Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

OCT 29 2002

Docket No. 50-336
B18766

Re: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Power Station, Unit No. 2
Response to a Request for Additional Information
Technical Specifications Change Request (TSCR) 2-10-01, Revision 2
Fuel Pool Requirements

In a letter dated November 6, 2001,(1) Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request in the form of changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes would: (1) increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core; (2) allow fuel to be located in 40 Region B Storage Cells, which are currently empty and blocked, and (3) credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$. Additionally, as a result of a subsequent discussion with the Nuclear Regulatory Commission (NRC) staff, DNC provided a revision to the Significant Hazards Consideration (SHC) discussion (Attachment 1) in a letter dated December 27, 2001.(2) The revised SHC discussion did not affect the conclusion of the Safety Summary or the original SHC determination. Additionally, in a letter dated July 15, 2002,(3) DNC responded to a Request for Additional Information (RAI) from the NRC related to the aforementioned license amendment request. Also, in a letter dated August 6, 2002,(4) DNC provided a revision to the aforementioned license amendment request, which revises proposed Design Features Technical Specification (TS) 5.6.1. The additional information provided did not affect the

(1) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," dated November 6, 2001.

(2) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements, Revised Significant Hazards Consideration Discussion," dated December 27, 2001.

(3) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Response to a Request for Additional Information Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," dated July 15, 2002.

(4) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Revision 1, Fuel Pool Requirements," dated August 6, 2002.

A001

conclusions of the Safety Summary or SHC discussion of the DNC November 6, 2001 and December 27, 2001 letters.

In a facsimile dated July 25, 2002,[5] a second RAI was received from the NRC which contains eleven (11) questions related to the aforementioned license amendment request.

Attachment 1 provides the DNC response to the July 25, 2002 RAI. As described in Attachment 1 (response to question No. 11), Attachment 2 contains the marked-up page and Attachment 3 contains the retyped page for the proposed Technical Specification changes to sections 5.6.1.c) and 5.6.1.d). The additional information provided in this letter will not affect the conclusions of the Safety Summary and SHC discussion in the DNC November 6, 2001 and December 27, 2001 letters.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 29 day of October, 2002

Notary Public

My Commission expires WM. E. BROWN
NOTARY PUBLIC
MY COMMISSION EXPIRES MAR. 31, 2006

cc: See next page

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(5) R. Ennis (NRC) facsimile to R. Joshi, "Issues For Discussion in Upcoming Telephone Conference Regarding Proposed Amendment to Technical Specifications, Spent Fuel Pool Requirements, Millstone Nuclear Power Station, Unit No. 2, Docket No 50-336," dated July 25, 2002.

Attachments (3)

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Millstone Unit No. 2
Millstone Senior Resident Inspector

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Docket No. 50-336
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Attachment 1

Millstone Power Station, Unit No. 2

Response to a Request for Additional Information
Technical Specifications Change Request (TSCR) 2-10-01, Revision 2
Fuel Pool Requirements

Millstone Power Station, Unit No. 2
Response to a Request for Additional Information
Technical Specifications Change Request (TSCR) 2-10-01, Revision 2
Fuel Pool Requirements

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(1) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," dated November 6, 2001.

(2) J. A Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements, Revised Significant Hazards Consideration Discussion," dated December 27, 2001.

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(5) R. Ennis (NRC) facsimile to R. Joshi, "Issues For Discussion in Upcoming Telephone Conference Regarding Proposed Amendment to Technical Specifications, Spent Fuel Pool Requirements, Millstone Nuclear Power Station, Unit No. 2, Docket No 50-336," dated July 25, 2002.

Question 1

How long have RACKLIFE and destructive examinations of poison panels been conducted? At what frequency are each completed?

Response

The Millstone Unit No. 2 (MP2) spent fuel racks have removable Boraflex poison boxes which is a unique design feature. Each spent fuel storage location has its own Boraflex poison box. The poison box is securely attached inside each storage cell, and the fuel assembly rests inside the poison box. Attached are Figures 4-1, 4-4, 4-5 and 4-6 which show the spent fuel rack module (Figure 4-1), Boraflex poison box (Figure 4-4), and Boraflex poison box details (Figures 4-5 and 4-6). These figures were previously transmitted to the NRC in a letter dated July 24, 1985.[6] A total of three Boraflex poison boxes have been removed from the MP2 Spent Fuel Pool (SFP). In 1991, two poison boxes were removed for visual examination to correlate blackness testing gap measurements and to provide a general examination of Boraflex performance in cooperation with the Electrical Power Research Institute (EPRI). These first two poison boxes removed in 1991 were not part of our formal Boraflex monitoring program. The third poison box was removed in the year 2000 as part of our current Boraflex monitoring test program. Future poison box removals as part of this Boraflex monitoring program are planned at approximately five year time intervals, and are controlled by plant procedures. The poison box removal program functions as the primary means of monitoring for Boraflex dissolution. Testing of this Boraflex is performed by a qualified vendor 10 CFR 50, Appendix B quality assurance (QA) program.

Thus, the current method of Boraflex monitoring by poison box removal has been performed only once, in the year 2000. Prior to that, Boraflex coupons were used. However, as documented in DNC's letter dated May 7, 1997,[7] the Boraflex coupons were not representative of the Boraflex in the racks and any testing results were of questionable value. Therefore, MP2 switched to the current method of removing actual Boraflex panels from the racks.

The MP2 RACKLIFE model was first issued in 1997. The RACKLIFE model was revised in 1999, and again in 2001. The MP2 RACKLIFE model is updated approximately every 2 years, in accordance with plant procedures, to incorporate the latest information available into the model. Each RACKLIFE model is

(6) J. F. Opeka letter to the U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, Proposed Changes to Technical Specifications, Modifications to Spent Fuel Storage Pool," dated July 24, 1985.

(7) M. L. Bowling letter to the U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, Spent Fuel Rack Poison Surveillance Coupon Program," dated May 7, 1997.

documented in detail as a non-QA calculation. The calculations are non-QA because there has not been an attempt to use the RACKLIFE results to estimate Boraflex loss. RACKLIFE is used only as an independent check of Boraflex performance. Removal and testing of Boraflex from the spent fuel racks is the QA means of testing. As described in the response to question 2, RACKLIFE results are used as a screening tool to help in the decision process to determine which poison box should be removed for destructive examination.

Question 2

What were the initial assumptions (including the "escape coefficient") in RACKLIFE used to support the selection for the first poison panel subjected to destructive examination?

Response

At this time, since there is no Boraflex degradation of any significance in the MP2 spent fuel storage racks, the determination of maximum gamma dose/Boraflex degradation as calculated by RACKLIFE is used only as a screening tool to decide which poison box should be removed for destructive examination. RACKLIFE assumptions for the RACKLIFE model that was used prior to the first poison box examination included:

- Escape coefficient was 0.045 for all panels in the SFP, from initial rack installation to current date. Measured MP2 SFP reactive silica data was used to determine this escape coefficient.
- All irradiated fuel movements in the Boraflex racks were modeled since the initial installation of the Boraflex racks.
- Nominal fuel burnups were input for all irradiated fuel resident in the racks.
- Nominal rack geometry input was used, including a description of the racks walls, poison box, Boraflex and any water gaps.
- A SFP temperature of 100°F and PH of 4.5 were used.
- Demineralizer and filter models were not included. Past experience showed that they have essentially no effect on reactive silica removal, and excluding them resulted in better agreement between measured versus predicted reactive silica behavior in the spent fuel pool.

Question 3

What assumptions in RACKLIFE have been modified to ensure that the predictions reflect the changes in the SFP parameters (e.g. chemistry) and actual Boraflex degradation?

Response

The latest RACKLIFE model was updated in 2001 after the poison box examination that was completed in 2000. This model incorporates the most recent EPRI version of RACKLIFE, and also incorporates the most recent measured silica data, including dilution of silica during refuel. These changes increased the escape coefficient, which is used for all Boraflex panels in the MP2 SFP, slightly from 0.045 to 0.060. This slight increase is due to the model changes, and does not represent any change in Boraflex performance itself. No data from the Boraflex poison box surveillance resulted in RACKLIFE modification. The results of Boraflex poison box examination completed in 2000 didn't show any significant Boraflex degradation. Therefore, there was no Boraflex degradation data that can be used to update the RACKLIFE models. The RACKLIFE models also predicted that no significant Boraflex degradation would be found.

Question 4

The SFP racks in Regions A and B contain Boraflex in a flux trap design. However, Region B has 40 storage locations which are empty and blocked for reactivity control. How are the differences in these two regions modeled in the RACKLIFE code?

Response

There is no difference in the RACKLIFE region A or B model due to the use of a cell blocker. The cell blocker prevents fuel insertion in certain locations, so the accumulated gamma dose, and therefore Boraflex degradation, lags behind in the blocked locations.

Question 5

Are RODLETS used only in Region C of the SFP and not in Regions A and B?

Response

RODLETS are credited in TS only for use in Region C. RODLETS are not credited in the TS for Region A or B. Fuel with RODLETS can be physically present in Region A or B, but such RODLETS would not fulfill any credited function.

Question 6

RACKLIFE is used to predict the remaining B-10 in the panels. In addition, the Boraflex panel with the highest predicted gamma dose and predicted Boraflex degradation value is destructively tested to determine the amount of B-10 remaining in the panel.

a. How do the results of the destructive tests compare with the RACKLIFE predictions? How are these results tracked?
b. How are the results of the destructive tests factored into future RACKLIFE projections?

Response

a. RACKLIFE predicted 0.8% B-10 loss on average, and 0.9% B-10 loss for the most degraded Boraflex panel. The average measured B-10 areal density from the Boraflex which was removed from the storage racks was ≥ 0.03532 grams B-10/cm^2. The manufactured B-10 areal density was 0.033 grams B-10/cm^2, +/- 0.003. Therefore, the B-10 loss, if any, is too small to be seen. The procedure governing Boraflex testing requires Boraflex testing results be obtained from a qualified (10 CFR 50, Appendix B) quality assurance program vendor, and then reviewed by the Reactor Analysis organization to evaluate the acceptability of continued use of Boraflex at MP2.

   DNC provided in a letter dated November 6, 2001 (Attachment 1, page 16) a description of the MP2 SFP Boraflex Material Condition, and the Boraflex monitoring program. The following is a re-iteration of the DNC conclusion from that discussion:

   > "Based on the testing to date, DNC concludes that the Boraflex contained in the Millstone 2 spent fuel racks has performed acceptably to date. There has been no detectable loss in Boraflex thickness to date. While axial Boraflex gaps are present due to irradiation caused shrinkage, the size of the gaps are small and have no appreciable reactivity impact to date. The proposed criticality analysis makes far more conservative assumptions on the Boraflex condition than the existing criticality analysis of record, in case future in-service testing detects degradation.
   >
   > DNC believes that the Boraflex in MP2 racks has performed acceptably to date for 2 reasons: (1) because the Boraflex material is 110 mills thick, which is thicker than typical, and (2) the Boraflex material is well protected from interaction with water due to its design."

   Attached for information are seven (7) pictures which show the poison box which was removed from the MP2 SFP in the calendar year 2000. These pictures show the following:

   Picture 1: This picture shows a view from the top of the poison box. The side of the poison box facing toward the top of the page shows the exposed

Boraflex. The stainless steel cover which would normally protect the Boraflex has been cut away on this face. The other 3 faces of the poison box still have the stainless steel protective cover installed.

Picture 2: This picture looks down the length of the poison box, showing the exposed Boraflex. This view is from the top of the poison box. The nominal Boraflex is quite thick, approximately 0.11 inch.

Picture 3: This picture shows a close-up of the Boraflex at the top of the poison box. The hole in the picture is the original inspection hole.

Picture 4: This picture is similar to picture 3, but at a slightly different angle.

Picture 5: This picture is similar to picture 3, but at a slightly different angle.

Picture 6: This picture looks down the length of the poison box, showing the exposed Boraflex. This view is from the bottom of the poison box.

Picture 7: This picture shows a close-up of the Boraflex at the bottom of the poison box.

b. There is a current procedural requirement for RACKLIFE model updates to be performed approximately every two (2) years. These model updates are documented in a detailed calculation file, and are also documented with a memo to appropriate plant personnel summarizing the RACKLIFE results. Any results from Boraflex examinations would be factored into RACKLIFE at the time of the two year update, however, there has been no Boraflex degradation of significance to date to incorporate into RACKLIFE.

<u>Question 7</u>

After determining the Boraflex panel to be removed and destructively tested, the licensee does not discuss its replacement.

a. Is the destructively tested Boraflex panel that is removed replaced? If so, with what material?
b. How is the removal of the Boraflex panel destructively tested factored into RACKLIFE?
c. How does the removal of this Boraflex panel affect the future placement of fuel in this storage rack?

Response

a. Boraflex removed to date has been replaced with Boraflex poison boxes manufactured to the same dimensions as the original poison boxes. These replacement Boraflex poison boxes were purchased at about same time as the original rack purchase, and like the racks, were purchased as QA category 1. A Certificate of Compliance, and associated Boraflex data sheets, were provided to Millstone from the manufacturer (BISCO) which states that the Boraflex used in the replacement/spare poison boxes meets the same limits (Boraflex dimensions and B-10 loadings) as the Boraflex used in the racks.

b. The removal of the three Boraflex poison boxes and the replacement of them with new boxes has not been modeled in RACKLIFE. The RACKLIFE models these three replaced poison boxes as if they have never been replaced. These three replaced poison boxes represent three out of a total of 384 poison boxes in the SFP, and this approximation will have a negligible effect on the RACKLIFE model results. Since the three locations which have replaced poison boxes contain relatively new Boraflex, these three locations are far from being limiting locations for Boraflex degradation.

c. There is no effect. Once a poison box has been replaced, the storage location is treated no differently than any other storage location.

   - When the first two poison boxes were replaced in 1991, a procedure was utilized to control the removal of the old poison boxes, and installation of the new poison boxes. This procedure required verification by Quality Services that the installed poison boxes were properly oriented and that the locking tabs were in place to secure the poison boxes in the storage rack. It was also documented that Boraflex was present in the inspection holes of each poison box. After installation of the new poison boxes, a free path check of the new poison boxes was successfully performed with a dummy fuel assembly.
   - When a poison box was replaced in the calendar year 2000, a procedure was utilized to control the removal of the old poison box, and installation of the new poison box. This procedure required verification by Quality Services that the installed poison box was properly oriented and that the locking tabs were in place to secure the poison box in the storage rack. It was also documented that Boraflex was present in the inspection holes of the poison box. No special free path check was performed after installation of this poison box, since a free path check is required as part of the manufacture of the poison box. A fuel assembly has since been moved into and out of this spent fuel location without any unusual indications.

Question 8

With respect to your submittal dated July 15, 2002, the response to question 1d states that 4 sections of near full Boraflex width were removed for B-10 density measurements. In addition, the submittal states "4 sections were chosen to avoid the top and bottom 2 feet of the Boraflex... Also sections of Boraflex were chosen if they showed thinned areas... testing was performed by selecting 32 random locations, and 1 specific location for a section that showed some minor thinning."

a. What were the constraints that formed the basis for the 32 random locations tested on the 4 Boraflex panels?
b. What was the tested area length for each of the 33 locations tested?

Response

a. There were no constraints placed on measured locations of the four Boraflex panels. Each of the four Boraflex panels was considered as a grid, with the size of each grid being 1 inch square. A random drawing was performed over all of the 1 inch square grids on each Boraflex panel to pick the 8 grids selected for measurement. Since 8 grids were randomly selected on each of the four panels, a total of 32 locations were randomly tested in all.

b. For B-10 areal density measurements, the neutron beam hole size was approximately 3/8 inch in diameter, and this therefore reflects the area that was measured at each of the 33 locations. For the Boraflex thickness measurements, the anvil area of the micrometer used was approximately ¼ inch in diameter, and this therefore reflects the area that was measured at each of the 33 locations.

Question 9

With respect to your submittal dated July 15, 2002, the response to question 2 states: "Cell blockers serve no function other than to provide a visible cue to the fuel handler that the fuel should not be inserted in that location."

a. What is the purpose of the cell blocker if fuel will be stored beneath it?
b. Will all fuel storage cells be blocked if fuel is stored beneath them?

Response

a. Since the cell blockers were already installed and available, it was thought prudent to retain them to provide additional protection against a misloading event.

b. No, not all fuel storage cells will be blocked if fuel is stored beneath them. Only 40 storage cells in Region B would have TS controlled cell blockers. Those 40 locations are shown in proposed TS Figure 3.9-2. As a practical matter, when this proposed TS is implemented, all 40 of these locations will contain fuel with a cell blocker over them. The 40 locations are needed for spent fuel storage, so the intent is to fill the 40 locations immediately and restore the cell blockers above them. The proposed (and current) TS allow removal of a cell blocker if the requirements of the TS are met.

Question 10

Placing spent fuel underneath the cell blockers is something that hasn't been done before at MP2. The cell blockers weigh approximately 20 pounds and are made of stainless steel. Has the licensee evaluated the possibility and subsequent consequences of cell blockers falling on the spent fuel?

Response

The drop of a cell blocker has been evaluated and it was found not to cause any fuel failure if dropped onto spent fuel.

Question 11

The staff would also like to discuss the following:

a. Criticality analysis for the new fuel storage vault and transfer machine as described on pages 11 and 12 of Attachment 1 in your submittal dated November 6, 2001; and
b. Responses to questions 6 and 8 in your submittal dated July 15, 2002.

Response

a. During the August 7, 2002 conference call, the NRC Staff requested to amplify our statement related to the criticality analysis for the new fuel storage vault and transfer machine as described on pages 11 and 12 of Attachment 1 of the DNC submittal dated November 6, 2001. The following information is provided in support of our statement that the Advanced Nuclear Fuel Corporation (ANF) analysis included in our submittal dated April 10, 1990[8] is still valid relative to the proposed TS change to Section 5.6.1. Specifically, the proposed TS change would increase the allowed nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry) and the reactor core.

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(8) E. J. Mroczka letter to the U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, Proposed Changes to Technical Specifications, Fuel Enrichment Limits," dated April 10, 1990.

In a letter dated April 10, 1990, TS changes were submitted to the NRC. The proposed changes would revise TS 5.6.1 to allow fuel with an enrichment up to 4.5 w/o U-235 to be stored in the new fuel storage racks and the reactor core and spent fuel pool. The supporting analysis, performed by ANF, evaluated the loading of fuel of enrichments up to 5.0 w/o U-235 in the new fuel storage racks, the reactor core or the spent fuel pool. The NRC reviewed the analysis and concluded that the methods and models used in the analysis were acceptable. In addition, the NRC's Safety Evaluation for Amendment 146[9] which approved the proposed TS changes, indicates that the criticality analysis presented in the ANF analysis meets the applicable NRC acceptance criteria.

DNC concludes that the proposed enrichment increase from 4.5 w/o U-235 to 4.85 w/o U-235 (as described in our submittal dated November 6, 2001) is covered under the ANF analysis (which covers enrichment up to 5.0 w/o U-235) submitted in letters dated April 10, 1990, and accepted by the NRC in a letter dated June 13, 1990, and therefore is still valid for the proposed changes described in our submittal dated November 6, 2001.

b. As discussed with the NRC staff during the August 7, 2002, conference call, the purpose of the proposed changes to TS 3.9.18 is to remove ambiguity and improve usability of existing TS. There are no technical changes associated with the proposed changes. However, these changes are made using the guidance of NUREG-1432, Revision 2. Therefore, it is concluded that no additional changes to the proposed changes included in our submittal dated November 6, 2001, are required.

   Our response (July 15, 2002, submittal) to Question 8 indicated that we could not add a reference to the proposed TS 5.6.1.c) and d) because there was none previously approved. We also noted that the specific Westinghouse report (Attachment 5 to our November 6, 2001, submittal) which is under NRC review as part of this request, contains the uncertainty values. Therefore, we did not see how we could reference an unapproved request in a license amendment request and as a result we could not add the non-approved reference to the proposed TS 5.6.1.c) and d). During a conference call on August 7, 2002, the NRC stated that it was acceptable to reference the Westinghouse report in the proposed TS even though it is currently under NRC review as part of this amendment request.

   Accordingly, TS 5.6.1.c) and d) will be revised to add the reference. In addition, an error was discovered in proposed TS section 5.6.1.c). Specifically the sign

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(9) G. S. Vissing (USNRC) letter to E. J. Mroczka, "Issuance of Amendment 146 (TAC NO. 76473)," dated June 13, 1990.

"≤" was used instead of the correct sign "<" for the term K_{eff}. This error occurred in transcription of the standard NRC words in our proposed TS change (our submittal dated November 6, 2001). Therefore, the new proposed TS 5.6.1.c) and 5.6.1.d) become as follows:

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Westinghouse Report A-MP- FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq 0.95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties as described in Westinghouse Report A-MP-FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

Attachment 2 contains the marked-up pages (please note that the marked-up and retyped pages do not reflect the recently issued License Amendment 270) and Attachment 3 contains the retyped pages of the proposed changes.

Question 12

Please provide additional information regarding the Design Basis Temperature for the spent fuel pool water.

Response

The design basis temperature for the spent fuel pool bulk water temperature is 150°F under normal conditions.

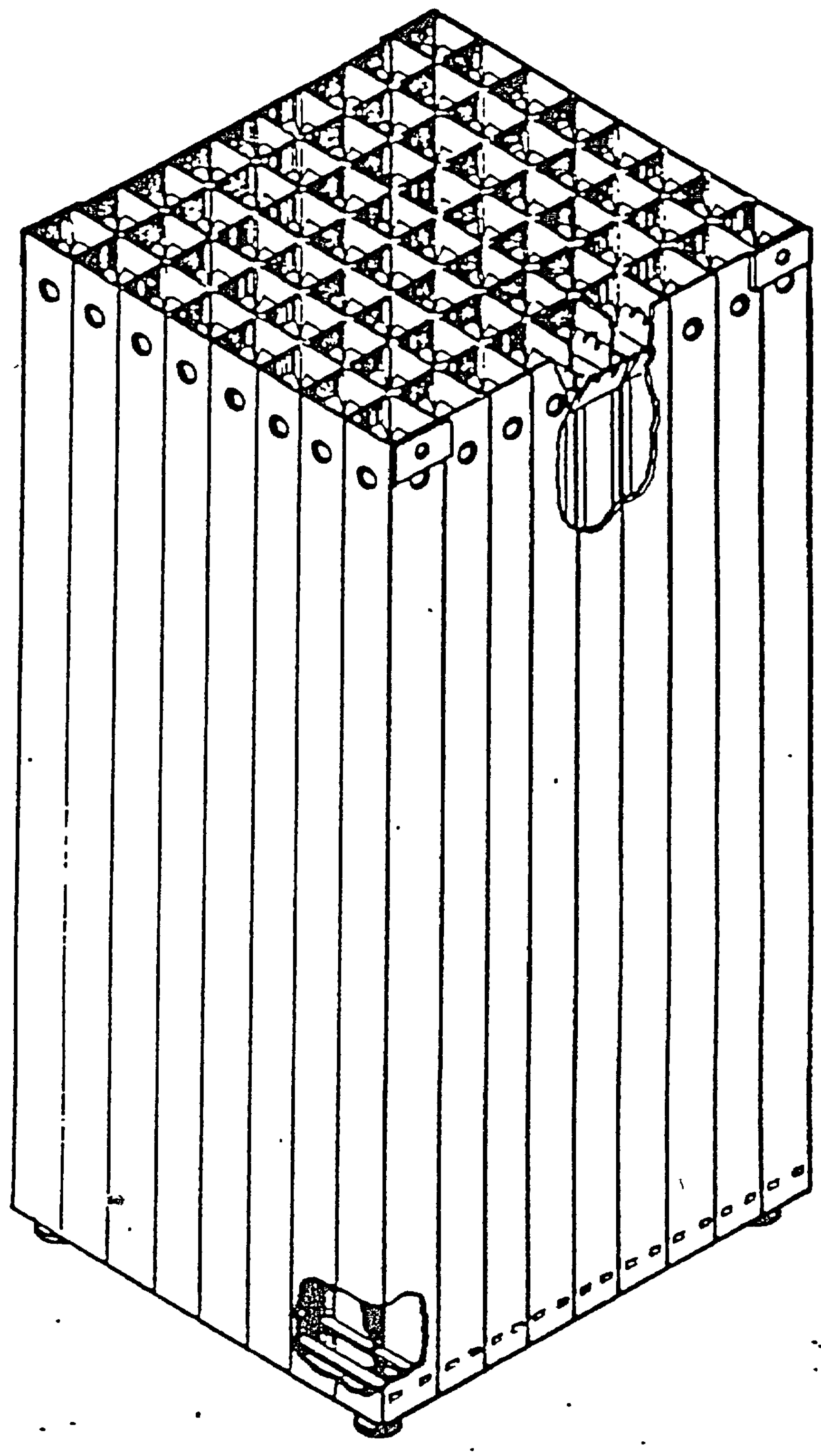


FIGURE 4-1

Typical Spent Fuel Rack Module/
Poison Box
(Region I)

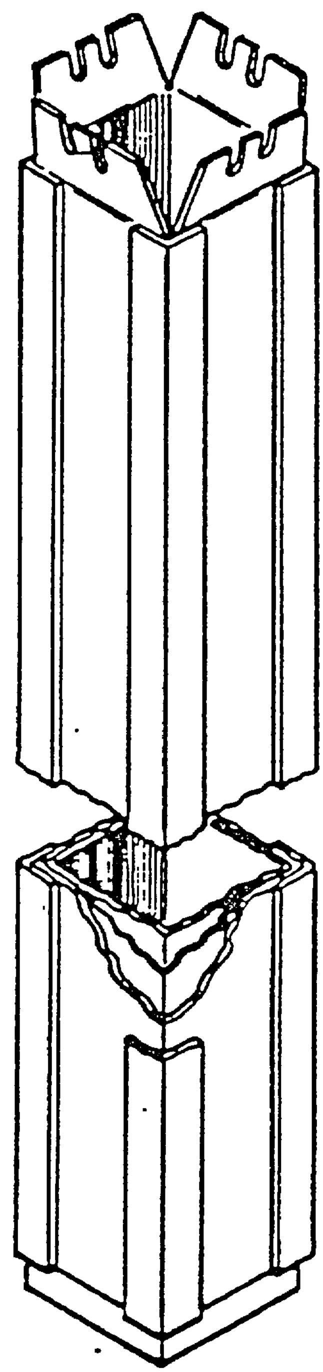


FIGURE 4-4

Poison Box

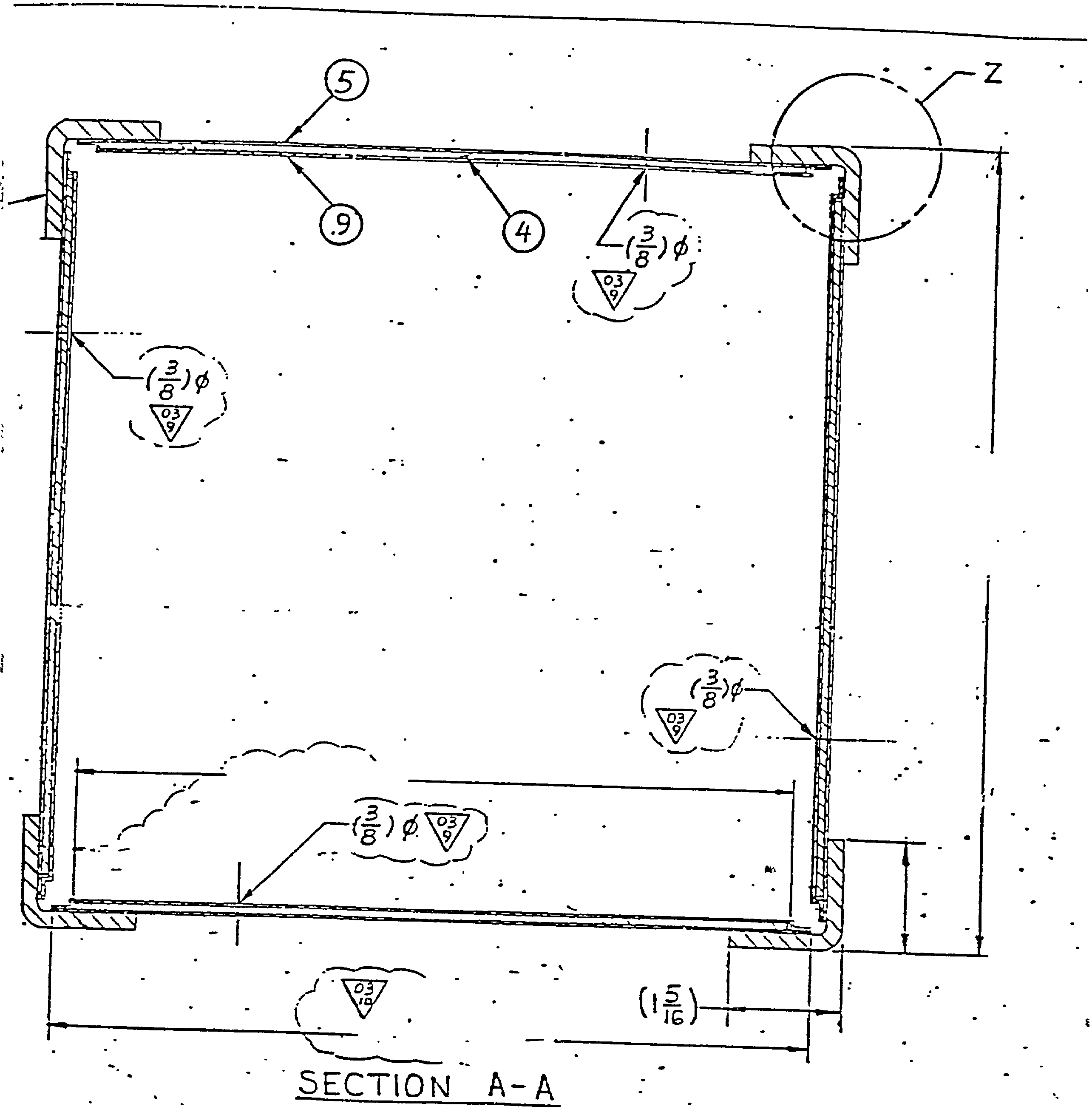


FIGURE 4-5

Poison Box Section View

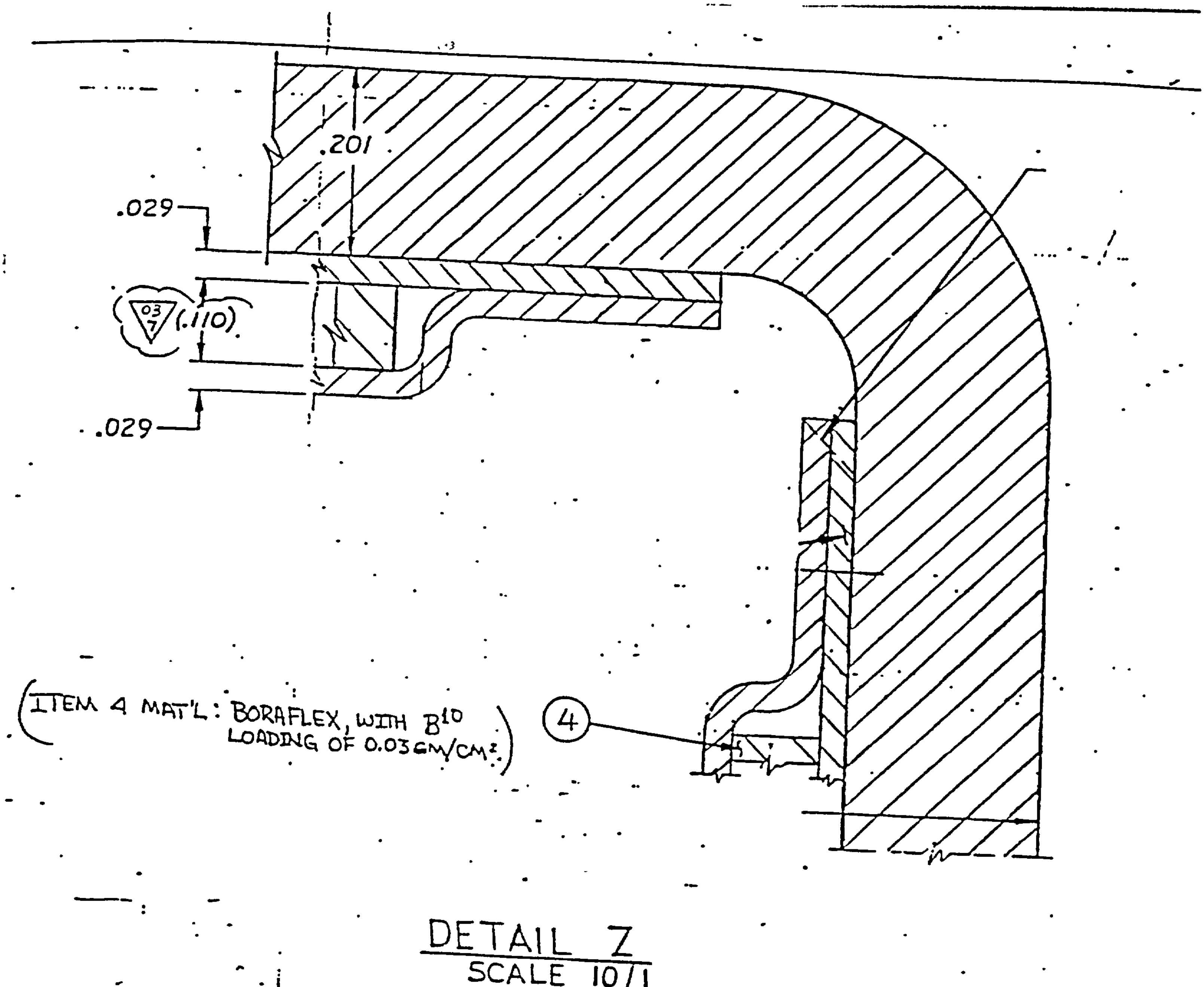
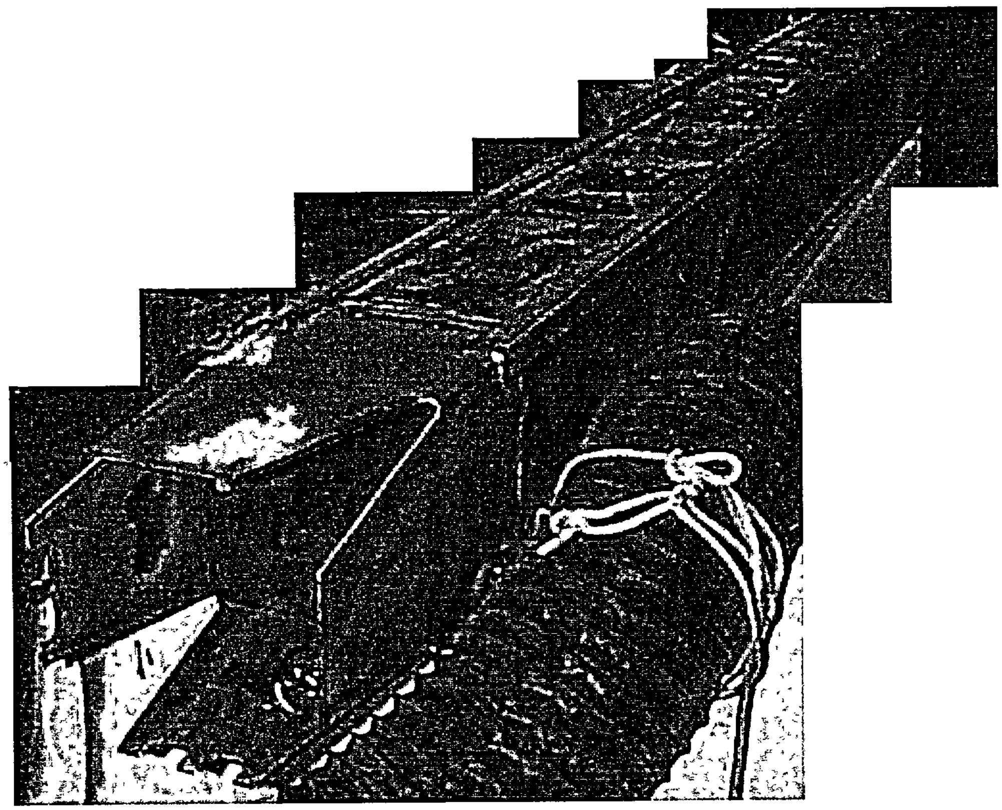


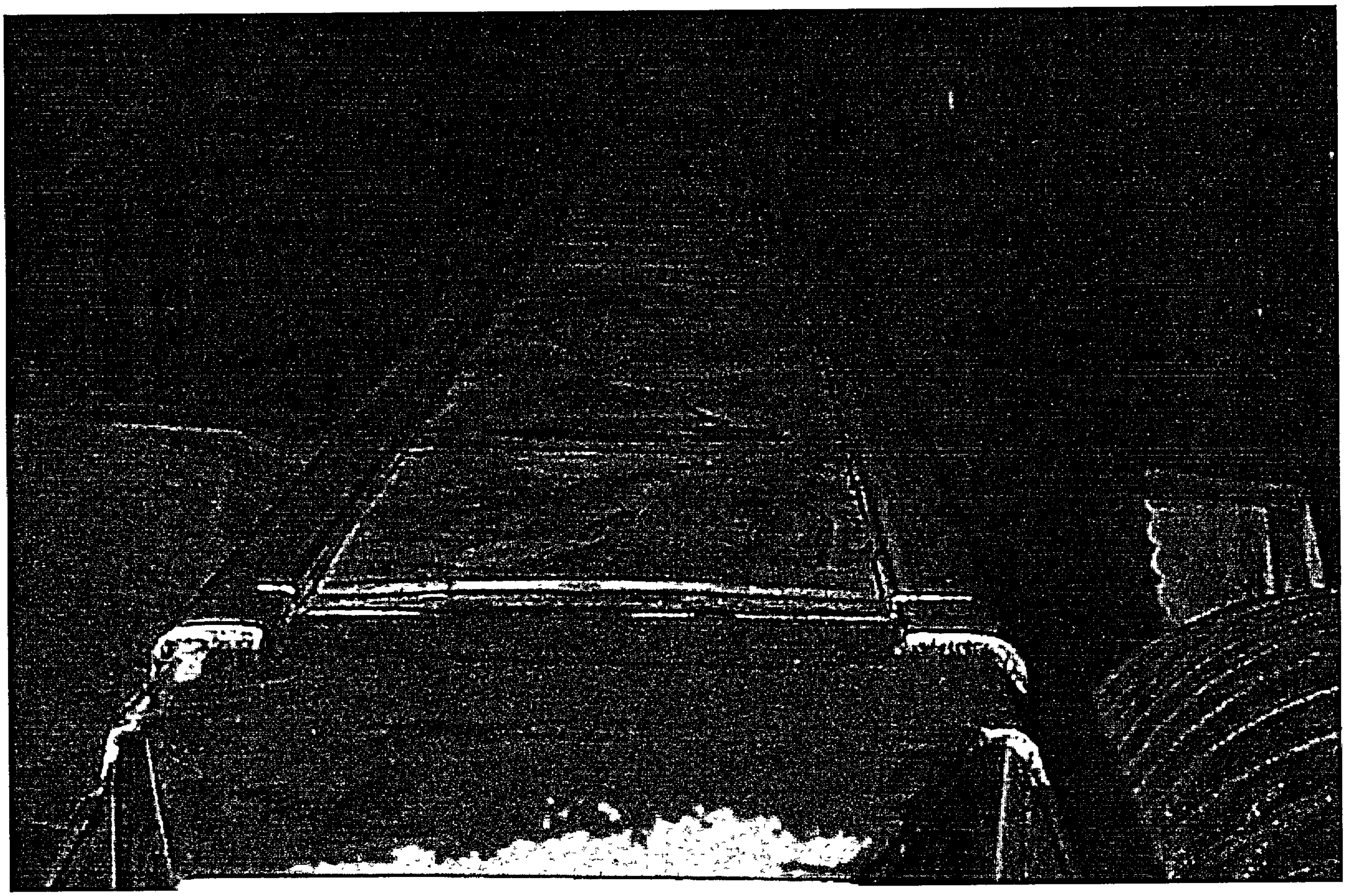
FIGURE 4-6

Poison Box Section Detail

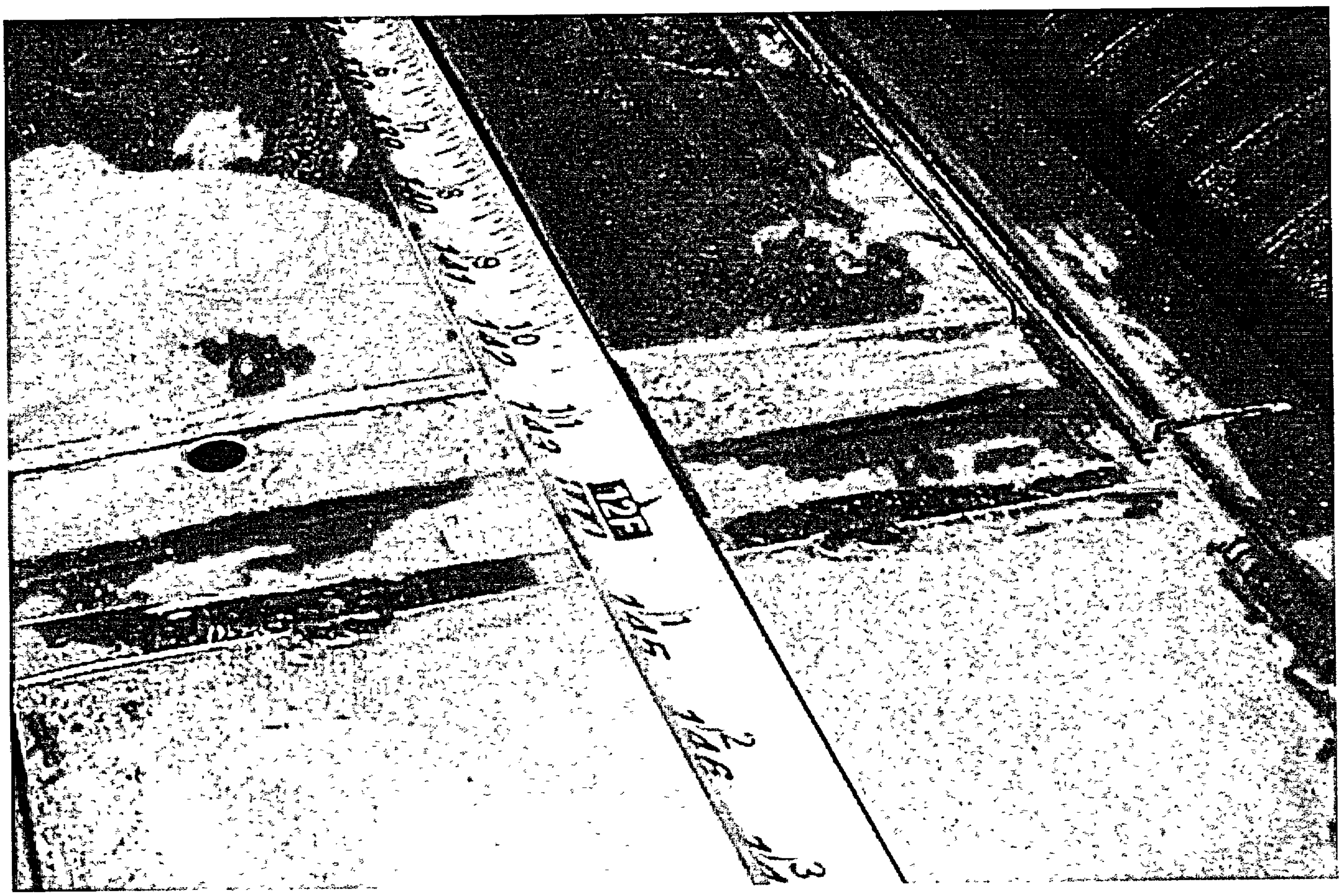
Picture 1



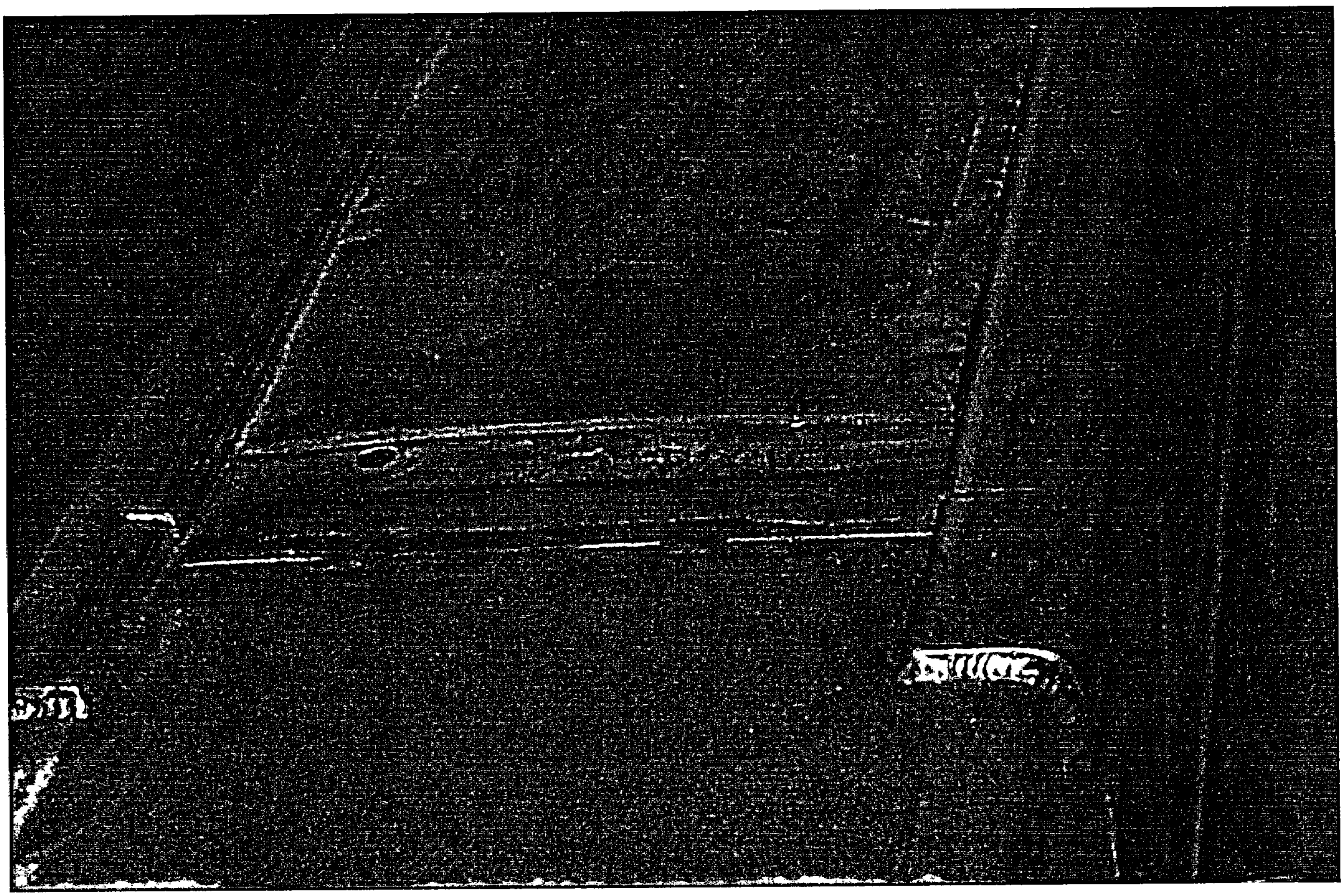
Picture 2



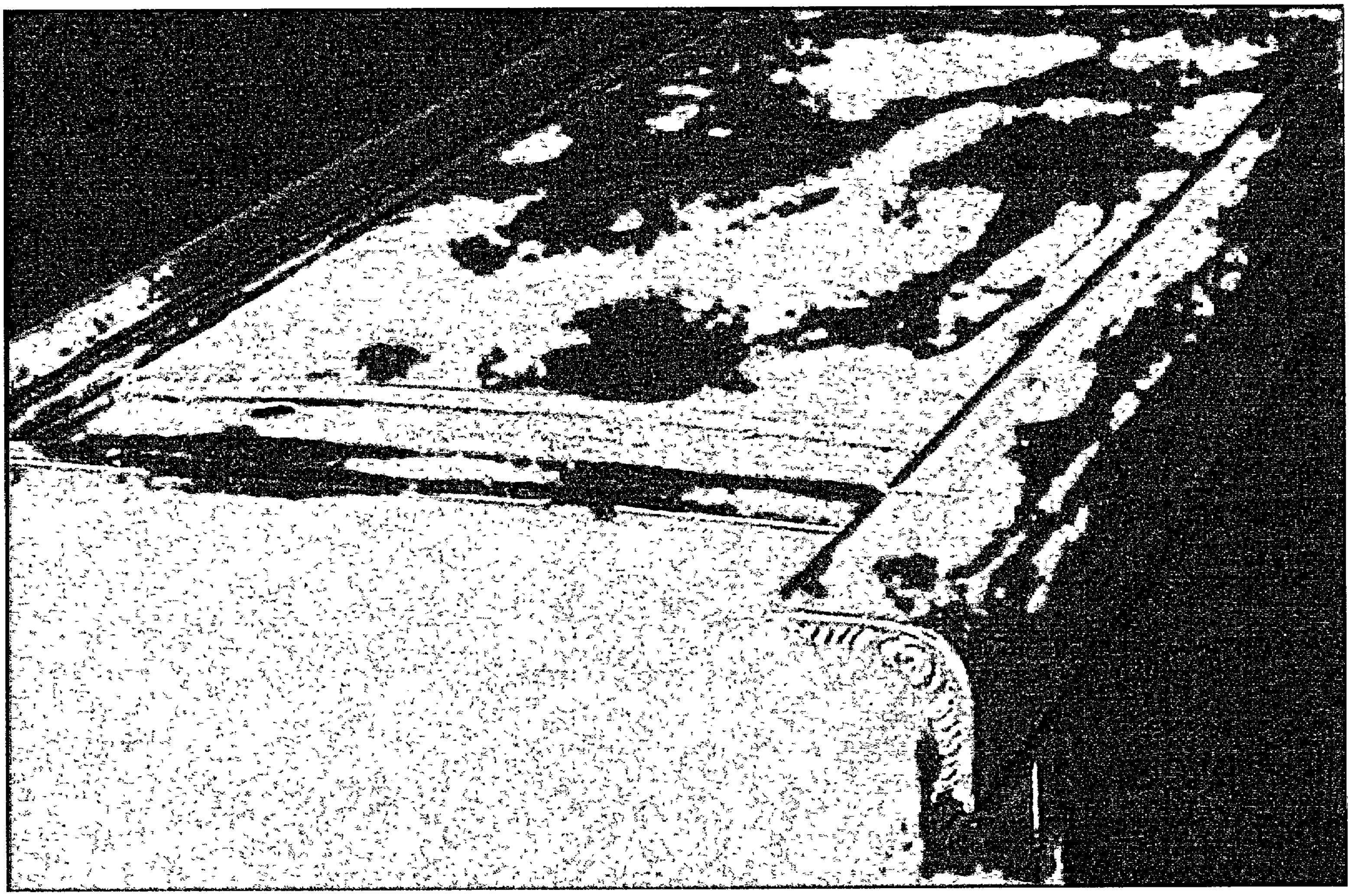
Picture 3



Picture 4

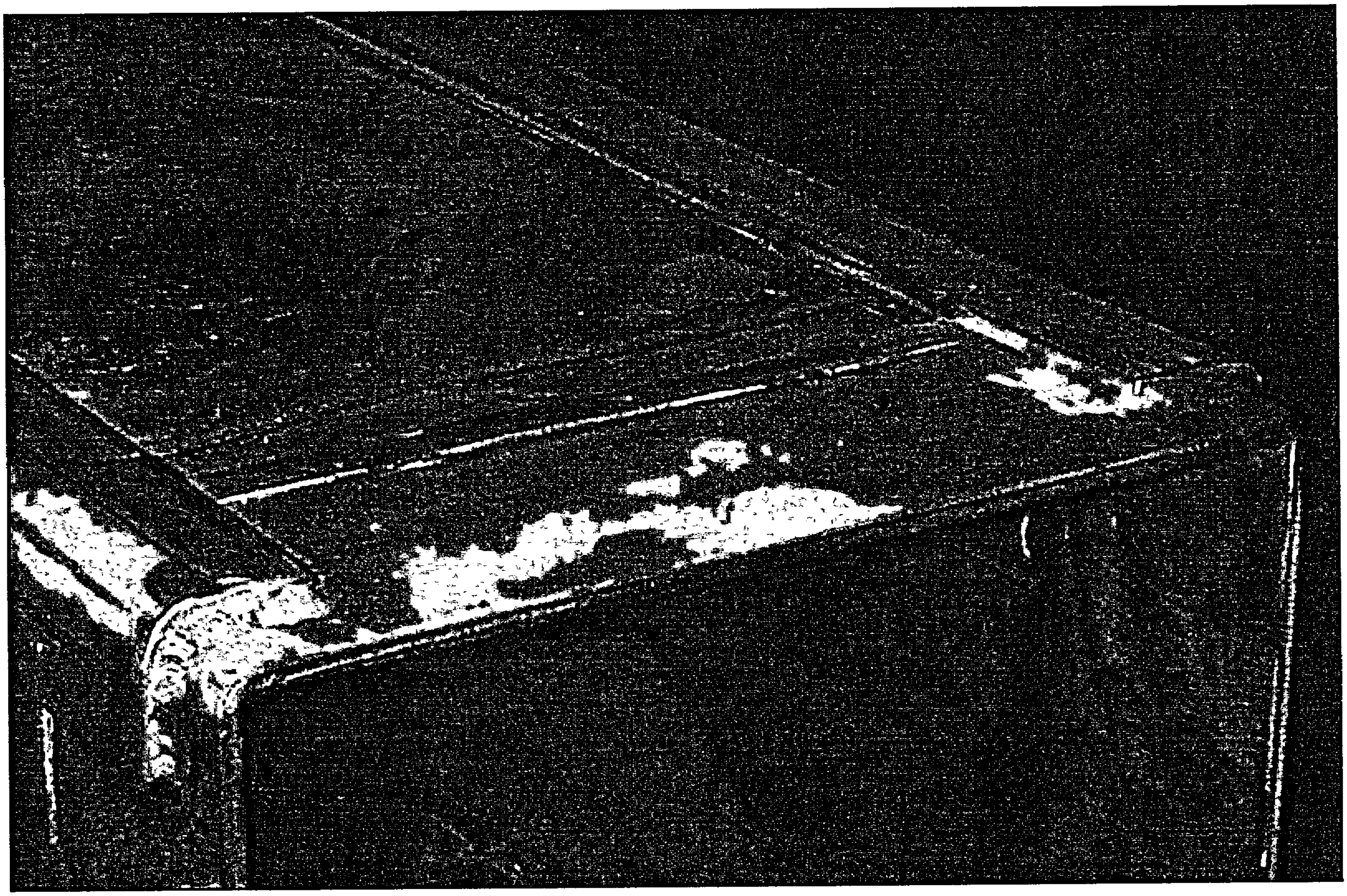


Picture 5



Picture 6

Picture 7



Docket No. 50-336
B18766

Attachment 2

Millstone Power Station, Unit No. 2

Technical Specifications Change Request (TSCR) 2-10-01, Revision 2
Proposed Design Features TS 5.6.1
Fuel Pool Requirements
Marked-Up Page

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is a nominal 10,981 ft^3.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $k_{eff} \leq .95$. The maximum nominal average fuel assembly enrichment to be stored in these racks is 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum nominal average enrichment of 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties, as described in Westinghouse Report A-MP-FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties, as described in Westinghouse Report A-MP-FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

e) Region A of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-4 to ensure that the design burnup has been sustained.

f) Region B of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region B contains both blocked and un-blocked storage locations, shown in Figure 3.9-2. Fuel having a maximum nominal enrichment of 4.85 weight percent U-235, may be stored in un-blocked locations. Fuel stored in blocked locations must be Batch B fuel assemblies.

g) Region C of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figures 3.9-1a or 3.9-1b to ensure that the design burn-up has been sustained. Additionally, fuel assemblies utilizing Figure 3.9-1b require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron.

h) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel. The contents of the consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3 to ensure that the design burnup has been sustained.

Docket No. 50-336
B18766

Attachment 3

Millstone Power Station, Unit No. 2

Technical Specifications Change Request (TSCR) 2-10-01, Revision 2
Proposed Design Features TS 5.6.1
Fuel Pool Requirements
Retyped Page

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is a nominal 10,981 ft^3.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $k_{eff} \leq .95$. The maximum nominal average fuel assembly enrichment to be stored in these racks is 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum nominal average enrichment of 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Westinghouse Report A-MP-FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties as described in Westinghouse Report A-MP-FE-0011, Revision 1, "Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit."

e) Region A of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-4 to ensure that the design burnup has been sustained.

f) Region B of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region B contains both blocked and un-blocked storage locations, shown in Figure 3.9-2. Fuel having a maximum nominal enrichment of 4.85 weight percent U-235, may be stored in un-blocked locations. Fuel stored in blocked locations must be Batch B fuel assemblies.

g) Region C of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figures 3.9-1a or 3.9-1b to ensure that the design burn-up has been sustained. Additionally, fuel assemblies utilizing Figure 3.9-1b require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron.

h) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel. The contents of the consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3 to ensure that the design burnup has been sustained.