October 4, 1977

NOTE TO: Karl R. Goller, Assistant Director for Operating Reactors, DOR

FROM: A. Schwencer, Chief, Operating Reactors Branch #1, DOR

SUBJECT: GETR REREVIEW - ABILITY TO COOL CORE ASSUMING COOLING WATER ABSENCE OF FULL SEISMIC DESIGN CAPABILITY

Since it appears likely that both the staff and the licensee will be required to expend substantial amounts of manpower in the review of the geological faults and seismic design of GETR, to assure the availability of containment or of water to cool the reactor fuel in the event of an earthquake, you asked that the ability of the core to be adequately coled to prevent melting following a seismic event be investigated to determine whether the radiological consequences would be acceptable in the absence of full seismic design capability of the facility.

During a recent discussion in my office with Fred Burger, Brian Grimes and members of the Reactor Safety Branch, it was felt by both Brian and Fred that, although the GETR core is small by comparison to power generating reactors, it would be extremely unlikely that the core could depend on being air cooled only any time soon after loss of cooling water without risk of meltdown if one assumed the initiating event to happen while the reactor is at power (50 MWt).

A brief literature search by Fred Burger has confirmed this feeling. T. J. Thompson's "The Technology of Nuclear Reactor Safety" on pages 692-693 (copy attached) describes the results of experiments on fuel temperatures reached on loss of coolant in the Low Intensity Training Reactor (LITR) using 15 aluminum alloy MTR-type fuel elements (of the type used in GETR and the results of later experiments using 21 elements in stead of 15). Based on these experiments as shown on figure 5.1 and in the text of the reference, the potential for fuel melt in air "exists for any reactor which is light water moderated and cooled and uses plate-type MTR elements and operates at power levels of over 1.5 to 2.0 MW. Thus a sudden loss of coolant from such cores during fuel power operation should be regarded as a potentially serious accident."

For comparison GETR, which has 21 MTR plate type fuel elements, operates at 50 MWt at full power while the T. J. Thompson article states that maximum power from 21 elements before melting would occur is about 2250 kw (or 2.25Mw). This threshold value is about 22 times lower than the GETR's 50 MW rating.

Since the GETR SAR, starting on page 9068, analyzes an accident which asumes fuel melt, it is useful to see what dependence that analysis makes on seismically susceptible structures remaining functional from the standpoint of limiting the radiological consequences of a fuel melt.

1. It assumes one of the six 3" diameter reactor pressure vessel bottom head nozzles fail,

2. It assumes an immediate reactor scram on low pressure,

3. It assumes the reactor pool remains intact.

4. It assumes that core melt and subsequent release of its radioactivity would be delayed one hour by making up water lost due to the nozzle break. This water would be supplied from other sources including the emergency pool recirculation systems, the Vallecitos site storage tank and a source of demineralized water,

5. It assumes no breach of the containment building (other than a 1.4%/day leak rate).

Using the above assumptions and applying TID-14844 assumptions, the licensee calculated 272 Rem as the limiting organ dose to a person at the site boundary (206 Rem to a person 2 miles away).

It will be noted that assumption 3, 4 and 5 depend on the structures and systems involved being able to withstand the effects of the initiating event. In order for those assumptions to be valid, and the resulting doses to be within 10 CFR 100 guidelines, substantial credit must be given for the structural integrity of several structures, components and pipe runs. Because of this, it would appear that the staff must have reasonable assurance of the ability of key items to withstand seismic forces.

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cc: V. Stello
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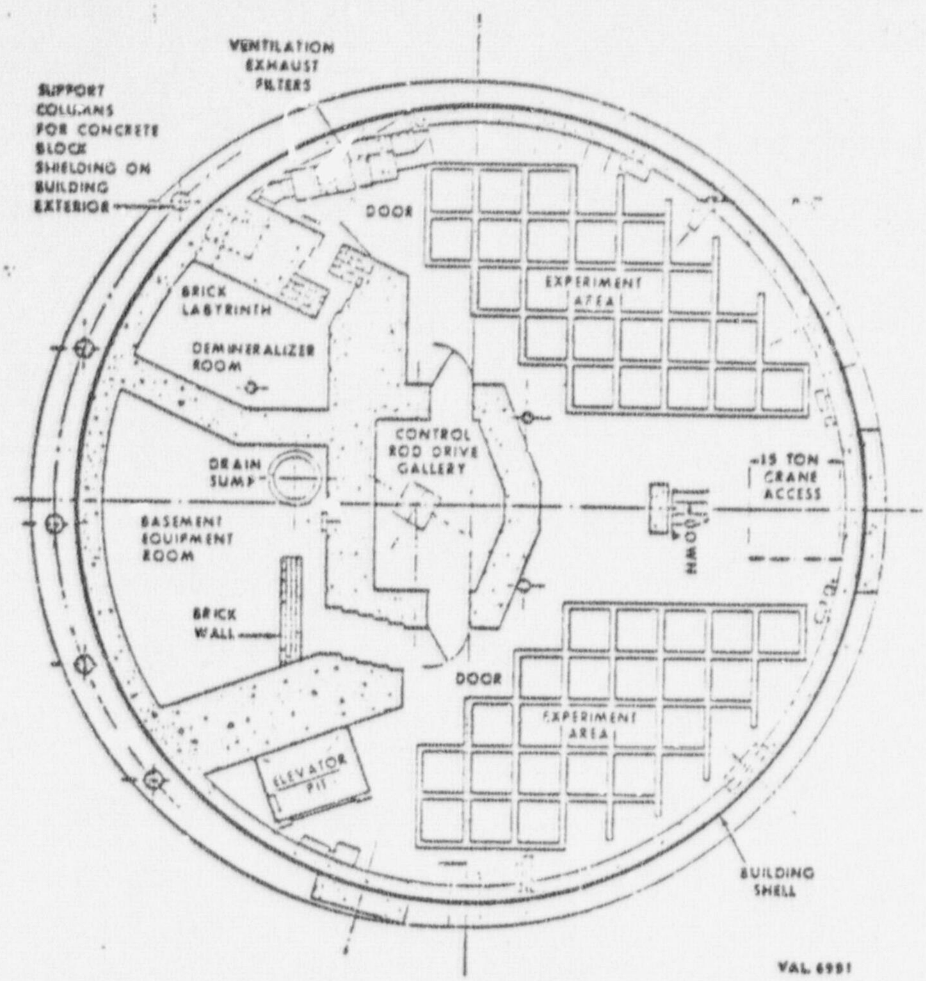


FIGURE 4-3. REACTOR BUILDING BASEMENT PLAN

### 4.2.3 Vacuum Relief System

The containment building liquid leg vacuum relief system is an engineered, safety-related system designed to relieve air automatically into the containment building should the containment shell limit for vacuum be reached. This system is essentially a large-scale manometer and is part of the containment barrier.

The liquid leg vacuum relief system is located outside the containment building at the ventilation inlet. The system (Figure 4-7) consists of a 4 ft by 4 ft by 0.5 ft tank, a 6-in. diameter "vacuum" stack connected to the containment building, a 6-in. atmospheric stack open directly to the atmosphere, a 5-gal reservoir which contains the manometer liquid (fully concentrated ethylene glycol antifreeze fluid), and associated liquid level instrumentation and sight gages. The tank, two stacks, and supply reservoir are fabricated of 6061-T6 aluminum.

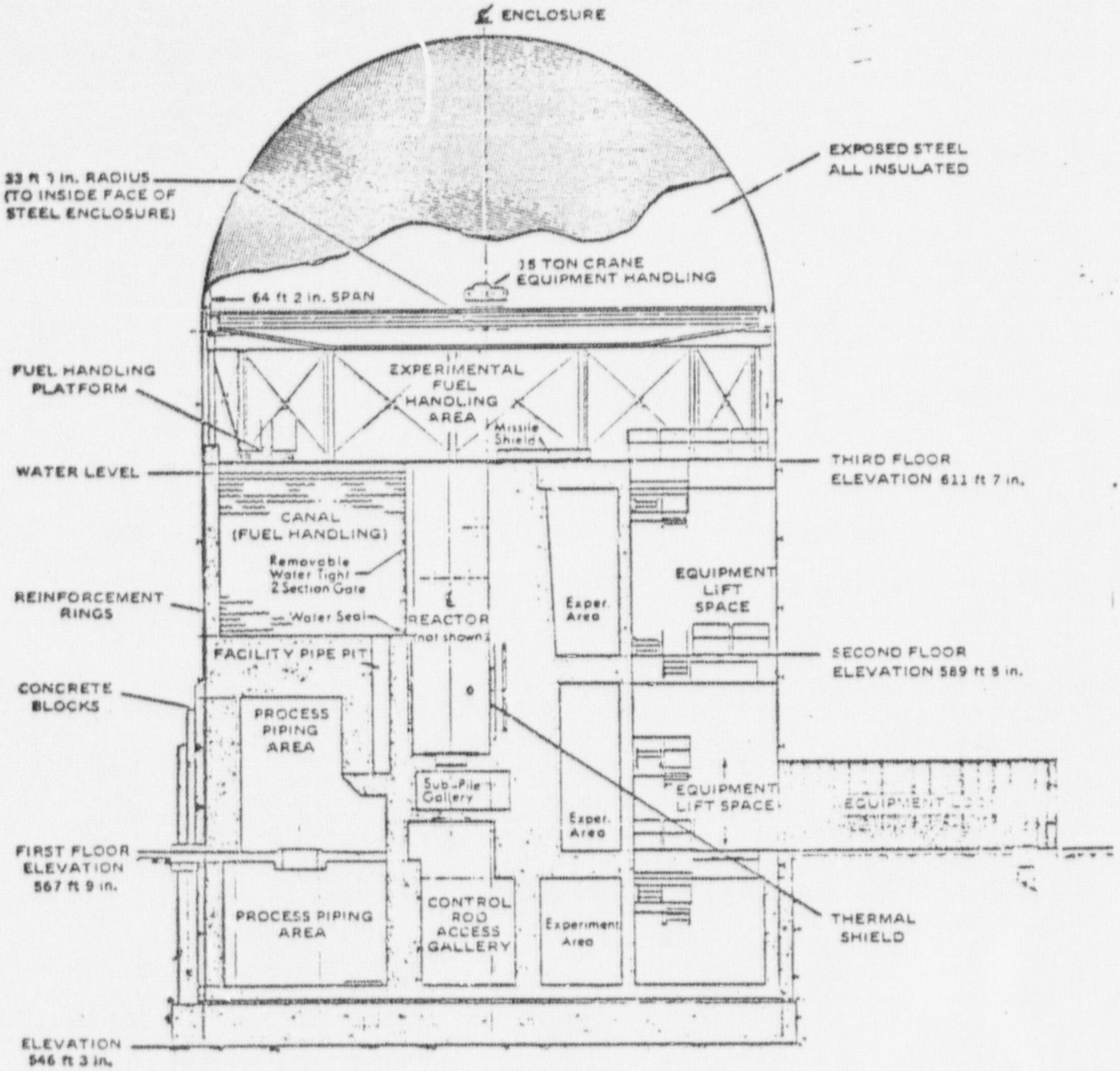


FIGURE 4–4. REACTOR BUILDING CENTERLINE SECTION

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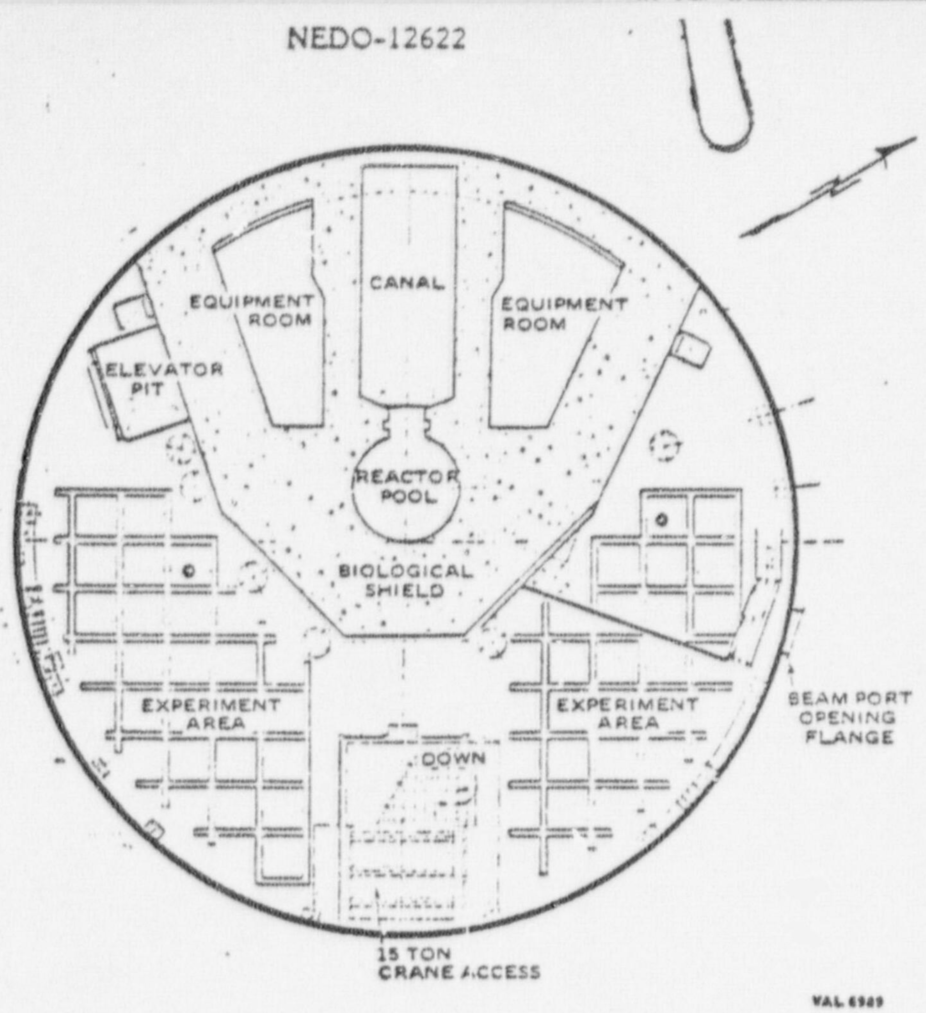


FIGURE 4–5. REACTOR BUILDING SECOND FLOOR PLAN

If the pressure becomes sufficiently subatmospheric in the containment building, the liquid level will drop to the bottom of the atmospheric stack and air will bubble into the system through the open atmospheric stack. The system is designed to relieve at a vacuum of approximately −0.1 psi to ensure the building design limit negative pressure differential of −0.20 psi is never reached. In the event of a pressure buildup in the containment, the pressure would force the liquid into the atmospheric stack. The system is designed to contain pressures to 8 psig.

Redundant, hermetically sealed magnetic switches are installed to monitor the liquid level continuously. These switches actuate an alarm in the control room if a high or low liquid level is reached. Glass sight gages are connected to both stacks to provide a visual indication of the system liquid level. The complete system is checked every 12 months to ensure its operability.

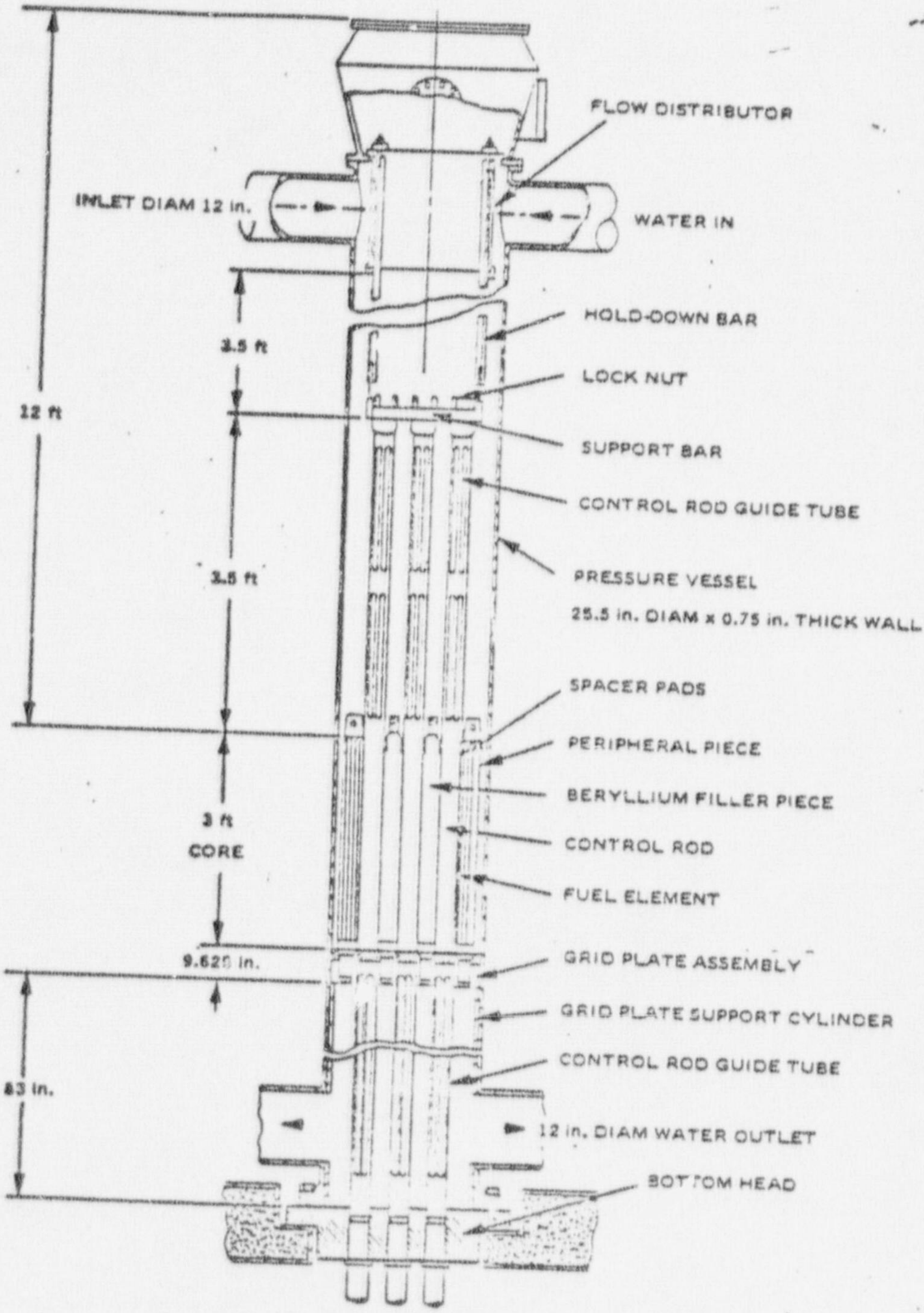


FIGURE 4-11. REACTOR AND CORE ARRANGEMENT

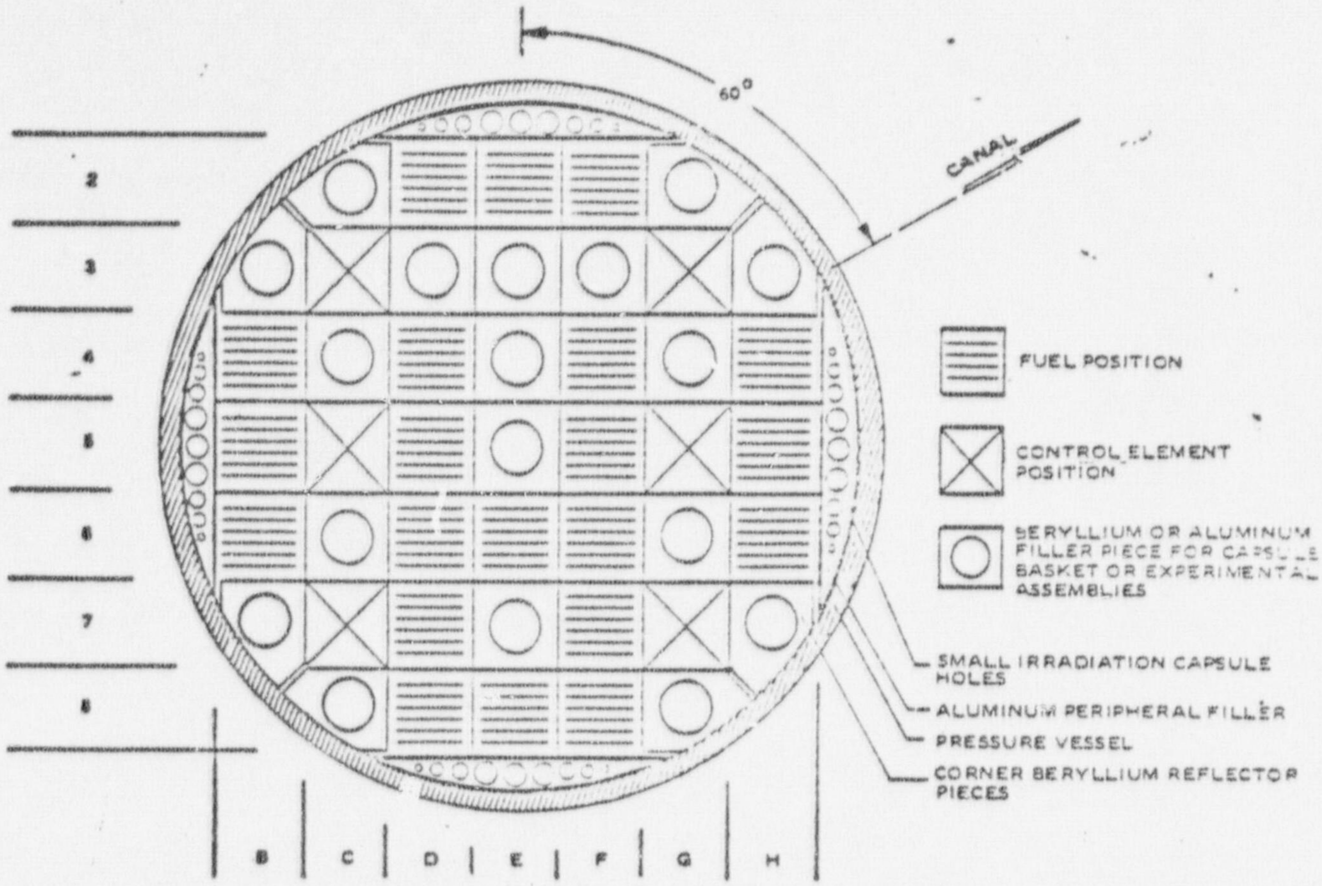


FIGURE 4–12. REACTOR CORE PLAN FOR TYPICAL 21-ELEMENT CORE

Depending upon the in-core and pool experiment loadings, the number and location of fuel elements may be modified as necessary to provide an adequate reactivity balance or flux distribution or both. The normal fuel loading for the past several years has consisted of 20 or 21 fuel elements and 6 control rod follower sections. Figure 4–12 shows the 21-fuel element, skewed-core loading used most frequently in recent years. A 20-element loading gives a symmetrical core arrangement with the fuel elements in the F-7 and D-7 positions removed and replaced with beryllium filler pieces and a fuel element in the E-4 position replacing a single-hole beryllium filler piece.

Square beryllium or aluminum filler pieces and associated capsules or experiments occupy matrix positions not used by control rods or fuel elements. Specially shaped aluminum and beryllium peripheral pieces surround the fuel-filler array and round the core into a cylinder.

The eight standard beryllium corner peripheral pieces have 1.5-in. diameter holes to accommodate experiment or isotope capsules. The four aluminum side peripheral pieces

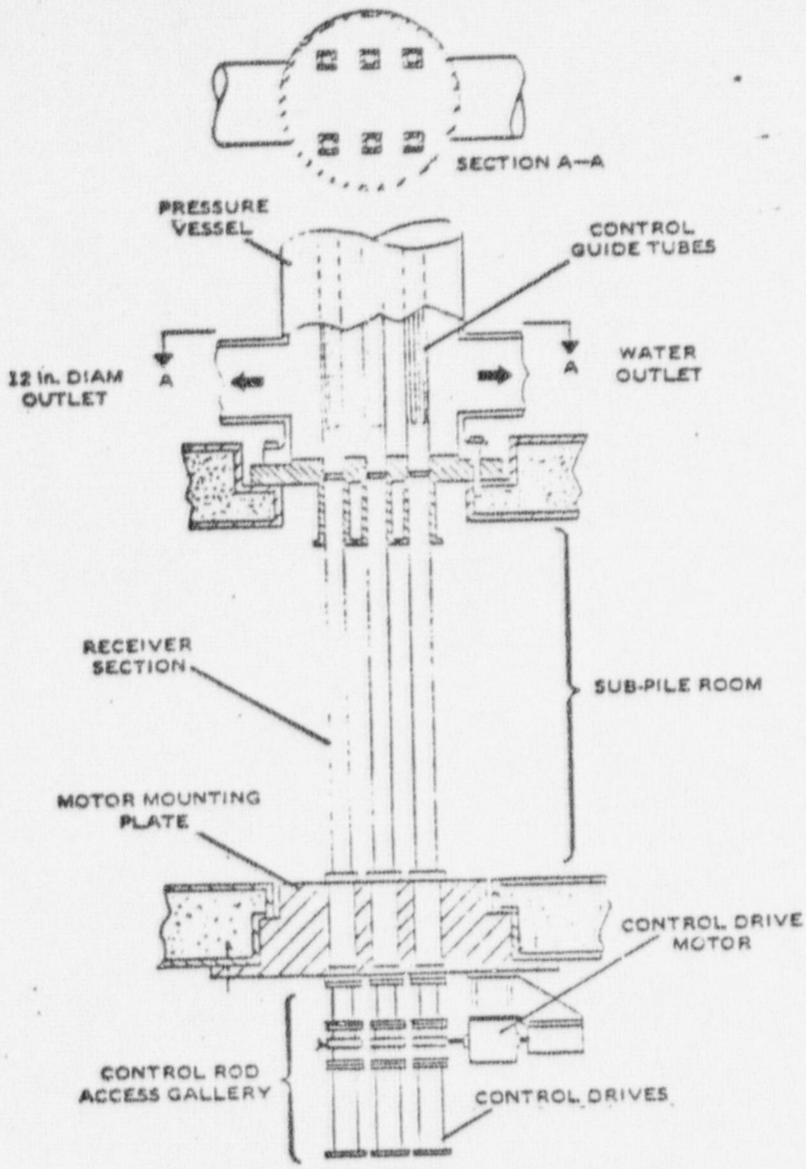


FIGURE 4–13. CONTROL ROD DRIVE ARRANGEMENT

tolerances for finished fuel and control rod followers. Sampling and testing plans are described in the specifications to ensure the actual fabricated fuel meets the design and quality requirements. Chemical, radiographic, and visual examination methods are used during various stages of fuel manufacture to maintain rigid quality control.

Fuel elements are replaced or changed primarily on the basis of reactivity worth. An axial average burnup level of 50% is not exceeded for this type fuel assembly. New fuel is usually inserted around the periphery and, after partial burnup, it is moved to the central region of the core. Normally, each fuel element is used in the core for about 6 to 8 power runs where each power run is typically 10 to 18 days of full power operation. Fully enriched uranium-aluminum, plate-type fuel has a long trouble-free record of performance in the GETR as well as in other test reactors.

had been actuated. The operation of the float shut off the demineralizer pump, but the leakage continued by back flow through the pump into the demineralizer room. Even if back flow had continued as long as possible, the level would have been maintained several feet above the active core.

In November 1963 a supervisor at the Texas Agricultural and Mechanics College Reactor entered the reactor building to find that the pool water level had dropped eight feet [50]. A gasket in the demineralizer tank access hole had failed and water had flowed, ultimately, to the hot sump. From there it was automatically pumped to a holdup tank from whence it overflowed, letting most of the water spill onto the ground. From there it went to a dry gully. The pool water level had an activity of 43 $\mu\mu c$/ml. No serious consequences resulted.

### 5.3.2 Fuel Temperatures Reached on Loss of Coolant in the LITR [81, 82, 83]

An early set of experiments done at Oak Ridge has provided some idea of the power levels at which melting might be expected in a tank-type or a swimming pool reactor due to loss of coolant [81, 82]. The Low Intensity Training Reactor (LITR) was employed for these tests using 15 MTR-type elements and three control-safety rods in a 3 × 6 lattice reflected by beryllium on all but one face. The experimental procedure consisted of running the reactor at a constant specified power for the desired length of time, shutting the reactor down by draining out the water (no drop of control rods), and then following the fuel plate temperature in the hottest element until a maximum was passed.

About five minutes before the run was to be terminated, the cooling water pumps were stopped and the inlet and outlet valves closed. A port was opened in the top of the tank as a vacuum vent. Then a 6 in. (15.2 cm) remotely operated valve was opened. It required 2.5 min to lower the level from the tank top to a point 1 ft (30.5 cm) above the fuel plates. At that point the drop in water level began to affect the reactivity and rapidly shut the reactor down. It required only 12 sec more for the water level to drop below the fuel plates and 30 sec more for all water to drain from the tank. The valve and the port in the top were then closed. Temperature readings were continued for two hours, which in every case up to 300 kw was time enough for the maximum temperature to be reached and passed. Nine 2-hr power runs at specified levels from 0.5 kw to 500 kw were carried out. The effect of longer runs was measured by runs of 2, 6.5, and 24 hours at 150 kw.

Later experiments extended these measurements to 1250 kw and 150 hr operation. These produced fuel element temperatures as high as 249°C (480°F) and end-box temperatures above boiling. This later set of experiments used 21 elements instead of 15 and the results had to be normalized by comparing neutron flux measurements in the central elements (1.5 times the maximum temperature observed in the 21-element case).

The observed points extrapolated to infinite running time for a 15-element core are plotted in Fig. 5-1 [82]. It is evident that the 15-element core should melt on loss of water after operating at power levels above about 1500 kw. With a 21-element core the maximum power before melting on loss of coolant would be about 2250 kw.

The added points in Fig. 5-1 show the central fuel plate temperatures reached in tests in which water was sprayed over the fuel at a rate of 2 to 6 gpm (0.13 to 0.38 liter/sec) until a 500 gal (1893 liter) capacity tank was empty. Spraying kept the fuel plate temperatures below 212°F during the 2.5 hr it continued. It was demonstrated in these tests that the LITR could reject heat continuously without melting fuel at a rate of 8 kw. Beall [82] estimates that at least twice this rate is possible before melting occurs.

The equation obtained for the fission product power in a single MTR-type element with 140 g of $U^{235}$ was [81]:

$$q = 14P\,[t^{-0.2} - (t + T)^{-0.2}]\,,$$

where q is in Btu/hr, P is the reactor power (kw) before shutdown, t the time (sec) after shutdown, and T the operating time (sec) before shutdown.

Beall concludes that as much as 75% of the fission product heat is lost by conduction to other parts of the reactor before the fuel reaches a maximum temperature. He also points out that calculations given by Poppendiek and Claiborne [83] estimate melting at much lower power levels. He believes that the difference is primarily due to improvement in thermal conductivity across heat conduction gaps due to water in the gaps. Thus, most of the heat seems to be carried away by aluminum conduction to the metal base plate (weight = 1000 lb or 454 kg).*

### 5.3.3 Control Rod Experience [78]

In the University of Michigan pool reactor [78], and in other reactors as well, it has been observed that some of the relatively flat blade-type control rods have become waterlogged and have expanded to assume a more cylindrical shape. The expansion appears sometimes to be due to gas formation in boron-containing compounds and sometimes to corrosion gas evolution.

This expansion has led to binding of control rods within their guides in the fuel elements. There then exists the possibility that the fuel element within which the control rod moves will be picked up as the rod is picked up, only to be later dropped back into the core resulting in a nuclear transient. This same type of control rod expansion might also prevent a control rod from

*Recently R. Panter, AERE Harwell, has completed work for the "Dido"-class reactors (which use MTR-type fuel plates in various configurations) on fuel element temperatures during loss-of-coolant accidents and during fuel transfer operations.

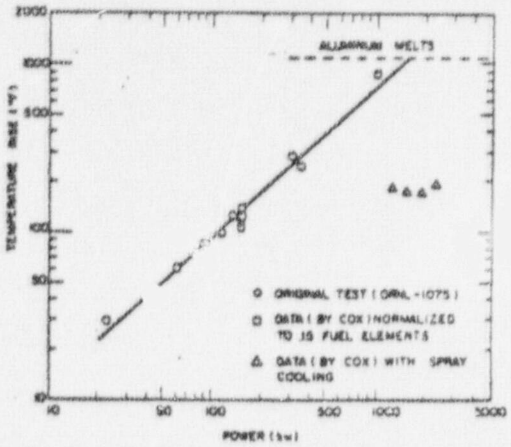


FIG. 5-1 Temperature rise of center fuel element vs. total reactor power for infinite running time. LITR—15-element core, 3 control rods.

dropping back into the core. This problem is discussed further in the Mechanical Design chapter.

Comments, Conclusions, and Recommendations

(1) It is wise in any reactor where loss of water can occur relatively easily (mostly in research and test reactors) to provide alarms to alert personnel to the problem and to take preventive action. If the reactor is to be unattended for an extended period perhaps the alarm should give a switchboard or telephone alert.

(2) It is well to remember that in many reactors—especially research and test reactors—the potential exists for ejection of plugs from reactor access holes. They may be ejected either by water pressure under relatively normal operating conditions or during the course of an accident. Such plugs should be secured in the shield by means of some strong latching device. It is also possible that beam tubes can collapse causing serious reactivity increases.

(3) At least one radiation alarm should remain in its sensitive state in each potential radiation area around a reactor—even when the reactor is off.

(4) The potential to melt fuel exists for any reactor which is light-water-moderated and -cooled and uses plate-type MTR elements and operates at power levels of over 1.5 to 2.0 Mw. Thus, a sudden loss of coolant from such cores during full power operation should be regarded as a potentially serious accident. The beta-gamma after-heat can give core melting. It is likely that single elements and widely spaced elements will not melt until somewhat higher power levels are reached since there is less interaction heating between elements.

(5) Special care should be taken that control rods are well designed, fabricated, and used so as to ensure that no expansion or distortion occurs in such a manner as to reduce reactor safety.

## 5.4 Material and Mechanical Failures

Evidence continues to show that mechanical and material failures can and do occur. At least four of these failures have had the potential for causing a serious accident, although none actually did occur. Other failures of a more minor nature occur more frequently and are usually not reported. Five well-documented examples of this more minor type are briefly described as typical.

SM-1 (APPR) Closure Bolt Failure (84). After 10.5 Mw-years of operation of this 10 Mw(t), 1200 psig (82 atm) pressurized water reactor, the vessel head was removed for a core examination. At that time it was discovered that two adjacent vessel head studs were br[illegible] additional studs had failed, a serious acc[illegible]ld have resulted. The breaks were det[illegible]ave been caused by stress corrosion [illegible]he Type 410 stainless steel studs [illegible] heat-treated by heating to the ra[illegible]°F (934 to 982°C) and held for 0.5 [illegible]empered at 1000°F (538°C) fo[illegible] Brinell hardness numb[illegible] Hardness ($R_C$) = 26.5). [illegible] showed that hardness r[illegible] BHN (18 to 41 $R_C$). (Durin[illegible]tion it was also found that the [illegible]e boron absorbers in four shim rods [illegible]ely cracked in the high burnup region [illegible]s attributed to helium gas formation.)

Vallecitos Boiling Water Reactor Main Steam Line Valve Failure (85). On March 9, 1960, the valve plug separated from the valve stem in the main steam pressure-reducing valve. The failure almost instantaneously stopped the flow of steam while the reactor was operating at 30 Mw(t). The reactor pressure rose, and the core void fraction decreased. This gave a positive reactivity effect and the reactor power rose to 35 Mw(t), the set point for an overpower scram, and the reactor scrammed in 5-10 sec after the valve failure. The reactor pressure had risen by about 50 psig (2.7 atm) above the nominal 1000 psig (68 atm) operating pressure. No serious consequences resulted.

SPERT-III Pressurizer Failure (86, 87). On October 26, 1961, during a series of test runs on the SPERT-III pressurized water reactor, the system was brought to 221°C (430°F) and 2460 psig (167 atm). About three hours later, smoke was observed coming from the vicinity of the pressurizer. A normal shutdown was started and the fire department was alerted. The plant was cooled down and depressurized without need for the fire department or further incident.

Inspection showed that the smoke came from the fabric covering of the blowdown-line riser, which was in line with a major steam leak in the pressurizer. A 3/8 in. (0.95 cm) wide, 2-3/4 in. (6.99 cm) long hole was found in the central girth seam weldmetal. A 1 in. (2.54 cm) diameter bolt which tightened a stabilizing band around the vessel was broken. Corrosion on the broken face of the bolt and subsequent investigation indicated that the bolt broke in an earlier and separate expansion of the vessel.

The pressurizer is a 33 in. (83.82 cm) inner diameter, 16-2/3 ft (5.08 m) high all-welded vessel of ASTM A-264, grade 3, 0.04% max. carbon steel with 304 L stainless steel fittings and internal cladding. The backing plate was ASTM A-212,