NRC PDR

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

DEC " 1979

MEMORANDUM FOR: Stephen H. Hanauer, Director
Unresolved Safety Issues Program

THRU: Ashok C. Thadani, Task Manager, A-9 Task

FROM: K. I. Parczewski, Reactor Safety Branch, DOR

SUBJECT: ATWS MEETING WITH CE AND CE PLANT OWNERS GROUP

A meeting was held in Bethesda on November 9, 1979, with Combustion Engineering (CE) and with the representatives of the CE plant owner's group to discuss the draft of the proposed report on ATWS in the CE plants.

The meeting was opened by the NRC staff outlining the purpose of the meeting and indicating the importance for establishing a firm schedule for issuing the report. This was followed by the presentation made by a representative of the CE plant owner's group who described the nature and the function of the group. He indicated that this group is separate from the group formed previously to handle the Bulletin and Order concerns and it consists of executives from participating utilities. He also stated that the ATWS report, although prepared by CE, will be reviewed and issued by the owner's group rather than by CE. The NRC staff agreed with this approach. Following these introductory remarks, CE outlined the content of the report and discussed in more detail the material presented in its different sections.

(1) The report will address the BIN #1 issues raised in Mattson's letter of February 15, 1979, and will describe the early verification efforts required by Alternative 3 of NUREG-0460, Vol. 3. The material provided in the report will fulfill the commitments made by CE during the August 17, 1979 meeting (Slide 2). The CE plants will be grouped in three classes and the report will address generically ATWS concerns for each individual class. It will be demonstrated that during ATWS structural integrity of the primary coolant pressure boundary (PCPB) and the functionability of individual components will be maintained. Also the leakage through the reactor vessel "O" ring seal will be evaluated. The report will also contain a discussion of the conformance of the CE plants to the ATWS criteria other than RCS pressure and will

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describe the systems used for ATWS mitigation. The evaluation presented in the report will show that the requirements of NUREG-0460, Alternative 3 are met by the operating CE plants. CE has indicated that some of the information presented at the meeting is of the proprietary nature and should be treated as such. Therefore the proprietary data have been deleted from the enclosure 2 copy of the slides presented at the meeting. These proprietary slides are included as enclosure 3, which is to be withheld from the PDR.

(2) The following criteria used in evaluating post-ATWS condition will be defined in the report (Slide 4).

(a) PCPB integrity and functionability for cooldown
(b) Radiological releases smaller than 10 CFR 100 limits
(c) Maintenance of coolable geometry
(d) Peak fuel enthalpy less than 280 cal/gm
(e) Low probability of DNB or limited clad degradation
(f) Containment pressure smaller than design

The staff questioned use of the 280 cal/gm enthalpy limit as a criterion for fuel failure, however, they agreed with CE that the DNB criterion is overconservative.

(3) The report defined the initial conditions and assumptions used in the evaluation (Slide 5). They are:

(a) Nominal initial condition
(b) Most positive moderator temperature coefficient (MTC) for 95% of core life. This is going to be shown by parametric study performed for individual plants.
(c) Automatic auxiliary feedwater actuation as a mitigating feature
(d) Reactor vessel flange "O" ring seal leakage. This leakage will be evaluated for different classes of plants.
(e) Manual actuation of SIAS required for injection of boric acid solution needed for reactivity control.

(4) The responses to ATWS of plants in each class will be evaluated in the report (Slide 6). The evaluation will consist of the following steps:

(a) Analysis of the models used in the evaluation
(b) Analysis of complete loss of normal feedwater event
(c) Discussion of the Three Mile Island concerns related to ATWS

CE discussed different aspects of the complete loss of feedwater event analyzed in the report and illustrated it with the viewgraphs representing the change of pressurizer pressure and reactivity with time

(Slides 7 and 8) for 3800 MWt class of plants and the change of pressurizer peak pressure with total pressurizer relief area and with moderator temperature coefficient for all three classes of plants (Slides 9, 10, 11, 12, 13 and 14). CE analyses show that some negative reactivity was introduced even before boric acid injection due to the increase in primary coolant temperature (decrease of moderator density) and the generation of voids in the core region due to boiling. The staff expressed its concern about the generation of voids which may raise problems similar to those in TMI. The staff suggested that this question should be addressed in a manner similar to the response to the Bulletin and Order Task Force request. CE responded that a proper analysis of this problem will be included in the report. The staff again asked CE to provide analyses of a stuck open PORV which would result in isolation signal actuation and a subsequent increase in the primary system pressure. The staff asked CE to pay close attention to the applicability of the codes to this type of ATWS event.

CE presented a list of TMI concerns which will be addressed in the report (Slide 15).

(a) Failure of PORV to open or close
(b) Loss of offsite power (LOOP)
(c) RC pumps operation and natural circulation
(d) Reflux boiling
(e) Boron precipitation
(f) Radiological releases
(g) Operator action

The staff expressed their interest in the pump performance at high pressure when a slight change in tolerances is expected and in pump cavitation, natural circulation and reflux boiling problems. They felt that these problems should be discussed in the report. In response, CE indicated that the last two effects are insignificant because of a relatively small loss of primary coolant inventory expected during ATWS event. This small inventory loss would also preclude boric acid precipitation. The staff also noted that the earlier LOOP ATWS analyses may be invalid because of change in MTC.

The report will demonstrate that the radiological releases are within the 10 CFR 100 criteria and that the operator action is not required immediately after the beginning of ATWS.

(5) CE told that the RCPB post-ATWS structural integrity will be demonstrated in the report (Slide 16). The report will describe the models used and then will provide the analyses of the response of RCPB components (Slides 17 and 20) to 4000 psia pressure transient. It will be demonstrated that in most cases level C and D criteria are satisfied. These analyses will include the reactor vessel shell (Slides 18 and 19). The staff questioned if RCPB components in all plants will be bounded by the analyses provided in the report. CE assured that the CE plant owners will verify the results for individual plants and will provide their findings to NRC. At the present moment, it is known that all piping, with the exception of the line leading to the letdown heat exchanger, meet the level D criteria.

(6) CE described the method of analysis used to evaluate the leakage which occurs at the reactor vessel "O" ring seal (Slides 21 and 22) and presented the results of the analyses. The analyses were performed for 2650 MWt and System 80 classes of plants. Analytical results discussed included the system pressure at which seal leakage initiates and the variation in leakage as a function of system pressure. CE noted that at all calculated ATWS pressures, even though there is leakage at the vessel flange seal, the closure bolts stress level is below the material yield strength.

The staff asked several questions on the hydraulic mechanism postulated in calculating the leakage flow and discussed the assumptions made by CE.

(7) CE summarized its presentation by stating that the analysis presented in the report will demonstrate that the criteria specified in NUREG-0460 for Alternative 3 plants are met by the CE plants.

At the conclusion of the meeting, the staff stated that the outline of the proposed ATWS report has indicated several new areas which would have to be carefully evaluated by the staff after the report is submitted for their review. Especially the sections of the report dealing with the discussion of models and codes and with stress considerations would have to be carefully reviewed. The staff also asked the owners to provide schedule for responses to the remaining questions in the February 15, 1979 Mattson letter.

It was agreed by CE and by the representatives of the owner's group that the report will be submitted to NRC by December 1, 1979.

An attendance list is provided in Enclosure 1.

K. I. Parczewski

K. I. Parczewski
Reactor Safety Branch
Division of Operating Reactors

Enclosure:
As stated

cc: ATWS Distribution
Meeting Attendees

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ENCLOSURE 1

MEETING ATTENDEES

ATWS EARLY VERIFICATION STATUS

NAME	REPRESENTING
R. C. L. Olson	CE Owners Group
Chris H. Poindexter	CE Owners Group
D. A. Kreps	CE
C. L. Kling	CE
W. E. Burchill	CE
C. R. Musick	CE
D. J. Ayres	CE
Dennis L. Terrill	TVA
K. I. Parczewski	NRC/DOR/RSB
F. Odar	NRC/DSS/AB
Medhat El-Zeftawy	NRC/SD
Francis Akstulewicz	NRC/DSE/AAB
M. Srinivasan	NRC/DSS/ICSB
F. C. Cherny	NRC/DSS/MEB
M. D. Stolzenberg	NRC/RES/RSB
D. K. James	FP&L (CE Owners Group)
S. H. Hanauer	NRC
H. VanderMolen	NRC/DOR/RSB
Kulin D. Desai	NRC/DSS/MEB

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Enclosure 2

Slide #1

AGENDA

C-E/NRC ATWS MEETING

NOVEMBER 8, 1979

INTRODUCTION	D. A. KREPS
ATWS REPORT OVERVIEW	C. L. KLING
TRANSIENT ANALYSES	C. L. KLING
RCPB ANALYSES	D. J. AYRES
O-RING SEAL LEAKAGE	D. J. AYRES
SUMMARY	C. L. KLING

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Slide #2.

ATWS ANALYSES: RESPONSE TO 2/15/79 NRC LETTER
CENPD - 263 (DRAFT)

1.0 INTRODUCTION

2.0 EVALUATION OF RESPONSE TO ATWS FOR EACH PLANT CLASS

3.0 DEMONSTRATION OF RCPB POST-ATWS STRUCTURAL INTEGRITY AND FUNCTIONABILITY FOR EACH PLANT CLASS

4.0 ASSURANCE OF CONFORMANCE TO ATWS CRITERIA OTHER THAN RCS PRESSURE

5.0 DESCRIPTION OF SYSTEMS USED FOR ATWS MITIGATION

6.0 SUMMARY AND CONCLUSIONS

APPENDIX A SUPPLEMENTARY PROTECTION SYSTEM

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INTRODUCTION

- BACKGROUND
- CRITERIA
- SPECIFICATION OF PLANT CLASSES
- PRINCIPAL PLANT PARAMETERS
- INITIAL CONDITIONS AND ASSUMPTIONS

Slide #4

ATWS CRITERIA

- FOR PEAK RCS PRESSURE DEMONSTRATE RCPB INTEGRITY AND FUNCTIONABILITY FOR COOLDOWN
- RADIOLOGICAL RELEASES LESS THAN 10CFR100 LIMITS
- MAINTAIN COOLABLE GEOMETRY
- PEAK FUEL ENTHALPY LESS THAN 280 CAL/GM
- LOW PROBABILITY OF DNB OR LIMITED CLAD DEGRADATION
- PEAK CONTAINMENT PRESSURE LESS THAN DESIGN

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INITIAL CONDITIONS AND ASSUMPTIONS

- NOMINAL INITIAL CONDITIONS
- MOST POSITIVE MTC FOR 95% OF CORE LIFE
- AUTOMATIC AUXILIARY FEEDWATER ACTUATION
- REACTOR VESSEL FLANGE O-RING SEAL LEAKAGE
- MANUAL SIAS AT 10 MINUTES

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Slide #6

EVALUATION OF RESPONSE TO ATWS FOR EACH PLANT CLASS

- ANALYSIS MODELS

- ANALYSIS OF COMPLETE LOSS OF NORMAL FEEDWATER

- THREE MILE ISLAND CONCERNS RELATED TO ATWS

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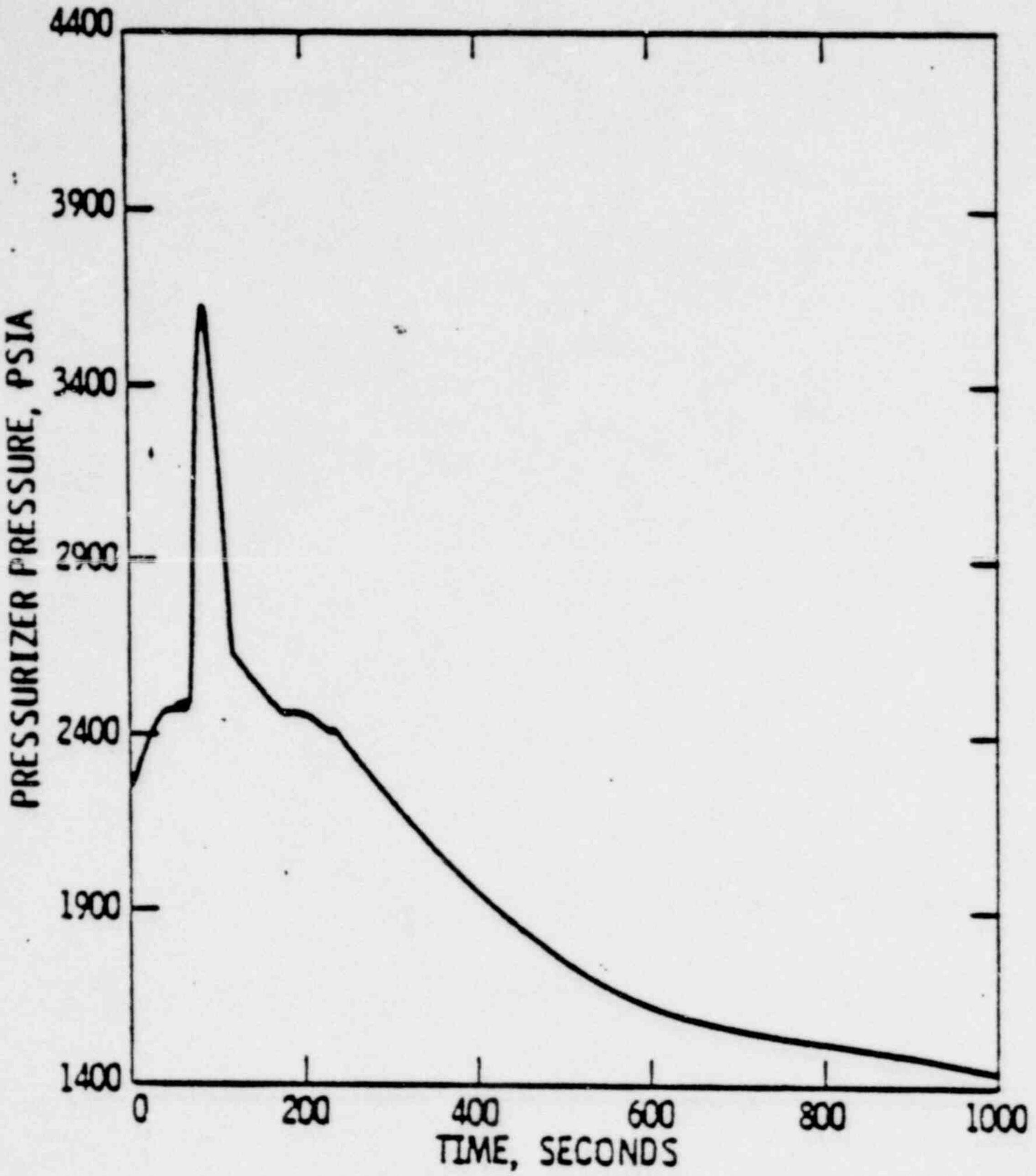
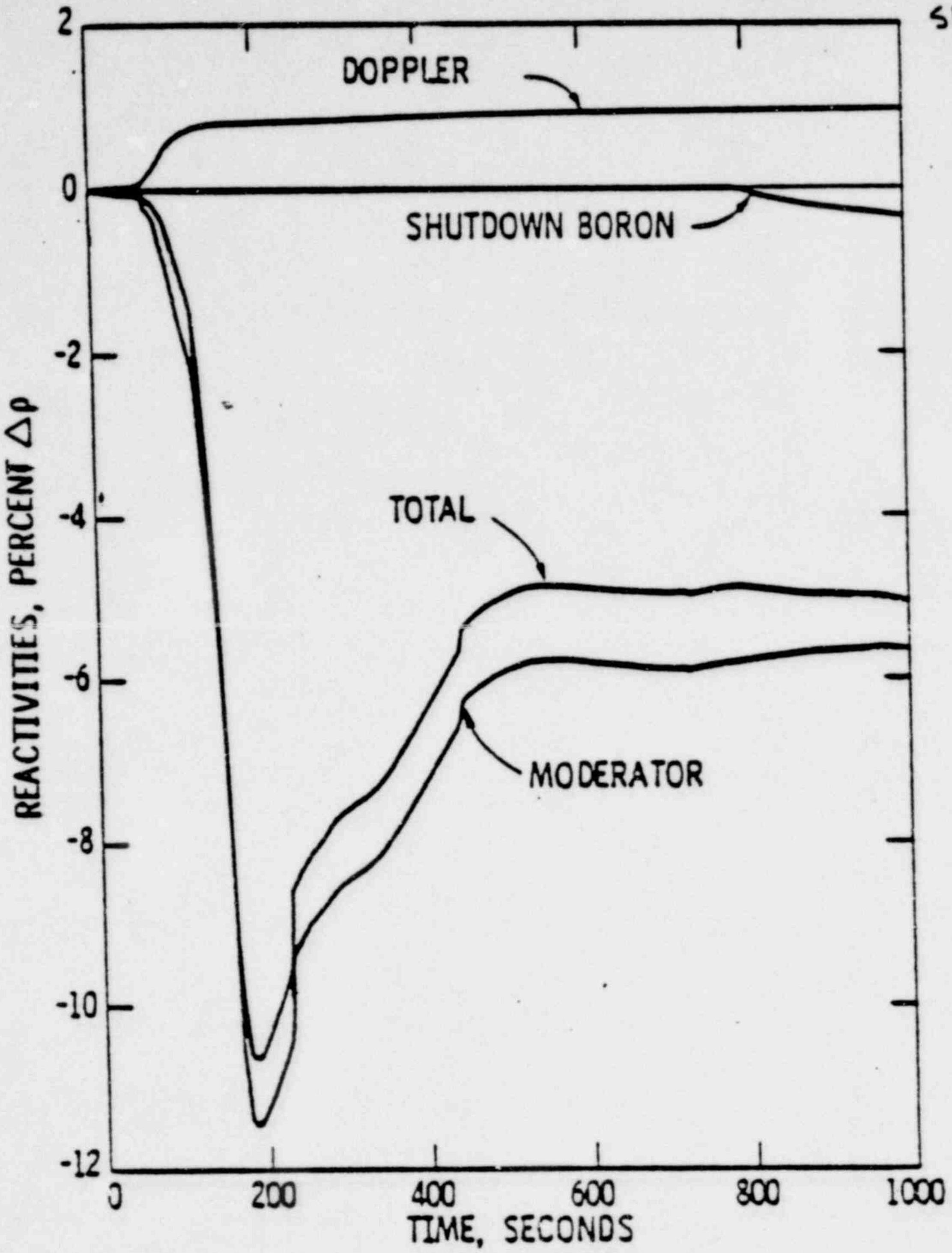


C-E ATWS GENERIC REPORT	PRESSURIZER PRESSURE TRANSIENT DURING 3800 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-3

Slide #8



C-E ATWS GENERIC REPORT	REACTIVITY COMPONENT TRANSIENTS DURING 3800 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-6

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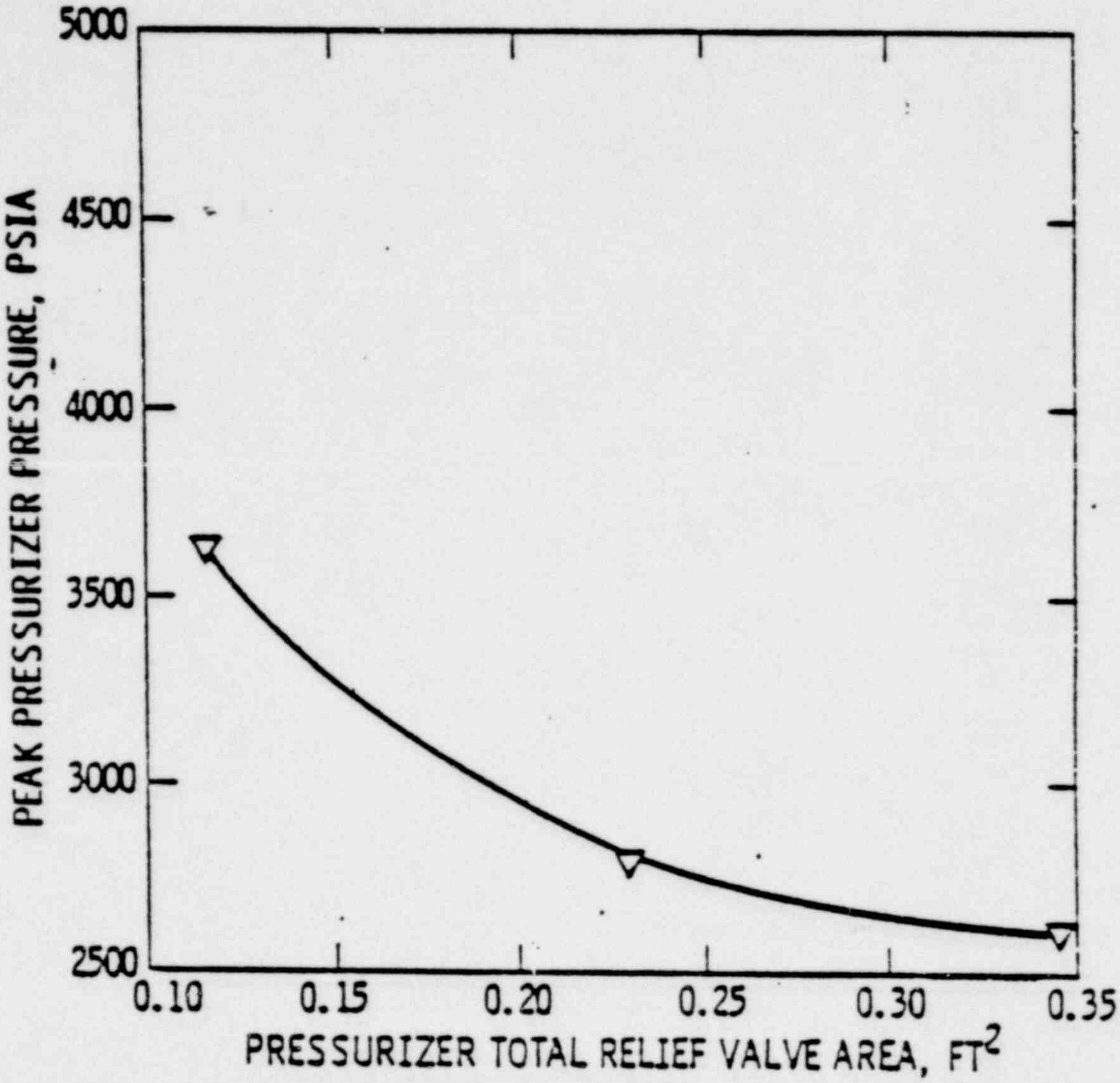
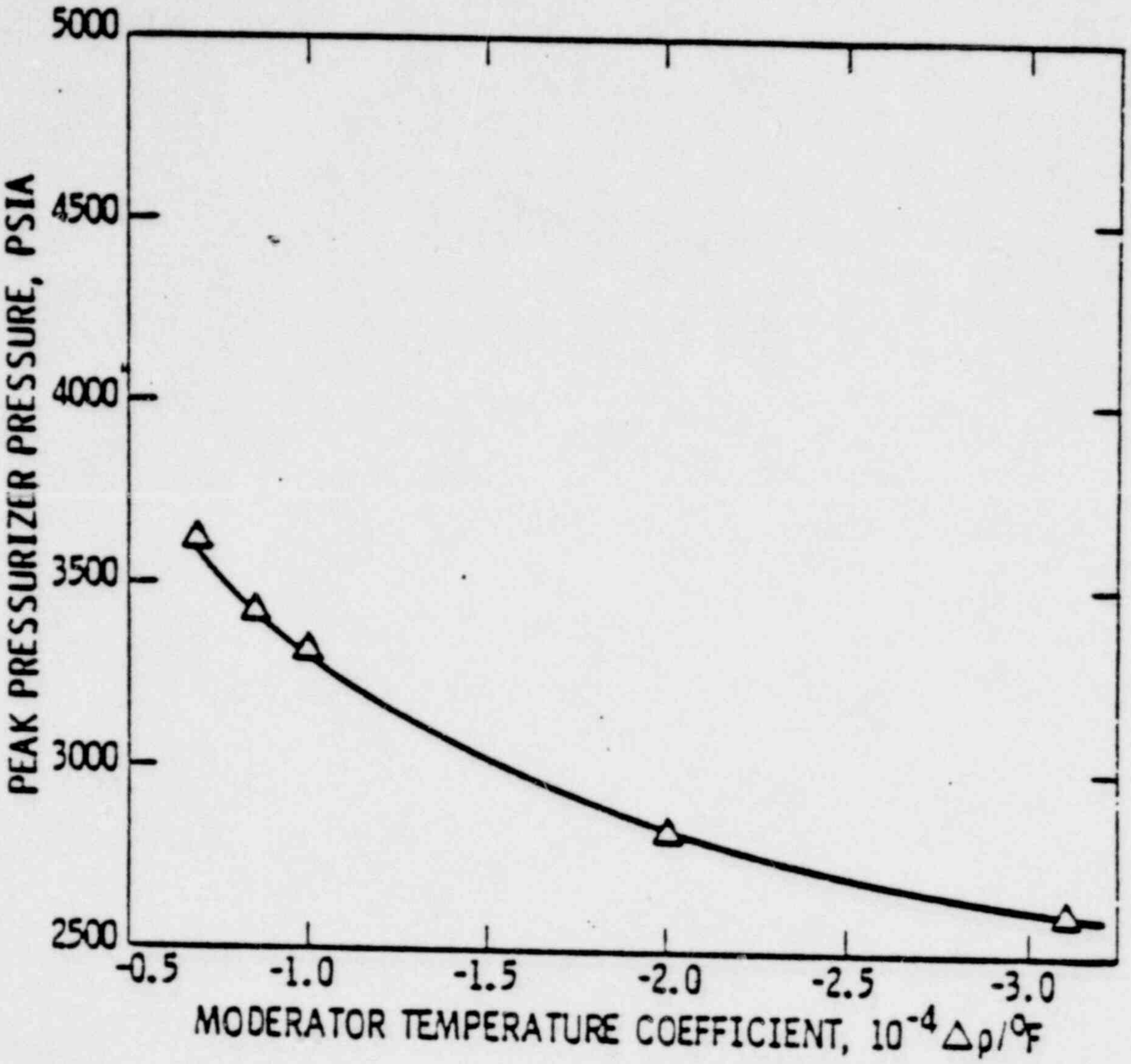
Slide # 9



C-E ATWS GENERIC REPORT	EFFECT OF PRESSURIZER RELIEF AREA ON PEAK PRESSURIZER PRESSURE DURING 3800 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-35

Slide # 10



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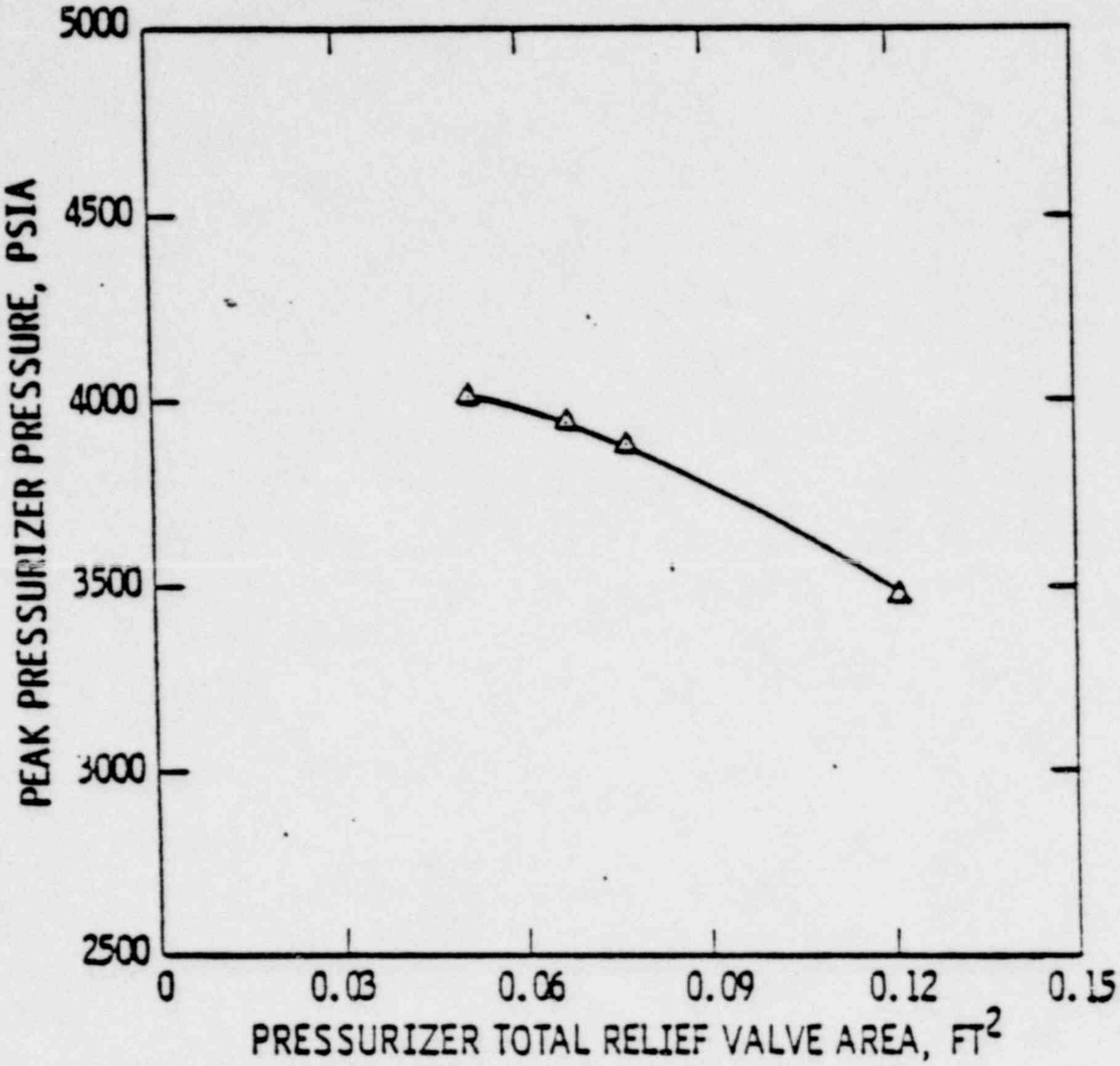
C-E ATWS GENERIC REPORT	EFFECT OF MODERATOR REACTIVITY FEEDBACK ON PEAK PRESSURIZER PRESSURE DURING 3800 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-32

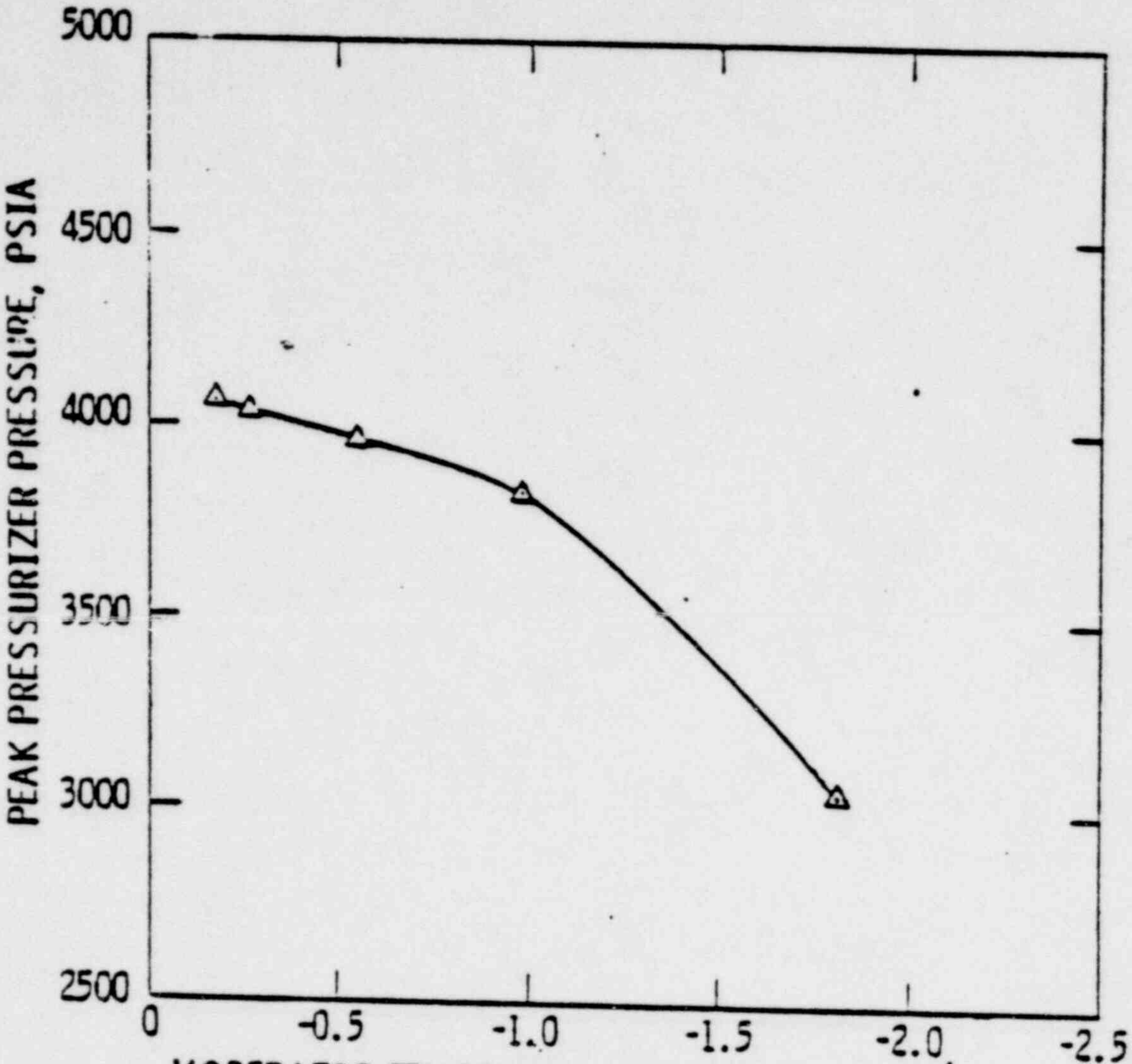
POOR ORIGINAL



C-E ATWS GENERIC REPORT	EFFECT OF PRESSURIZER RELIEF VALVE AREA ON PEAK PRESSURIZER PRESSURE DURING 2560 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-37

POOR ORIGINAL

Slide #12



C-E ATWS GENERIC REPORT	EFFECT OF MODERATOR REACTIVITY FEEDBACK ON PEAK PRESSURIZER PRESSURE DURING 2560 MWt COMPLETE LOSS OF FEEDWATER	Figure 2-34

POOR ORIGINAL

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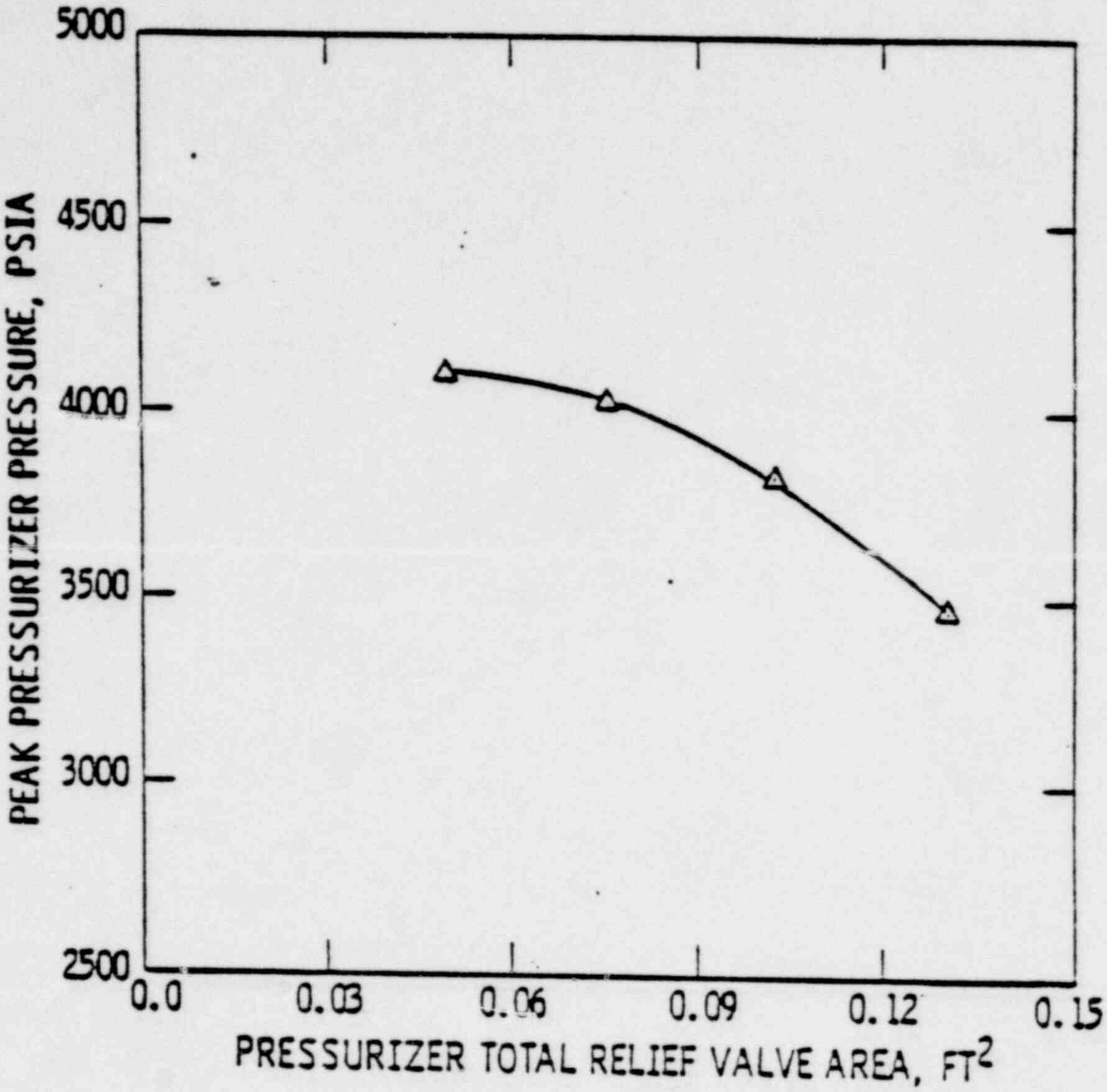
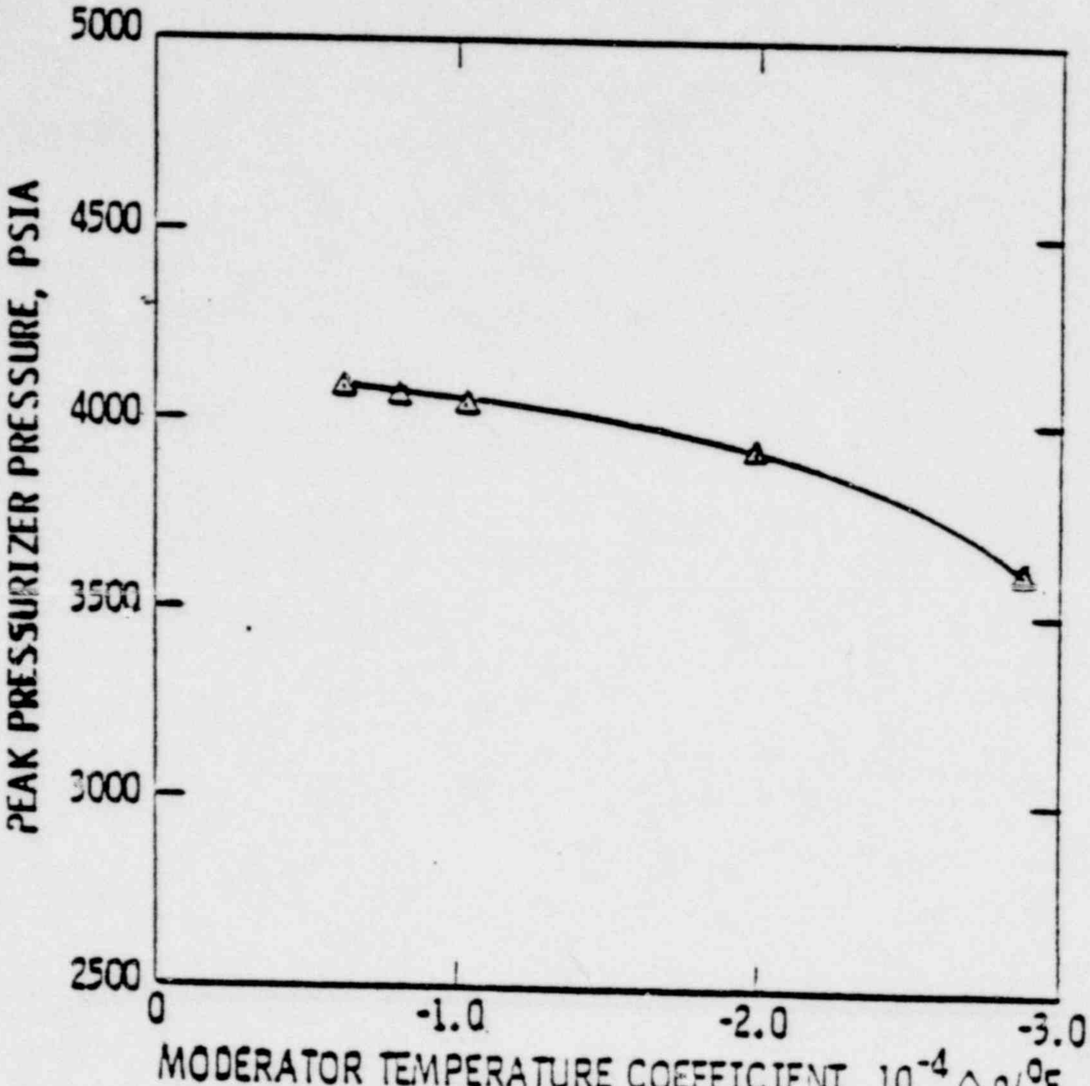


C-E ATWS GENERIC REPORT	EFFECT OF PRESSURIZER RELIEF VALVE AREA ON PEAK PRESSURIZER PRESSURE DURING 34XX MWt COMPLETE LOSS OF FEEDWATER	Figure 2-36

Slide# 14



C-E ATWS GENERIC REPORT	EFFECT OF MODERATOR REACTIVITY FEEDBACK ON PEAK PRESSURIZER PRESSURE DURING 34XX MWt COMPLETE LOSS OF FEEDWATER	Figure 2-33

POOR ORIGINAL

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Slide # 15

POST TMI CONCERNS ON ATWS

FAILURE OF PORV TO OPEN

FAILURE OF PORV TO CLOSE

LOSS OF OFFSITE POWER

RCP OPERATION AND NATURAL CIRCULATION

REFLUX BOILING

BORON PRECIPITATION

RADIOLOGICAL RELEASES

OPERATOR ACTION

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Slide # 16

DEMONSTRATION OF RCPB POST-ATWS
STRUCTURAL INTEGRITY AND FUNCTIONABILITY
FOR EACH PLANT CLASS

- INTRODUCTION
- METHOD OF ANALYSIS
- ASME III ANALYSIS OF RCPB COMPONENTS
- VESSEL FLANGE "O" RING SEAL LEAKAGE

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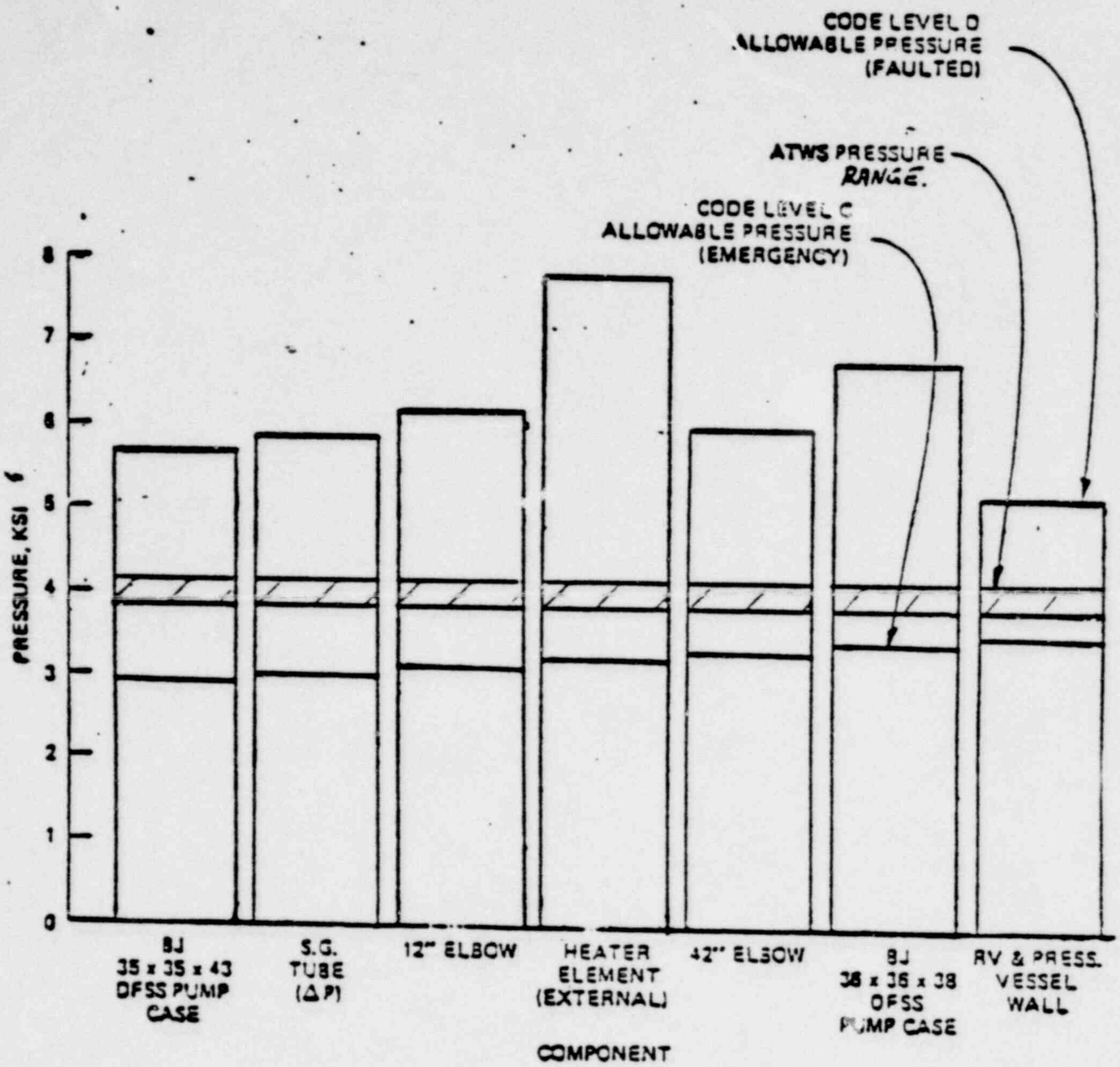
Slide # 17



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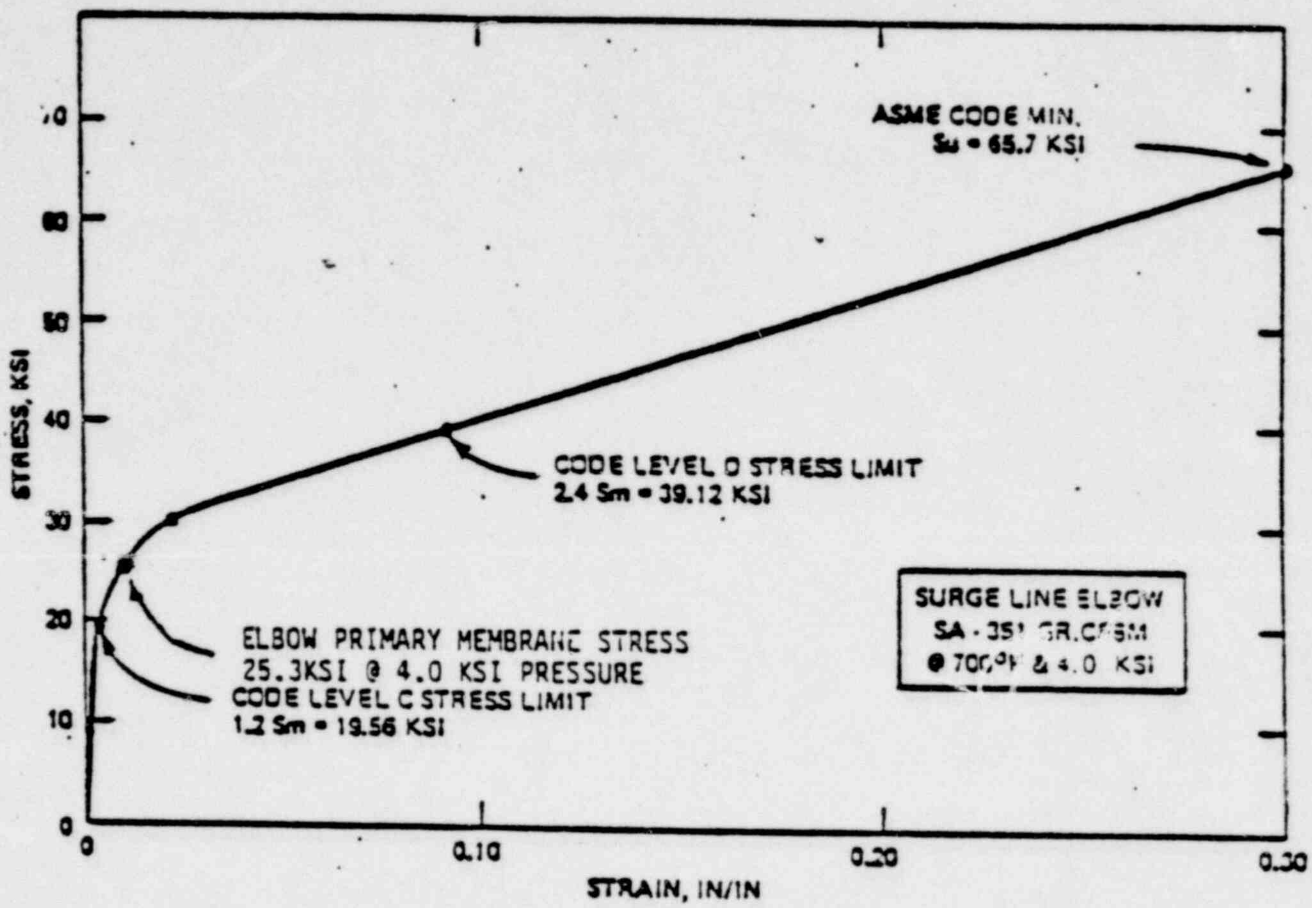
C-E ATWS GENERIC REPORT	PRIMARY SYSTEM COMPONENT LOADS COMPARED TO CODE LEVEL C AND D STRESSES	Figure 3-2

Slide # 18



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C-E ATWS GENERIC REPORT	STRESS STRAIN CURVE FOR REACTOR VESSEL SHELL	Figure 3-1

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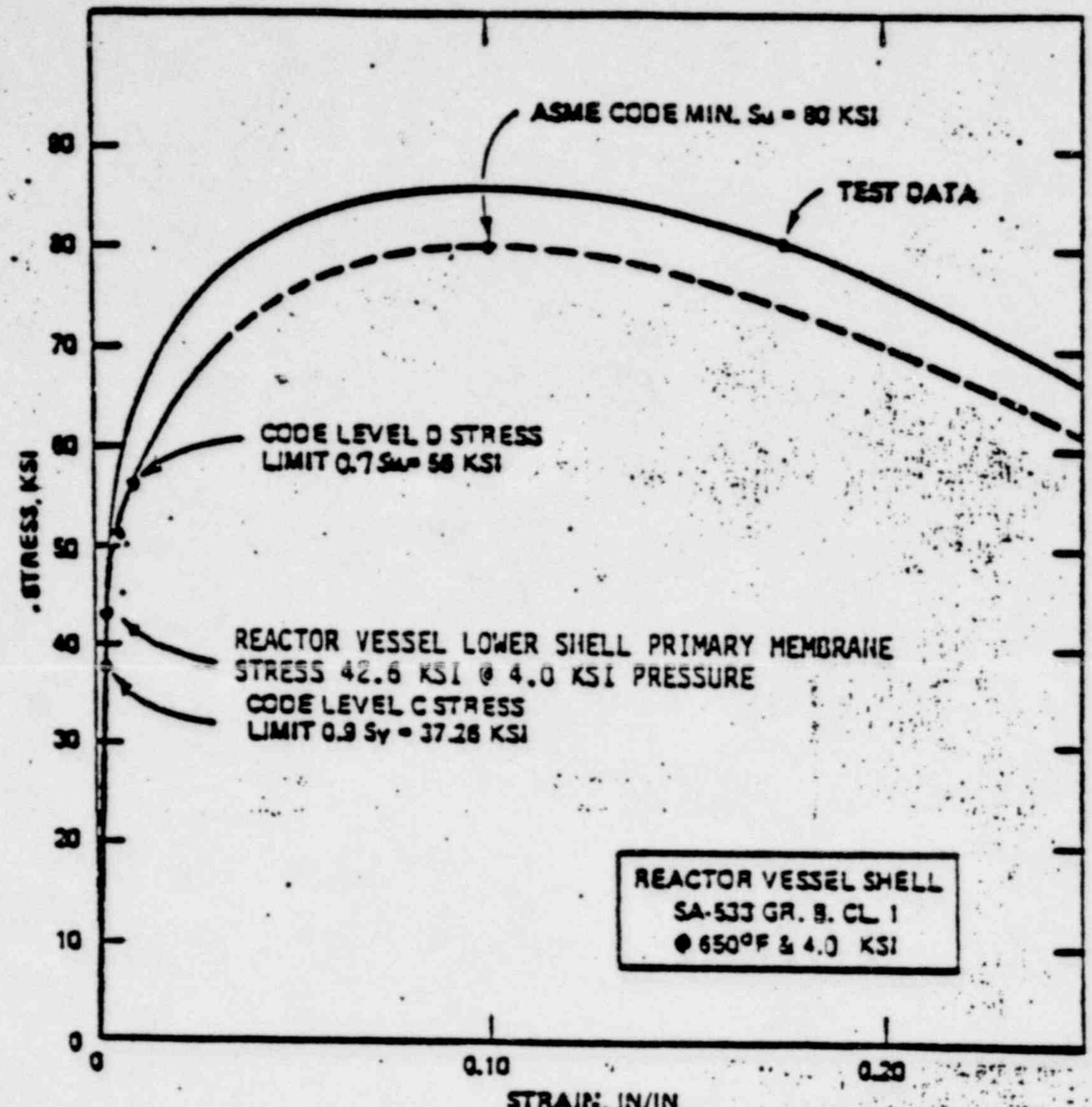
Slide #19



POOR ORIGINAL

C-E ATWS GENERIC REPORT	STRESS STRAIN CURVE FOR REACTOR VESSEL SHELL	Figure 3-3

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Slide #20

SUMMARY OF ATWS EVALUATION AT 4000 PSI

1. SURVEY OF PRIMARY COMPONENTS - BAR CHART
2. SURGE LINE ELBOW: STRESS - STRAIN CURVE (FIG. 1)
3. SHUTDOWN COOLING ISOLATION VALVE: 16x12x16 MOTOR OPER.

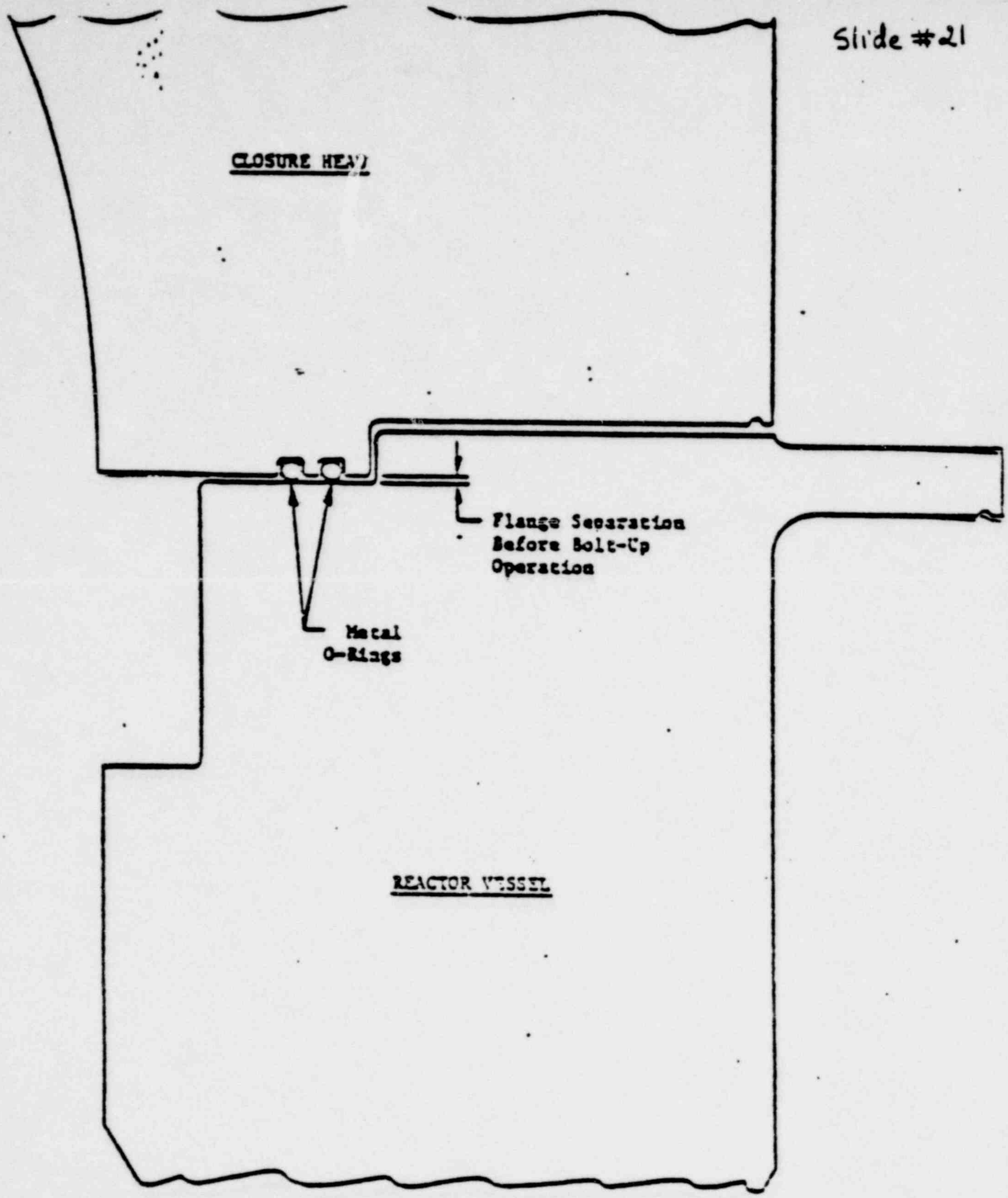
   $STRESS_{MAX}$ = 17,100 PSI AT PIPE

   $STRESS_{YIELD}$ = 18,500 PSI
4. PRESSURIZER SAFETY VALVE
   - A. DISC AND NOZZLE: FINITE ELEMENT ANALYSIS;
     DISC BEHAVES ELASTICALLY WITH SLIGHT LOCAL PLASTICITY.
     NOZZLE REACHES YIELD AT 6350 PSI.
   - B. INLET FLANGE AND BOLTING
     FLANGE REACHES LEVEL C STRESS LIMITS AT 4535 PSI
     (BASED ON DIRECT PRESSURE RATIO OF DISCHARGE FORCE).
     BOLT STRESS REACHES LEVEL C STRESS LIMITS AT 7500 PSI.
5. TYPICAL 1500 LB. VALVE EVALUATION

   STRESS LEVELS ARE WITHIN LEVEL C LIMITS AT 4000 PSI.
6. REGENERATIVE HX

   STRESS LEVELS ARE WITHIN LEVEL C LIMITS AT 4000 PSI.

Slide #21



REACTOR VESSEL & HEAD ASSEMBLY
PRIOR TO BOLT-UP OPERATION

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Slide #22

FIGURE 3-4. VESSEL FLANGE O-RING SEAL LEAKAGE MODEL

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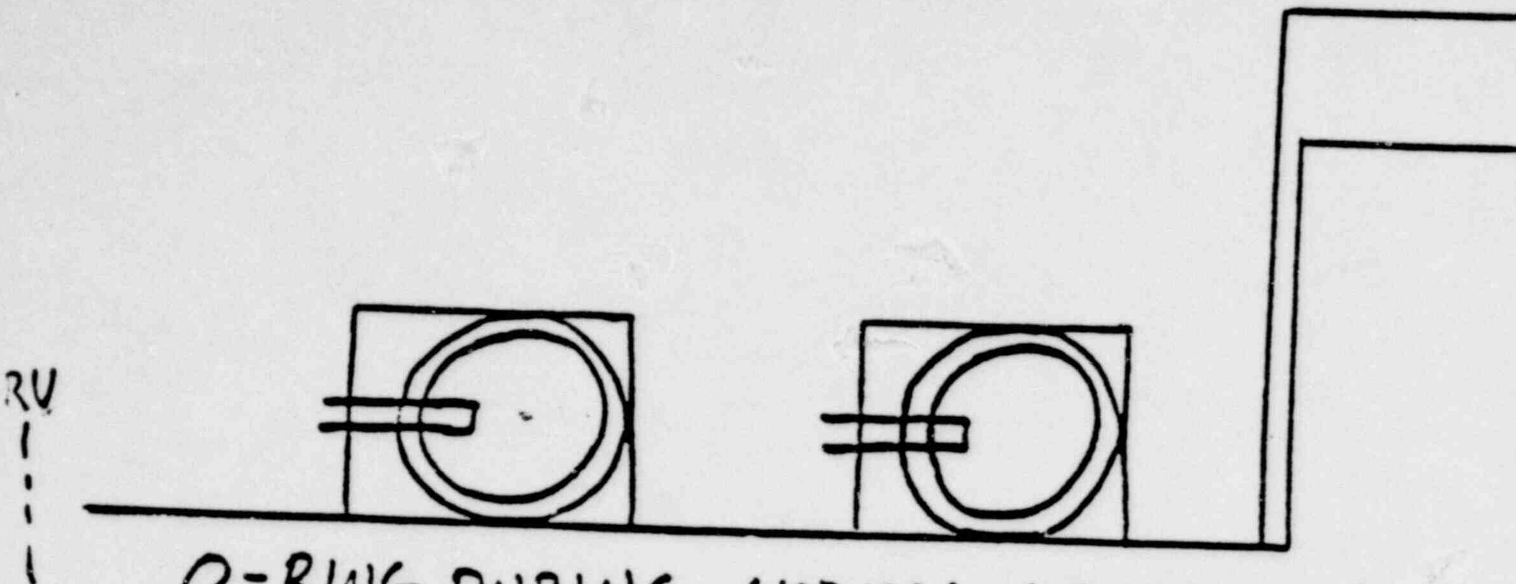
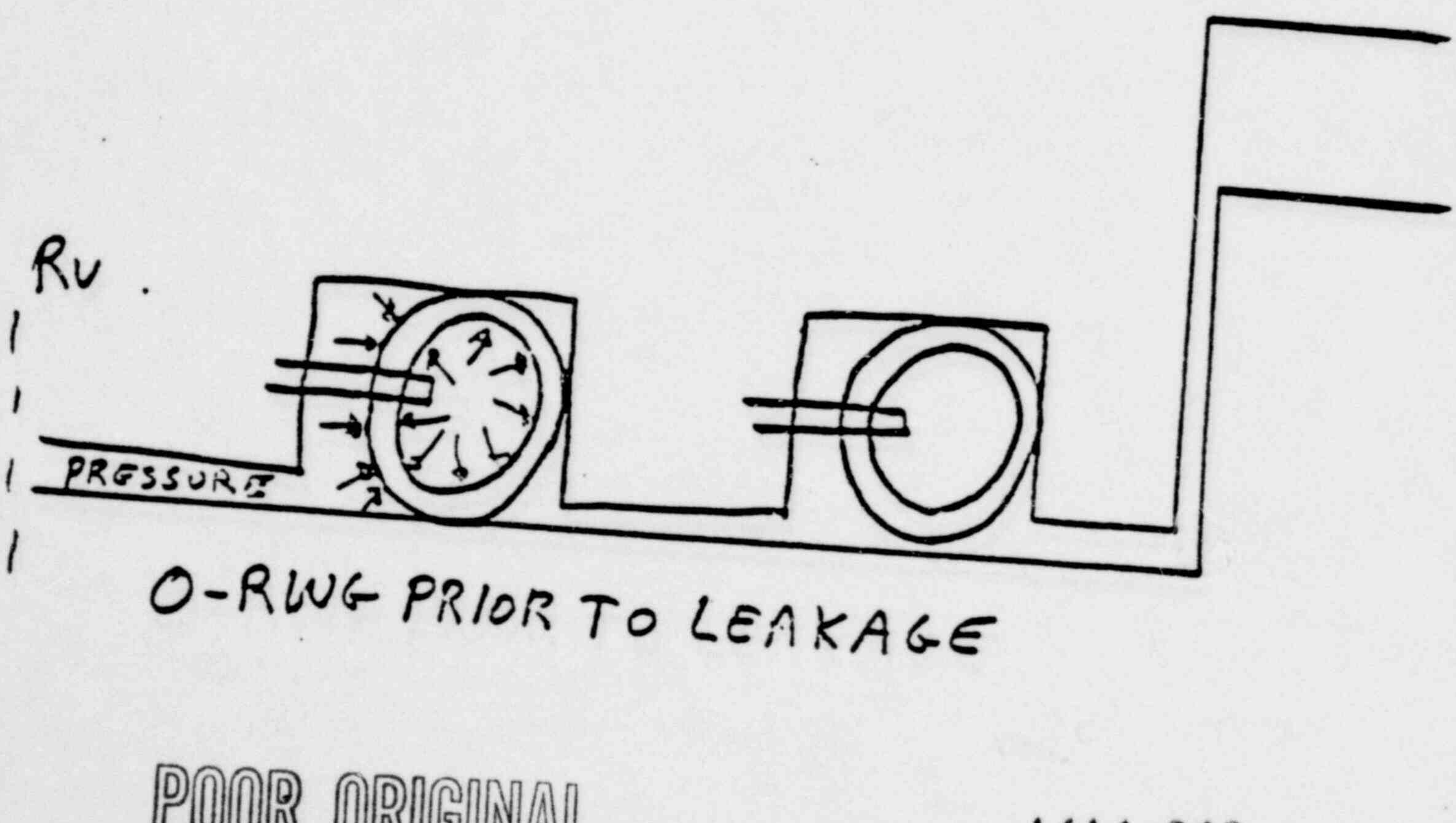
Slide # 23



O-RING DURING NORMAL OPERATION



O-RING PRIOR TO LEAKAGE

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Slide #24

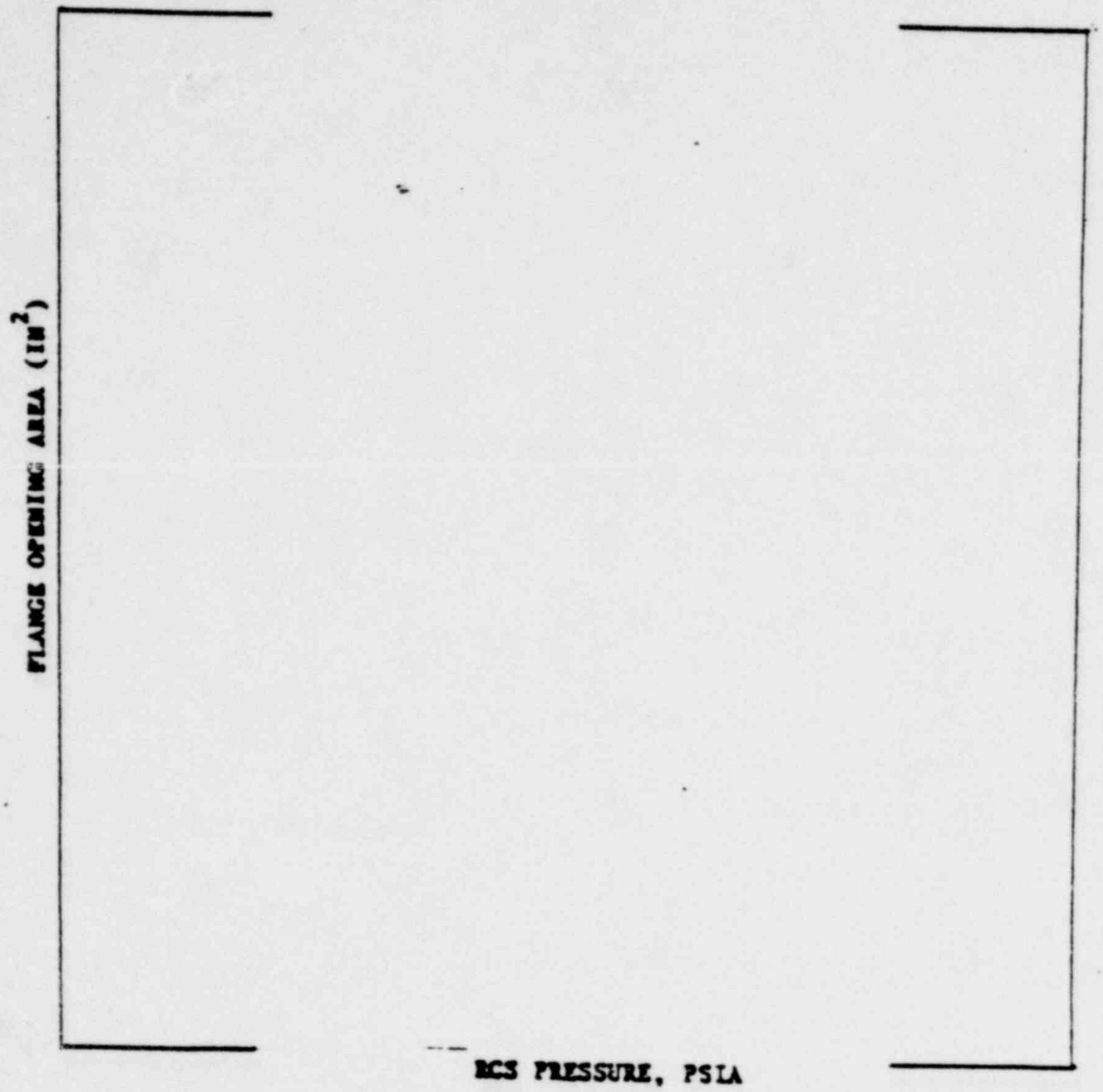

FIGURE 3-6 O-RING SEAL OPENING AREA AS A FUNCTION OF PRESSURE FOR THE 3800 MWT PLANT CLASS

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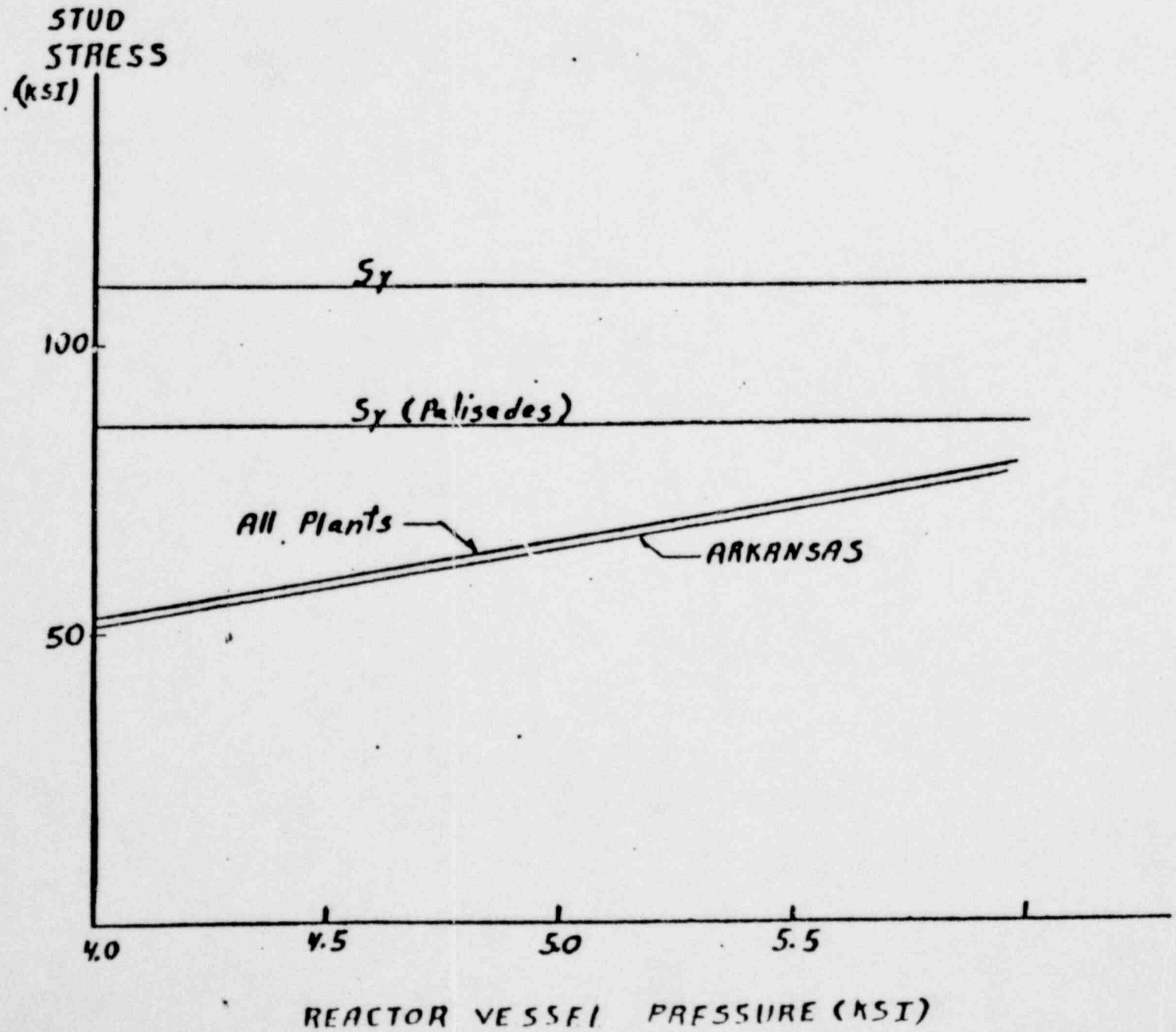
Slide #25

FLANGE OPENING AREA (IN^2)

RCS PRESSURE, PSIA

FIGURE 3-5 O-RING SEAL OPENING AREA AS A FUNCTION OF PRESSURE FOR THE 2560 MWT AND 34XX MWT PLANT CLASSES

slide #26



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