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October 6, 1989

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

Gentlemen:

Subject: Docket No. 50-206
Conceptual Design and Repair Plan for the
Reactor Vessel Thermal Shield
San Onofre Nuclear Generating Station
Unit 1

This letter provides the conceptual design and repair plan for the reactor vessel thermal shield to the NRC for approval. The proposed design modifications and repair will be performed during the Cycle XI refueling outage scheduled to begin by June 30, 1990. SCE will request a meeting with the NRC staff in the near future to discuss and respond to any questions concerning the information provided below and in the enclosure.

By letter dated May 15, 1989, the NRC required SCE to develop a conceptual design and a plan for the repair of the thermal shield. The May 15, 1989 letter requested that SCE address the plans for the existing support block bolts that are not broken, the design of new flexures, and the modifications to the limiter keys. Details regarding the thermal shield conceptual design and repair plan are provided in the enclosure. We are also following the thermal shield experience at Connecticut Yankee and will ensure that lessons learned are appropriately incorporated into our design and repair plan.

In the letter of May 15, 1989 it was stated that three inspections should be performed of the core barrel as discussed below:

1. An ultrasonic test of the support block ledge should be performed from the inside of the core barrel.

   SCE Response
   After careful consideration of how an ultrasonic examination in this area could best be accomplished, SCE has concluded that it should be performed from the outside of the core barrel because access to the support block ledge area from the inside is severely restricted and deemed to be impractical due to the presence of the lower core support plate and the intermediate flow distribution plate. The examination will be performed in the areas below each of the six support block locations.

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2. A visual examination of the core barrel-to-lower support plate weld should be performed both from the inside and outside of the core barrel.

   SCE Response
   A 100% visual examination of this weld both from the inside and outside of the core barrel, similar to the inspection performed at Connecticut Yankee, will be performed in accordance with ASME Boiler and Pressure Vessel Code, Section XI.

3. An ultrasonic examination should be performed on a representative section of the core barrel-to-lower support weld.

   SCE Response
   SCE has concluded that the 100% visual inspection described above will adequately verify the condition of this weld and that an ultrasonic examination should not be attempted. This conclusion is based on the following considerations.

   a) The surface of this weld has not been prepared for ultrasonic examination, and the work required to achieve adequate surface conditions to conduct an examination is not warranted by the benefits which would be obtained.

   b) The loads from the thermal shield lower supports are concentrated at the block locations. These loads are dispersed and distributed throughout the full core barrel circumference and are not concentrated at particular points of the weld.

   c) A fatigue analysis was performed on this weld by Westinghouse which demonstrated that the cumulative fatigue usage factor would be less than 1.0 for one fuel cycle of operation in the worst credible degraded condition (all bolts and flexures failed). This analysis will be updated to determine the cumulative fatigue usage factor of the weld for the expected current condition of the thermal shield supports, and future operation with the modified design.

If the ultrasonic examination of the core barrel support block ledge reveals any degradation, SCE will revise the inspection plan and obtain NRC approval of the revised plan.

SCE was originally planning to perform only an inspection of the thermal shield during a Cycle 10 mid-cycle outage to begin no later than June 30, 1990. SCE committed to this inspection in a letter dated May 3, 1989. By letter dated September 12, 1989, we submitted the acceptance criteria for the inspection to the NRC for approval. SCE now plans to perform the thermal shield repair during the June 1990 outage, and therefore requests that the

September 12, 1989 submittal regarding the inspection of the thermal shield be disregarded.

If you have any questions or desire further information, please contact me.

Very truly yours,

F. R. Nandy
Manager of Nuclear Licensing

Enclosure

TSRPR2

cc: J. B. Martin, Regional Administrator, NRC Region V
C. Caldwell, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

bcc: See attached sheet

Enclosure

THERMAL SHIELD CONCEPTUAL DESIGN AND REPAIR PLAN

The inspections to be performed prior to the repair, the design modifications, and the repairs to be implemented are as follows:

1. After the core barrel has been removed from the reactor vessel, a detailed visual inspection of the following will be performed; A) the upper six flexures and fasteners, B) exterior of the specimen tubes and baskets at the expansion joint (see Figure 1), C) the four existing limiter keys and keyways (perform gap measurements if possible), D) the six lower support blocks front and back views, E) sides and the bottom of each of the support blocks in the area where they fit into the core barrel machined groove.

2. In place of the ultrasonic examination of the support block ledge area from the inside of the core barrel suggested by the NRC, an ultrasonic examination of the core barrel will be performed from the outside of the core barrel near the support block ledge just below the thermal shield support blocks at all six support block locations . This examination is performed from the outside of the core barrel because access to the support block ledge area from the inside of the core barrel is severely restricted and deemed to be impractical due to the presence of the lower support plate and the intermediate flow distribution plate as shown in Figure 2.

3. A 100% visual examination of the core barrel-to-lower support plate weld will be performed both from the inside and outside of the core barrel in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Category BN3. This inspection will be performed as a part of the ten year Inservice Inspection (ISI) program scheduled during the June 1990 outage.

4. The NRC has indicated that an ultrasonic examination should be performed on a representative section of the core barrel-to-lower support weld.

   SCE has concluded that the 100% visual inspection described above will adequately verify the condition of this weld and that an ultrasonic examination should not be attempted. This conclusion is based on the following considerations.

   a) The surface of this weld has not been prepared for ultrasonic examination, and the work required to achieve adequate surface conditions to conduct an examination is not warranted by the benefits which would be obtained.

   b) The loads from the thermal shield lower supports are concentrated at the block locations. These loads are dispersed and distributed throughout the full core barrel circumference and are not concentrated at particular points of the weld.

c) A fatigue analysis was performed on this weld by Westinghouse which demonstrated that the cumulative fatigue usage factor would be less than 1.0 for one fuel cycle of operation in the worst credible degraded condition (all bolts and flexures failed). This analysis will be updated to determine the cumulative fatigue usage factor of the weld for the expected current condition of the thermal shield supports, and future operation with the modified design.

If the ultrasonic examination of the core barrel support block ledge reveals any degradation, SCE will revise the inspection plan and obtain NRC approval of the revised plan.

5. An ultrasonic examination will be performed on all of the dowel pins in the lower support blocks. If indications are present or welds are cracked, the dowel pins will be replaced.

6. All thermal shield lower support block threaded fasteners including the hidden bolts (a total of 30 bolts) will be replaced with new fasteners.

7. The six flexures will be completely removed and discarded. In their place at the top of the thermal shield a new limiter key support system will be installed at six locations. The new limiter keys and keyways will be attached to the core barrel and thermal shield by threaded fasteners and dowel pins (see Figures 3 and 4). This improved design provides for both tangential and radial support at six locations at the top of the thermal shield.

The new limiter keys and keyways will be manufactured from ASME SA-420 type 304 stainless steel. This material will be ultrasonically examined to ASME SA-435 and supplementary requirement No.1. The new fasteners will be fabricated from ASME SA-479 (1986 edition) type 316 stainless steel.

The visual inspections will be performed using an underwater camera calibrated in accordance with the guidelines of the National Bureau of Standards. The camera is equipped with a calibration chart which when used at the correct distance from the camera lens would indicate a resolving power from 12 to 80 lines per millimeter.

The repair and modifications to the thermal shield support system will be performed with the core barrel removed from the reactor vessel. In order to keep the core barrel under water, it will be placed on a support fixture at the reactor vessel flange or suspended from above. The thermal shield will be attached to the core barrel at all times. Removable tooling will be lowered via elevator mast to the work area. From this location remote tooling will be used to replace the lower support block fasteners, remove the old flexures, and perform the necessary machining for the new upper limiter keys and keyways. The lower support work will be performed prior to performing the alignment of the upper limiter keys and keyways. This will ensure that the thermal shield is in its correct position prior to the installation of the new limiter system. Diving operations will not be required for any of the repair tasks. Once the repair is complete the core barrel and the interior of the reactor vessel will be cleaned and inspected.

Analyses and Evaluations

Design qualification and implementation of the thermal shield support system requires various engineering analyses. The analytical methods and techniques used to perform the analyses for the thermal shield support and repair are the same as those which were used by Westinghouse in 1989 to justify continued operation with degraded support fasteners provided in Westinghouse report WCAP-12149, "Engineering Evaluation of the SONGS 1 Thermal Shield Supports." These methods will also be used to qualify the design modifications and the repair effort. Below is a list of the analyses that will be performed:

- Vibration Analysis
- Seismic Analysis
- Thermal Analysis
- Structural, Stress and Fatigue Analysis
- Qualification of the Design Modification
- Evaluation of Loads Generated During the Field Implementation

Vibration Analysis;
The primary flow induced vibratory loading is due to flow turbulence. A turbulent vibration analysis methodology and the required finite element models were previously developed by Westinghouse for use in analyzing the thermal shield. A three dimensional finite element model of the lower internals which includes simplified models of the upper and lower thermal shield supports will be used to perform the vibration analysis. A linear modal superposition technique will be used to compute the total vibratory response of the core barrel and thermal shield to determine the loads acting on the thermal shield support system. These models and analysis will be used to support the repair program. Additional vibration analysis will be performed using the same methodology to investigate the effects of the design changes and to evaluate the impact of potential modifications on the lower support block fasteners.

Seismic Analysis;
Analyses will be performed using simplified finite element beam models and conservative seismic input. Although the changes to the lower supports do not affect the seismic response of the core barrel and thermal shield, the modifications to the upper end of the thermal shield necessitates additional seismic analyses to determine the loads acting on the components of the upper and lower supports.

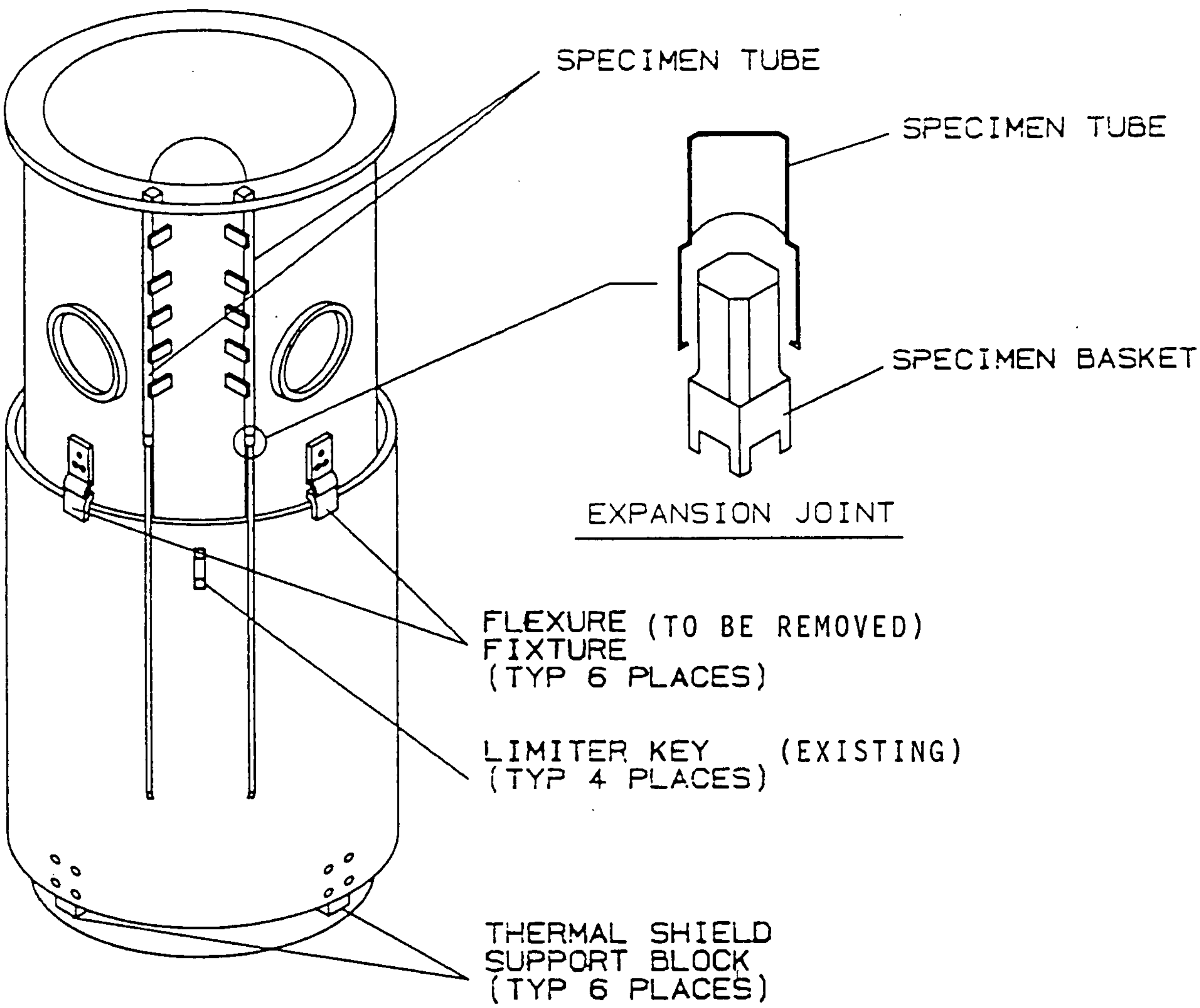
Thermal Analysis;
Differential thermal expansion between the thermal shield and the core barrel during steady state and transient conditions generates loads on the thermal shield support system. These stresses will be determined and incorporated in the design and qualification effort. Steady state and transient thermal analyses using finite element and or finite difference techniques will be used to determine the magnitude of the temperature differences. The temperature profile generated will be used in the structural model to determine loads at

the lower support blocks and the proposed upper limiters.

Structural, Stress and Fatigue Analysis;
Structural evaluations will be performed to qualify the repairs made to the lower support blocks and to qualify the new upper support design. The loads considered in the evaluations are Flow Induced Vibration, thermal, deadweight and seismic. The new upper support design, the connecting hardware to the core barrel, bolts and dowel pins, and the core barrel in this region will be evaluated to the guidelines of the ASME code, Section III. All appropriate loads will be considered in the evaluations and analyses.

Qualification Of The Design Modifications;
Qualification of the design modifications will be performed for the thermal shield modified upper and lower supports. Qualification will include tests performed to support the analytical evaluation. The analytical approach will use the ASME code, Section III, 1986 edition as a guideline.

Evaluation of Loads Generated During Field Implementation;
During the field implementation the tooling used will impart various loads on the thermal shield and the support system. Structural and stress analysis will be performed as required to evaluate the acceptability of these loads.



THERMAL SHIELD SUPPORT
SYSTEM CONFIGURATION

(PRIOR TO DESIGN MODIFICATION)

Figure 1
San Onofre Unit 1
Thermal Shield Repair

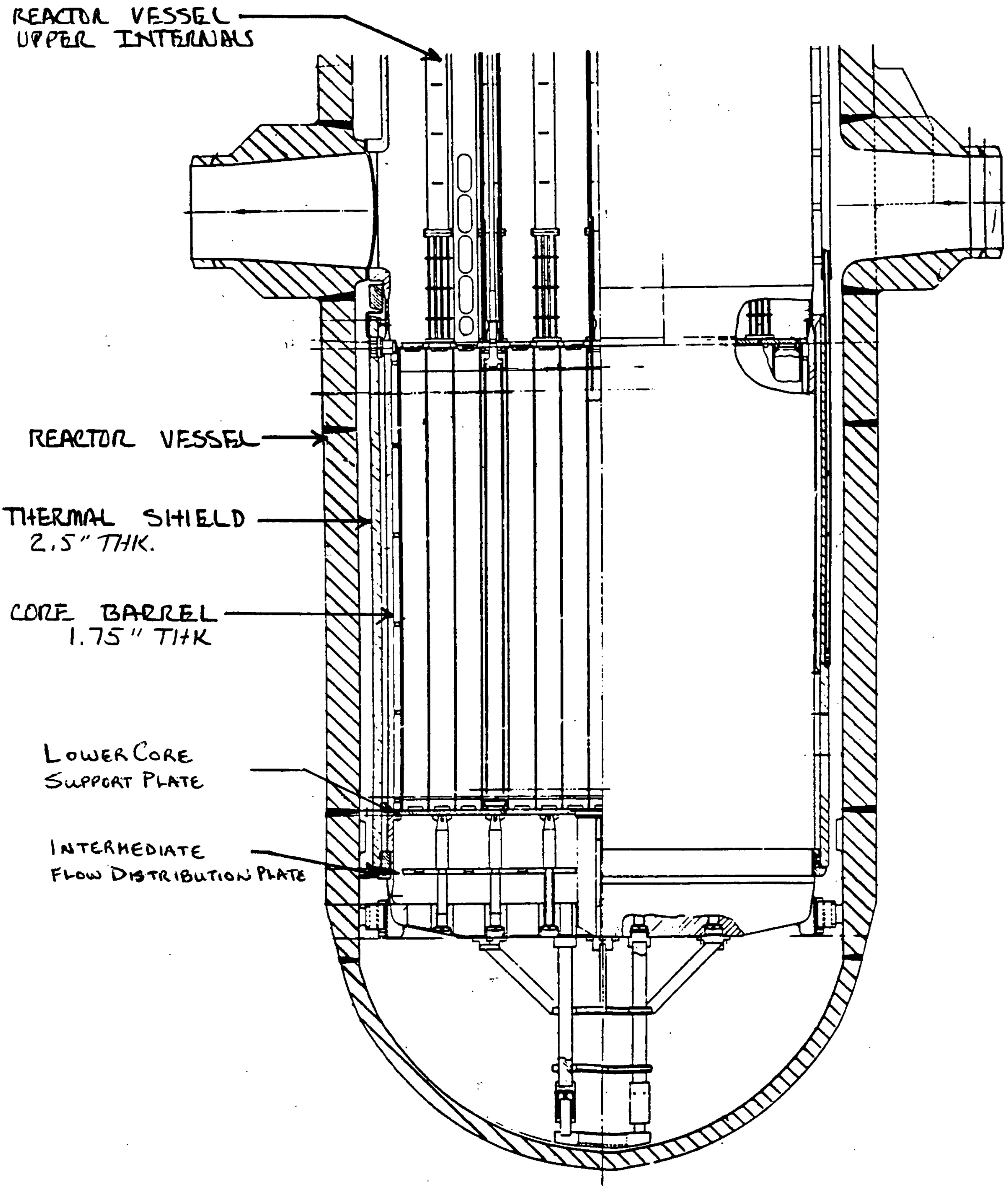
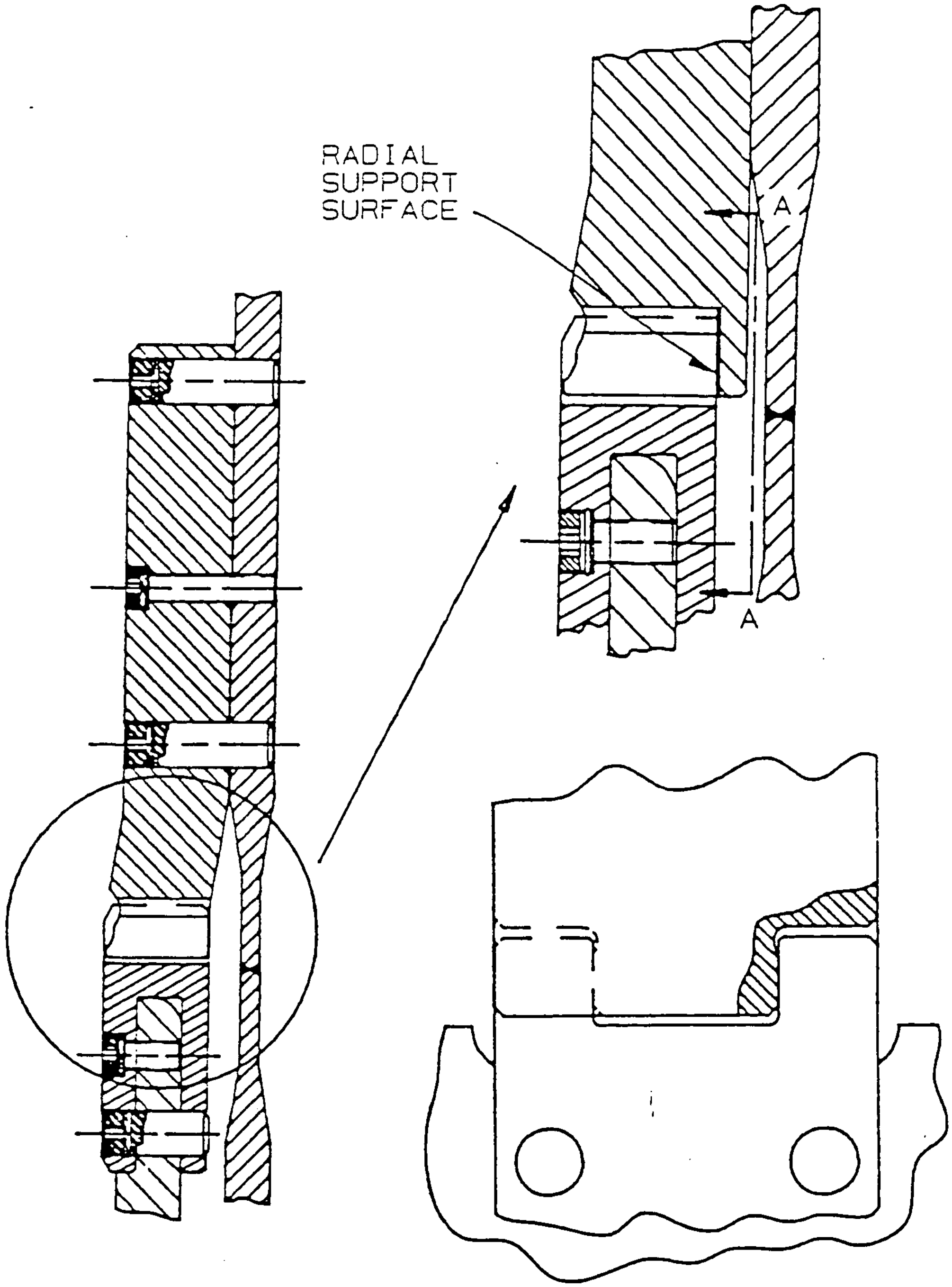
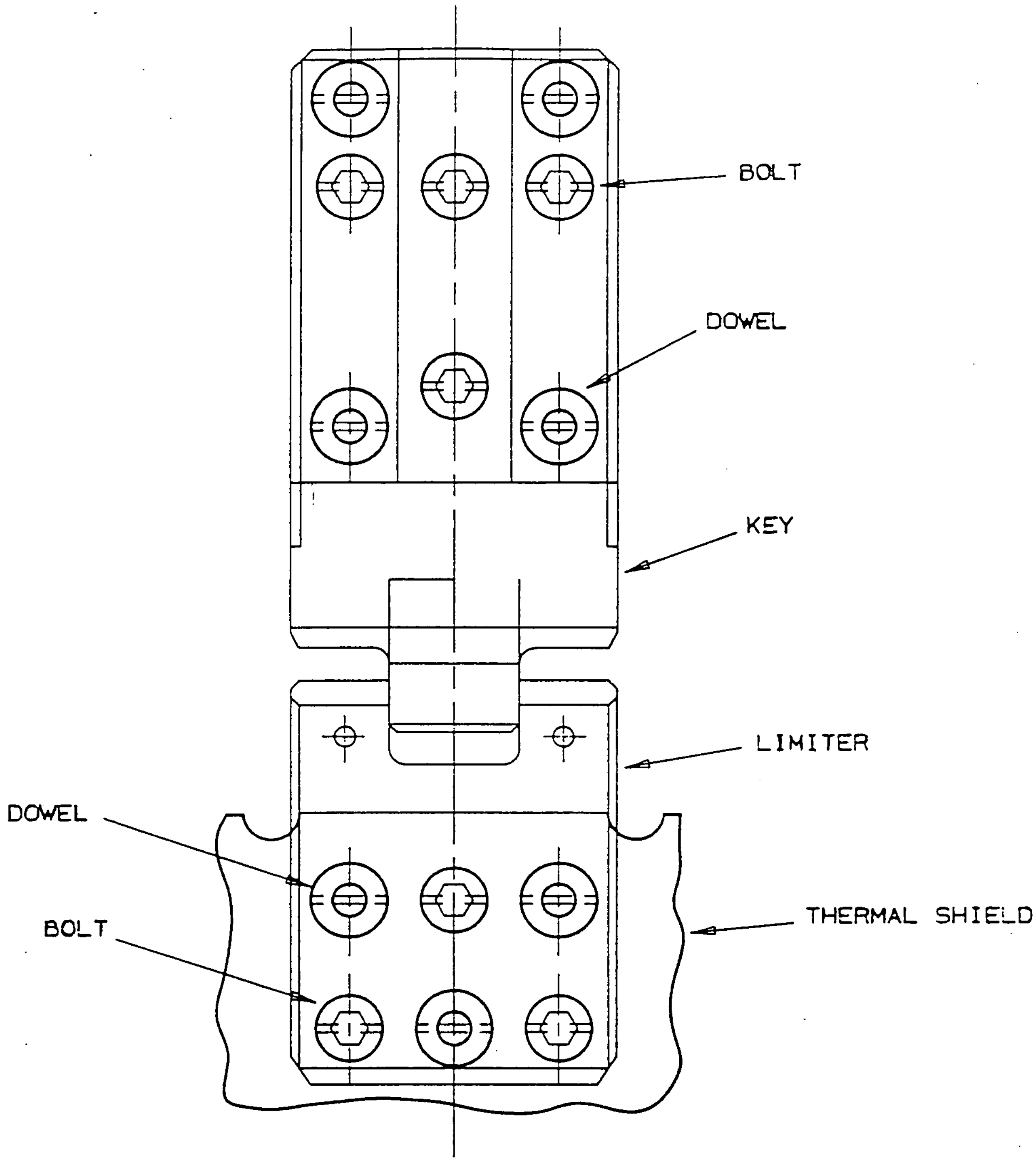


FIGURE 2 REACTOR VESSEL INTERNALS



DISPLACEMENT LIMITER DESIGN CONCEPTS
WITH TANGENTIAL AND RADIAL SUPPORTS

Figure 3
San Onofre Unit 1
Thermal Shield Repair



DISPLACEMENT LIMITER ASSEMBLY

Figure 4
San Onofre Unit 1
Thermal Shield Repair