in assembly 13536 actually experienced dryout but did not operate in a dryout condition for sufficient time for the clad to fail.

Alternate Failure Mechanisms Considered

While the examination results themselves are considered sufficient to establish dryout as the failure mechanism, alternate potential causes for the failures were evaluated and eliminated as possible causes.

The possibility of a manufacturing or core loading error which could lead to a fuel rod with an inappropriate enrichment for the position it occupies in the core was thoroughly investigated. The identification numbers for the minibundles and assemblies containing the failed rods were checked to confirm that the bundles occupied the correct locations. The identification numbers for fuel rods H8, G8, and E7 of assembly 13536, and H8 of assembly 13544, were compared with manufacturing records to confirm that the correct rods occupied these locations. The orientation of the minibundles and assemblies were confirmed to be correct. The gamma-scan and hot cell information was used to confirm the enrichments of the rods for which the measurements were made. A review of the pedigree of the failed rods revealed that they were not manufactured in a common lot. For example, assemblies 13543 and 13544 were delivered to the plant in 1986, and assemblies 14311 and 14312 were delivered in 1987. Based on these evaluations, manufacturing or positioning errors were eliminated as a potential cause for the failures.

Elevated activity levels were initially detected in the off gas and primary coolant water at about 90% power during the start-up on December 30, 1987. Since this plant evolution involved the movement of control rods at relatively high power, Pellet Clad Interaction (PCI) was immediately suspected as a possible failure mechanism. However, the subsequent investigation of plant operating records coupled with the relatively low burnup of the fuel demonstrated that the chances of a PCI failure were extremely remote.

The plant was taken critical at a relatively low pressure during the December 30, 1987, startup, and the return to full power was relatively rapid. Therefore, the possibility of the combined effects of relatively high local power peaks associated with the movement of control rods and low system pressure leading to dryout at low reactor power was hypothesized. A subsequent review of plant operating data, however, eliminated this as a likely mechanism for the failures.

The plant operating data were thoroughly reviewed for the possibility of a power transient in the start-up or operating range leading to fuel failures. It was concluded from this review that there were no plant transients which could have caused the failures.

The elimination of other potential causes for the fuel failures provides further confirmation of the conclusion that the failure mechanism was dryout.

CONDITIONS CAUSING DRYOUT AT OSKARSHAMN 2

Dryout of the rods in Oskarshamn 2 occurred because the Plant Process Computer predicted a value of the Minimum Critical Power Ratio (MCPR) of the core which was higher than that which actually existed. The Plant Process Computer predicted values of MCPR in excess of 1.35 for the entire period between plant startup in September, 1987, and the final increase in coolant activity in mid-February, 1988. However, examination of the failures showed that the failed rods operated under dryout conditions, or at an assembly MCPR of about 1.0 to 1.06, for a period of about two to seven days.

The Plant Process Computer would mispredict the core MCPR if either the input to the computer models were incorrect or the computer models themselves were inadequate. ABB Atom has established that the primary cause for the misprediction of MCPR by the Plant Process Computer is that the input did not accurately describe the assembly configuration in the core as it actually existed. As a matter of fact, the ABB Atom methods predict that the dryout would have a high probability of occurrence in the failed fuel locations when applied to the geometric situation as it actually occurred in the reactor.

Inappropriate Assumptions Regarding the Geometric Core Configuration in the Plant Process Computer Input

The failed fuel rods were operating at higher powers than expected primarily because the widths of the control rod water gaps were substantially greater for the assemblies containing these four fuel rods than assumed in the Plant Process Computer calculations. The increased neutron moderation associated with the increased control rod gap widths led to the increase in fuel rod powers. The control rod gap widths were underpredicted in the Plant Process Computer input for two reasons.

The assemblies containing the rods which failed were located adjacent to assemblies with reused channels which had experienced exceptionally high radiation exposures. These channels had experienced excessive channel bow away from the control rod gap which had caused the width of the control rod gap to increase. Figure 3 shows this effect for the central core "supercell" where two of the four failed rods were located. The term "supercell" refers to a four assembly array surrounding the control rod water gaps. ABB Atom has performed extensive channel bow measurements in the Nordic plants, and the tendency of channels to bow away from the control rod gaps in the interior of D lattice cores has been well established. The rate of change of the channel bow as a function of burnup at these higher burnups was, however, somewhat more rapid than at lower burnups. Channel bow is a manifestation of differences in channel growth of opposite sides and is proportional to channel growth. Figure 4 shows ABB Atom channel growth data. As shown in Figure 4, the measured channel growth shows an accelerated trend above 50 MWd/KgU relative to the roughly monotonic behavior at lower exposures. This behavior is also reflected in the channel bow data for assembly 10232 shown in Table 1. Following the normal practice in the United States and Europe, the effects of channel bow were not included in the initial core management calculations.

A second contributing cause to the underprediction of the power in the fuel rods adjacent to the control rod gaps in these locations were the approximations used in the generation of the input to the Plant Process Computer even if there were no irradiation induced channel bow. When the fresh SVEA-64 fuel assemblies were installed in the reactor in August, 1987, the inlet pieces of the assemblies were modified to cause the assemblies to be shifted toward the center of the control rod gap. This modification is routinely performed in Nordic reactors when SVEA fuel is introduced. The purpose of this modification is to achieve a more uniform lattice when the entire core has been converted to SVEA fuel. The more uniform lattice achieved with this improvement provides improved reactivity characteristics and reduces the fast flux differential across the channel, which should reduce channel bow. This improvement is a routine practice for D-lattice Nordic plants. In the Oskarshamn 2 case, the input to the Plant Process Computer was calculated assuming the control rod gaps adjacent to all of the SVEA Assemblies were reduced as they would be in a full SVEA core. Therefore, the control rod gap widths for all supercells containing 8X8 fuel assemblies were assumed to be narrower for the SVEA assemblies in the Plant Process Computer predictions than they actually were in the reactor even in the absence of channel bow. This assumption caused the power in the fuel rods adjacent to the control rod gap to be underestimated even in the absence of channel bow.

In order to quantify the impact of the channel bow and the oversimplified input to the Plant Process Computer, ABB Atom performed four sets of calculations using the PHOENIX/POLCA system of codes:

<u>Reference Case</u> - The reference case was performed with the same assumptions utilized in the input calculations for the Plant Process Computer. A single SVEA cross section set with the reduced-width control rod gaps was applied to all SVEA assemblies. POLCA three-dimensional core simulator calculations for the cycle in which the fuel failed were performed which predicted Minimum Critical Power Ratio's (MCPR's) between 1.30 and 1.40.

<u>Wide Gap SVEA</u> - The CPR calculations were repeated with cross section and local power distribution data calculated for a SVEA assembly with wider control rod gaps than the reference case which approximated the correct amount of water in the control rod gaps in the absence of channel bow.

<u>Quadruple With No Channel Bow</u> - The CPR calculations were repeated for cross section and local power distribution data calculated with the geometry shown in Figure 5. This case provides the correct representation for the supercells containing the failed fuel in the absence of channel bow.

<u>Quadruple With Channel Bow</u> - The CPR calculations were repeated for cross section and local power distribution data for the geometry shown in Figure 6. The actual channel bow magnitudes vary axially and will be somewhat different for each of the 8X8 assembles. The geometry shown in Figure 6 represents a typical case intended to reflect the effect of the observed channel bows on CPR. The 0.5 mm (0.02 inch) channel bow applied to the SVEA bundle reflects the practice of installing assemblies with any manufacturing-related channel bow oriented away from the control rod.

The results of this sensitivity study are shown in Table 3. The predicted MCPR of 1.00 means that the ABB Atom analytical methods if used consistently predict the occurrence of the dryout observed in Oskarshamn 2 when the geometric situation as it existed in the reactor is appropriately described. The results in Table 3 confirm the adequacy of the ABB Atom methods for predicting dryout.

The Plant Process Computer predicted minimum MCPR's of at least 1.35 for the cycle in which the fuel failures occurred. As shown by the Reference Case in Table 3, a repeat of these calculations using the same assumptions and ABB Atom methods predicts a range of MCPR values between 1.30 and 1.40 with a minimum MCPR of 1.30 for the cycle.

Therefore, the CPR values predicted by the Plant Process Computer were higher than the results predicted by ABB Atom design methods in Table 3. The input to the Plant Process Computer was prepared by OKG using analytical methods which are somewhat different than the ABB Atom design methods. OKG and ABB Atom are currently working together to resolve this discrepancy.

EFFECT OF ASSEMBLY DESIGN ON OSKARSHAMN 2 TYPE FUEL FAILURES

There is no direct relationship between the SVEA watercross design and the Oskarshamn 2 fuel failure mechanism. The Oskarshamn 2 fuel failures occurred because significantly higher powers occurred in relatively undepleted fuel rods than were predicted by the Plant Process Computer used to monitor the plant thermal limits. It occurred in SVEA-64 fuel simply because the SVEA bundles were relatively undepleted and occupied locations in which the control rod gap widths were substantially larger than assumed in the Plant Process Computer input. Fuel failures did not occur in the PX8 assemblies adjacent to these control rod gaps because the assemblies were relatively highly depleted as shown in Table 1.

The SVEA design is no more susceptible to this type of failure than other designs which do not contain a watercross. In fact, there are some features of the design which can be expected to reduce its susceptibility to this type of failure.

For example, the watercross increases the neutron moderation in the center of the assembly. The improved neutron moderation in the center of the assembly leads to lower fuel rod relative powers than the traditional designs without a watercross. The Oskarshamn 2 fuel failures demonstrate that considerable care must be taken in the selection of the fuel rod enrichments adjacent to the water gaps to assure that their relative power, or local power peaking, does not become excessive. Since relative fuel rod powers are generally somewhat lower in the watercross fuel than other designs with the same number of fuel rods, the constraints required on the selection of enrichments for fuel rods adjacent to the water gaps to accommodate the effects of the water gaps should be more easily accommodated in the watercross design.

The second factor which tends to reduce the susceptibility of the SVEA fuel to the Oskarshamn 2 type failure is the extensive experimental and analytical work which has led to the capability to reliably predict dryout. The division of the assembly into four subchannels reduces the problem to a more tractable two-step process. The first step involves the use of test data to predict the occurrence of dryout in a subchannel. The limited variability in the subchannels relative to the entire assembly makes it more practical to obtain sufficiently extensive test data to assure that the entire operating range is covered. The second step involves the application of the CPR correlation derived from the data to the entire assembly in a manner which ensures that dryout is conservatively predicted considering subbundle power mismatch.

The dryout behavior of the SVEA-64 design has been established by extensive testing of sixteen-rod subbundles at the ABB Atom FRIGG Loop in Sweden and the Westinghouse Canada facility in Hamilton, Ontario. Data were obtained over the entire range of core flow, system pressure, inlet subcooling and local power expected in steady-state or transient operation. The data base for the sixteen-rod subassembly in the watercross design is based on over thirty-five 16-rod local power distributions with corner rod relative powers up to a value of 1.7. The relatively small size of the sub-bundles allowed the thorough understanding of the CPR behavior as a function of fuel rod position and relative power. This extensive CPR data base ensures that the ABB Atom methods can predict the occurrence of dryout in the Oskarshamn 2 case when the appropriate geometry is used in the analysis.

The CPR correlation for the sixteen-rod test data is applied to the SVEA-64 assembly in a manner which conservatively predicts dryout. Dryout is predicted at a slightly lower power than the test data would actually predict. Flow communication holes are provided in the SVEA assembly to equalize pressure between the sub-channels. A conservative estimate of dryout power is accomplished by assuming in the CPR calculations that there is no flow communication between the minibundles. Extensive subchannel analyses using the computer codes COBRA-IV and GOBLIN have confirmed that the assumption of no flow communication between the sub-channels leads to a conservative prediction of dryout power. In addition, ABB Atom has performed full-scale comparative 64-rod and 16-rod watercross assembly tests which confirm the results of the subchannel analyses.

The Oskarshamn 2 failures demonstrate that the uncertainties in the power level of the rods adjacent to the control rod gaps can be somewhat greater than for the interior rods. It is expected that the susceptibility for the Oskarshamn 2 type failures will be greater as the dryout sensitivity to changes in corner rod relative peaking increases. ABB Atom has performed a comparison of SVEA test data with full-scale 8X8 assembly test data using a similar spacer grid, which demonstrates, that the change in critical bundle power as the corner rod local peaking factor is changed is about equal for the SVEA design and the 8X8 design. Thus it has been shown that the SVEA design is not more sensitive to changes in corner rod peaking than fuels without a watercross.

CONCLUSION

The following conclusions regarding the fuel failures at Oskarshamn 2 can be drawn:

1. The Oskarshamn 2 fuel failures occurred due to overheating of the cladding associated with operation under dryout conditions.

2. The dryout of the failed rods occurred because of unanticipated channel bow in assemblies adjacent to those containing the failed fuel rods and an oversimplified treatment of cross sections and fuel rod power distributions in the input to the Plant Process Computer.

3. The fuel failures were not related to any design features specific to the SVEA-64 assemblies. The failures occurred in the SVEA-64 assemblies because the assemblies were fresh and were located adjacent to water gaps which were not accurately represented in the Plant Process Computer used to monitor thermal limits. Traditional BWR fuel designs without the watercross are affected in the same way as the watercross design with respect to this type of failure.

4. Analysis of the Oskarshamn 2 cycle in which the fuel failures occurred with ABB Atom analytical methods confirm that the ABB Atom methods are reliable for predicting this type of failure.

5. The overall conclusion is that current methods can predict this phenomenon without any added uncertainties, provided that the Plant Process Computer input is based on realistic best-estimate type data.

6. The ongoing investigations might provide additional refinements. For example, the analytical methods might predict results in even closer agreement with the actual operating conditions. These refinements are not expected to change the conclusions presented above.

TABLE 1

EXPOSURE OF CHANNELS AND ASSEMBLIES ADJACENT TO ASSEMBLIES WITH FAILED CORNER RODS

ASSEMBLY	BOW MEASUREMENTS DATE	EXPOSURE (MWD/KgU)	BOW[1] mm	9/88 CHANNEL EXPOSURE (MWD/KgU)	9/88 BUNDLE BURNUP (MWD/KgU)
10232	1/84	31.7	-1.22	67.9	36
	3/84	31.7	+1.13[2]		
	6/85	42.9	-0.59		
	8/87	59.4	-4.71		
	9/88	67.9	-7.31		
10230	3/84	19.5	-1.31	55.2	34
10223	6/83	29.3	-0.97	65.5	36
	9/88	65.5	-7.43		
12562	11/84	26.8	-1.21	54.4	26
12605	1/84	26.2	-1.17	53.8	26
10213	1/82	21.5	-1.41	57.1	34

[1]A negative value indicates bow away from the control rod channel. The magnitude of the values is maximum deflection of a channel face based on several measurements axially.

[2]Assembly 10232 bowed away from the control rod while in the reactor prior to 1/84. It was removed from the reactor in 1/84 and reinstalled 3/84 with the bow toward the control rod. The difference in measured values of 0.11 mm is due to measurement uncertainty.

TABLE 2

HOT CELL EXAMINATIONS

ASSEMBLY	ROD	REMARKS
13544	H8	-Primary failure occurred just below the top spacer facing the control rod gap. -Metallography revealed a phase change in the failure area demonstrating that a temperature of 860 deg. C. had been exceeded. -Profilometry revealed a cladding collapse on to the pellets just below the top spacer.
13536	H8	-The cladding was not breached. -Clad collapse just below the top spacer indicates elevated temperatures in this region. -Fission gas release measurements indicate that the rod was operated at significantly higher powers than expected. -Cesium migration measurements indicate elevated temperatures in the area of the top spacer.
	E7	-Burnup measurements provide a reference for gamma scan measurements.

13535	G7	-This rod contained Gd203/UO2 fuel, and the correct enrichments were confirmed.
		-The cladding was not breached. The rod had been suspect based on initial sipping data.
		-Fission gas results indicated that the rod had operated at significantly higher powers than expected.

TABLE 3

EFFECT OF GEOMETRIC REPRESENTATION AND CHANNEL BOW ON CPR

CASE	INCREMENTAL PERCENT CPR DECREASE	CUMULATIVE MULTIPLICATIVE EFFECT ON CPR	PREDICTED CORE MCPR
Reference	0	1.0	1.30 - 1.40
Wide Gap SVEA	7	0.93	1.21 - 1.30
Quadruple - No Channel Bow	3	0.9	1.17 - 1.26
Quadruple - Channel Bow	13	0.77	1.00 - 1.08

FIGURE 1

Oskarshamn 2: Failed Fuel Core Locations

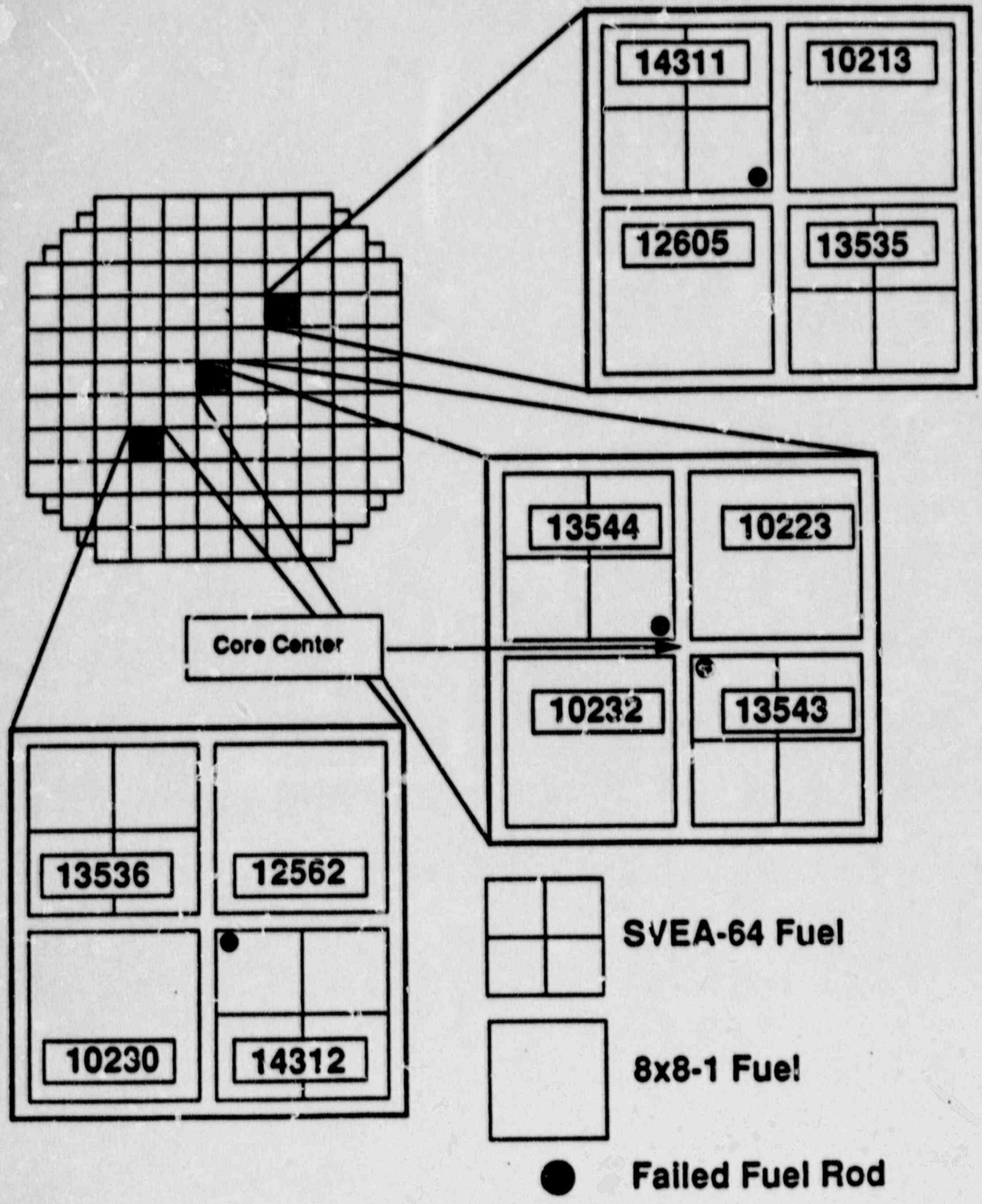

Figure 2

SVEA-64 Fuel Rod Identification

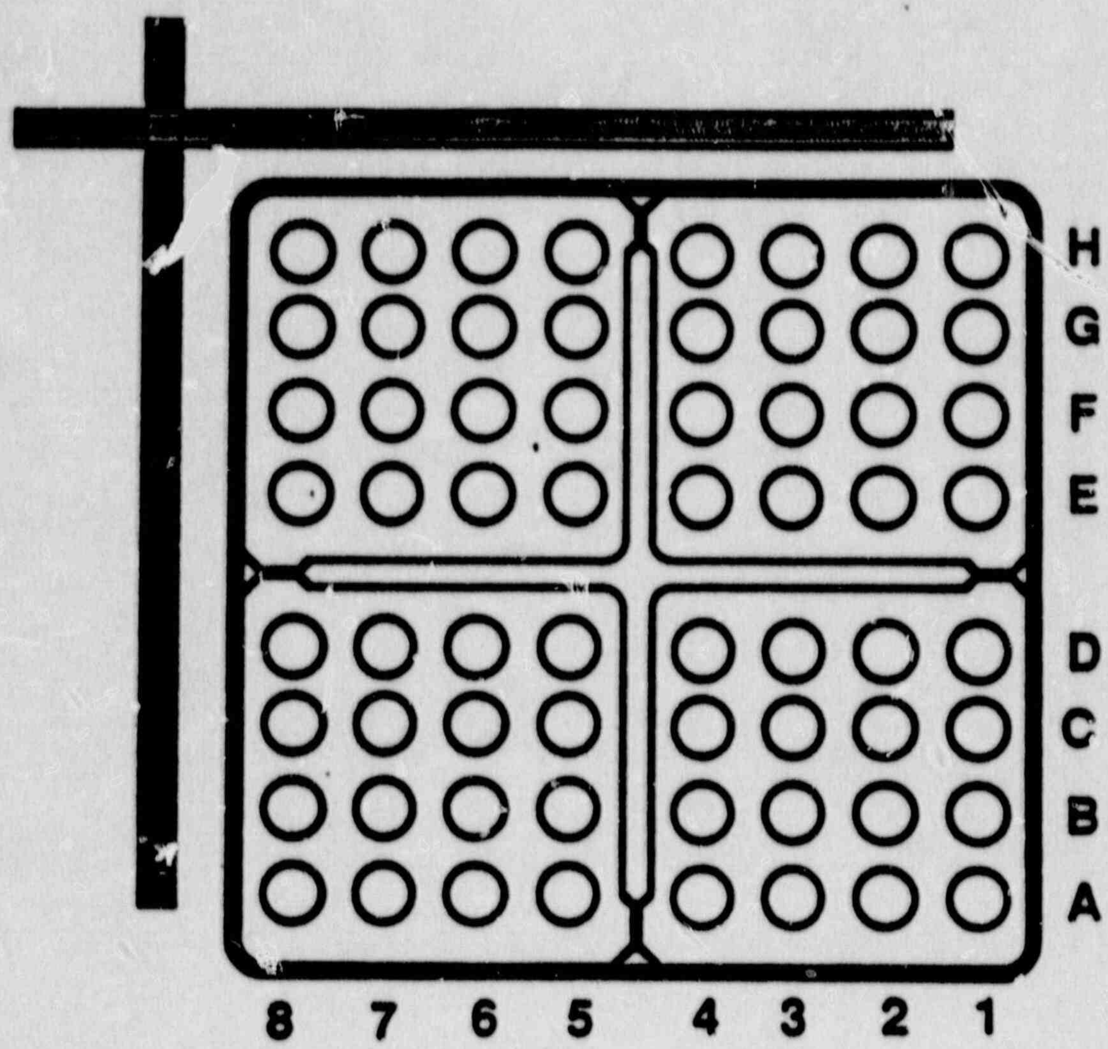

Figure 3

Channel Bow in Oskarshamn 2 Central "Supercell"

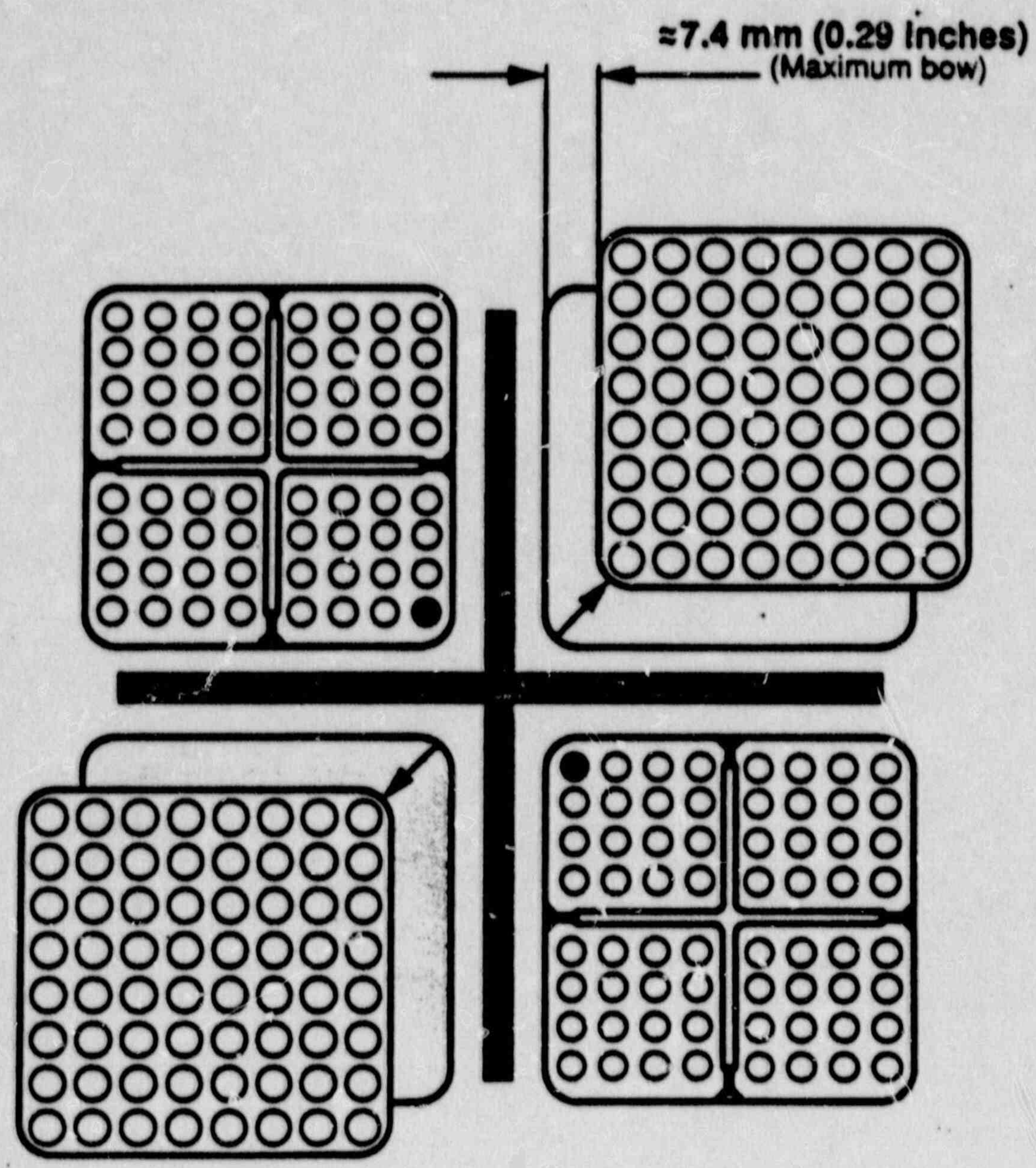

FIGURE 4

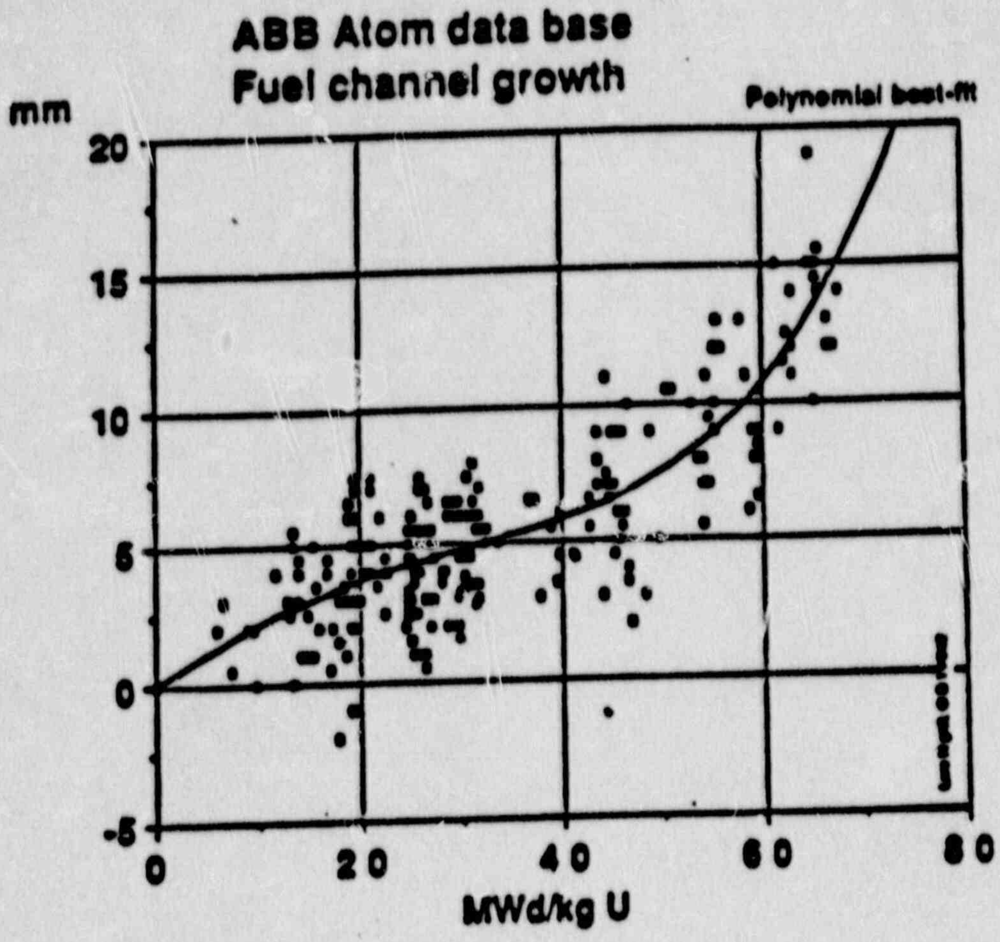

ABB Atom data base
Fuel channel growth
mm
Polynomial best-fit
20
15
10
5
0
-5
0
20
40
60
80
MWd/kg U

Figure 5

Sketch Showing the Displacement of Fresh Assemblies Toward the Control Rod Gap

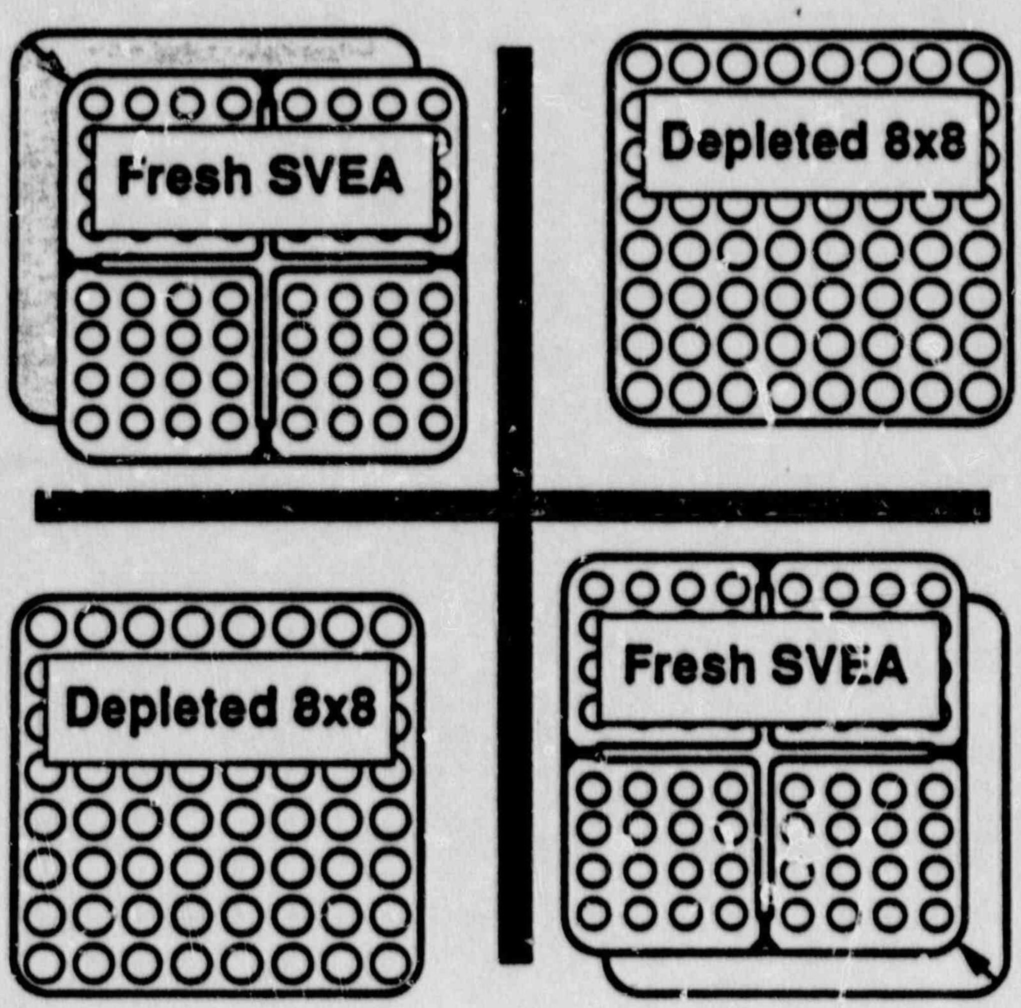

Figure 6

Geometry Used to Evaluate the Effect of Channel Bow