WESTINGHOUSE PROPRIETARY CLASS 3

Supplemental Information for NRC Approved Version of WCAP-9401/9402 and WCAP-9500

PREPARED BY:

S. L. Davidson
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February, 1983

APPROVED BY:

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Westinghouse Electric Corporation
Nuclear Energy Systems
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Pittsburgh, Pennsylvania 15230

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 12 1982

Westinghouse Electric Corporation
ATTN: Mr. E. P. Rahe, Manager
Nuclear Safety Department
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: Supplemental Acceptance Number 1 for Referencing of Licensing Topical Report WCAP-9500A

The Nuclear Regulatory Commission (NRC) accepted for referencing Westinghouse Electric Corporation licensing topical report WCAP-9500 entitled "Reference Core Report 17x17 Optimized Fuel Assembly" by letter from R. L. Tedesco to T. M. Anderson dated May 22, 1981.

The NRC review, which culminated in the acceptance of the report WCAP-9500 for referencing, considered a core containing only Optimized Fuel Assemblies (OFA's). The review of the use of OFA's mixed with standard assemblies had not been completed at that time. One of NRC's concerns regarding mixed cores of standard and optimized assemblies involved structural considerations--namely that structural component changes between the standard assembly and OFA can result in a change in mechanical responses to LOCA and seismic loads. NRC expected to evaluate these structural considerations for mixed core reloads on a case by case basis.

Westinghouse in their letter from E. P. Rahe to L. S. Rubenstein on August 11, 1981 indicated a desire to further demonstrate generically that analyses for a full core of standard fuel and for a full core of optimized fuel bound all mixed core combinations. The Westinghouse March, 1982 letter from E. P. Rahe to J. R. Miller provided the results of several mixed core loading configurations and concluded that allowable limits are not exceeded for any of the plants encompassed of by the verification testing and analysis program of WCAP-9401/9402.

The NRC has completed its review of the seismic and LOCA loading and structural response considerations provided in the above submittals. Our safety evaluation is enclosed.

Based on our review, we have concluded that mixed cores of Westinghouse standard assemblies and OFA's subjected to seismic and LOCA forces given in WCAP-9401 are acceptable with respect to meeting the requirements of Appendix A to Standard Review Plan Section 4.2.

As a result of our review, we find that Westinghouse licensing topical report WCAP-9500 is acceptable for referencing in mixed cores license applications with respect to structural considerations, provided it is shown that the applied forces considered in WCAP-9401 bound the plant in question. Otherwise, additional analysis will be required. It should be noted that this acceptance pertains to the structural considerations previously stated in the introduction to the Safety Evaluation Report on WCAP-9500 relative to mixed cores. The core physics and thermal hydraulic reservations expressed in that SER remain applicable and will be addressed in the near future.

We do not intend to repeat the review of the safety features described in the topical report as augmented by responses to staff questions and found acceptable in the attachment. Our acceptance applies only to the features described in the topical report and the auxiliary documents and under the conditions discussed in the enclosure.

In accordance with established procedure (NUREG-0390), it is requested that Westinghouse Electric Corporation publish an accepted version of this report, proprietary and non-proprietary. The accepted version is to incorporate this letter, including the attached topical report evaluation, following the title page and thus just in front of the abstract. The report must appropriately include all supporting information submitted relevent to NRC's mixed core structural concerns. The report identifications of the approved reports are to have a -A suffix.

Should NRC criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Westinghouse Electric Corporation and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Acting Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

cc: Mr. Bruce Lorenz
Nuclear Safety Department
P. O. Box 355
Pittsburgh, Pennsylvania 15230

SAFETY EVALUATION OF MIXED CORES OF WESTINGHOUSE STANDARD AND OPTIMIZED FUEL SUBJECTED TO SEISMIC AND LOCA LOADING

Structural Evaluation

The earlier NRC Safety Evaluation Reports (Refs. 1 and 2) for WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly," and WCAP-9401, "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," concluded that the Optimized Fuel Assembly (OFA) meets all the requirements of the Appendix A to SRP Section 4.2 with respect to fuel assembly structural response to Seismic-and-LOCA forces. It was also concluded that for each individual plant it must be demonstrated that the applied forces considered in WCAP-9401 bound the plant in question. These SERs in addition to approving the OFA subjected to Seismic-and-LOCA forces, pointed out that mixed cores of Westinghouse standard assembly and OFA will be evaluated on a case-by-case basis.

In order to take care of mixed cores, Westinghouse, however, has submitted the results of a generic study (Ref. 3). In this study Westinghouse has analyzed fives cases of mixed cores subjected to Seismic-and-LOCA loading given in WCAP-9401. These cases, as shown in Figure 1 of the submittal, are: (1) homogeneous standard fuel assembly, (2) 2/3 standard and 1/3 OFA, (3) a different row of 2/3 standard and 1/3 OFA, (4) 1/3 standard and 2/3 OFA, and (5) homogeneous OFA. These cases cover a broad spectrum of the mixed cores and, therefore, are representative of mixed cores of standard and OFA. The response of these cases has been characterized by the grid impact forces which are the most critical response elements. Results of these analyses have been summarized in a table in the submittal. The maximum grid impact force on any grid from these analyses is only 75 percent of the allowable grid impact strength. The mixed cores of standard and OFA, subjected to Seismic-and-LOCA forces given in WCAP-9401 are, therefore, acceptable with respect to meeting the requirements of Appendix A to SRP Section 4.2. For each individual plant, however, it must be shown that the applied forces considered in WCAP-9401 bound the plant in question or else additional analysis will be required.

References:

1. Memorandum for L. S. Rubenstein, USNRC, to R. L. Tedesco, "Safety Evaluation Report on WCAP-9500," dated May 15, 1981.

2. Letter from R. L. Tedesco, USNRC, to T. M. Anderson, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9500," dated May 22, 1981.

3. Letter from E. P. Rahe, Westinghouse, to J. R. Miller, USNRC, "WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items," dated March 19, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 24 1983

Mr. E. P. Rahe, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: Supplemental Acceptance Number 2 for Referencing of Licensing Topical Report WCAP-9500

The Nuclear Regulatory Commission (NRC) accepted for referencing Westinghouse Electric Corporation licensing topical report WCAP-9500 entitled "Reference Core Report 17x17 Optimized Fuel Assembly" by letter from R. L. Tedesco to T. M. Anderson dated May 22, 1981.

The NRC review, which culminated in the acceptance of the report WCAP-9500 for referencing, considered a core containing only Optimized Fuel Assemblies (OFA's). The review of the use of OFA's mixed with standard assemblies had not been completed at that time. One of NRC's concerns regarding mixed cores of standard and optimized assemblies involved the effects on diversion crossflow between assemblies due to different axial pressure losses. NRC expected to evaluate this consideration for mixed core reloads in conjunction with our review of WCAP-9272 entitled "Westinghouse Reload Safety Evaluation Methodology."

Westinghouse in their letter from E. P. Rahe to L. S. Rubenstein on August 11, 1981 indicated a desire to further demonstrate generically that analyses for a full core of standard fuel and for a full core of optimized fuel bound all mixed core combinations. The Westinghouse March, 1982 letter from E. P. Rahe to J. R. Miller provided the results of several mixed core loading configurations and concluded that allowable limits are not exceeded for any of the plants encompassed by the verification testing and analysis program of WCAP-9401/9402.

We have completed our review of the diversion crossflow effects considerations provided in the above submittals. Our safety evaluation is enclosed.

Based on our review of the information provided in the above submittals and our independent audit, we conclude that the adjustment to the departure from nucleate boiling ratio (DNBR) limit and the method used to thermal-hydraulically analyze mixed cores of 17x17 OFAs and 17x17 standard assemblies are acceptable. For transition cores containing different fuel arrays, e.g., 14x14 or 15x15, the DNBR adjustment must be re-analyzed or Westinghouse must demonstrate that the present adjustment bounds these other fuel types.

RECEIVED
JAN 28 1983
Nuclear Safety Department

As a result of our review, we find that Westinghouse licensing topical report WCAP-9500 is acceptable for referencing in mixed cores license applications with respect to diversion crossflow effects considerations. It should be noted that this acceptance pertains to the crossflow considerations previously stated in the introduction to the Safety Evaluation Report on WCAP-9500 relative to mixed cores. The structural reservations expressed in that SER have been previously addressed in supplemental acceptance number 1 and physics considerations will be addressed on a case by.case basis.

We do not intend to repeat the review of the safety features described in the topical report as augmented by responses to staff questions and found acceptable in the attachment. Our acceptance applies only to the features described in the topical report and the auxiliary documents, and under the conditions discussed in the enclosure.

In accordance with established procedure (NUREG-0390), it is requested that Westinghouse Electric Corporation publish an accepted version of this report, proprietary and non-proprietary. The accepted version is to incorporate this letter, including the attached topical report evaluation, following the title page and thus just in front of the abstract. The report must appropriately include all supporting information submitted relevant to NRC's mixed core structural concerns. The report identifications of the approved reports are to have a -A suffix.

Should NRC criteria or regulations change, such that our conclusions as to the acceptability of the report are invalidated, Westinghouse Electric Corporation and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure: As stated

cc: Mr. Bruce Lorenz
Nuclear Safety Department
P.O. Box 355
Pittsburgh, Pennsylvania 15230

1.0 Thermal-Hydraulic Design

1.1 Introduction

From a thermal-hydraulic standpoint, the staff required in our WCAP-9401 safety evaluation report (Rubenstein, April 1981) that Westinghouse provide additional submittals which quantified the effects on interbundle diversion crossflow of the different grid heights and fuel pin diameters and the consequential effects on departure from nucleate boiling. In response to this requirement, Westinghouse performed a series of sensitivity studies which were intended to address the staff's concern on a mixed core reload, (Rahe; August 17, 1982). As a result of these analyses, Westinghouse has recommended an adjustment to the OFA departure from nucleate boiling ratio (DNBR) limit when there is a mixed core configuration. This penalty would conservatively bound the hydraulic incompatibility of fuel assemblies having different axial pressure loss profiles and the increase in the uncertainty of the THINC-IV code (WCAP-7956) when it is used to predict the local coolant conditions in a mixed core.

1.2 Summary of Submittal

The sensitivity studies on a mixed core reload were performed using the THINC-IV computer code and the methodology presented in WCAP-9500. Twenty-six different analyses were performed on an analogous core model using different loading patterns, pressures, inlet temperatures, powers, flows and axial power distributions. In addition, an investigation on the effects of the different rod diameters on the lateral friction factor and the resultant crossflow was performed.

Based on the results of these analyses, Westinghouse has requested an adjustment to the DNBR limit of the 17x17 OFA when it is placed in a transition core. This adjustment is intended to encompass any additional uncertainties which may be present in a mixed core reload.

Finally, Westinghouse presented their approach to analyzing mixed core reloads.

1.3 Staff Review

Since the methodology and analytical tools used to perform the sensitivity analyses have been previously approved by the staff (Rubenstein; May 15, 1981) our review centered mainly on the proposed adjustment and method of analyzing mixed cores.

During our review, the staff orally requested that Westinghouse justify the cases used in assessing the DNBR penalty. Westinghouse responded that the axial power distributions used were those expected throughout a transition core and the range of parameters varied were consistent with previously approved submittals.

We also asked Westinghouse to justify using the model reported. Their response was that the model was sufficient to define the adjustment and a full core model was too detailed and could not be constructed.

As part of the review effort the staff performed an audit calculation of a full core OFA and a mixed core with an OFA as the limiting assembly. The COBRA-IV code was used in the analyses and the results of these calculations are presented in Table 1. The difference in the full and mixed core MDNBRs is (approximately 1.7%) well within the adjustment proposed by Westinghouse.

Table 1
Comparison of Staff Audit Calculations

Case	Elevation (inches)	MDNBR (-)	Enthalpy (BTU/lbm)	Mass Flux (Mlb/hr-ft^2)
Full Core	101.9	2.642	658.37	2.4404
Mixed Core	101.9	2.596	659.09	2.4180

Based on our review of the additional information submitted by Westinghouse to address our WCAP-9401 concerns and our audit calculations using COBRA-IV, we conclude that the methodology and the adjustment to the DNBR limit described in August 17, 1982 submittal is acceptable for 17x17 transition cores. Transition cores containing different fuel rod arrays must be re-analyzed or Westinghouse must demonstrate that the present adjustment for a 17x17 transition core bounds these different fuel designs.

4.0 References

4.1 Topical Reports

WCAP-7956, "THINC-IV-An Approved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," Westinghouse Electric Corporation, June 1973.

WCAP-9500, "Reference Core Report 17x17 Optimized Fuel Assembly," Westinghouse Electric Corporation, July 1979.

4.2 Other References

Letter, E. P. Rahe (Westinghouse) to James R. Miller (NRC), Subject: Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information," August 17, 1982.

Memorandum, L. S. Rubenstein, to Robert L. Tedesco, Subject: "Review of Topical Report WCAP-9500," August 23, 1981.

Memorandum, L. S. Rubenstein to Robert L. Tedesco, Subject: "Safety Evaluation Report on WCAP-9500," May 13, 1981.

Memorandum, L. S. Rubenstein to Robert L. Tedesco, Subject: "Review of Topical Report WCAP-9401," April 23, 1981.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 24 1983

Mr. E. P. Rahe, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: Supplemental Acceptance Number 2 for Referencing of Licensing Topical Report WCAP-9401/9402

The Nuclear Regulatory Commission (NRC) accepted for referencing Westinghouse Electric Corporation licensing topical report WCAP-9401/9402 entitled "Verification Testing and Analyses of 17x17 Optimized Fuel Assembly" by letter from R. L. Tedesco to T. M. Anderson dated May 7, 1981.

The NRC review, which culminated in the acceptance of the report WCAP-9401/9402 for referencing, considered a core containing only Optimized Fuel Assemblies (OFA's). The review of the use of OFA's mixed with standard assemblies had not been completed at that time. One of NRC's concerns regarding mixed cores of standard and optimized assemblies involved the effects on diversion crossflow between assemblies due to different axial pressure losses. NRC expected to evaluate this consideration for mixed core reloads in conjunction with its review of WCAP-9272 entitled "Westinghouse Reload Safety Evaluation Methodology."

Westinghouse in their letter from E. P. Rahe to L. S. Rubenstein on August 11, 1981 indicated a desire to further demonstrate generically that analyses for a full core of standard fuel and for a full core of optimized fuel bound all mixed core combinations. The Westinghouse March, 1982 letter from E. P. Rahe to J. R. Miller provided the results of several mixed core loading configurations and concluded that allowable limits are not exceeded for any of the plants encompassed by the verification testing and analysis program of WCAP-9401/9402.

We have completed our review of the diversion crossflow effects considerations provided in the above submittals. Our safety evaluation is enclosed.

Based on our review of the information provided in the above submittals and our independent audit, we conclude that the adjustment to the departure from nucleate boiling ratio (DNBR) limit and the method used to thermal-hydraulically analyze mixed cores of 17x17 OFAs and 17x17 standard assemblies are acceptable. For transition cores containing different fuel arrays, e.g., 14x14 or 15x15, the DNBR adjustment must be re-analyzed or Westinghouse must demonstrate that the present adjustment bounds these other fuel types.

As a result of our review, we find that Westinghouse licensing topical report WCAP-9401/9402 is acceptable for referencing in mixed cores license applications with respect to diversion crossflow effects considerations. It should be noted that this acceptance pertains to the crossflow considerations previously stated in the introduction to the Safety Evaluation Report on WCAP-9500 relative to mixed cores. The structural reservations expressed in that SER have been previously addressed in supplemental acceptance number 1 and physics considerations will be addressed on a case by case basis.

We do not intend to repeat the review of the safety features described in the topical report as augmented by responses to staff questions and found acceptable in the attachment. Our acceptance applies only to the features described in the topical report and the auxiliary documents, and under the conditions described in the enclosure.

In accordance with established procedure (NUREG-0390), it is requested that Westinghouse Electric Corporation publish an accepted version of this report, proprietary and non-proprietary. The accepted version is to incorporate this letter, including the attached topical report evaluation, following the title page and thus just in front of the abstract. The report must appropriately include all supporting information submitted relevant to NRC's mixed core structural concerns. The report identifications of the approved reports are to have a -A suffix.

Should NRC criteria or regulations change, such that our conclusions as to the acceptability of the report are invalidated, Westinghouse Electric Corporation and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure: As stated

cc: Mr. Bruce Lorenz
Nuclear Safety Department
P.O. Box 355
Pittsburgh, Pennsylvania 15230

DEC 1 4 1982

1.0 Thermal-Hydraulic Design

1.1 Introduction

From a thermal-hydraulic standpoint, the staff required in our WCAP-9401 safety evaluation report (Rubenstein, April 1981) that Westinghouse provide additional submittals which quantified the effects on interbundle diversion crossflow of the different grid heights and fuel pin diameters and the consequential effects on departure from nucleate boiling. In response to this requirement, Westinghouse performed a series of sensitivity studies which were intended to address the staff's concern on a mixed core reload, (Rahe; August 17, 1982). As a result of these analyses, Westinghouse has recommended an adjustment to the OFA departure from nucleate boiling ratio (DNBR) limit when there is a mixed core configuration. This penalty would conservatively bound the hydraulic incompatibility of fuel assemblies having different axial pressure loss profiles and the increase in the uncertainty of the THINC-IV code (WCAP-7956) when it is used to predict the local coolant conditions in a mixed core.

1.2 Summary of Submittal

The sensitivity studies on a mixed core reload were performed using the THINC-IV computer code and the methodology presented in WCAP-9500. Twenty-six different analyses were performed on an analogous core model using different loading patterns, pressures, inlet temperatures, powers, flows and axial power distributions. In addition, an investigation on the effects of the different rod diameters on the lateral friction factor and the resultant crossflow was performed.

Based on the results of these analyses, Westinghouse has requested an adjustment to the DNBR limit of the 17x17 OFA when it is placed in a transition core. This adjustment is intended to encompass any additional uncertainties which may be present in a mixed core reload.

Finally, Westinghouse presented their approach to analyzing mixed core reloads.

1.3 Staff Review

Since the methodology and analytical tools used to perform the sensitivity analyses have been previously approved by the staff (Rubenstein; May 15, 1981) our review centered mainly on the proposed adjustment and method of analyzing mixed cores.

During our review, the staff orally requested that Westinghouse justify the cases used in assessing the DNBR penalty. Westinghouse responded that the axial power distributions used were those expected throughout a transition core and the range of parameters varied were consistent with previously approved submittals.

We also asked Westinghouse to justify using the model reported. Their response was that the model was sufficient to define the adjustment and a full core model was too detailed and could not be constructed.

As part of the review effort the staff performed an audit calculation of a full core OFA and a mixed core with an OFA as the limiting assembly. The COBRA-IV code was used in the analyses and the results of these calculations are presented in Table 1. The difference in the full and mixed core MDNBRs is (approximately 1.7%) well within the adjustment proposed by Westinghouse.

Table 1
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Mixed Core	101.9	2.596	659.09	2.4180

Based on our review of the additional information submitted by Westinghouse to address our WCAP-9401 concerns and our audit calculations using COBRA-IV, we conclude that the methodology and the adjustment to the DNBR limit described in August 17, 1982 submittal is acceptable for 17x17 transition cores. Transition cores containing different fuel rod arrays must be re-analyzed or Westinghouse must demonstrate that the present adjustment for a 17x17 transition core bounds these different fuel designs.

4.0 References

4.1 Topical Reports

WCAP-7956, "THINC-IV-An Approved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," Westinghouse Electric Corporation, June 1973.

WCAP-9500, "Reference Core Report 17x17 Optimized Fuel Assembly," Westinghouse Electric Corporation, July 1979.

4.2 Other References

Letter, E. P. Rahe (Westinghouse) to James R. Miller (NRC), Subject: Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information," August 17, 1982.

Memorandum, L. S. Rubenstein, to Robert L. Tedesco, Subject: "Review of Topical Report WCAP-9500," August 23, 1981.

Memorandum, L. S. Rubenstein to Robert L. Tedesco, Subject: "Safety Evaluation Report on WCAP-9500," May 13, 1981.

Memorandum, L. S. Rubenstein to Robert L. Tedesco, Subject: "Review of Topical Report WCAP-9401," April 23, 1981.

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Technology Division
Box 355
Pittsburgh Pennsylvania 15230

August 11, 1981

NS-EPR-2498

Mr. Lester S. Rubenstein, Assistant Director
Core and Containment Systems
Division of Systems Integration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Laurence E. Phillips, Core Performance Branch
Dr. Ralph O. Meyer, Core Performance Branch

SUBJECT: WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items

Dear Mr. Rubenstein:

The NRC's SERs for WCAP-9500, "Reference Core Report - 17x17 Optimized Fuel Assembly" and WCAP-9401(P)/WCAP-9402(NP), "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly" contained two generic issues relating to the mixed core compatibility of Westinghouse Optimized Fuel Assemblies (OFA's) with standard fuel.

Quoting from Page 1 of the Staff's SER for WCAP-9500, the two items are summarized below:

1. "The major concern in the area of thermal-hydraulic compatibility of the two types of assemblies is the effect on flow diversion (cross-flow) by different grid heights and fuel pin diameters. Before the Staff can approve a mixed core reload, Westinghouse must provide additional information to resolve the thermal-hydraulic concern."

2. "Structural component changes between the standard assembly and OFA can result in a change in mechanical responses to LOCA and seismic loads. This issue also should be evaluated on a case-by-case basis."

The attached discussion addresses the NRC's concern with respect to the first item and is submitted for your review. We are available to meet with members of your staff to discuss any remaining questions or concerns.

The second item has full core OFA as well as mixed core implications. With respect to a full core application of optimized fuel, Westinghouse will either confirm that the analyses results given in Section 3.0 of WCAP-9401/9402 apply for a specific plant application on a plant-by-plant basis, or we will provide additional analyses or evaluations. For mixed cores of Westinghouse standard and optimized fuel, we are currently performing evaluations and analyses to further demonstrate generically that analyses for a full core of standard fuel and a full core of optimized fuel bound all mixed core combinations. We plan to submit the results of these evaluations and analyses to the NRC in the fourth quarter of 1981. As with item 1, we will be available to meet with members of your staff, if necessary, to resolve any remaining concerns.

As stated in the conclusion (Section 18.4) of the Reload Core Methods and Considerations (Chapter 18) of WCAP-9500, the standard reload methodology will be applied for any transition cores. Results of this analysis will be evaluated in accordance with the requirements of 10CFR50.59. Any items having potential safety impact will be considered for review in accordance with current procedures for reloads for which Westinghouse has design responsibility.

Please let us know if we can be of further assistance.

Very truly yours,

E P Rahe Jr.

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Attachment

ATTACHMENT TO NS-EPR-2498

SUBJECT: Applicability of W DNB Correlations for Mixed OFA/STD Cores

This note addresses the acceptability of applying W DNB critical heat flux correlations (e.g. WRB-1 correlation), which are based on experimental data from small rod bundles in a closed channel, to the evaluation of open lattice PWR cores. The redistribution of flow in PWR cores is a well documented and modelled phenomenon which occurs generally due to thermal-hydraulic fluid condition gradients within the core. In a mixed core of standard Inconel grid and optimized fuel assemblies (OFA) the local hydraulic impedance differences are also a mechanism for redistribution. This redistribution results in the fluid velocity vector having a lateral component as well as the dominant axial component. The lateral component is commonly referred to as "crossflow".

The lateral component of flow has been assumed to not degrade the critical heat flux. The predicted critical heat flux, based on the DNB correlation and the axial component of flow only, is conservative providing appropriate enthalpy conditions are utilized.

Rod bundle DNB tests with small localized perturbations (i.e. blockage, Reference 1) to the flow path have indicated no DNB penalty. In fact, the blockage test indicated that, for some conditions, the CHF was enhanced.

More closely related to the direct effect of combined axial and lateral flow are tests by Gaspari on misaligned rod clusters (Reference 2) showing no substantial effect and Bergles (Reference 3) and Whalley (Reference 4) on the CHF enhancement effects of swirl.

Numerous studies have been performed on determining CHF in pure crossflow. Reference 5, for example, indicates CHF enhancement by a factor of 2 to 3 over a comparable parallel flow tube bank. In a mixed core of standard Inconel grid assemblies and OFA Zircaloy grid assemblies the peak local lateral velocity component is on the order of 20 percent of the axial flow, corresponding to a deviation of 11 degrees from the vertical. It must be noted here that the angle of flow vector relative to the vertical induced by thermal-hydraulic redistribution or by localized hydraulic discontinuities would not be nearly large enough to hypothesize the separation of flow or downstream effects behind the rod when in pure crossflow.

It is thus concluded that the application of W DNB correlations to the analysis of a mixed core of standard Inconel grid assemblies and OFA Zircaloy grid assemblies is valid since 1) there is no adverse observed effect of crossflow to DNB critical heat flux (in fact there is evidence which implies enhancement) and 2) the dominant velocity component is the axial direction with a lateral (crossflow) component which is not large enough to induce the flow pattern characteristic of a rod in pure crossflow.

References

1. Hill, K. W., Motley, F. E., Cadek, F. F., Casterline, J. E., "Effects on Critical Heat Flux of Local Heat Flux Spikes or Local Flow Blockages in Pressurized Water Reactor Rod Bundles," ASME Paper 74-WA/HT-54, November, 1974.

2. Gaspari, G. P., "Dryout Measurements in Freon 12 in Aligned and Misaligned 19 Rod Clusters," ASME Paper 75-HT-23.

3. Bergles, A. E., "Enhancement of Heat Transfer," Sixth International Heat Transfer Conference, Paper KS-9, 1978.

4. Whalley, P. B., "The Effect of Swirl on Critical Heat Flux in Annular Two-Phase Flow," International Journal of Multiphase Flow, Vol. 5, pp. 211-217, 1979.

5. Coffield, R. D., Roliner, W. M., Tong, L. S., "An Investigation of Departure from Nucleate Boiling (DNB) in a Crossed-Rod Matrix with Normal Flow of Freon-113 Coolant", Nuclear Engineering and Design, Vol. 2, 1967.

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2573

March 19, 1982

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Mr. Laurence E. Phillips, Core Performance Branch
Dr. Ralph O. Meyer, Core Performance Branch

SUBJECT: WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items

REFERENCE: Westinghouse Letter No. NS-EPR-2498 (E. P. Rahe to L. S. Rubenstein) dated August 11, 1981

Dear Mr. Miller:

Enclosed are:

1. Twenty-five (25) copies of WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatibility Items (Proprietary).
2. Fifteen (15) copies of WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatibility Items (Non-Proprietary).

Also enclosed are:

1. One (1) copy of Application for Withholding, AW-82-14, (Non-Proprietary).
2. One (1) copy of original Affidavit (Non-Proprietary).

The NRC's SERs for WCAP-9500, "Reference Core Report - 17 x 17 Optimized Fuel Assembly" and WCAP-9401 (P)/WCAP-9402 (NP), "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly" contained two generic issues relating to the mixed core compatibility of Westinghouse Optimized Fuel Assemblies (OFA's) with standard fuel.

Quoting from page 1 of the Staff's SER for WCAP-9500, the two items are summarized below:

1. "The major concern in the area of thermal-hydraulic compatibility of the two types of assemblies is the effect on diversion (crossflow) by

different grid heights and fuel pin diameters. Before the Staff can approve a mixed core reload, Westinghouse must provide additional information to resolve the thermal-hydraulics concern."

2. "Structural component changes between the standard assembly and OFA can result in a change in mechanical responses to LOCA and seismic loads. This issue also should be evaluated on a case-by-case basis."

The enclosed discussions supplement the information provided in the reference submittal and are submitted for your review. With respect to item 1, several meetings and telephone discussions have been held with members of the Core Performance Branch - Thermal/Hydraulics Section to better define their concerns. The information provided is responsive to these concerns. With respect to item 2 the enclosed information gives the results of several mixed core loading configurations and concludes that the allowable limits are not exceeded for any of the plants encompassed by WCAP-9401/9402.

Plant specific OFA transition core licensing documentation will be provided to several utilities in 1982 for submittal to the NRC in support of their upcoming initial reload of optimized fuel. Therefore, your prompt consideration of the enclosed material is requested. We are available to discuss any remaining questions or concerns.

This submittal contains proprietary information of Westinghouse Electric Coporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-82-14 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Manager
Nuclear Safety Department

MDB/kk
Attachments

RESPONSE TO SER ITEM 1

This paper describes the 17x17 Standard Fuel Assembly (STD) to the 17x17 Optimized Fuel Assembly (OFA) Thermal-hydraulic transition core methods, the resulting transition core DNBR penalty relative to a 17x17 OFA full core analysis and Westinghouse's position on the application of this penalty.

The 17x17 OFA has an[ ]relative to the 17x17 STD due +(a,c)
to heat flux and equivalent diameter effects. [+(a,c)

]

[+(a,c)

]

The analysis which determined the transition core DNBR penalty used the THINC IV(1) code. The configuration used was a[+(a,c)

]

It should be noted that the THINC IV code uses the Novendstern-Sandberg axial friction factor correlation[4] which, under two-phase conditions, employs the homogeneous flow model proposed by Owens[5]. Also under two-phase conditions, the THINC IV code implicitly corrects the pressure drop at grid locations by using the bulk density rather than the saturated liquid density. This is equivalent to the APD Simplification Homogeneous Model[4] and conservatively over-predicts the pressure drop of expansion and contraction at two-phase conditions over the quality ranges of interest for PWR applications. Thus, the effect of localized hydraulic mismatches would be accentuated at two-phase conditions.

[+(a,c)

]

Also, for your information, the static pressure distributions of a full core of 17x17 STD and 17x17 OFA is presented in Figures 5 and 6. These figures were calculated isothermally. A representative best estimate flowrate was used. The static pressure distribution values for a transition core would be interpolated between the values on the two figures at each axial position as a function of the number of each assembly type in the core.

It is the position of Westinghouse to analyze 17x17 transition cores in the +(a.c)

]

REFERENCES

1. H. Chelemer, et.al., "THINC IV-An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

2. H. Chelemer, et.al., "Improved Thermal Design Procedure," WCAP-8567, July, 1975.

3. F. E. Motley, et.al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762, July 1976.

4. E. H. Novendstern, and R. O. Sandberg, "Single Phase, Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850, April, 1966.

5. W. L. Owens, Jr., "Two-Phase Pressure Gradient," in "International Developments in Heat Transfer," Part II, pp. 363-8, American Society of Mechanical Engineers, New York, 1961.

FIGURE 1

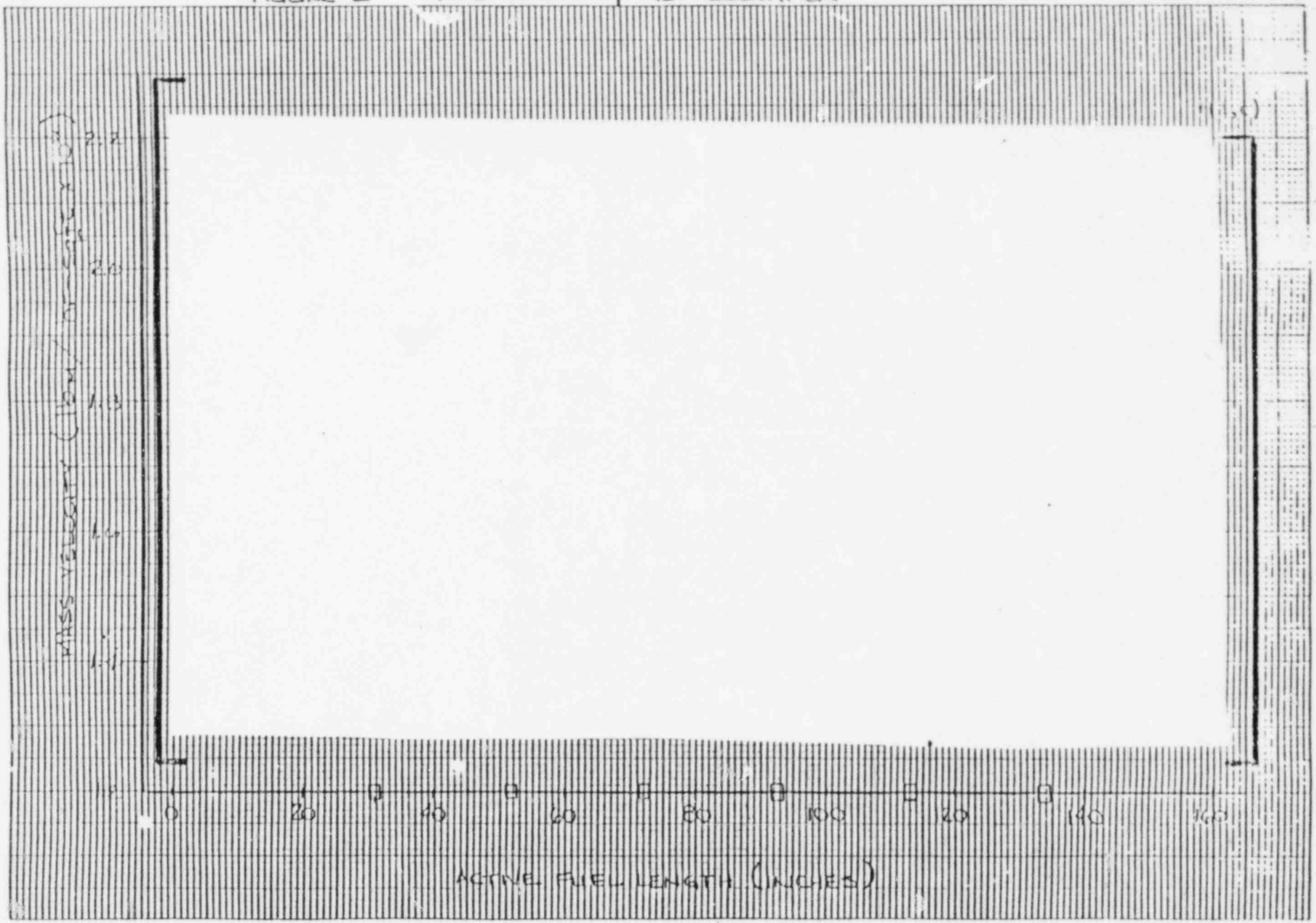
TRANSITION PATTERN

+(a,c)

STD - STANDARD FUEL ASSEMBLY

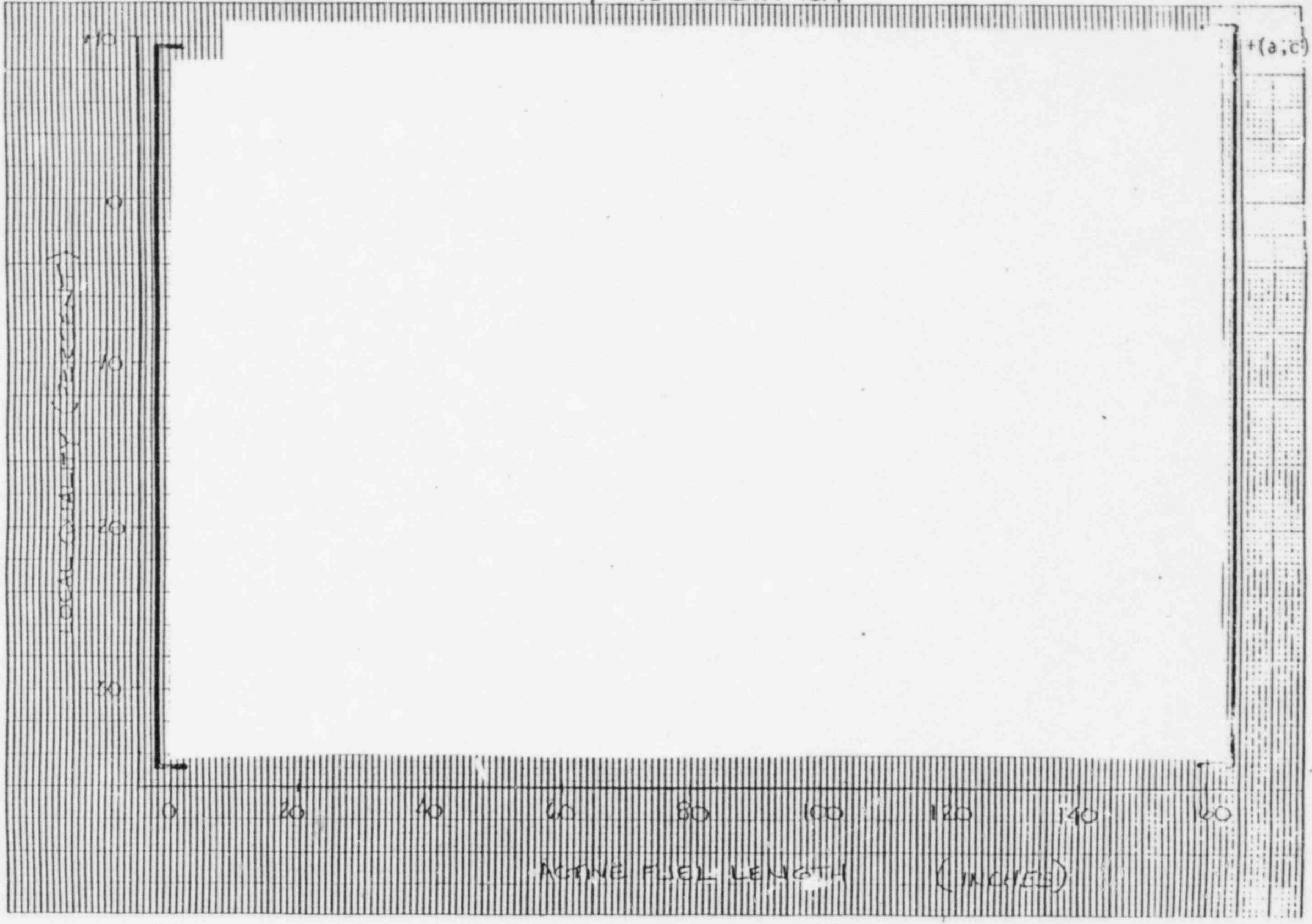
OFA - OPTIMIZED FUEL ASSEMBLY

FIGURE 2 MASS VELOCITY VS. ELEVATION



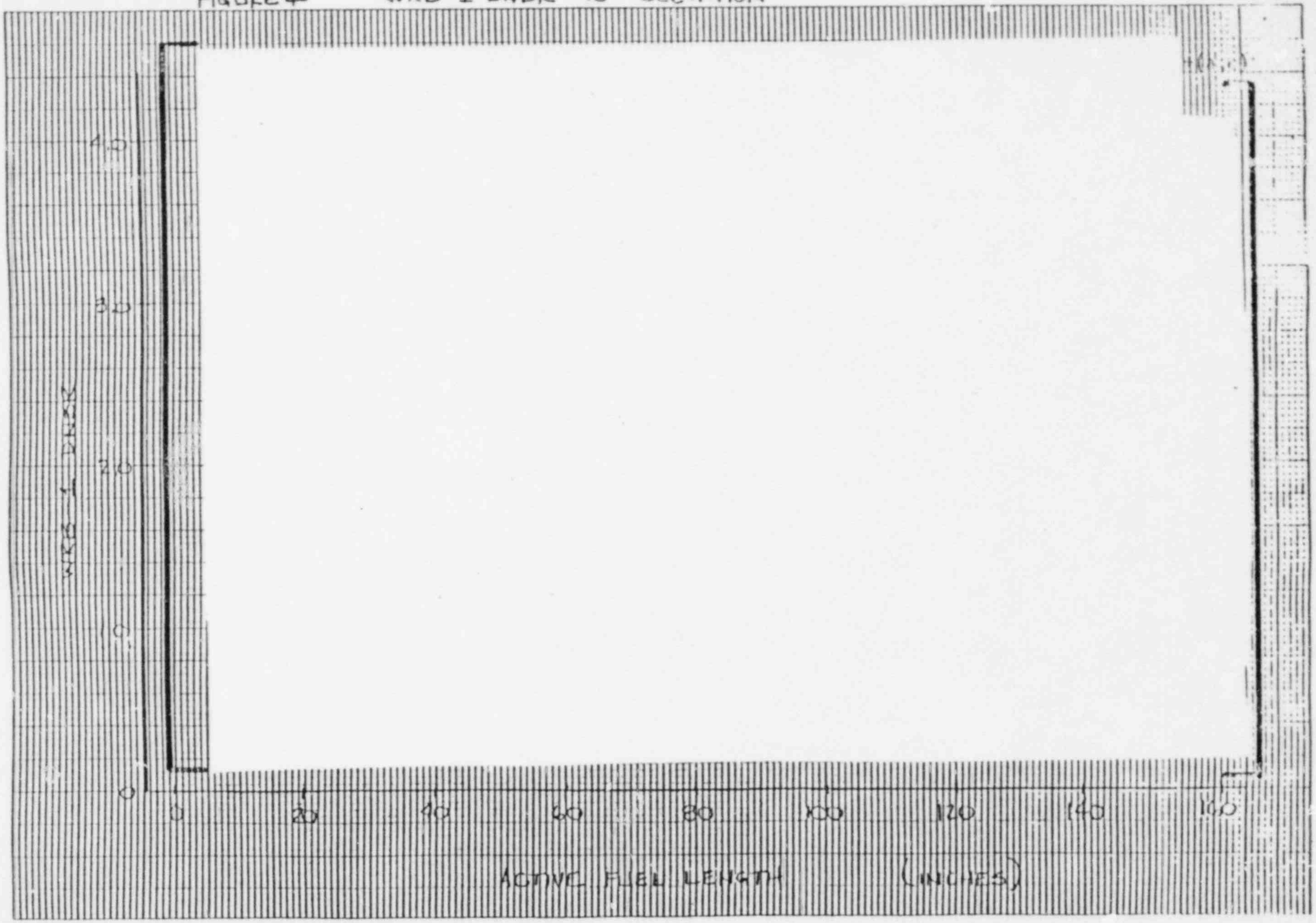
TLR 3/1/82

FIGURE 3 LOCAL QUALITY VS. ELEVATION



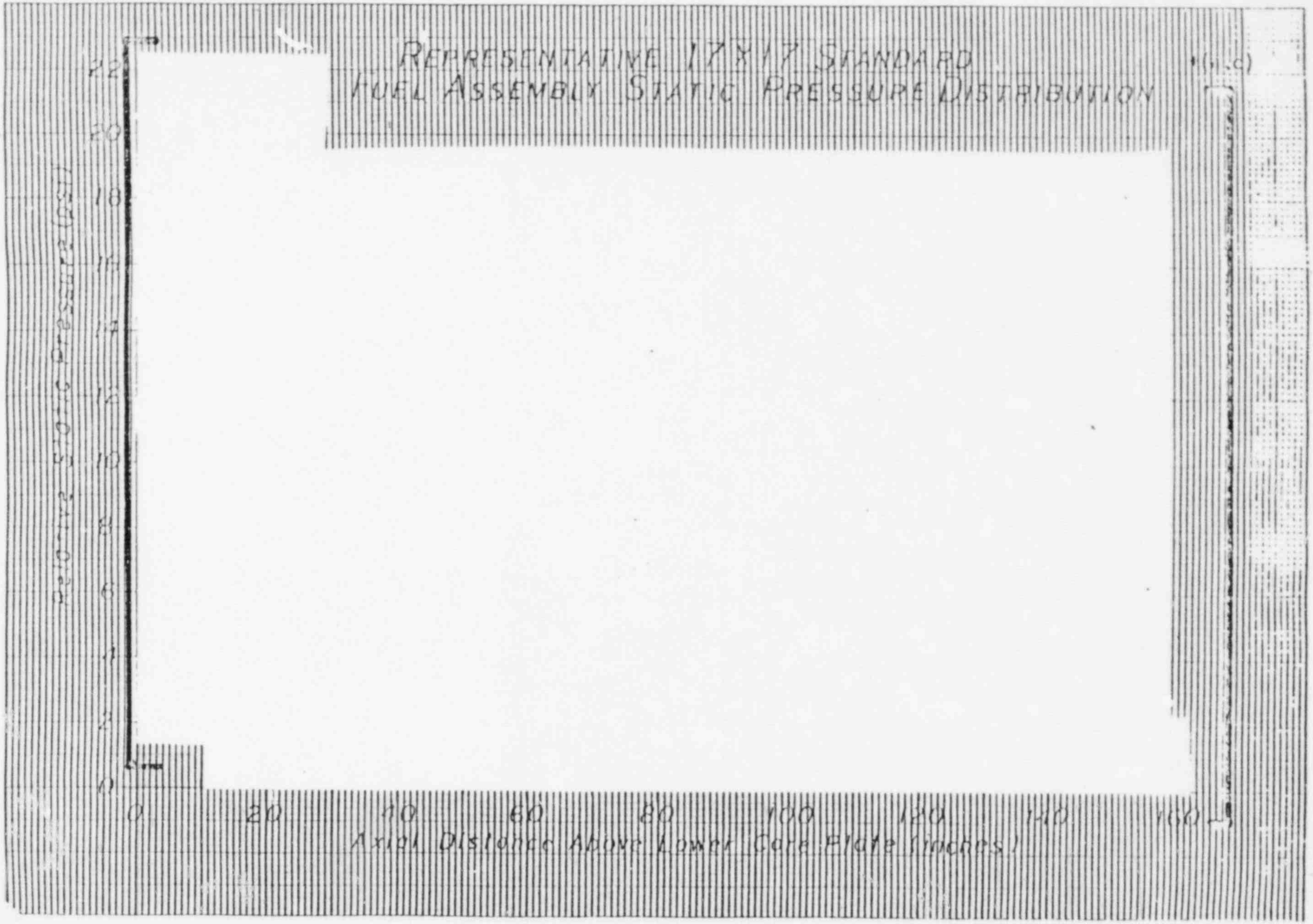
TLR 3/1/82

FIGURE 4 WRB-1 DNBR VS ELEVATION



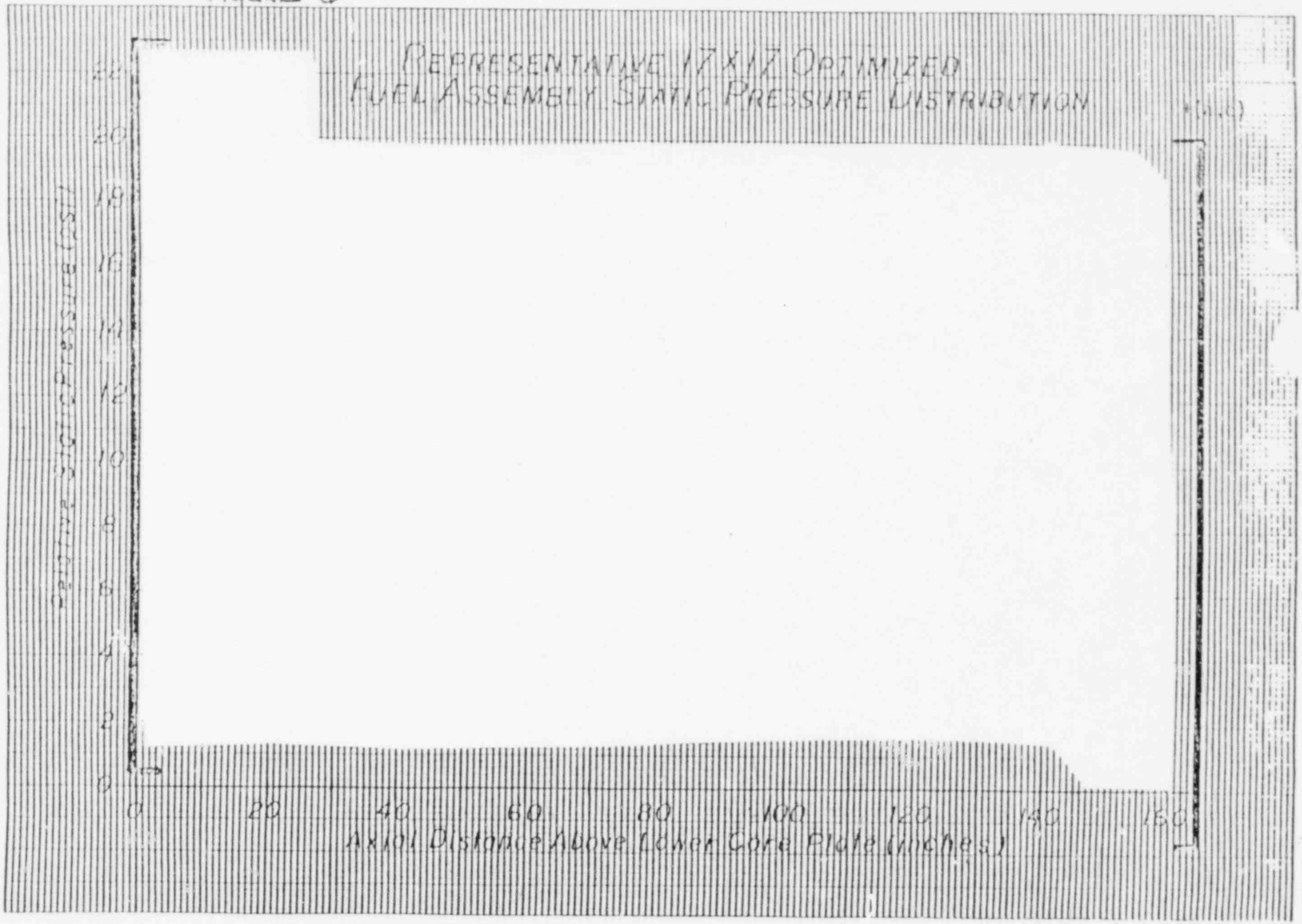
TLR 3/4/82

FIGURE 5



REPRESENTATIVE 17 X 17 STANDARD
FUEL ASSEMBLY STATIC PRESSURE DISTRIBUTION
(a,c)
Relative Static Pressure (psi)
0
2
4
6
8
10
12
14
16
18
20
22
0
20
40
60
80
100
120
140
160
Axial Distance Above Lower Core Plate (inches)

FIGURE 6


REPRESENTATIVE 17 X 17 OPTIMIZED
FUEL ASSEMBLY STATIC PRESSURE DISTRIBUTION
0
2
4
6
8
10
12
14
16
18
20
22
0
20
40
60
80
100
120
140
160
Axial Distance Above Lower Core Plate (inches)

RESPONSE TO SER ITEM 2

FUEL ASSEMBLY RESPONSE TO SEISMIC AND LOCA FORCES FOR WESTINGHOUSE STANDARD, MIXED AND OFA CORES

This discussion addresses the subject of seismic/LOCA analysis results for standard, mixed and optimized fuel assembly (OFA) cores for those plants which contain combined seismic and LOCA events in their licensing basis. It is in response to item 7 on page 3 of reference (1), i.e., "A determination that the appropriate seismic and LOCA forces are bounded by the cases considered in WCAP-9401 or additional analysis."

Reference (2) stated that, "For mixed cores of Westinghouse standard and optimized fuel, we are currently performing analysis to further demonstrate generically that analysis for a full core of standard fuel and a full core of optimized fuel bound all mixed core combinations". This has been demonstrated to be so in the case of all the standard core [

]+(a,c)

Based upon the results of post irradiation examination of Westinghouse demonstration fuel assemblies which underwent two cycles of irradiation, it was determined by [

]+(a,b,c)

The attached table shows the percentage of allowable crushing strength that the grids are subjected to, due to the combined seismic and LOCA loadings. The analysis takes into account the [
]. +(a,c) Five cases are considered and are illustrated in Figure 1. These cases start with a core composed of all standard fuel assemblies and continue through the various mixed reload patterns to a core composed of all optimized fuel assemblies (OFA's). It can be seen that in all cases the allowable crush strength is not exceeded.

The analysis presented here is based on 17x17 OFA and standard fuel assemblies and will bound all plants that fall within the applied force envelope as presented in WCAP-9401. Other 17x17 fueled plants that do not fall within this envelope will require plant specific evaluations or analysis dependent upon the relative similarity to other plants already analyzed. Results would be presented as part of a plant specific application for introduction of optimized fuel.

The incremental increase in applied grid loads resulting from the transition to optimized fuel is on the whole relatively small. As indicated by our earlier letter and the topicals submitted therewith, WCAPs 9283 and 9558, there is substantial technical basis for concluding these loads should not be combined.

Summary of Grid Load Response Results

Case	Reactor Core	Max. Grid Impact Force as % of Grid Impact Strength			
		SRSS Load Combination		SRSS Load Combination (w/1.3 LOCA)	
		OFA	Std. FA	OFA	Std. FA
1	Homogeneous Standard FA				+(a,c)
2	Transition 2/3 Std. & 1/3 OFA				
3	Transition 2/3 Std. & 1/3 OFA				
4	Transition 1/3 Std. & 2/3 OFA				
5	Homogeneous OFA				

References

1. Letter; R. L. Tedesco, NRC to T. M. Anderson, W, "Acceptance----WCAP-9500", May 22, 1981.

2. Letter; E. P. Rahe, W to L. S. Rubenstein, NRC, "WCAP-9500/9401/9402---", NS-EPR-2498, August 11, 1981.

Core Barrel

Core Barrel

CASE	FA NO. 1 2 3 4 5 10 11 12 13 14 15	CORE
	+(a,c)	
1		Homogeneous Standard FA
2		Transition Core 1/3 OFA, 2/3 Std OFA
3		Transition Core 1/3 OFA, 2/3 Std OFA
4		Transition Core 2/3 OFA, 1/3 Std OFA
5		Homogeneous Optimized FA

FIG 1 REACTOR CORE MODELS

Legend:

S - Standard Fuel Assembly
O - Optimized Fuel Assembly

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2643

August 17, 1982

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

SUBJECT: Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information

REFERENCE: Westinghouse Letter No. NS-EPR-2573 (E. P. Rahe to J. R. Miller) dated March 19, 1982

ATTENTION: Mr. Laurence E. Phillips, Core Performance Branch

Dear Mr. Miller:

Enclosed are:

1. Twenty-five (25) copies of WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatibility Items - Supplemental Information (Proprietary).

2. Fifteen (15) copies of WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatibility Items - Supplemental Information (Non-Proprietary version).

Also enclosed are:

1. One (1) copy of Application for Withholding, AW-82-48, (Non-Proprietary).

2. One (1) copy of original Affidavit (Non-Proprietary).

Enclosed is supplemental information to information submitted previously via the reference letter and responds to questions raised in recent telephone discussions held with members of the Core Performance Branch, Thermal/Hydraulic Section concerning the thermal/hydraulic performance of Westinghouse mixed OFA/Standard fuel cores.

It is our understanding that the enclosed information provides all remaining required information and, as such, should result in prompt resolution of this item.

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-82-48 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Enclosures

This paper describes the 17x17 Standard Fuel Assembly (STD) to the 17x17 Optimized Fuel Assembly (OFA) Thermal-hydraulic transition core methods, the resulting transition core DNBR penalty relative to a 17x17 OFA full core analysis and Westinghouse's position of the application of this penalty.

The 17x17 OFA has an [ ] relative to the 17x17 STD due +(a,c)
to heat flux and equivalent diameter effects. [+(a,c)

]

[+(a,c)

]

The analysis which determined the transition core DNBR penalty used the THINC IV[1] code. The configurations used were [+(a,c)

]

[+(a,c

]

Investigation was also done on the effect of differing rod diameters on the lateral friction factor. [+(a,c)

]

[+(a,c)

]

It should be noted that the THINC IV code uses the Novendstern-Sandberg axial friction factor correlation[4] which, under two-phase conditions, employs the homogeneous flow model proposed by Owens[5]. Also under two-phase conditions, the THINC IV code implicitly corrects the pressure drop at grid locations by using the bulk density rather than the saturated liquid density. This is equivalent to the APD Simplification Homogeneous Model[4] and conservatively over-predicts the pressure drop of expansion and contraction at two-phase conditions over the quality ranges of interest for PWR applications. Thus, the effect of localized hydraulic mismatches would be accentuated at two-phase conditions.

[+(a,c)

]

Also, for your information, the static pressure distributions of a full core of 17x17 STD and 17x17 OFA is presented in Figures 10 and 11. These figures were calculated isothermally. A respresentative best estimate flowrate was used. The static pressure distribution values for a transition core would be interpolated between the values on the two figures at each axial position as a function of the number of each assembly type in the core.

It is the position of Westinghouse to analyze 17x17 transition cores in the following manner: [+(a,c)

]

REFERENCES

1. H. Chelemer, et.al., "THINC IV-An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

2. H. Chelemer, et.al., "Improved Thermal Design Procedure," WCAP-8567, July, 1975.

3. F. E. Motley, et.al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762, July 1976.

4. E. H. Novendstern, and R. O. Sandberg, "Single Phase, Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850, April, 1966.

5. W. L. Owens, Jr., "Two-Phase Pressure Gradient," in "International Developments in Heat Transfer," Part II, pp. 363-8, American Society of Mechanical Engineers, New York, 1961.

FIGURE 1

TRANSITION PATTERN 1

+(a,c)

STD - STANDARD FUEL ASSEMBL

OFA - OPTIMIZED FUEL ASSEMBLY

FIGURE 2

TRANSITION PATTERN 2

+(a,c)

STD - STANDARD FUEL ASSEMBLY

OFA - OPTIMIZED FUEL ASSEMBLY

FIGURE 3

REPRESENTATIVE AXIAL POWER
DISTRIBUTION

+(a,c)

FIGURE 4
REPRESENTATIVE AXIAL POWER
DISTRIBUTION

+(a,c)

FIGURE 5

REPRESENTATIVE AXIAL POWER
DISTRIBUTION

+(a,c)

FIGURE 6
REPRESENTATIVE AXIAL POWER
DISTRIBUTION

+(a,c)

FIGURE 7 MASS VELOCITY VS. ELEVATION

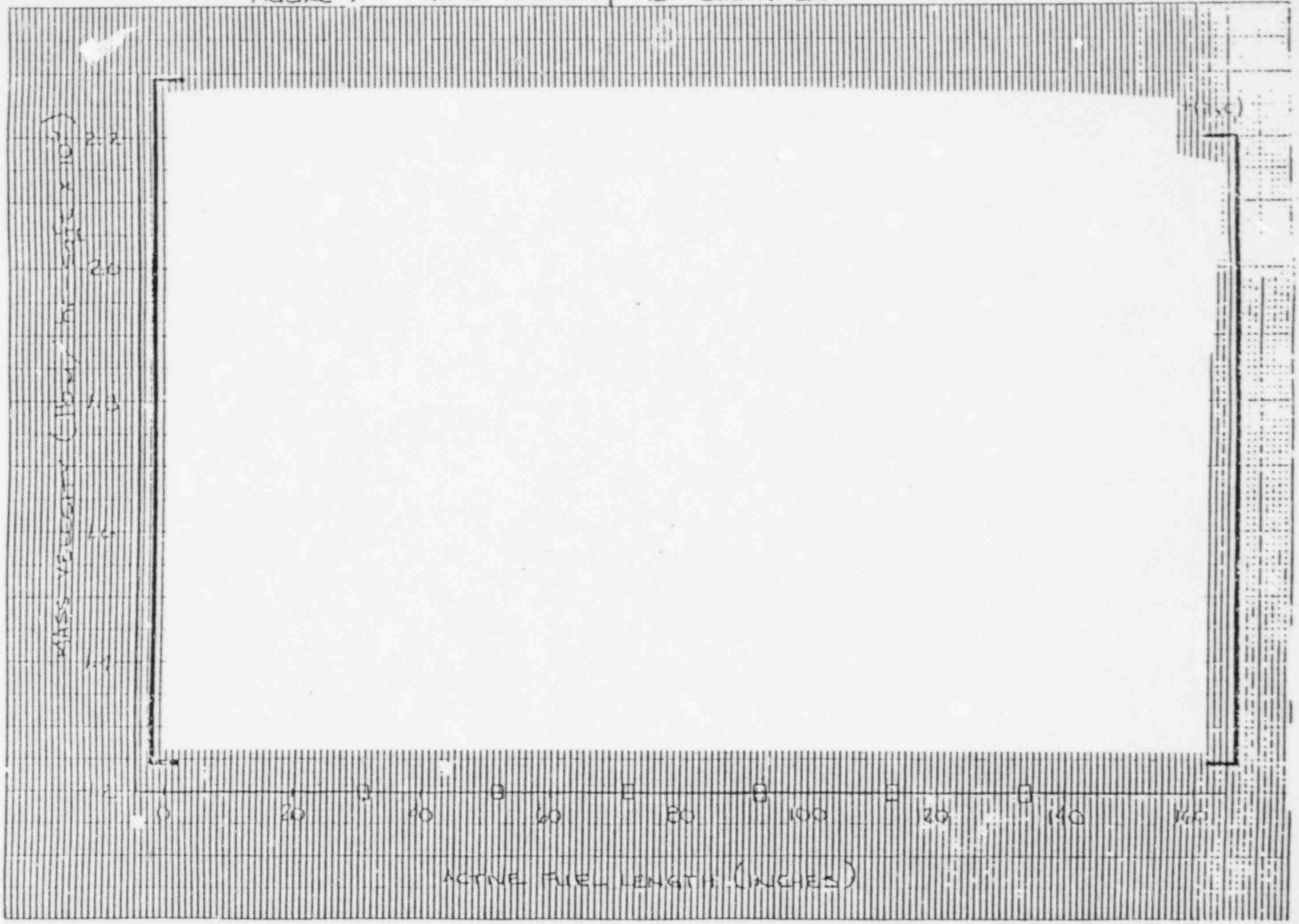


FIGURE 8 LOCAL QUALITY VS. ELEVATION

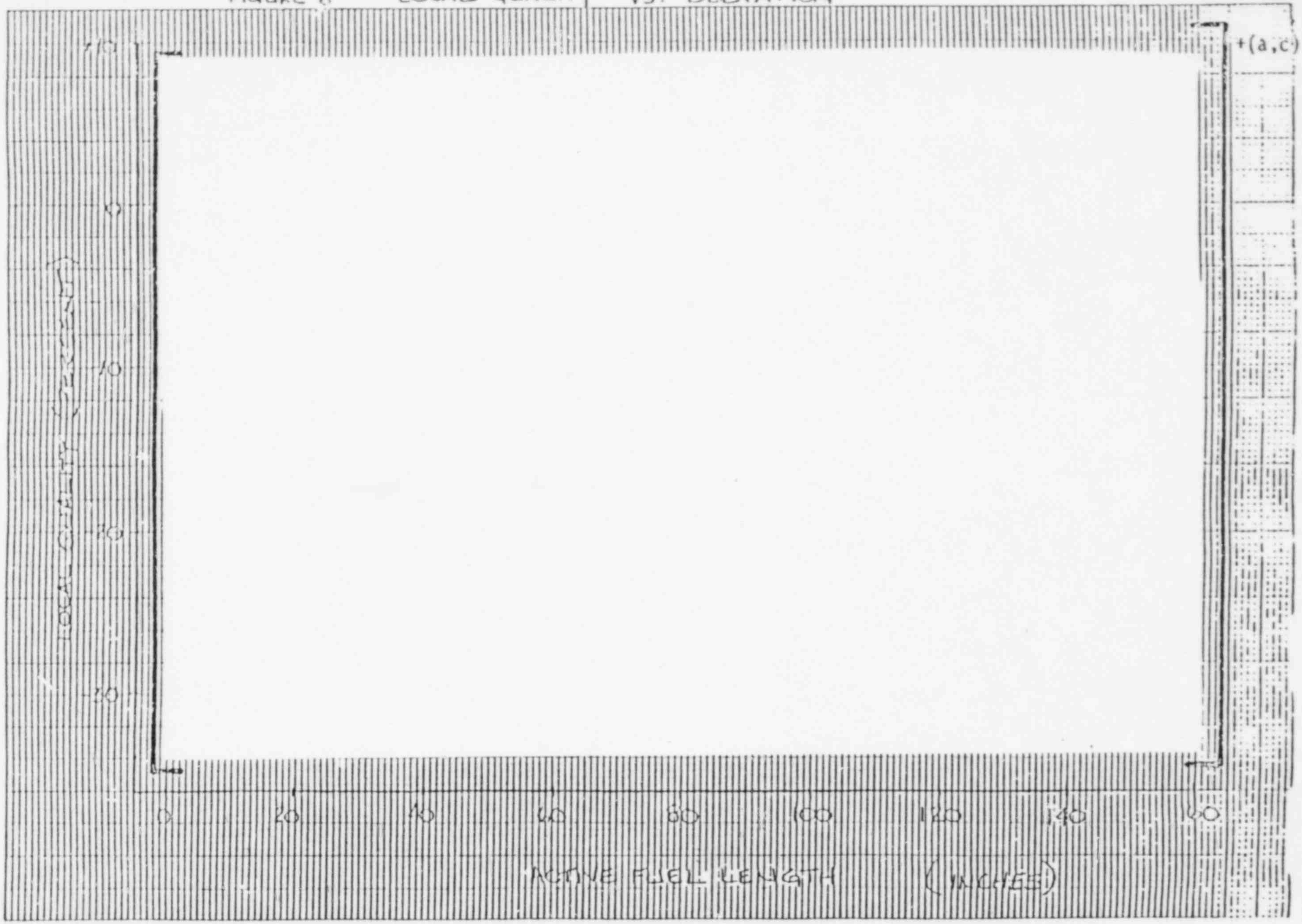


FIGURE 9 WRB-1 DNBR VS ELEVATION

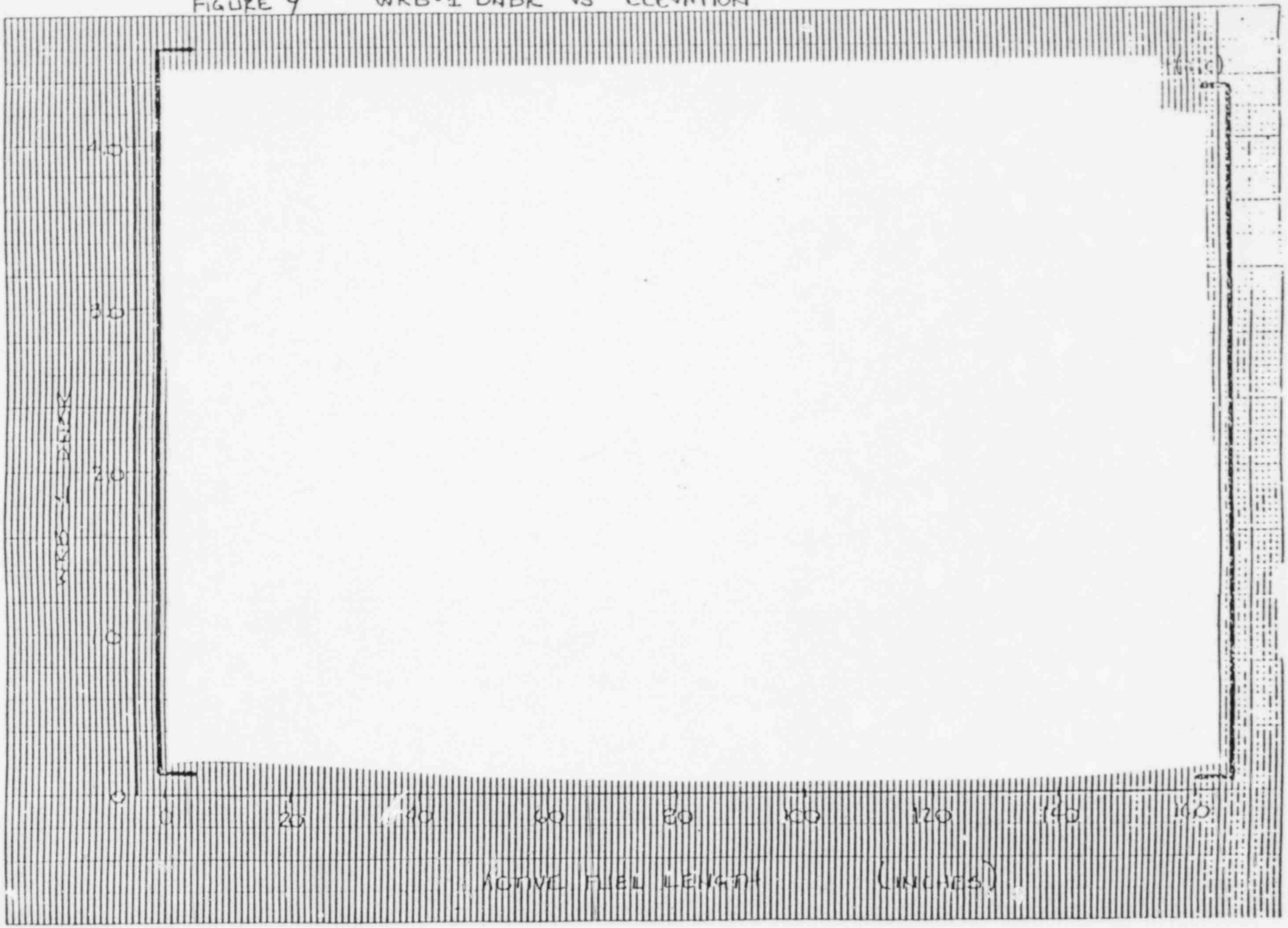
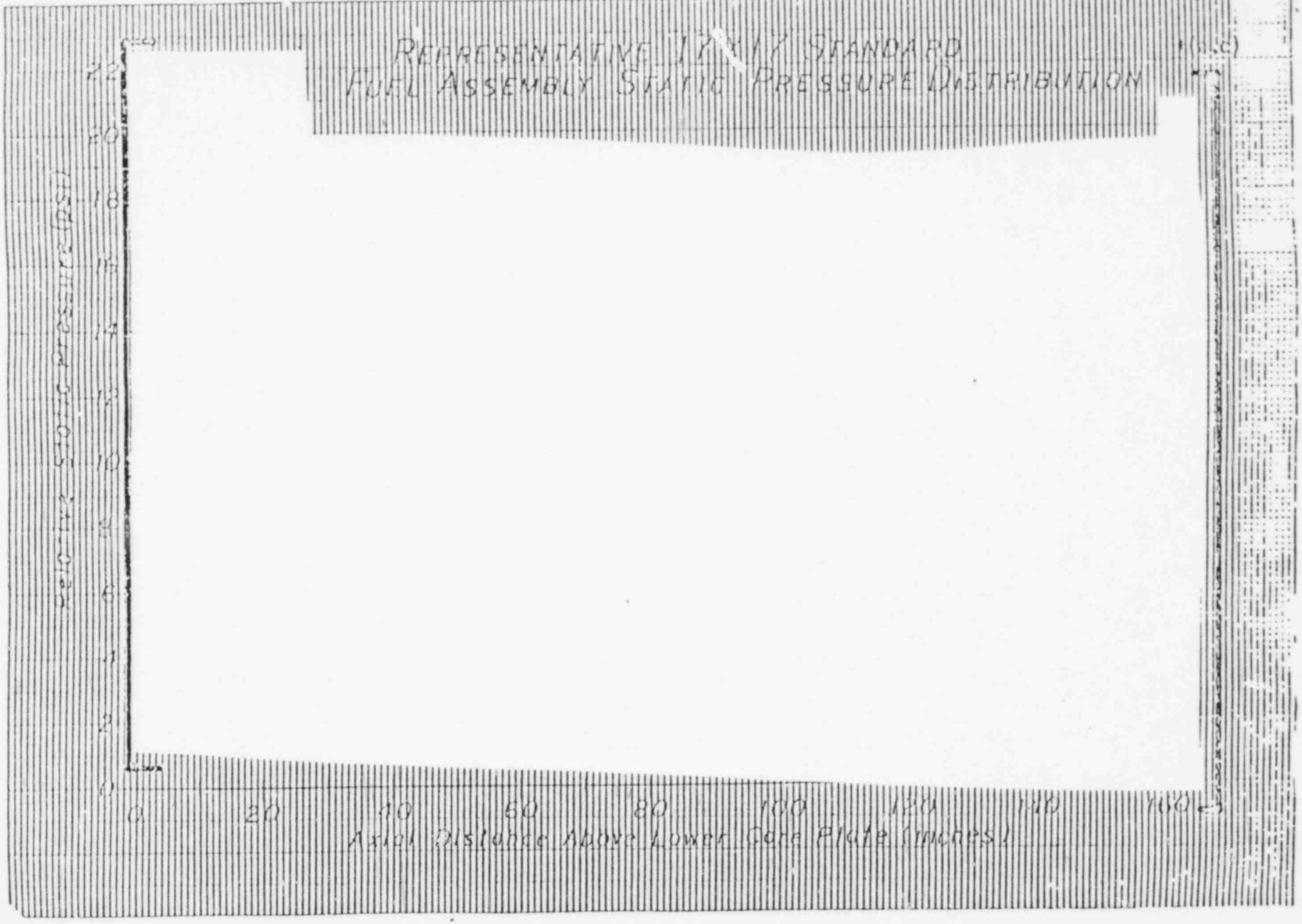
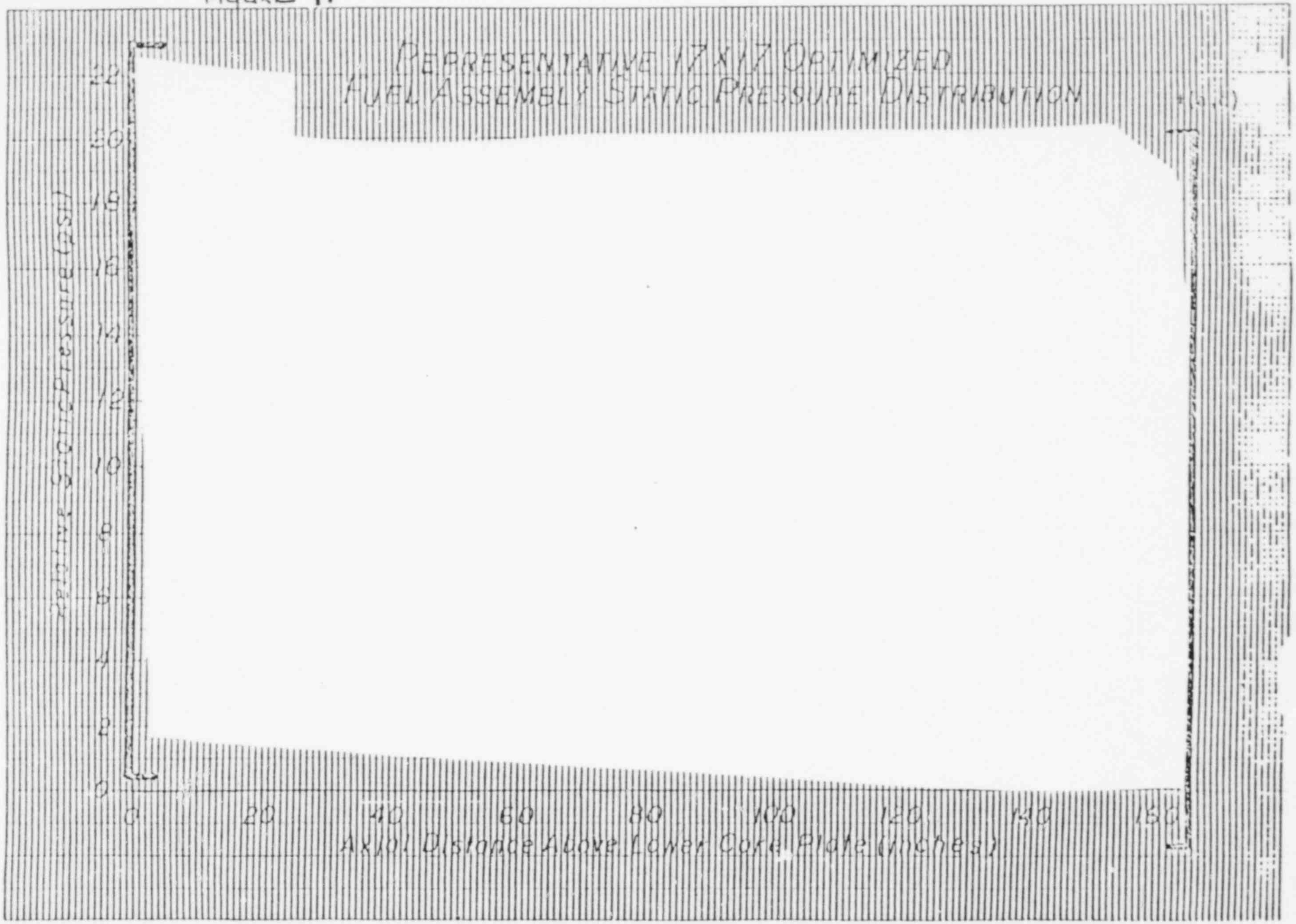


FIGURE 10



Representative 17x17 Standard
Fuel Assembly Static Pressure Distribution
(a,c)
0
2
4
6
8
10
12
14
16
18
20
22
0
20
40
60
80
100
120
140
160
Axial Distance Above Lower Core Plate (inches)

FIGURE 11



REPRESENTATIVE 17 X 17 OPTIMIZED
FUEL ASSEMBLY STATIC PRESSURE DISTRIBUTION
Relative Static Pressure (psi)
0
2
4
6
8
10
12
14
16
18
20
22
Axial Distance Above Lower Core Plate (inches)
0
20
40
60
80
100
120
140
160

TABLE 1

RUNS MADE TO JUSTIFY TRANSITION CORE METHODS

+(a,c)

Run	Configuration (Figure No.)	Pressure (psia)	Inlet Temperature (°F)	POWER (% of 17.7 MWt/Assy)	FLOW (% of 1750 $\frac{gpm}{assy}$)	Axial Power Distribution (Figure No.)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						
26						

TABLE 2 - COMPARISONS

Type	Runs	ΔDNBR (%)

+(a,c)