May 14, 1990

Docket No. 50-206

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

LICENSEE: SOUTHERN CALIFORNIA EDISON COMPANY

SUBJECT: SUMMARY OF MEETING HELD AT ONE WHITE FLINT NORTH ON MAY 7, 1990, RE: THERMAL SHIELD REPAIR

On May 7, 1990, the NRC staff met with representatives of Southern California Edison Company (SCE) to discuss SCE's thermal shield repair plans. Persons attending the meeting are identified in Enclosure 1.

The meeting was held to discuss the details of SCE's Amendment Application No. 181 dated April 20, 1990, on this subject to facilitate NRC review efforts. SCE has requested NRC approval of the thermal shield repair plans by July 15, 1990, to support the Cycle XI outage schedule.

As a result of the meeting, SCE has agreed to provide additional information to support NRC review efforts. The information required is included in Enclosure 2.

Viewgraphs used at the meeting are included in Enclosure 3.

original signed by James E. Tatum

James E. Tatum, Project Manager
Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Additional Information Required
3. Viewgraphs

DISTRIBUTION
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DRSP/PDV	(A)DRSP/PDV
JTatum:sg	JLarkins
5/15/90	5/ /90

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1/1

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

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James E. Tatum

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Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Additional Information Required
3. Viewgraphs

Mr. Harold B. Ray
Southern California Edison Company

San Onofre Nuclear Generating
Station, Unit No. 1

cc

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ENCLOSURE 1

LIST OF ATTENDEES

NRC

J. Larkins

D. Terao

C. Trammell

J. Tatum

C. Hinson

L. Wharton

J. Rajan

R. Hermann

A. Wang

M. Hum

C. Sellers

SCE

F. Nandy

J. Reilly

R. Ashe-Everest

R. Ornelas

M. Motamed

G. Stawniczy

WESTINGHOUSE

C. Boyd

J. Goossen

B. Bevilacqua

D. Dominicis

D. Forsyth

OTHER

J. Porowski
(O'Donnell & Assoc)

T. Landgraf
(Bechtel-KWU-Alliance)

M. Mundt (SDG&E)

ENCLOSURE 2

Additional Information Required to Facilitate NRC Review

1. Details of ALARA considerations, including dose assessments and time-motion studies.
   Approximate Submittal Date: 5/20/90

2. Details and results of qualification testing for fasteners (torque vs. preload).
   Approximate Submittal Date: 5/30/90

3. Final drawing details.
   Approximate Submittal Date: 5/30/90

4. Material Specifications for KWU and Westinghouse fasteners.
   Approximate Submittal Date: Not Established

5. Details and justification for exception to ASME fatigue design requirements.
   Approximate Submittal Date: 5/30/90

6. Final design criteria.
   Approximate Submittal Date: 6/15/90

7. Description of measures being taken to assure RCS cleanliness during repair evolution and prior to returning the unit to service. (This item was discussed with SCE following the meeting).
   Approximate Submittal Date: 5/20/90

8. Description of followup inspections planned following repair of the thermal shield, and frequency for conducting inspections.
   Approximate Submittal Date: Not Established

TECHNICAL PRESENTATION

ON

THERMAL SHIELD SUPPORT SYSTEM REPLACEMENT

AND

CYCLE 11 MONITORING PROGRAM

FOR SAN ONOFRE UNIT 1

SOUTHERN CALIFORNIA EDISON

MAY 7, 1990

SONGS 1 REACTOR VESSEL
THERMAL SHIELD SUPPORT REPLACEMENT

AGENDA

May 7, 1990

I.	INTRODUCTION	R. ORNELAS
II.	TECHNICAL OVERVIEW	R. ASHE-EVEREST
III.	DESIGN OF NEW SUPPORT SYSTEM	R. ASHE-EVEREST
	A. Comparison of SONGS I and Haddam Neck Design	J. GOOSSEN
	B. Implementation	
	C. Design Criteria	
	D. Response to NRC Questions	
IV.	FUTURE INSPECTIONS	R. ASHE-EVEREST
V.	THERMAL SHIELD MONITORING SYSTEM (AS REQUESTED)	R. ASHE-EVEREST
VI.	LICENSING ACTIONS	R. ORNELAS
VII.	QUESTIONS AND SUMMARY	ALL

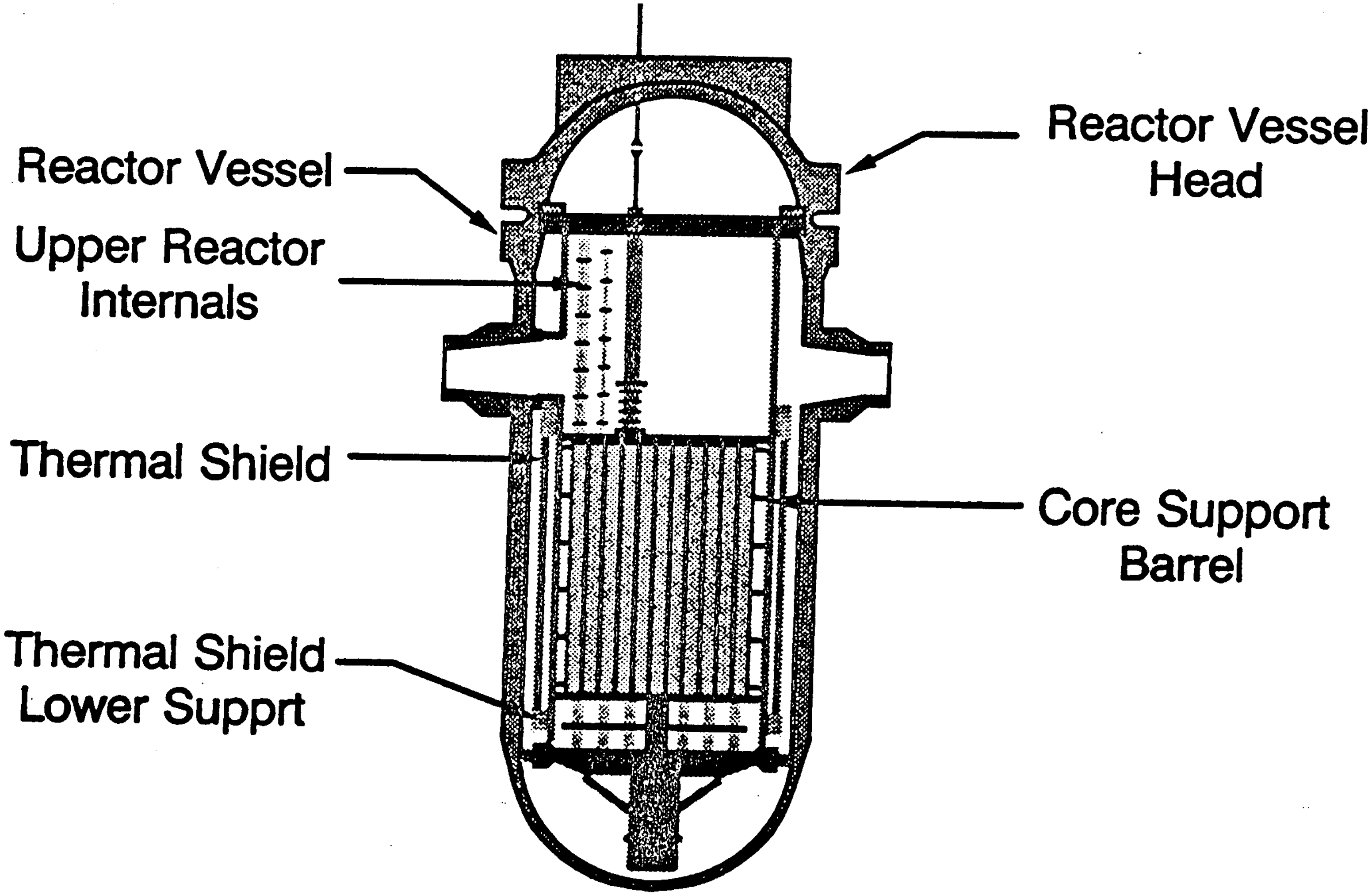
THERMAL SHIELD SUPPORT REPLACEMENT

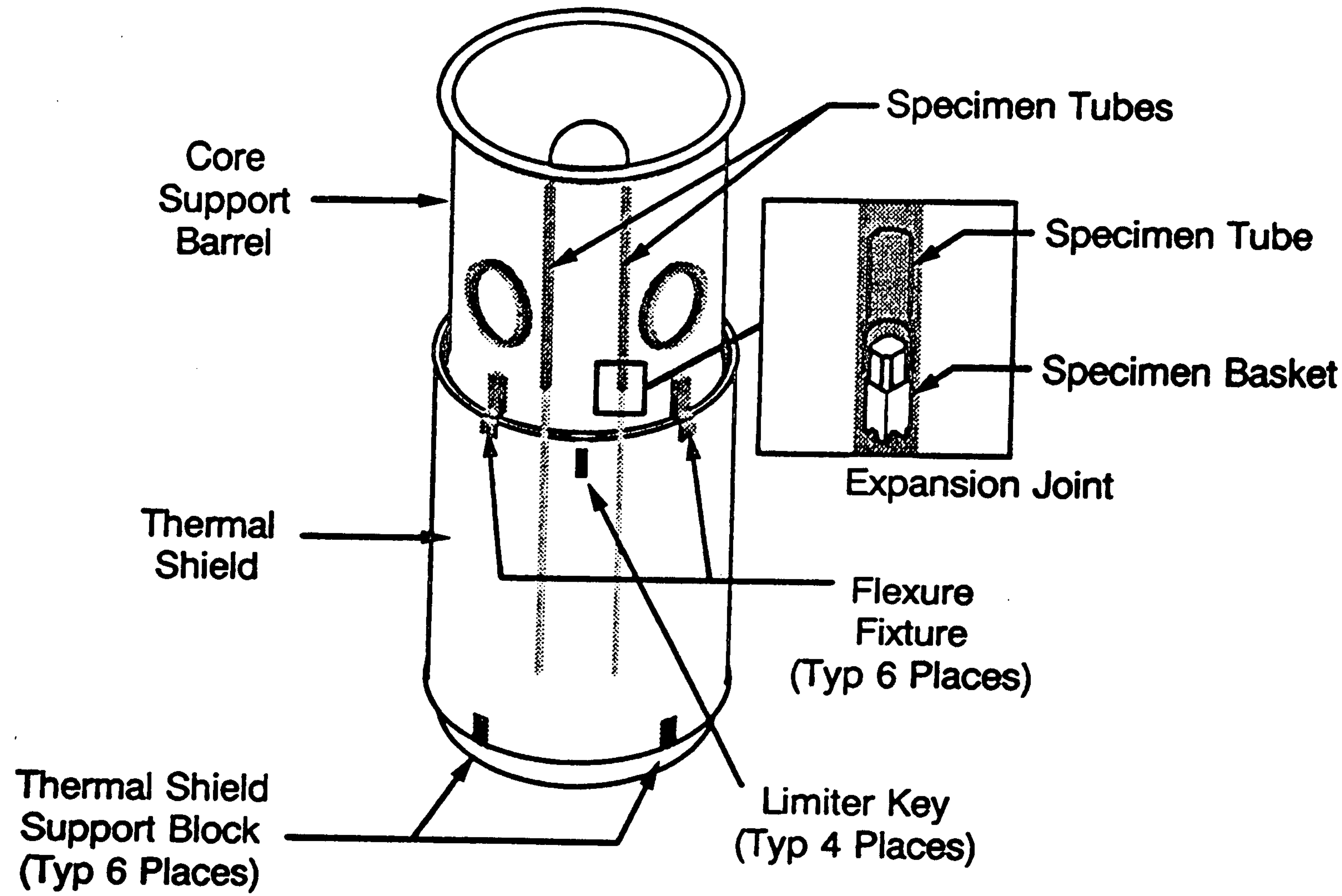
SAN ONOFRE NUCLEAR STATION - UNIT 1

OVERVIEW

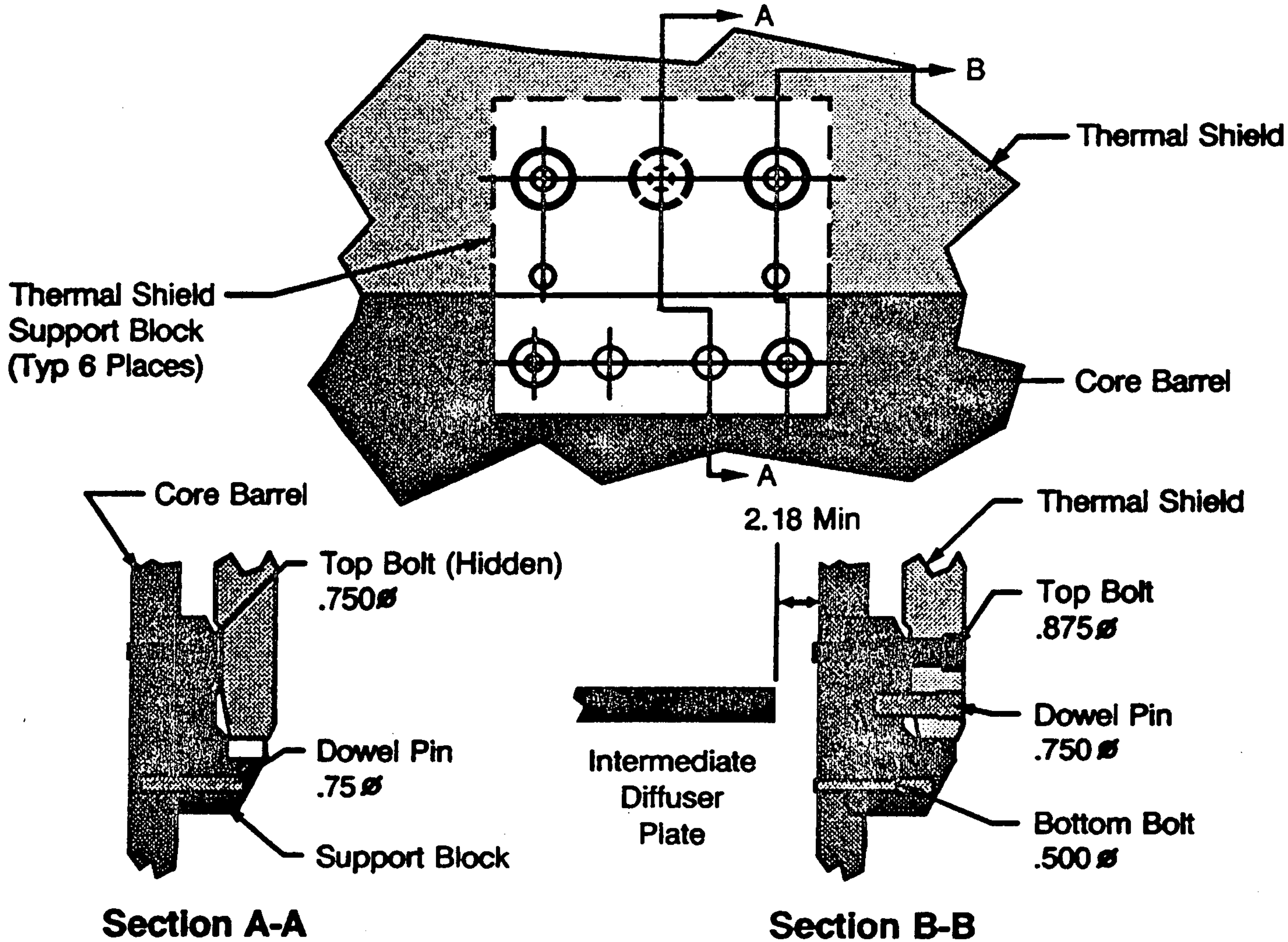
- Commercial Operation - January 1, 1968
- Effective Full Power Years - 12 Years
- Prior to Cycle X refueling, several thermal shield inspections performed with no unacceptable degradation observed
- Cycle X refueling thermal shield inspection revealed limited support degradation
    - Five of six flexures broken (four were known to be broken from prior inspections)
    - Three of thirty support block bolts broken
    - One dowel pin in lower support block had a broken tack weld
- Analysis performed and monitoring systems added to allow operation through Cycle X
- NRC required inspection of thermal shield support system
- SCE has elected to replace the support system during the 1990 actual outage
- Thermal shield material and core barrel material are 304SS SA240

SONGS Reactor Internals

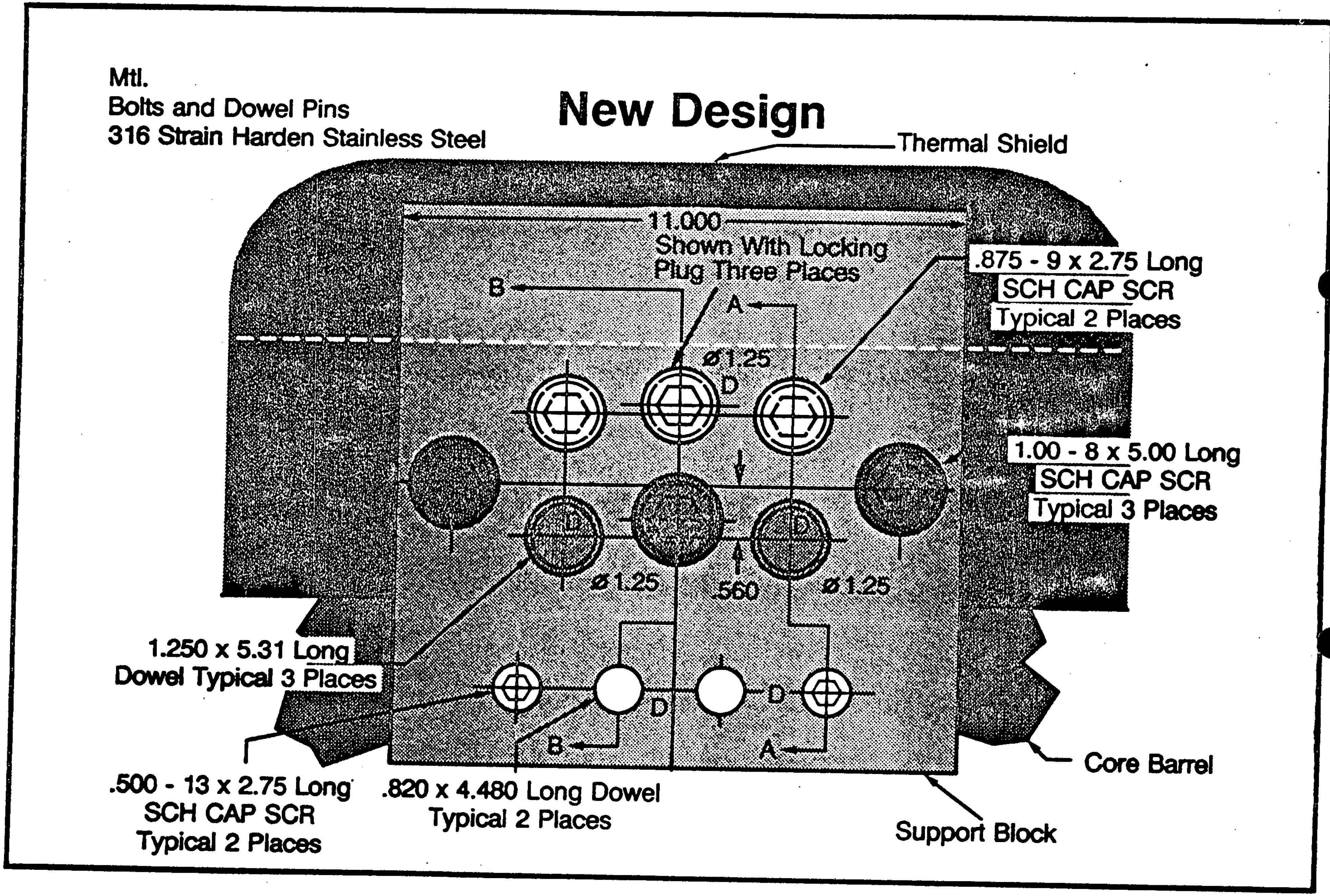



Specimen Tubes
Core Support Barrel
Specimen Tube
Specimen Basket
Expansion Joint
Thermal Shield
Flexure Fixture (Typ 6 Places)
Thermal Shield Support Block (Typ 6 Places)
Limiter Key (Typ 4 Places)

Existing Design



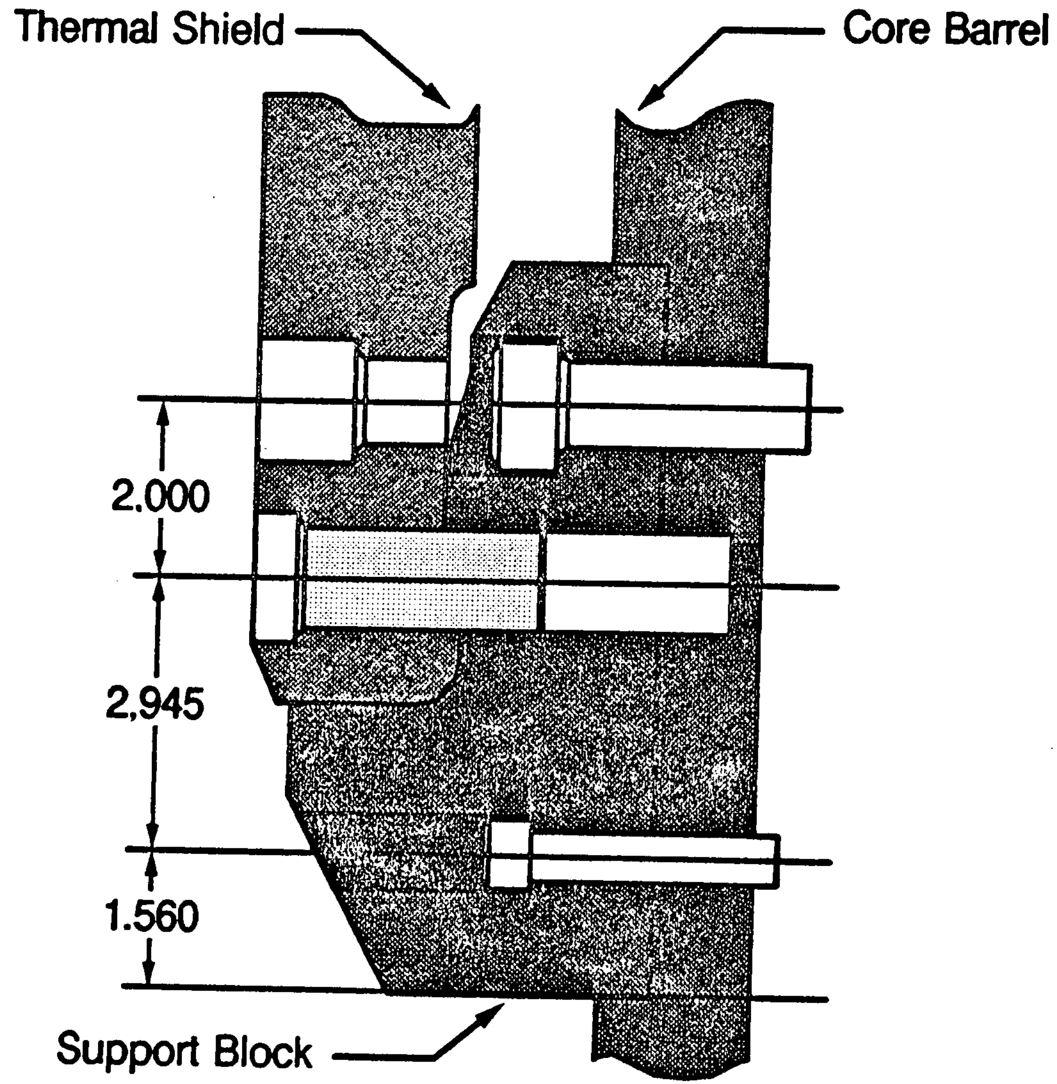
New Design



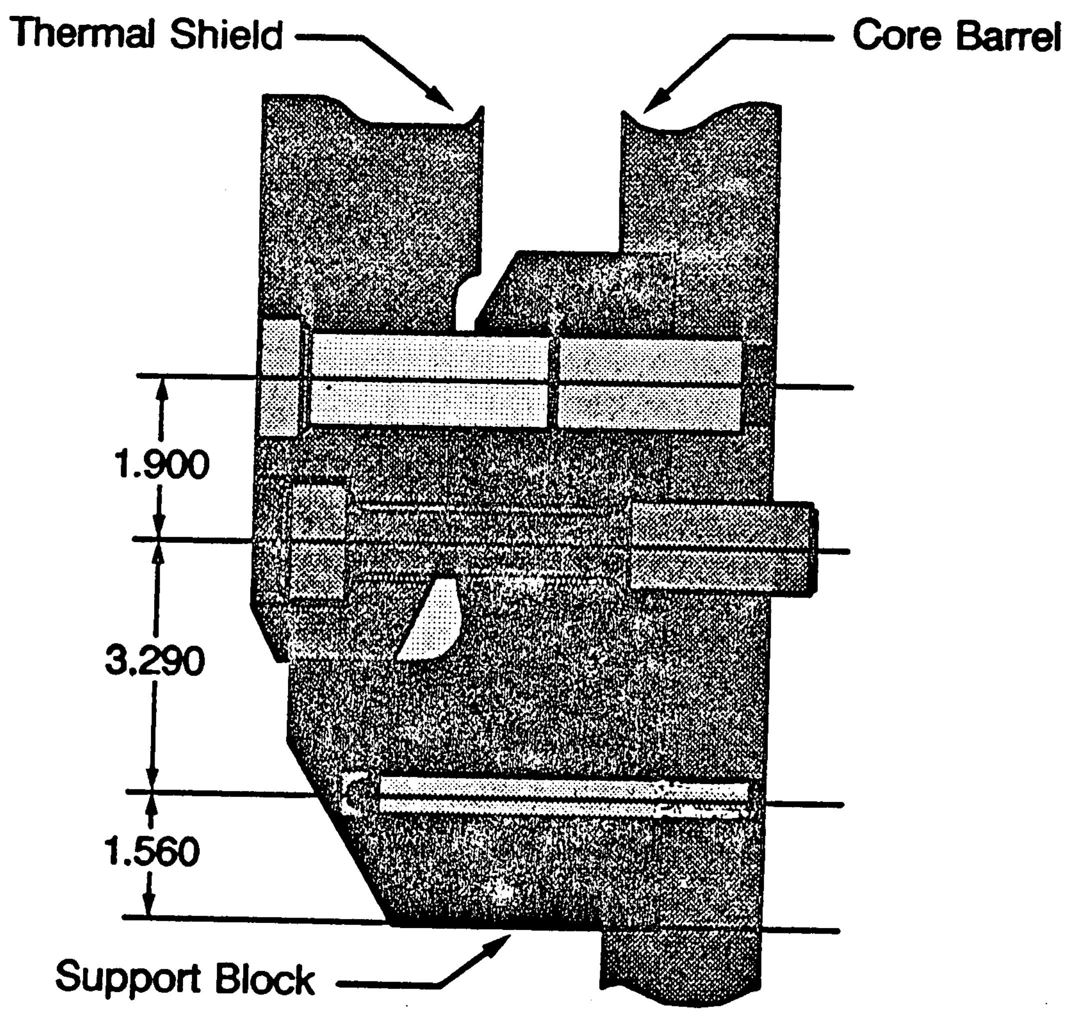
Mtl.
Bolts and Dowel Pins
316 Strain Harden Stainless Steel
Thermal Shield
11.000
Shown With Locking Plug Three Places
B
A
ø1.25
D
.875 - 9 x 2.75 Long SCH CAP SCR Typical 2 Places
1.00 - 8 x 5.00 Long SCH CAP SCR Typical 3 Places
ø1.25
.560
ø1.25
1.250 x 5.31 Long Dowel Typical 3 Places
D
D
B
A
Core Barrel
.500 - 13 x 2.75 Long SCH CAP SCR Typical 2 Places
.820 x 4.480 Long Dowel Typical 2 Places
Support Block

Section A-A

Support Block Mtl.
304 Stainless Steel



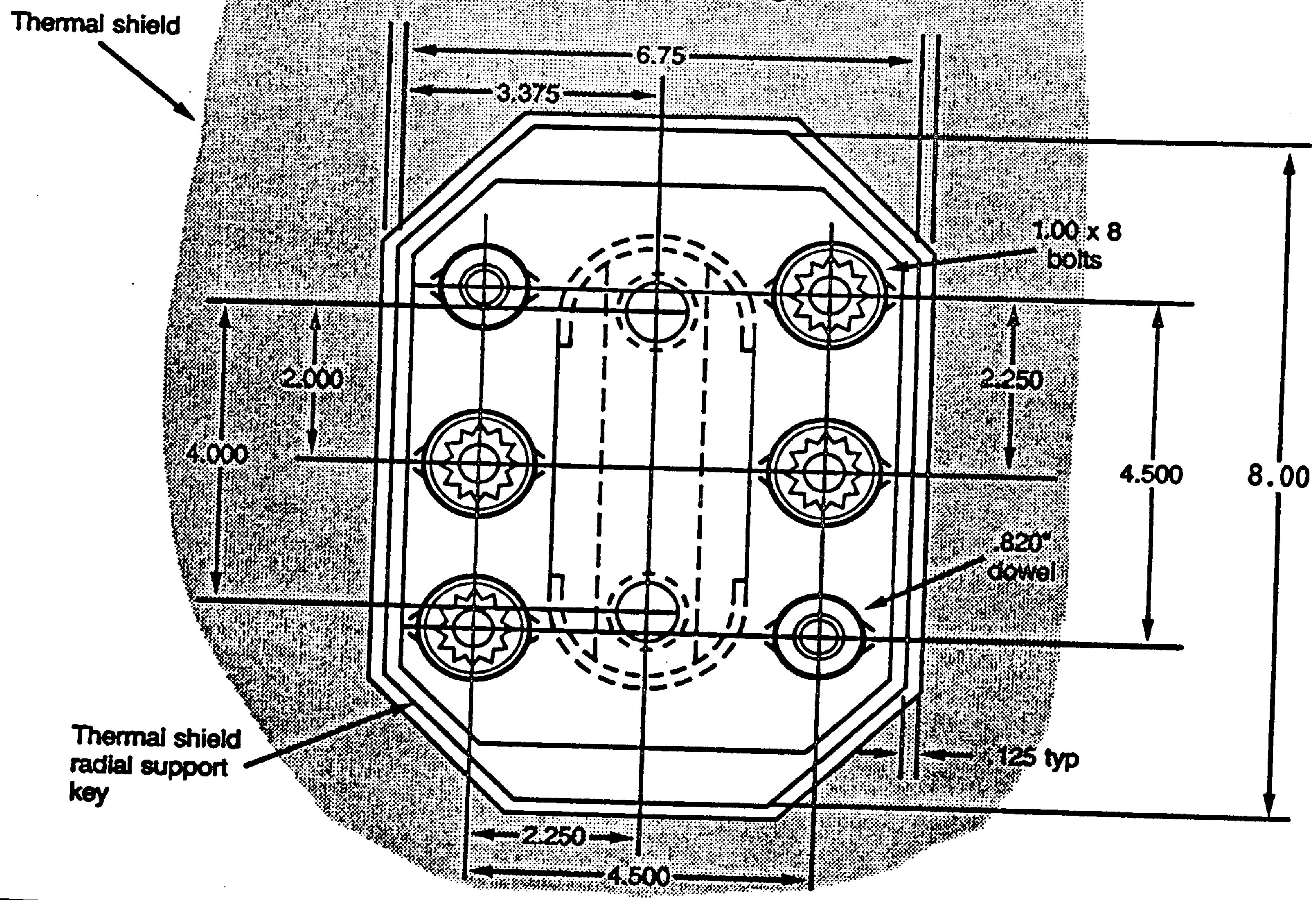
Section B-B



LOWER SUPPORT BLOCK

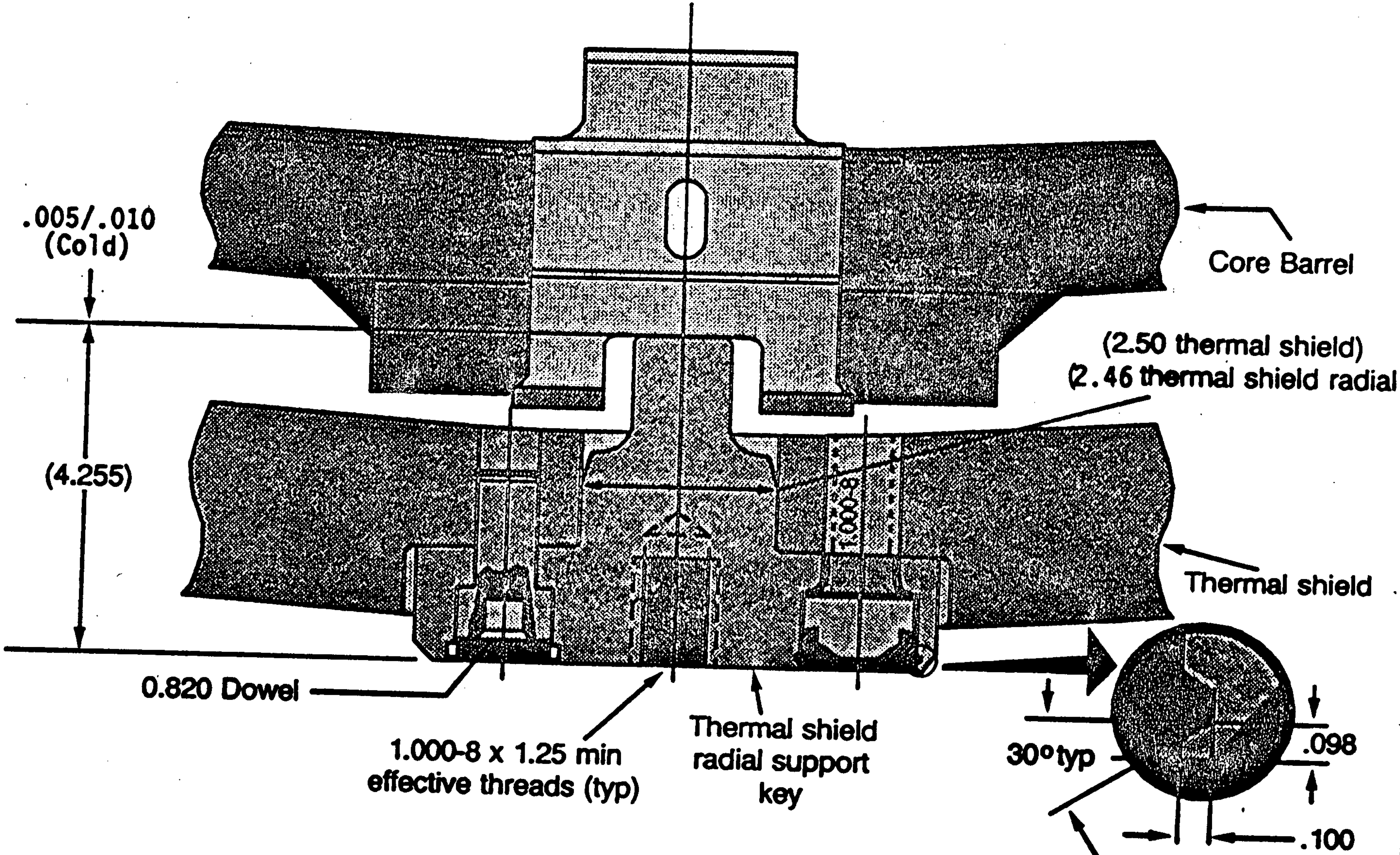
NEW DESIGN	OLD DESIGN
o 11" wide; more than three times contact area thermal shield and support blocks	8" wide
o Seven bolts 75% increase in area	Five bolts
o Five dowel pins 150% increase in area	Four dowel pins
o Dowel pins designed for interference fit	Shrink fit
o Bolts & dowel pins will be restrained by a locking nut or crimped head	Bolts - welded lock bar Dowels - welded

New Design



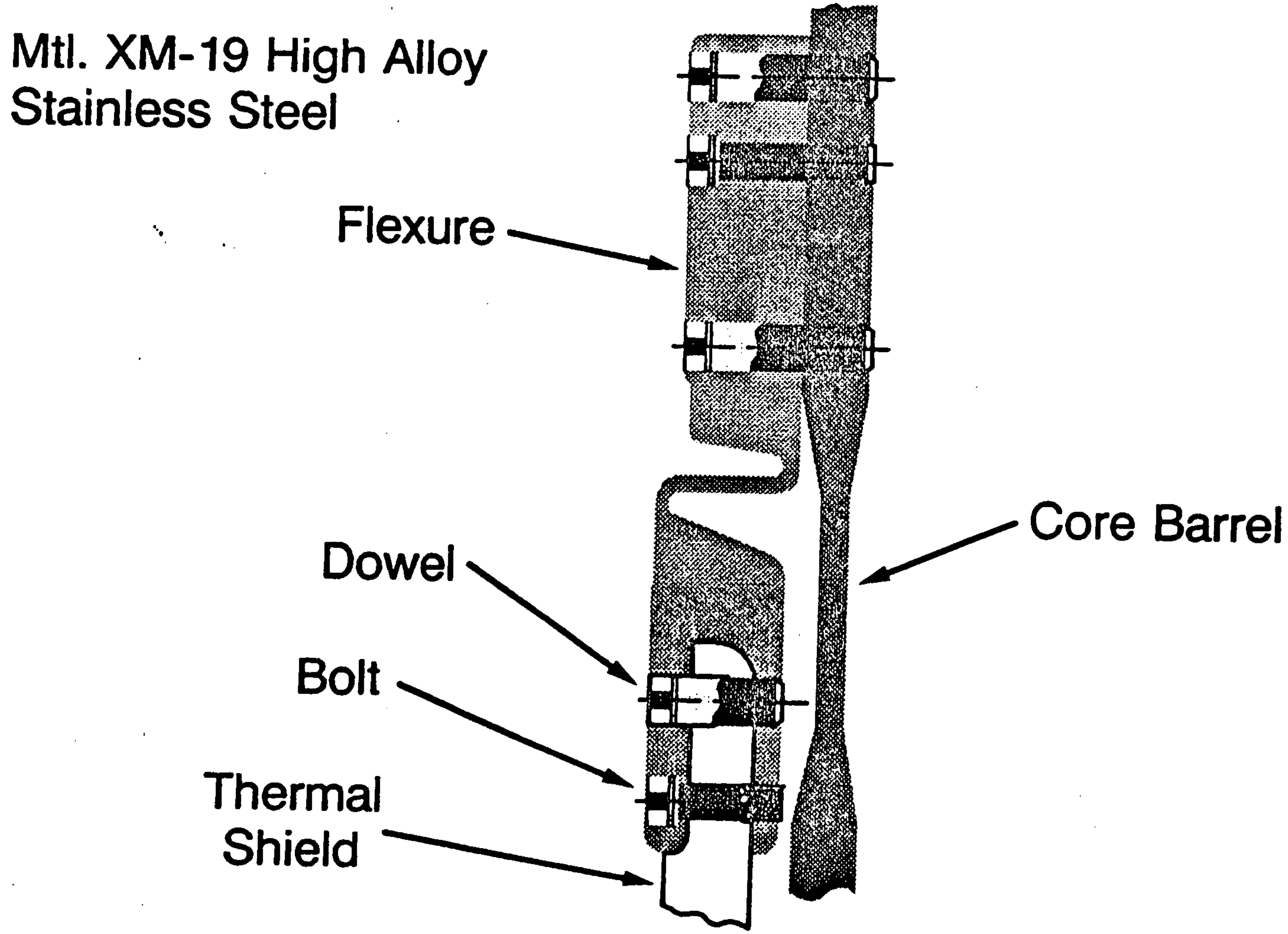
Thermal shield
6.75
3.375
1.00 x 8
bolts
2.000
4.000
2.250
4.500
8.00
.820"
dowel
Thermal shield
radial support
key
.125 typ
2.250
4.500

New Design

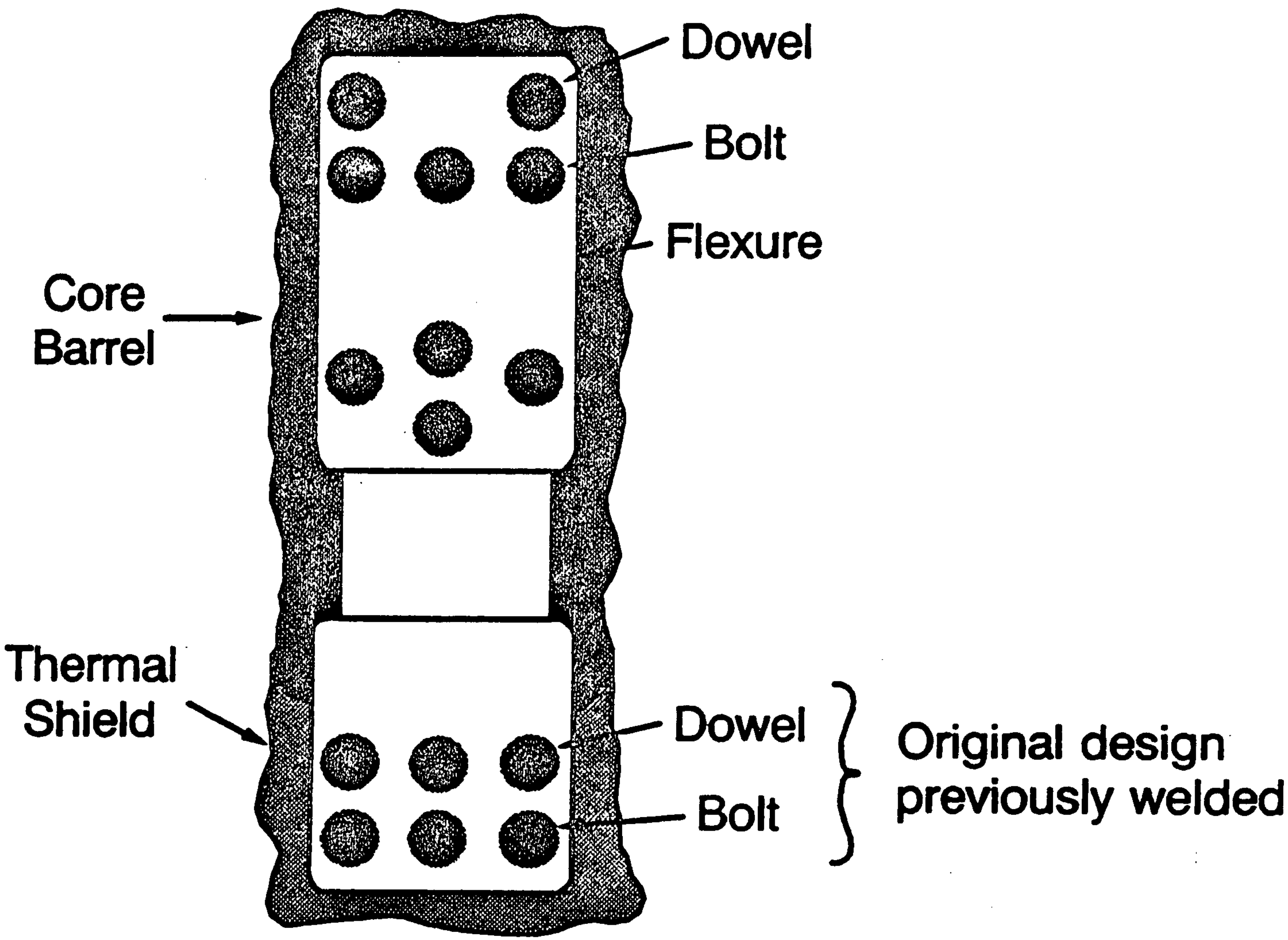


.005/.010
(Cold)
(4.255)
Core Barrel
(2.50 thermal shield)
(2.46 thermal shield radial
1.000-8
Thermal shield
0.820 Dowel
1.000-8 x 1.25 min
effective threads (typ)
Thermal shield
radial support
key
30° typ
.098
.100

New Thermal Shield Flexure Assembly



New Thermal Shield Flexure Assembly



FLEXURE DESIGN COMPARISON

NEW DESIGN	OLD DESIGN
o Material SA-479 XM-19 high alloy stainless steel (annealed) Tensile 100 KSI Yield 55 KSI Hardness 293 Brinell	ASTM A-276 type 304SS Tensile 75 KSI Yield 30 KSI Hardness 202 Brinell
o Flexure fabricated in one piece (no welds)	Welded in web area
o Preloaded flexure web area	No preload
o Web geometry improved to reduce stress	--
o One flexure will be moved to reduce loading	--

MATERIAL DATA COMPARSION
316ss vs XM-19

CHEMICAL REQUIRMENTS : 1986 EDITION ASME B&PVC

% COMPOSITION
SA-479

ELEMENT	316ss	XM-19
C	.03	.06
MN	2.0	4/6
P	.045	.04
S	.03	.03
SI	1.	1.
CR	16/18	20.5/23.5
NI	10/14	11.5/13.5
MO	2/3	1.5/3.0
N	.1	.2/.4
CB	–	.1/.3
V	–	.1/.3

MATERIAL DATA COMPARSION
316ss vs XM-19

CORROSION RESISTANCE
PROPERTIES
LABORATORY TEST DATA
(SUPPLIED BY ARMCO STEEL)

TEST MEDIUM	CORROSION RATE INCHES PER YEAR (UNLESS SPECIFICALLY INDICATED)	
	XM-19 (ANNEALED)	316ss (ANNEALED)
10%FeCL3 (25 C PLAIN)	<.001 G/IN2	0.011 G/IN2
10%FeCL3 (25 C CREVICED)	<.001 G/IN2	0.186 G/IN2
1%H2SO4 (80 C)	<.001	0.002
2%H2SO4 (80 C)	<.001	0.011
5%H2SO4 (80 C)	<.001	0.060
5%H2SO4 (BOILING)	<.194	0.26
1%HCL (35 C)	<.001	0.012
2%HCL (35 C)	0.024	0.021
65%HNO3 (BOILING)	0.010	0.012
70%H3PO4 (BOILING)	0.203	0.202
33%ACETIC ACID (BOILING)	<.001	<.001

TESTING PERFORMED ON 5/8" DIA X 5/8" LONG MACHINED CLINDERS. RESULTS AN AVE. OF FIVE 48 HR. PERIODS.

MATERIAL DATA COMPARSION
316ss vs XM-19

MECHANICAL REQUIRMENTS : SA-479 1986 EDITION ASME

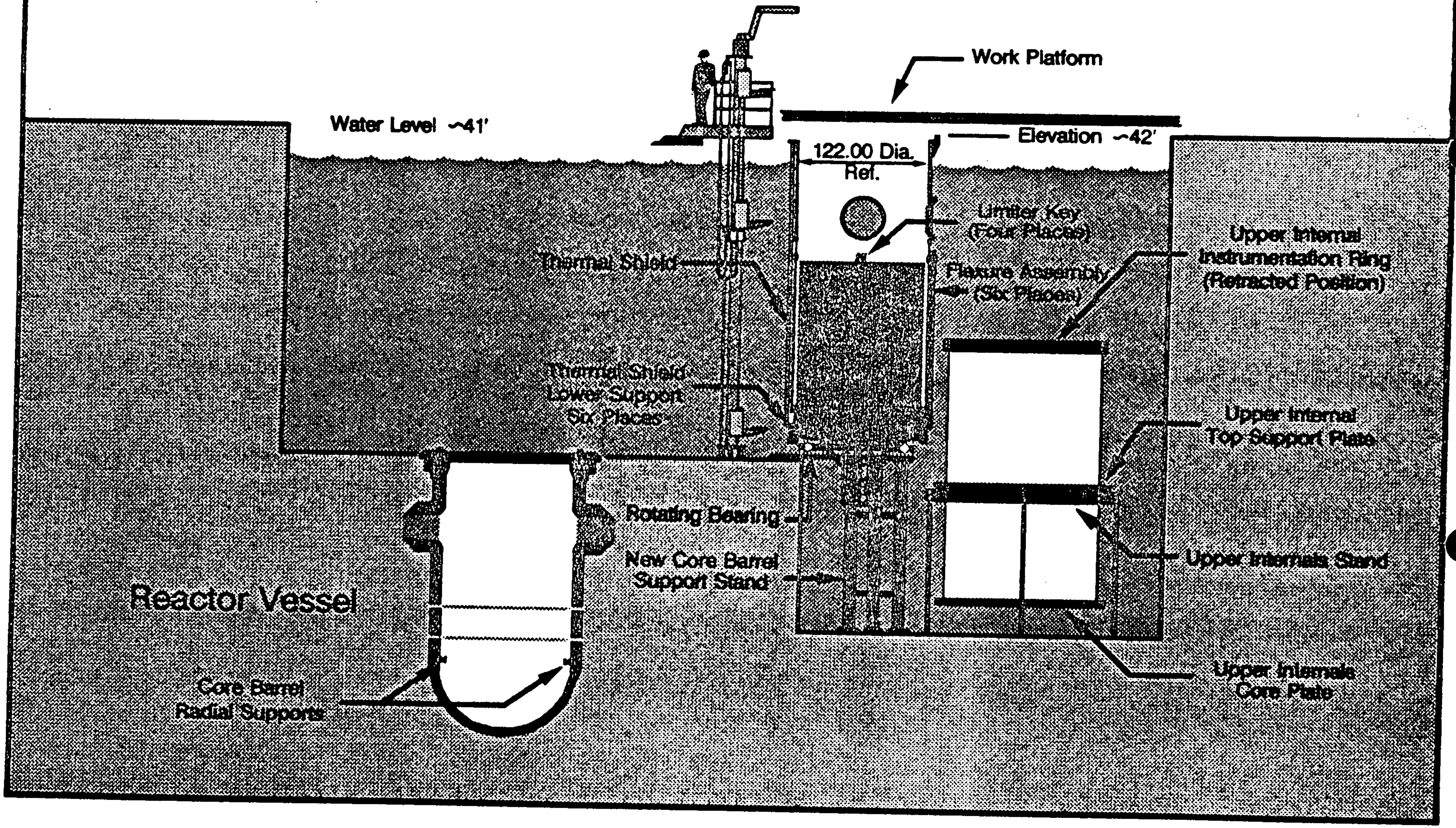
MECHANICAL PROPERTIES

MECHANICAL PROPERTIES (MAT TYPE)	TENSILE STRENGTH (KSI)	YIELD MIN (KSI)	ELONGATION 2 IN MIN %	REDUCTION AREA MIN %	BRINELL HARDNESS MAX
316ss	70	25	30	40	-
XM-19	100	55	35	55	293

THERMAL SHIELD SUPPORT

SONGS 1	HADDAM NECK
o Replace lower support with larger design.	Original blocks retained.
o Interference fit lower support between thermal shield and core barrel.	No adjustments made.
o Bolting and dowel pin system has been redesigned.	New bolts installed; only backed out dowel pins replaced.
o Dowel pins installed with interference fit.	Clearance fit installation of new dowels.
o Four new limiter keys.	No refurbishment of original keys.
o Six new, improved flexures (no wear surfaces).	Six new limiter key and keyways.
o New support system incorporates lessons learned from Haddam Neck.	

San Onofre Unit 1 Reactor Cavity With Core Barrel and Upper Internals Removed



PERFORMANCE OF THE QUALIFICATION TESTS

INTERFERENCE FIT

Test No.	Diameter	1	2	3	4
D1	.8125"	0.0002"	0.0005"	0.0008"	0.0011"
D2	1.25"	0.0003"	0.0007"	0.001"	0.0013"
D3	1.75"	0.0004"	0.0008"	0.0011"	0.0014"

LOCKING SYSTEM QUALIFICATION TEST

TEST NO.	BOLT 1 DOWEL PIN DIMENSION	DIMENSIONAL CHECK	VISUAL EXAM.	DETORQUING OF LOCKED SPECIMEN	METALLURGICAL TEST SECTION A-A	METALLURGICAL TEST SECTION B-B	PRESS IN AND PUSH OUT	ORIGINAL MATERIAL
B1	(4) 1.00-8x2 long SCH CAP SCR	(4)	(4)	(3)	(1)	--	--	x
B2	(4) 0.500-13x2.75 long SCH CAP SCR	(4)	(4)	(3)	--	(1)	--	x
D4	(2) 1.75"-4." long dowel pin	(2)	(2)	--	--	--	(2)	--
D5	(2) 1.25"-5.219" long dowel pin	(2)	(2)	--	--	--	(2)	--
D6	(2) .8125"-4.045" long dowel pin	(2)	(2)	--	--	--	(2)	--
B3	(2) 1.00-8x2 long SCH CAP SCR	(2)	(2)	Detorque measurement on unlocked locked bolts 1)				--
D7	(2) 1.25-5.215 long dowel pin	(2)	(2)	Cyclic loading test 2)			(2)	x

NOTES: 1) Measurement of effect of crimping on bolt preload by detorquing the nuts instead of bolt head.

2) Test parameters see paragraph 3.2e.

QUALIFICATION TEST
PRELIMINARY TOOLING TESTS

1. Removal of the old parts:
   - EDM of locking on fasteners (bars, washers, dowel pins).
   - Pull out of dowel pins.
   - Detorque of bolts.
   - Removal of remaining parts of broken fasteners.

2. Preparation for installation of new design:
   - EDM pilot holes and counterboring.
   - EDM recess for mechanical locking of fasteners.
   - EDM recess for flexure and key.
   - EDM thread 1.5".
   - Thread cutting 1".
   - Reaming.

3. Final installation:
   - Press in dowels.
   - Torque bolts.
   - Locking of fasteners.

Tool testing in addition with original tools and equipment on mock ups.

SCE THERMAL SHIELD SUPPORT SYSTEM REPLACEMENT <u>DESIGN CRITERIA</u>

- DESIGN LIFE
  - Goal of 15 operating years

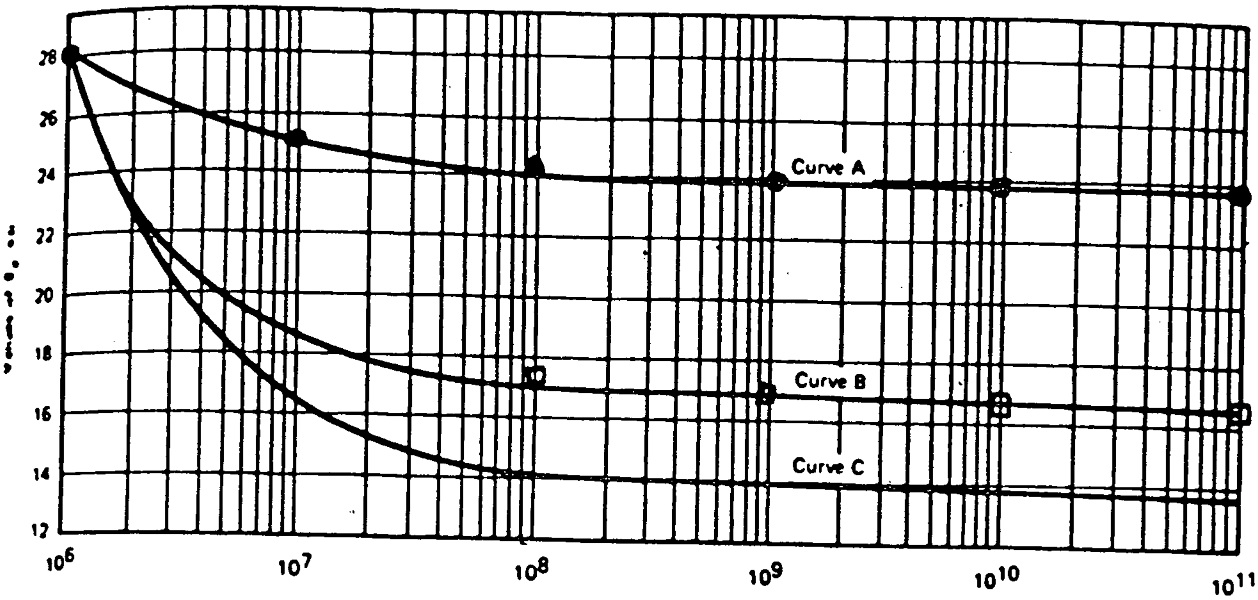
- ASME CODE
  - The ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Section III, Subsection NG, does not apply to the thermal shield.
  - However, for conservatism the 1986 edition of the ASME Code, Section III, Subsection NG, will be used.

- LOAD COMBINATIONS
  - Normal and Upset Conditions
    - Deadweight + pressure + FIV
    - Deadweight + pressure + seismic + FIV
    - Deadweight + pressure + thermal + FIV
    - Deadweight + pressure + seismic + thermal + FIV
  - Faulted Conditions
    - Deadweight + pressure + seismic

- NSSS OPERATING CONDITIONS
  - Flexure preload will be established based on 100% full power with reduced Tavg conditions.
  - Evaluations will also be performed to show acceptability at 92% full power at reduced Tavg conditions.

- SEISMIC
  - Seismic condition loads will be considered in the design of the Thermal Shield Support System. The San Onofre Design Basis Earthquake Spectra used at the Reactor Vessel Supports is commonly referred to as the .67 g Modified Hausner Earthquake.

● MJM 1990 EQUS. STRAIN CONTROL
□ " " " LOAD CONTROL



NOTE:
$E = 28.3 \times 10^6$ psi

Criteria for the Use of the Curves in This Figure[1,2,3,4]

Curve	Elastic Analysis of Material Other Than Welds and Heat Affected Zones	Elastic Analysis of Welds and Heat Affected Zones
A	$(P_L + P_b + Q)_{Range}$ < 44 ksi	. . .
B	. . .	$(P_L + P_b + Q)_{Range}$ < 44 ksi
C	$(P_L + P_b + Q)_{Range}$ > 44 ksi	$(P_L + P_b + Q)_{Range}$ > 44 ksi

NOTES:

(1) Range applies to the individual quantities P_L, P_b, and Q, and applies to the set of cycles under consideration.

(2) Thermal bending stresses resulting from axial and radial gradients are excluded from Q.

(3) Curve A is also to be used with inelastic analysis with $S_a = \frac{1}{2} \Delta \epsilon_t E$, where $\Delta \epsilon_t$ is the total effective strain range.

(4) The maximum effect of retained mean stress is included in curve C.

FIG. I-9.2.2 DESIGN FATIGUE CURVE FOR AUSTENITIC STEELS, NICKEL—CHROMIUM—IRON ALLOY, NICKEL—IRON—CHROMIUM ALLOY, AND NICKEL—COPPER ALLOY FOR $S_a \leq 28.2$ ksi, FOR TEMPERATURES NOT EXCEEDING 800°F
(For $S_a > 28.2$ ksi, use Fig. I-9.2.1.)
Table I-9.2.2 Contains Tabulated Values for Accurate Interpolation of This Curve

DRAFT

PROPOSED CODE CASE

Alternate limit on primary plus secondary stress intensity range permitted for austenitic stainless steel to use Design Fatigue Curve A in Figure I-9.2.2, Section III, Division 1.

Inquiry: When using the Design Fatigue Curve on Figure I-9.2.2 for austenitic stainless steel, is it permissible to limit the primary plus secondary stress intensity range excluding thermal bending to the elastic portion of the applicable cyclic stress-strain curve in lieu of meeting the specified value of 27.2 ksi?

Reply: It is permitted to use the minimum elastic limit of 44 ksi for austenitic stainless steel in the fatigue analysis in lieu of the specified value of 27.2 ksi to determine the acceptability of using Design Fatigue Curve A on Figure I-9.2.2. Keeping the primary plus secondary stress intensity range below this value of 44 ksi assures the fatigue analysis is independent of whether the applied loading is strain or load controlled.

NRC QUESTIONS

Q What is the material for the fastener locking devices?

A Bolts & dowel pins 316 strain harden SS. 304 SS threaded locking plug for the 1.25" diameter dowel pins.

Q Have these fasteners been corrosion tested after locking process? Has some surveillance or procedure documented results with respect to corrosion properties? Is there a qualification program?

A No corrosion testing is performed because of extensive use in Europe. Approximately 4500 bolts have been installed in higher temperature and higher radiation fields than seen in this design. No failures observed to date.

Q What is the material for the limiter keys, support blocks, flexures, and the thermal shield? Have they been corrosion tested after peening?

A Limiter keys 304SS, support block 304SS, flexure XM-19, thermal shield SA-240 304 SS - no peening will be used.

Q What is the method for ISI of new material? Have they been proven by industry? What confidence do we have that they won't break?

A The lower support block fasteners will be ultrasonically tested next refueling. Future inspections will be incorporated into the ISI Program (VT3).

Q Baffles - Westinghouse had problems with lateral flow, cross-flow through mechanical joints. Westinghouse did some peening to impede flow. Have these peened baffles exhibited cracking? Has anyone (Westinghouse) looked at this?

A
1. No peening will be performed to lock the fasteners.
2. No baffle cross flow fuel damage observed at SONGS 1.
3. SONGS is an up-flow, not a down-flow design.
4. SONGS does not have a high delta pressure drop across the baffle plates.

Q Is there a contingency plan to remove the thermal shield?

A Yes, Westinghouse is doing a feasibility study. Also, Pacific Nuclear is doing preliminary engineering on cask preparations.

Q Why are neck down bolts used as fasteners in the support blocks?

A Higher fatigue strength - more flexibility in the joint lower stresses in the bolt (1" bolts and .881" shank).

Q Preloading will minimize stress in the flexure web area at operating temperatures and power levels. Is the operating condition used 92% power and reduced Tavg, or rated conditions? Will this impact the licensed power rating of SONGS 1 ?

A Reduced temperature conditions were used for preload of the flexure. Preload will be established at 100% power with reduced Tavg. The design stress criteria will be met for 92% power and reduced Tavg.

Q How does moving the location of one flexure reduce loading?

A Analysis showed that moving the 21 degree flexure to 352 degrees reduced the stresses more than 10%.

Q If water from the spent fuel pool drained to the cavity, could water level drop below the top of fuel assemblies in the pool?

A The reactor cavity water level will be higher than the spent fuel pool water level.

Both the fuel transfer gate valve and the transfer gate will be closed. Hence, a double isolation is established.

Upon a postulated failure of the double isolation, and if the cavity water level should drop, approximately 3' of water will be maintained above the top of the fuel assemblies due to the presence of the wier wall.

CYCLE 12 REFUELING
THERMAL SHIELD INSPECTIONS

During the Cycle 11 support replacement, additional inspection ports will be added through the core barrel flange. These inspection ports will allow camera access through ports in the flange when the core barrel is in the reactor vessel. The planned inspections during the Cycle 12 refueling are as follows:

A) Visually inspect the six upper flexures.

B) Visually inspect the four upper limiter keys.

C) Visually inspect the six lower supports.

D) Ultrasonically examine the lower support fasteners at all six support blocks.

E) Visually inspect a representative sample of the surveillance capsules and baskets.

F) Visually inspect the lower vessel head region for debris.

This inspection plan, in conjunction with the continued monitoring system, will enable the condition of the thermal shield to be assessed.