NUCLEDYNE

ENGINEERING CORPORATION

728 West Michigan Avenue
Jackson, Michigan 49201

May 15, 1979

Represented by
O. B. Falls, Jr.
Consultant

Mr. Victor Stello, Director
Reactor Operations Division
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

The energy supply situation in our country needs immediate attention to prevent electrical blackouts in the near future which could lead to extreme hardships to homelife and business. Majority public opinion supports nuclear power as a necessary energy source.

The country needs a firm, definitive statement of support and encouragement by President Carter, Energy Secretary Schlesinger and the several Congressional Committees, having an interest in our energy policy and supply, as a basis for rejuvenation of the nuclear power industry. Also, there is a current need for substantial safety improvements for light-water reactor (LWR) power plants. Consequently, we sent a Mailgram to President Carter with copies to Secretary Schlesinger and NRC Chairman Joseph Hendrie. A copy of this Mailgram is enclosed. We would call your attention to the references to the Three Mile Island incident. Your known interest in the energy situation in our country has prompted us to write to you.

To support the claims made in the Mailgram a NucleDyne document is enclosed. This is a copy of a paper presented at the American Power Conference in Chicago on April 24, 1979; "Passive Containment System - A New Concept to Solve Safety Concerns". This paper responds specifically to the five safety research projects recommended to the Congress by the NRC in Report NUREG-0438, dated April 12, 1978. Also, some of the benefits that are derived from a licensed nuclear power plant with the new safety features are enumerated in the enclosed "Application of the PCS produces the following results". Extra copies of these publications are available on request.

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Mr. Victor Stello May 15, 1979

The PCS incorporates the substantial improvements needed for the LWR power plants. We request your urgent attention to our claims as stated in the Mailgram and discussed in the enclosed document. Furthermore, we request an opportunity to visit you to validate our claims. Your support in the application of these safety improvements will enable the LWR power plants to become a viable basic source of energy. This, in turn, may encourage President Carter and others to a firm statement in support of nuclear power.

We await your response to our request to meet with you.

Sincerely,

O.B. Falls, Jr.

O. B. Falls, Jr.
Consultant

OBF/mr
Enclosures

MAILGRAM SERVICE CENTER
MIDDLETOWN, VA. 22645

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NUCLEDYNE ENGINEERING CORP
728 WEST MICHIGAN AVE
JACKSON MI 49201

THIS MAILGRAM IS A CONFIRMATION COPY OF THE FOLLOWING MESSAGE:

5177874742 MGM TDBN JACKSON MI 187 05-07 0207P EST
ZIP
JIMMY CARTER, PRESIDENT UNITED STATES OF
AMERICA
WHITE HOUSE DC 20500

THREE MILE ISLAND (TMI) INCIDENT NECESSITATES URGENT ATTENTION TO NEW DESIGN CONCEPTS THAT IMPROVE SAFETY OF PRESENT AND FUTURE NUCLEAR PLANTS.

NRC HAS KNOWN FOR MORE THAN THREE YEARS OF THE UNIQUE PASSIVE CONTAINMENT SYSTEM. (PCS) FOR LIGHT WATER REACTOR (LWR) PLANTS WHICH WOULD HAVE PREVENTED CORE DAMAGE AND THE RELEASE OF RADIOACTIVITY UNDER TMI CONDITIONS; PLANT RECOVERY WOULD HAVE BEEN IMMEDIATE. PCS RESPONDS TO SAFETY RESEARCH PROJECTS/TOPICS RECOMMENDED IN NRC REPORT TO CONGRESS, NUREG-0438.

NRC REFUSES CONSIDERATION OF THIS NEW CONCEPT ON GROUNDS SAFETY EVALUATION IS TO DEMANDING FOR AVAILABLE NRC STAFF. RECENT ADVERSE EVENTS FOCUS ATTENTION ON THE NEED FOR NEW SAFETY CONCEPTS FOR NEXT GENERATION OF LWR NUCLEAR PLANTS WHICH ELIMINATE POSSIBLILITY OF ANOTHER TMI TYPE INCIDENT.

WE SOLICIT YOUR SUPPORT OF ACTION BY NRC AND DOE TO REVIEW PCS SO INDUSTRY IS ASSURED OF TIMELY REGULATORY LICENSING OF PLANTS USING PCS. A MEETING IS REQUESTED WITH APPROPRIATE MEMBERS OF YOUR STAFF AND COMMITTEES INVESTIGATING TMI INCIDENT TO FULLY VALIDATE CLAIMS REGARDING PCS FOR NEW PLANTS AND RETROPROFIT OF PCS EMERGENCY CORE COOLING SYSTEM ON EXISTING PLANTS.

NUCLEDYNE ENGINEERING CORP
BY O B FALLS JR, CONSULTANT

14:07 EST

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PASSIVE CONTAINMENT SYSTEM

A NEW CONCEPT TO SOLVE SAFETY CONCERNS

AMERICAN POWER CONFERENCE
Chicago, Illinois
April 24, 1979

Authors: O. B. Falls, Jr.
F. W. Kleimola

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Jackson, Michigan 49201

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REFERENCES

1. NUREG - 0438, Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants - A Report to the Congress of the United States of America, Nuclear Regulatory Commission - April, 1978.

2. NucleDyne Engineering Corporation, Safety Improvements With The Passive Containment System - Prepared in Response to NUREG - 0438, June 12, 1978.

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PASSIVE CONTAINMENT SYSTEM

A NEW CONCEPT TO SOLVE SAFETY CONCERNS

O. B. Falls, Jr. & F. W. Kleimola
NUCLEDYNE ENGINEERING CORPORATION

AMERICAN POWER CONFERENCE
Chicago, Illinois
April 24, 1979

INTRODUCTION

In the Passive Containment System (PCS-2) the innovative design features incorporate alternate and advanced engineering safety features for light-water reactors. These features interface on the five research projects suggested by the Nuclear Regulatory Commission (NRC) in NUREG-0438 to improve the safety of light-water nuclear plants (Ref. 1). A preliminary response to NUREG-0438 on the safety improvements offered by the PCS was prepared and submitted to the NRC (Ref. 2).

This presentation is specifically directed toward the improvements offered by PCS-2 to the five research projects for a four-loop pressurized water reactor (PWR). The improved safety features are equally applicable to other PWRs and to the boiling water reactors (BWR).

PROJECT A - ALTERNATE CONTAINMENT CONCEPTS

The Primary Reactor Containment System in PCS-2 is a thick-walled steel structure composed of interconnected cells. Individual cells house the components of the reactor coolant system (RCS) and the engineered safety systems. These components include: reactor vessel, steam generators, reactor coolant pumps, reactor coolant pipes, and the pressurizer; also the deluge, refill, and quench tanks. A typical cell arrangement for two loops in a four-loop PWR is shown in Figs 1 to 9 inclusive.

For accident purposes the air within the free volume of the containment is evacuated to less than 2 psia. Electrical equipment requiring cooling is housed in separate compartments; namely, the reactor coolant pump motors, the control rod drives, and the greater portion of the pressurizer.

The containment is designed with a free volume of 250,000 cu ft. The deluge and refill tanks along with the steam generator secondaries contain sufficient liquid to flood the free volume to an elevation above any postulated RCS pipe break.

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An evaluation of the engineered safety systems (Fig. 10) in the loss of coolant accident (LOCA) is made for the worst case, a double-ended, guillotine-type, pump-suction pipe break. Steam carryover into the deluge tanks (Fig. 6) initiates at less than 5 psia. Each of the four deluge tanks is designed with twelve (12) or more 12-inch vent pipes. Each vent pipe penetrates the top head of the deluge tank and extends almost the length of the tank. These vent pipes are perforated by thousands of small orifices to facilitate an immediate quench of the steam carryover by the borated deluge water. Each vent pipe is encircled by a shroud pipe to promote thermal circulation past the orifices and within the tank. Sufficient freeboard space is provided for the steam mass carryover and for thermal expansion in the LOCA.

Additional heat sink capacity for the LOCA is provided by the quench tank fluid in the post-accident time period (Fig. 8). One or more vent pipes (patterned after the vent pipes in the deluge tanks) are installed at each quench tank. Thus, the quench tanks provide for a vented containment with a heat capacity equal to that of the deluge tanks.

RCS blowdown in the LOCA initiates steam flow through the deluge tank vents and the quenching of the steam at the orifices. Steam flow into the deluge tanks for the postulated pump suction break is traced in Fig. 11. Maximum steam carryover occurs at a little over four seconds into the accident with about 13,300 lb/sec of steam, representing 15.7 million Btu/sec of energy, quenched by the deluge water.

The steam carryover increases the liquid volume within the four deluge tanks as traced in Fig. 12. The total volume increases from 53,400 cu ft to about 58,000 cu ft and a corresponding increase in temperature from 50 F to about 128 F. With initial vacuum conditions, both in the containment free volume and at the freeboard, the liquid volume increase as a result of steam carryover does not impose an added pressure on the containment for the post-accident period. Approximately 500 cu ft of post-accident freeboard is allotted for each deluge tank. The post-accident freeboard is at a vapor pressure of 2.1 psia (the saturation pressure of the 128 F water) toward the end of the containment pressure transient.

The evacuated containment and steam carryover into the deluge tanks have a decided beneficial effect in the LOCA. A curve of the containment pressure response to the pump suction break is shown in Fig. 13. The containment pressure peaks at about 75 psia. At this point the amount of energy in the steam flowing into the deluge tanks, plus the energy retained in the saturated water in the containment, starts to exceed the RCS blowdown energy and the containment pressure reducers. By the end of the RCS blowdown, approximately 27 seconds into the accident, the containment atmosphere has reduced to sub-atmospheric pressure.

In the post-accident period of a LOCA, any radiolytic hydrogen released from the borated water flooding the containment is safely handled by a vacuum pumping system. A cold trap positioned at the intake to the vacuum pump removes

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water vapor carryover. The hydrogen is pumped into the holding tanks of the gaseous radwaste system for processing through recombiners.

Plant recovery from a LOCA is immediate. With the reactor refueling enclosure removed from the reactor containment (Figs. 3 & 4), the fuel is removed from the core before containment decontamination proceeds. After the fuel is retrieved, the borated water flooding the containment is processed through the radwaste demineralizers and stored for reuse. Containment decontamination is more readily accomplished in the thick-walled steel structure; there is no contaminated thermal insulation to be removed, and there are not concrete surfaces requiring decontamation. Equipment within the containment requiring decontamination is at a minimum. All moisture is readily removed with the vacuum pumps present. Any faulted RCS component can be replaced through access openings (Fig. 4), or through roof closures (Fig. 5). Steps, to recommission the plant, can be undertaken during recovery operations.

In summary, PCS-2 offers an alternate containment concept that has an inherent venting feature. As stated by the NRC (Ref.1), "the objective of alternate containment designs is to reduce the probability of containment failure and subsequent release of airborne radioactivity." The heat sink capacity of the deluge, refill, and quench tanks is over 1300 million Btu. This heat capacity is more than 3.5 times that required for the pump suction pipe break (367 million Btu). There is no danger of subsequent release of airborne radioactivity in that the venting is contained and the containment does not become pressurized above atmospheric even in consideration of the 1300 million Btu of energy.

PROJECT B - ALTERNATE EMERGENCY CORE COOLING CONCEPTS

PCS-2 employs steam from a stored energy source for initiating emergency core cooling in the LOCA. In the PWR the enormous amount of stored energy in the steam generator secondary is employed as the motive fluid for steam jet injectors positioned within refill tanks (Fig. 9). With the reactor in operation the refill tanks - completely filled with chilled, borated water - are maintained at secondary system pressure.

Depressurization of the RCS in the postulated LOCA passively initiates ECCS. The check valves (Fig. 10) positioned in series at the piping interconnecting the refill tanks to the RCS automatically open as soon as the RCS is depressurized below secondary system pressure. Safety injection piping is routed from the bottom of each refill tank through an adjoining deluge tank cell. From the deluge tanks the piping is routed to a "cold leg" within a reactor coolant pump cell or to a "hot leg" within the steam generator cell.

Depressurization of the refill tanks automatically initiates steam flow to the injectors within the refill tanks. A steamline is routed from each steam header through the top end of an adjoining quench tank cell. From the quench tank cell each steamline is directed into a refill tank where the line branches to a number of injectors in a parallel array. Steam flow through the injectors,

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entrains the borated water, providing rapid safety injection at high pressure. Each refill tank is paired to a steam generator. Each "cold leg" and each "hot leg" in the reactor system has an interconnecting pipe from a refill tank for safety injection purposes.

A more detailed analysis of the spectrum of pipe breaks may prove that safety injection to each "cold leg" only is more effective than injection into both cold and hot legs.

Detailed analyses of ECCS requirements for the spectrum of pipe breaks in the intermediate range may show further advantages for a separate stored energy source at a temperature and pressure higher than the secondary system for passive safety injection utilizing injectors. This stored energy source would entrain borated water from a separate refill tank also maintained at a higher pressure. but at a lower temperature than the secondary system.

The economy of the steam jet injectors utilized in safety injection is defined in terms of the pounds of water entrained by each pound of steam flow (Ref. 3). In Fig. 14 the economy shown ranges from about one and one fourth at 1000 psia back pressure to about seven at 10 psia back pressure; this economy is based on 1000 psia steam and 50 F intake water. Suppliers of injectors anticipate better economy than is shown in Fig. 14. Performance tests on injectors are required to establish the actual economy at the higher steam pressures and high back pressures.

For the four-loop PWR in the postulated double-ended, guillotine-type pump suction break, the safety injection mass flow rate is traced for an 80 second time period after the LOCA (Fig. 15). The back pressure curve plotted in respect to "time after LOCA" shows RCS depressurization. The safety injection mass flow rate corresponds to the back pressure at any point in time.

The ECCS design is based on a core reflood rate equivalent to six inches per second at 100 psia RCS back pressure. As may be noted in Fig. 15, the mass injection rate almost doubles as the back pressure decreases from 100 psia to 14.7 psia.

A rapid reduction in the RCS back pressure as shown in Fig. 15 is a point of interest. This stems from a refill system that overwhelms the LOCA. The heat sink capacity of the injection fluid as displayed in Fig. 16 provides the basis for the statement that the refill system overwhelms the LOCA; this results in the rapid reduction of the RCS system pressure. The heat sink capacity shown does not take credit for the injection fluid lost (spillage) through the pipe break.

During RCS depressurization the safety injection fluid entering the RCS through the intact legs is at the liquid saturation temperature corresponding to the back pressure. This injected fluid is subject to rapid heating (boiling) on contact with the coolant remaining, and by the stored energy in the RCS system components (i.e., the reactor vessel internals and the core elements) and by the

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steam generator secondary fluid remaining. Blowdown of the injected fluid requires increasing its internal energy by the amount of latent heat required for its evaporation.

The safety injected mass provides significant heat sink capacity (Fig. 16). As shown, the mass injected within 30 seconds after the LOCA has a capacity of 54 million Btu, which is equivalent to the stored energy in the core elements and the reactor ve. el interals. The capacity increases to 246 million Btu at 65 seconds and to 328 million Btu at 80 seconds after the LOCA.

Another way to depict the refill system's capability in overwhelming the LOCA is by looking at the time required to refill given volumes (Fig. 17). The mass injected into the reactor system during 30 seconds after the LOCA - the blowdown time for the most part - refills the reactor vessel to the bottom of the core. The entire core is flooded within 43 seconds and an overflow through the postulated pipe break starts within 50 seconds after the LOCA.

In consideration of the safety injection fluid energy capacity (Fig. 16), and the refill capability (Fig. 17), emergency cool. g of the core fuel is effective. The high turbulence resulting from reactor coolant blowdown enhances energy transfer from the fuel; this continues with the rapid injection of emergency cooling water. With highl, borated water starting to reflood the core within 30 seconds after the LOCA the fuel is rapidly quenched preventing an excessive temperature rise.

The steam generators contain an adequate amount of stored energy (steam) for safety injection and continued post-accident decay heat removal (Fig. 18). With the core reflooded 43 seconds into the accident, the secondary system pressure is at 523 psia. At 50 seconds with an overflow out the pipe break the steam pressure is at 486 psia. As shown, at 80 seconds after the LOCA, the secondary pressure is still above 300 psia.

The refill tanks have an adequate supply of borated water for emergency core cooling and core reflood (Fig. 19). During the 80 second time period shown, the stored volume of water has reduced by only 30 percent, from 24,000 to 16,667 cubic feet.

In summary, the refill system in PCS-2 provides emergency core cooling water passively injected into the reactor system to flood the core in a timely and effective way. Rapid quenching prevents fuel temperatures from rising much above the temperatures existing during reactor operation.

As stated (Ref. 1), the NRC "concern about ECC effectiveness has been related mainly to the difficult and complex calculations needed for analyzing the performance of ECC systems in large pipe-break accidents in pressurized water reactors." The response of the ECC systems presented herein were analyzed with straightforward calculations performed by "hand". A computer program was not utilized.

PROJECT C - ALTERNATE DECAY HEAT REMOVAL CONCEPTS

PCS-2 encompasses alternate decay heat removal systems that are bunkered (Figs. 6, 8, and 9). These systems transfer decay heat for an extended time period after the nuclear chain reaction has been stopped. These systems are effective in a postulated pipe break accident condition in the RCS or in the secondary system; also, whenever the normal feedwater sources are unavailable as on the loss of electric power or some other malfunction.

LOSS OF COOLANT ACCIDENT

As described under Project B in the postulated LOCA, the reactor vessel refill system provides emergency core cooling (Fig. 9). After core reflood, the refill system continues post-accident heat removal for a number of minutes.

As soon as the secondary system pressure is expended by the refill system, residual heat removal automatically continues with gravity flow of borated water from the deluge tanks (Fig. 10).

Each deluge tank is interconnected to a high pressure safety injection pipe leading to a "cold leg" in the RCS. Each pipe leading from the bottom of a deluge tank branches into a safety injection pipe at a point between the refill tank and the first valve. Each pipe from a deluge tank has an isolation valve and two check valves in series.

Residual heat removal initiates as the refill system injection pressure decreases below the static head at the deluge tanks. This deluge tank water, heated from 50 F to 130 F by steam carryover from the containment, has over 50 feet static head. In that the containment-free volume and the deluge tank freeboard are at about the same pressure, the driving force continuing emergency cooling for the core is over 20 psi. The stored mass in the deluge tanks continues passive decay heat removal for about four hours into the accident. During this time, the containment is flooded with borated water to an elevation above any postulated pipe break in the RCS.

Heat exchange units (not shown) located at the bottom end of the deluge tank cells can be interconnected to heat exchange units in an outdoor ultimate heat sink at a higher elevation. This could provide residual heat transfer for the balance of the accident period. The four hour deluge water flow period provides time for this passive heat exchange system to "take over".

A vented containment is provided for the post-accident time period (Fig. 8). Pipe vents leading from the containment-free volume into the chilled fluid within the quench tanks prevent the vapor pressure in the free volume from rising to atmospheric pressure.

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The research project on alternate decay heat concepts recommended by the NRC (Ref. 1) places emphasis on passive systems for high reliability. The PCS-2 post-accident decay heat removal system is passive and bunkered thus providing high reliability.

LOSS OF NORMAL FEEDWATER

A second alternate decay heat removal system is presented (Fig. 20). This system is effective in core decay heat removal for an extended time period whenever the normal feedwater sources are unavailable.

This alternate system also enables RCS cooldown at 50 F per hour. Emergency feedwater is automatically injected into the secondary system along with steam blowdown to the contained heat sinks. Decay heat is transferred by conduction and natural convection from the core elements to the secondary system for rejection from the RCS.

On a loss of normal feedwater flow, power-operated relief valves on the steam header for each steam generator automatically open. One set of relief valves initiate steam blowdown to both the deluge and quench tanks (Figs. 6 and 8). A second set of relief valves initiate steam flow through steam jet injectors which entrain chilled water from the quench tanks for injection into the feedwater headers. The steam heats the entrained water to the saturation temperature; thus, the initial injection is about 545 F corresponding to the 1000 psia secondary system pressure.

Steam flow to the deluge and quench tanks rejects the energy resulting from decay heat generation, sensible energy flow from (50 F per hour) RCS cooldown, and secondary system temperature and pressure reduction. The latter enables continued thermal conduction and natural convection of energy from the RCS to the secondary system. The initial mass flow of steam into the deluge and quench tanks is in the range of 80 pounds per second rejecting 95,000 Btu/sec. The steam is dissipated in the deluge tanks through small orifices with an encircling shroud promoting circulation of water past the orifices.

Steam flow to the injectors positioned within the quench tanks is used to replenish the mass lost through secondary system steam blowdown; also the added amount required for the change in the specific volume during steam generator cooldown. Steam flow through the injectors entrains the chilled water, and develops a velocity head with sufficient resultant pressure to open the downstream check valves for emergency feedwater injection into the adjacent feedwater headers. Initially, the high pressure steam entrains about 1.24 pounds of water per pound of steam. The starting feedwater flow rate is in the range of 100 pounds per second. As the secondary system pressure decreases, the economy of the injector improves essentially as shown in Fig. 14. In this application the steam pressure and secondary system back pressure decrease simultaneously.

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A beneficial feature of the rejection of the energy to the heat sinks provided within the deluge and quench tanks is that the release of potentially radioactive steam from the secondary system is completely contained. After the first hour of cooldown, steam blowdown into the deluge tanks has increased the water temperature from 50 F to about 99F. This produces a vapor pressure slightly under 2 psia in the containment through evaporation at the deluge tank vents.

As shown in Fig. 21, the heat sink temperature increases to about 177 F in four hours. The pressure in the containment is at 7 psia, over 7 psi below atmospheric.

At this point in time the RCS pressure and temperature has reduced to 400 psia and 350 F. Decay heat removal can be switched to the normally provided residual heat removal system.

As shown in Fig. 21, the heat sink temperature increases from 177 F to about 200 F from the fourth to the sixth hour; the containment pressure increases from 7 psia to about 12 psia. If there are reasons to continue operation of the alternate decay heat removal system it can effectively continue decay heat transfer as long as required by cooling the liquid in the deluge and quench tanks with the heat transfer systems provided for these tanks.

PROJECT D - IMPROVED IN-PLANT ACCIDENT RESPONSE

This research project as recommended by the NRC (Ref. 1) "deals with what the plant operators can and should do in a developing accident situation". The NRC believes that human factors have a major influence on the availability of safety systems when needed under stress conditions. Safety system availability is also influenced by performance tests and maintenance operations along with components left in an unavailable state through an oversight.

The engineered safety systems in PCS-2 inherently respond to the project proposed.

1. A response from the plant operators in a developing accident situation is not required;
2. Operator action under stress conditions during an accident is not required;
3. Test performance and maintenance operations, which may jeopardize safety system availability, are not required for passive systems.
4. Passive systems do not require test operations which may initiate accidents as is the case for active systems.

PROJECT E - ADVANCED SEISMIC DESIGNS

This recommended NRC research project entails a study of various concepts for improved seismic resistance (Ref. 1).

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In PCS-2, embedment of the reactor building to a depth whereby the refueling floor is at grade elevation lowers the center of gravity substantially. This places the major portions of the massive components in the nuclear steam supply system below grade. In addition, the massive reinforced - concrete required in the supporting and shielding structure, the large water tanks for the engineered safety systems, and the refueling structures; along with the large inventory of water therein, below grade, further lowers the center of gravity. The compact arrangement of the reactor building makes undergounding feasible.

This embedment of the reactor building, lowering the center of gravity reduces amplification of the ground motion up through the structure, moderating the structural forces and the subgrade bearing pressures and reducing the amplified respose spectra on the equipment. This deep embedment also provides stability against sliding and overturning and moderates the toe pressure at the soil-foundation interface.

The heavy reinforced - concrete mat foundation for the reactor building is common to all engineered safety features. In this manner all radioactive and all safety class structures, systems and components are commonly based. Interstructure relative displacements are not a concern with the reactor building being the basic seismic Category I structure. Faulting displacements of safety related umbilicals are limited to the underground piping to the ultimate heat sink. These are not required for at least four or more hours into the LOCA.

The reactor building is not subject to the pressure and temperature transient in the LOCA; therefore, the building does not require a structural acceptance test (SAT) at the LOCA conditions. In the LOCA the reactor building provides secondary containment and is not subject to the mass and energy release from RCS blowdown. This postulated accident is accommodated by the primary reactor containment's thick-walled, steel structure.

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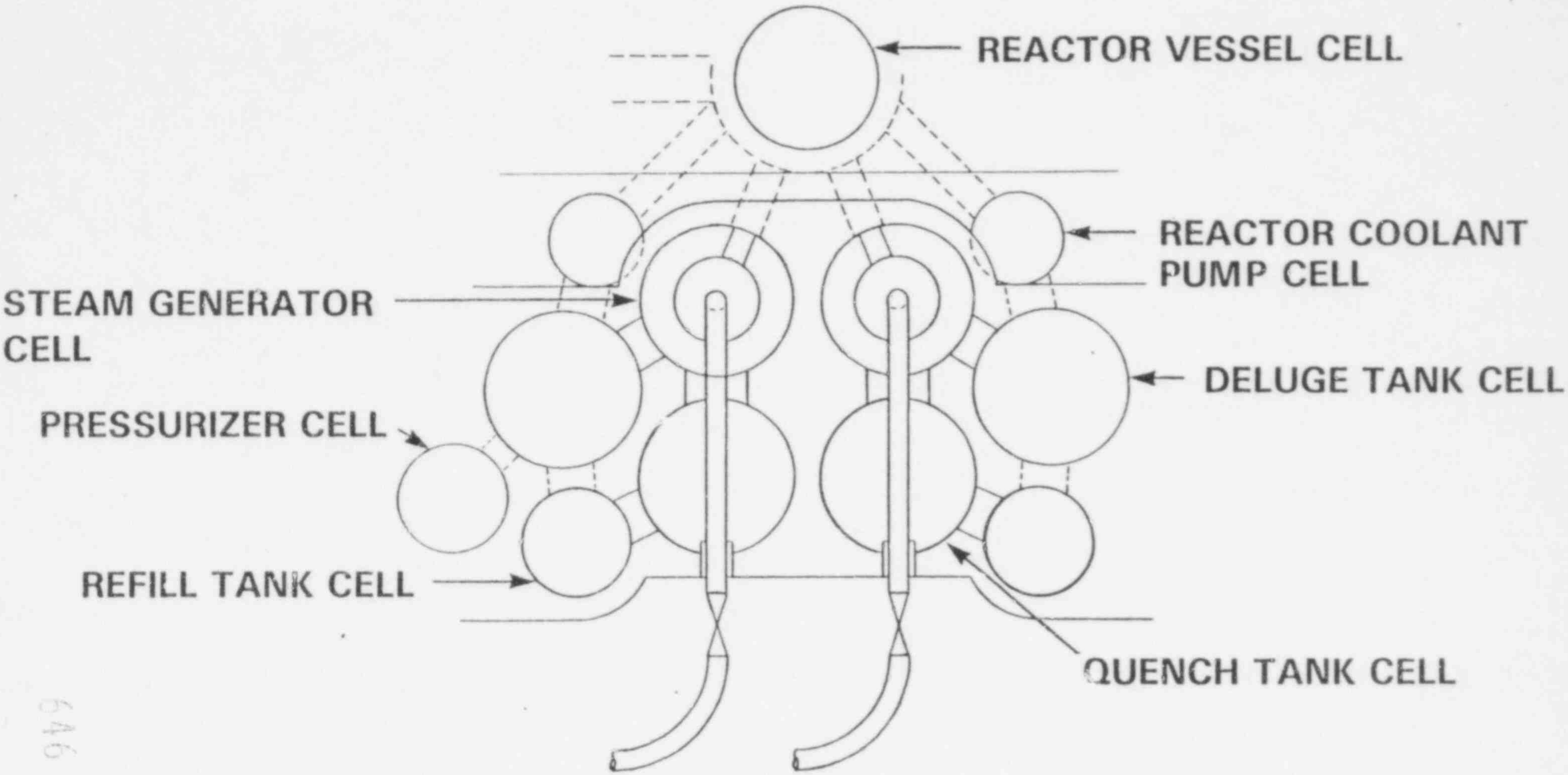

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FIGURE 1

CELL ARRANGEMENT

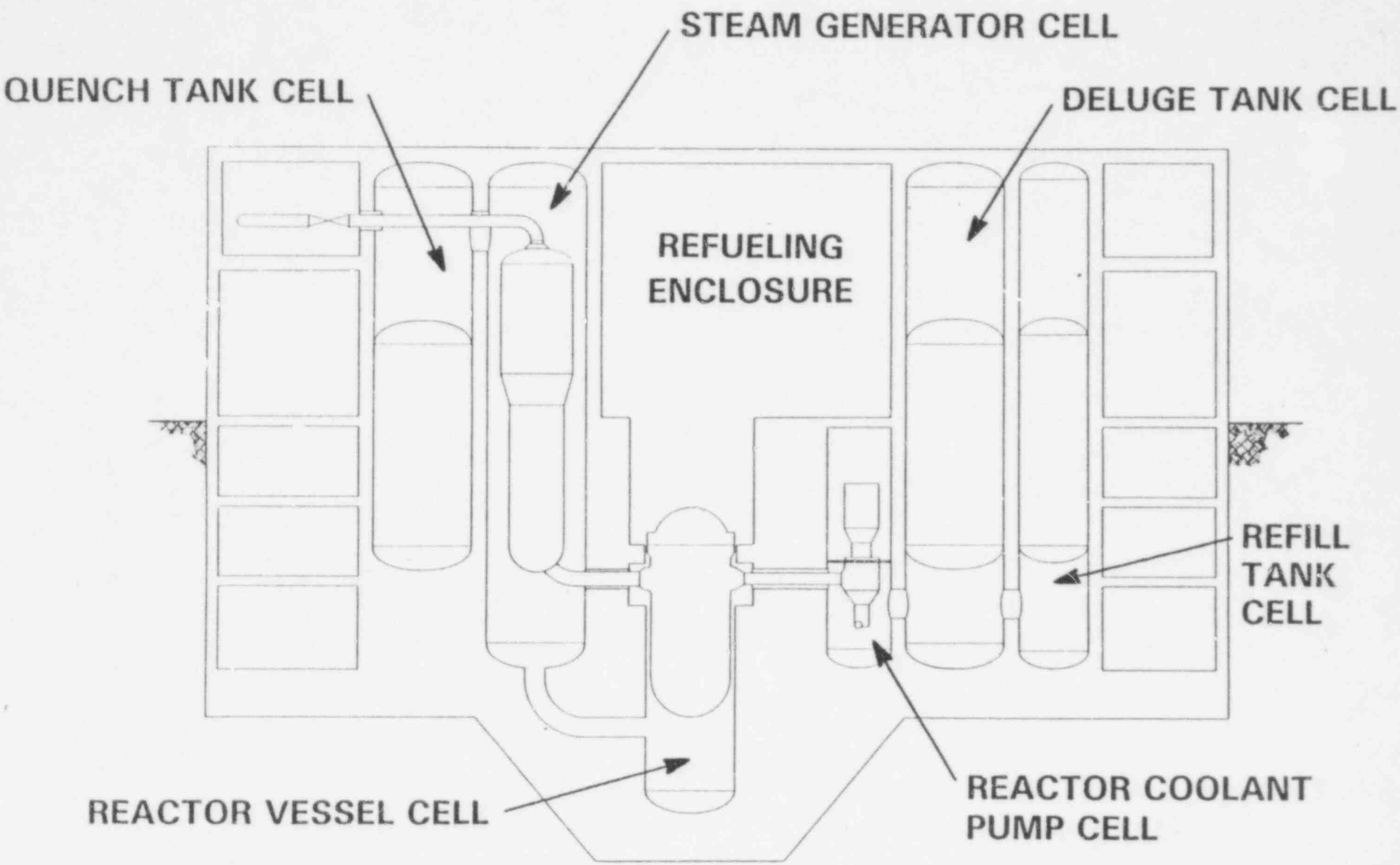

FIGURE 2

CELL ELEVATION

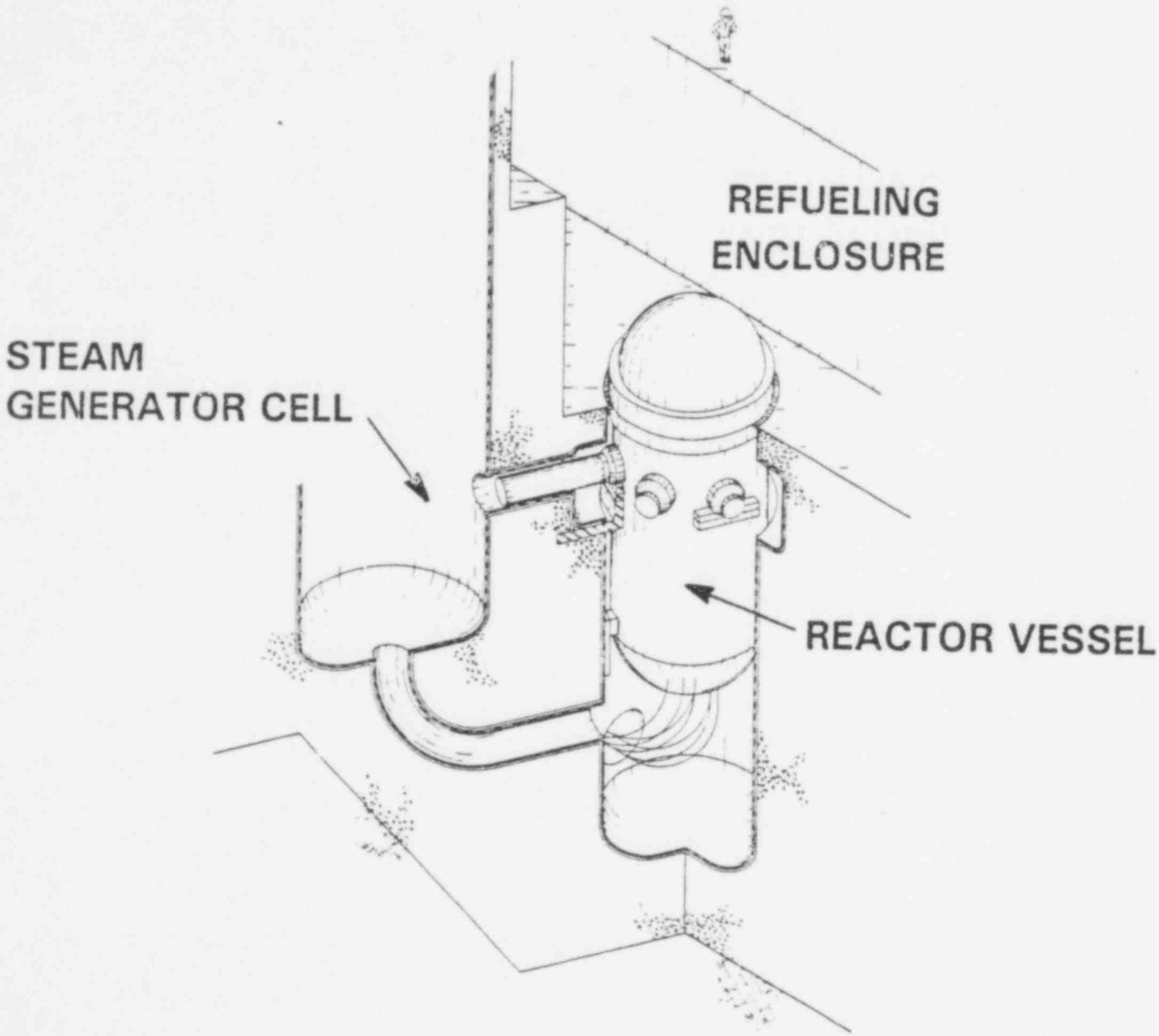

FIGURE 3
REACTOR VESSEL CELL

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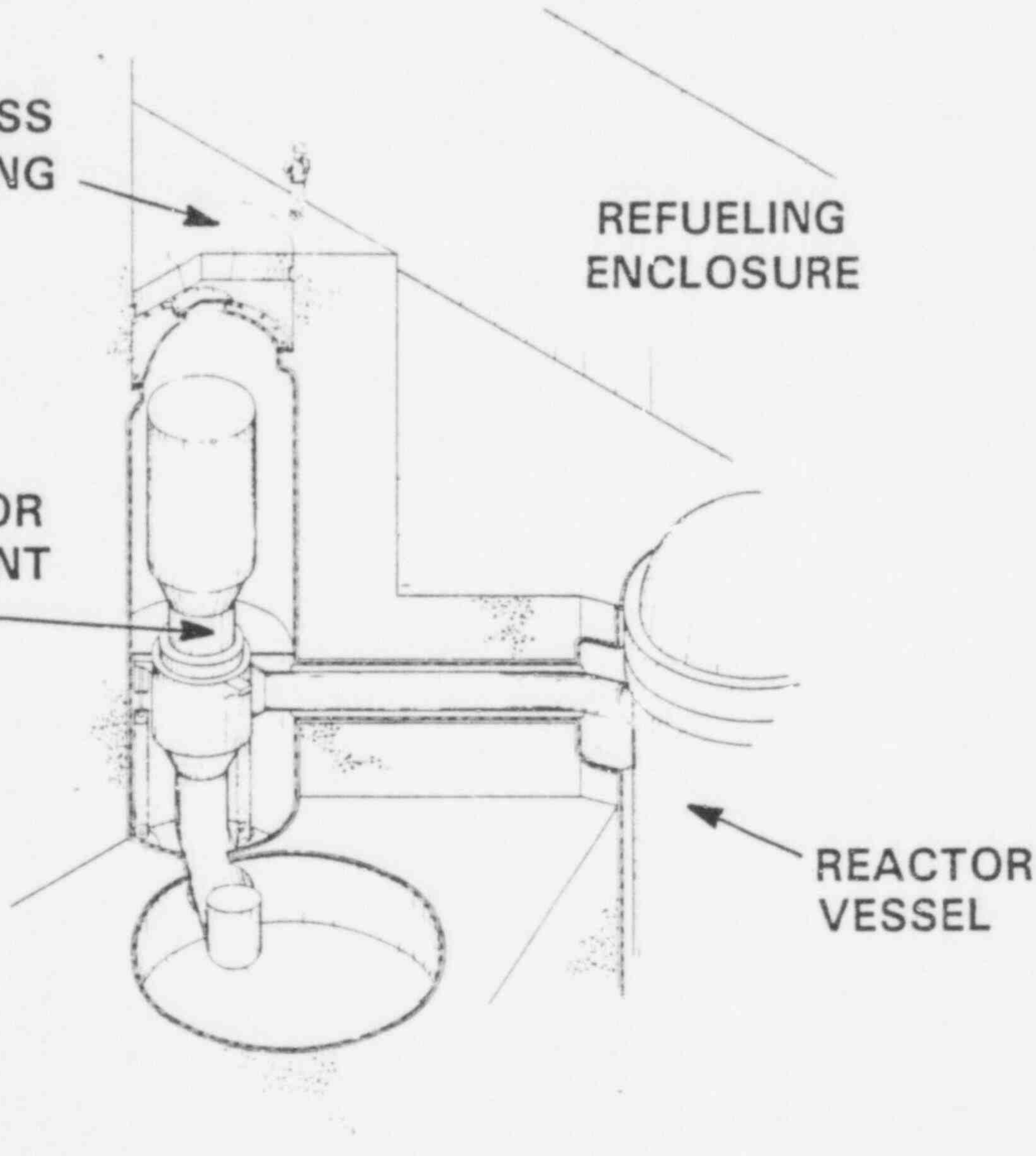

FIGURE 4

REACTOR COOLANT PUMP CELL

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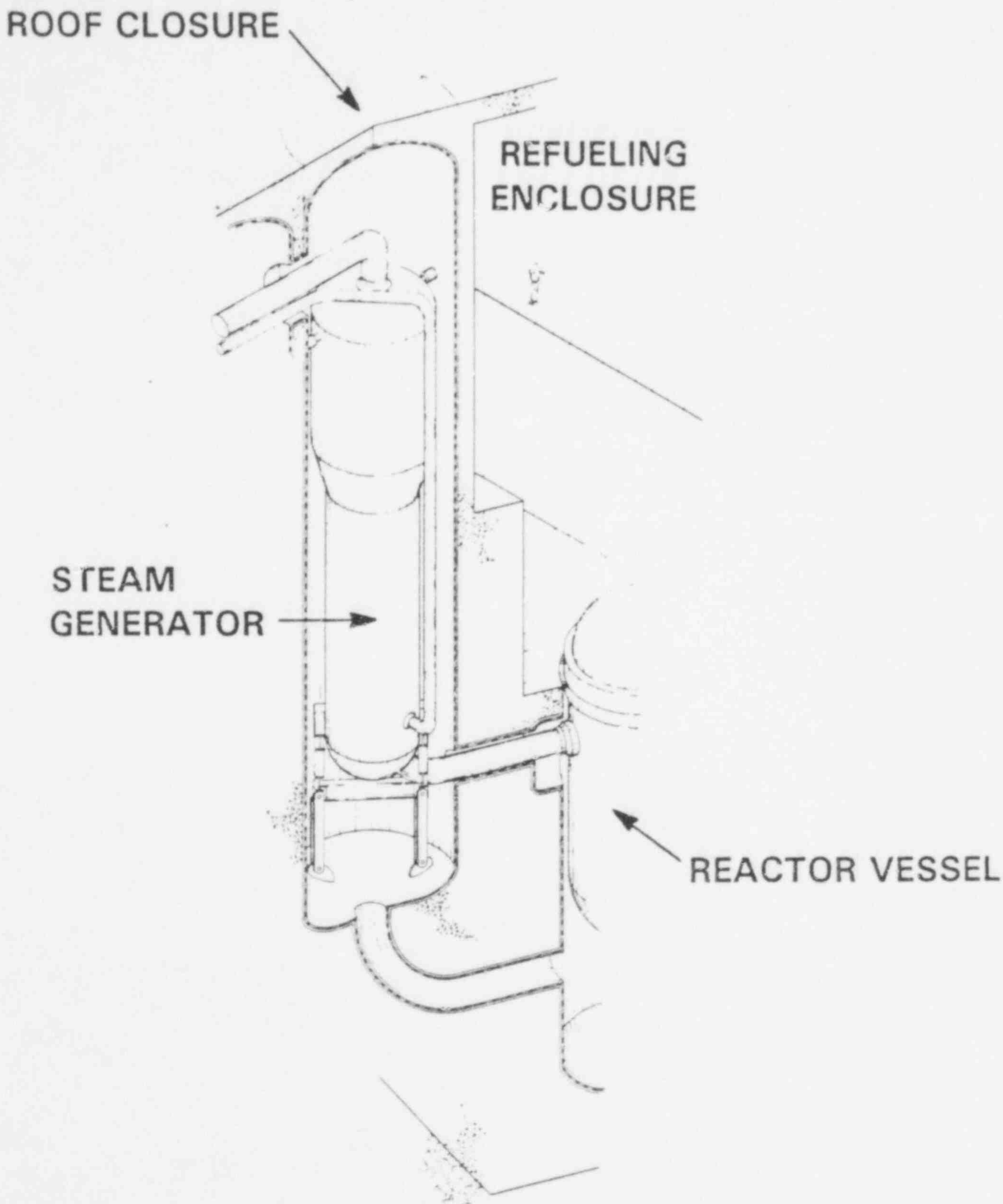

FIGURE 5
STEAM GENERATOR CELL

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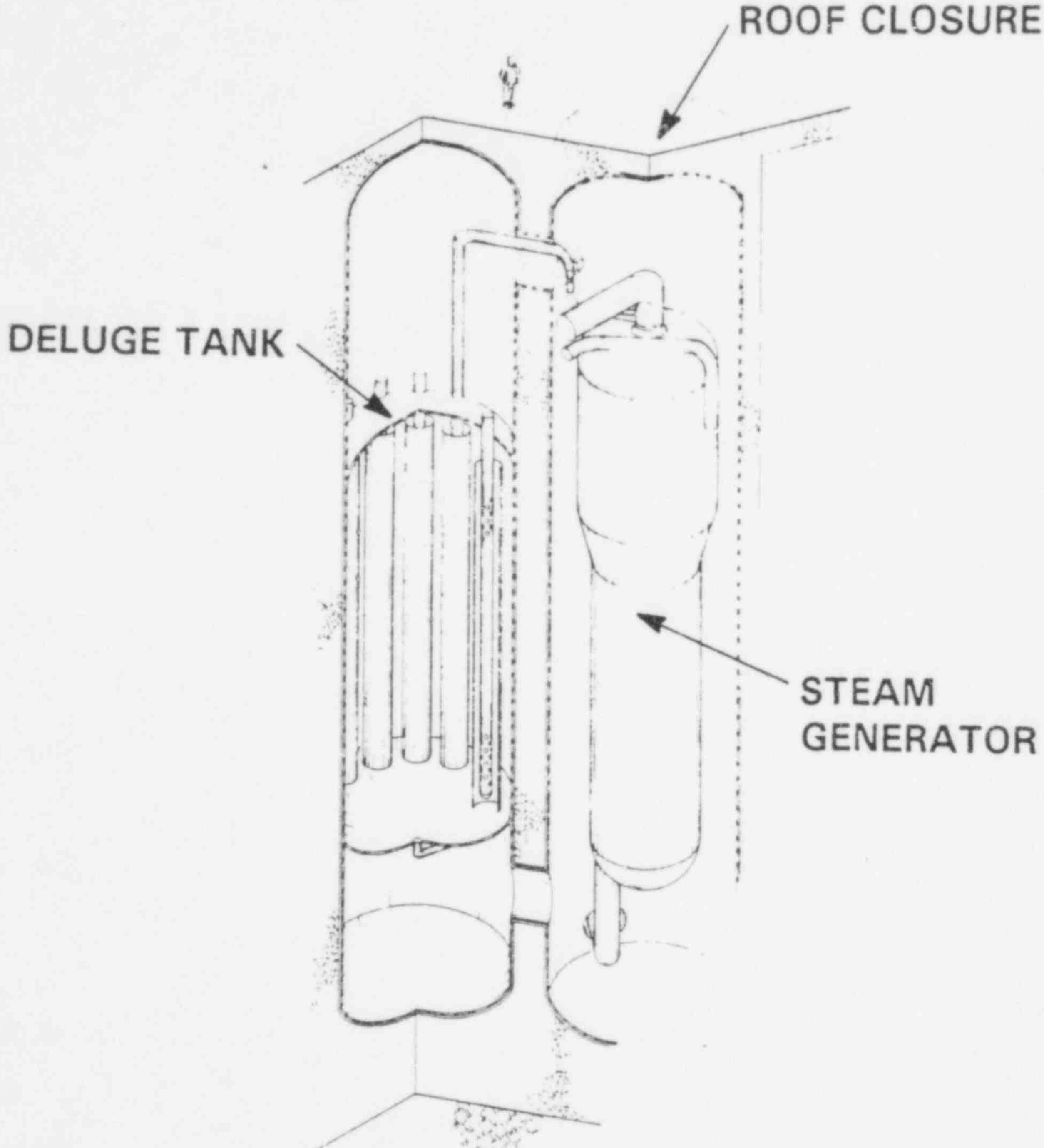

FIGURE 6

DELUGE TANK CELL

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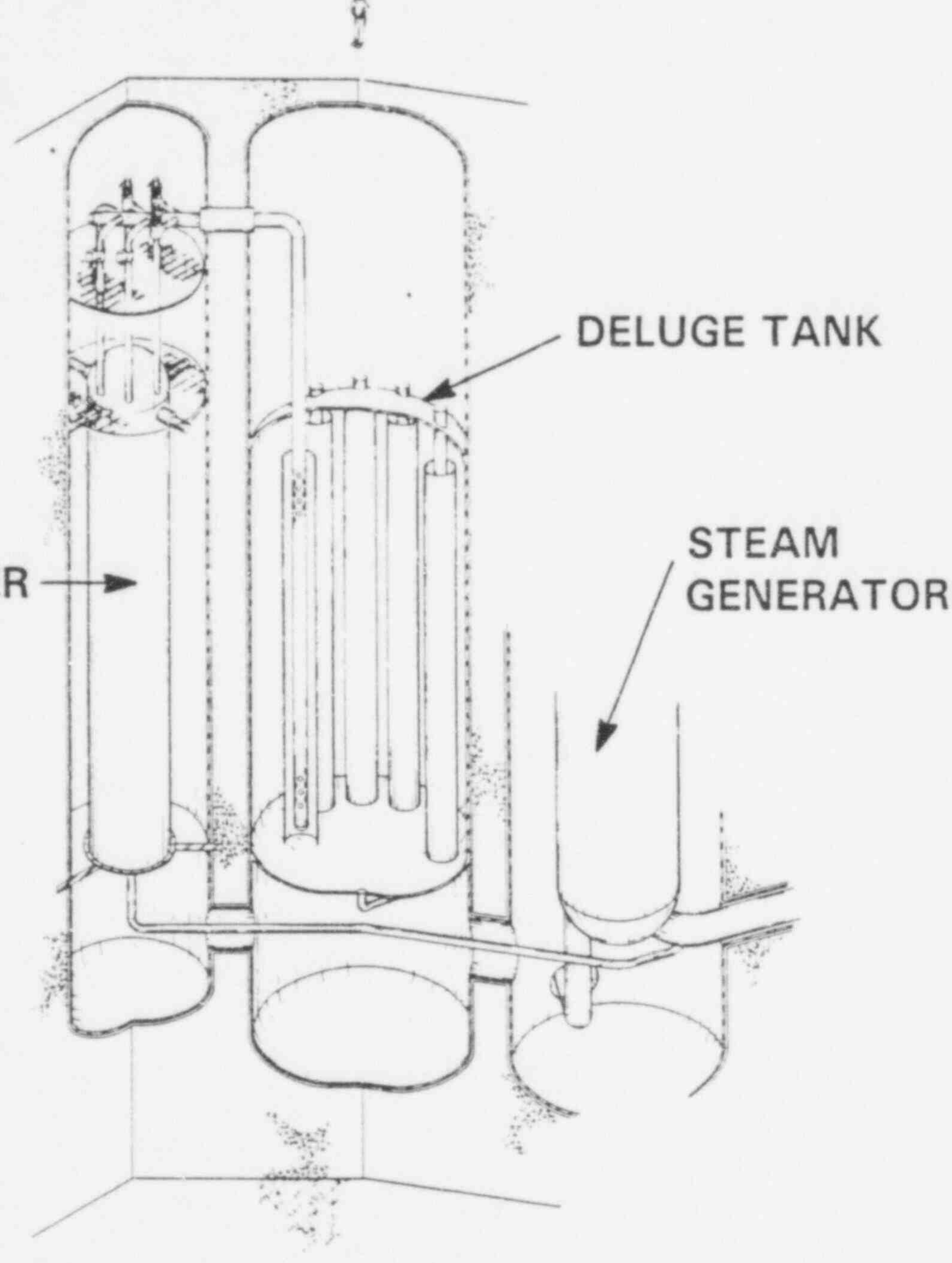

FIGURE 7
PRESSURIZER CELL

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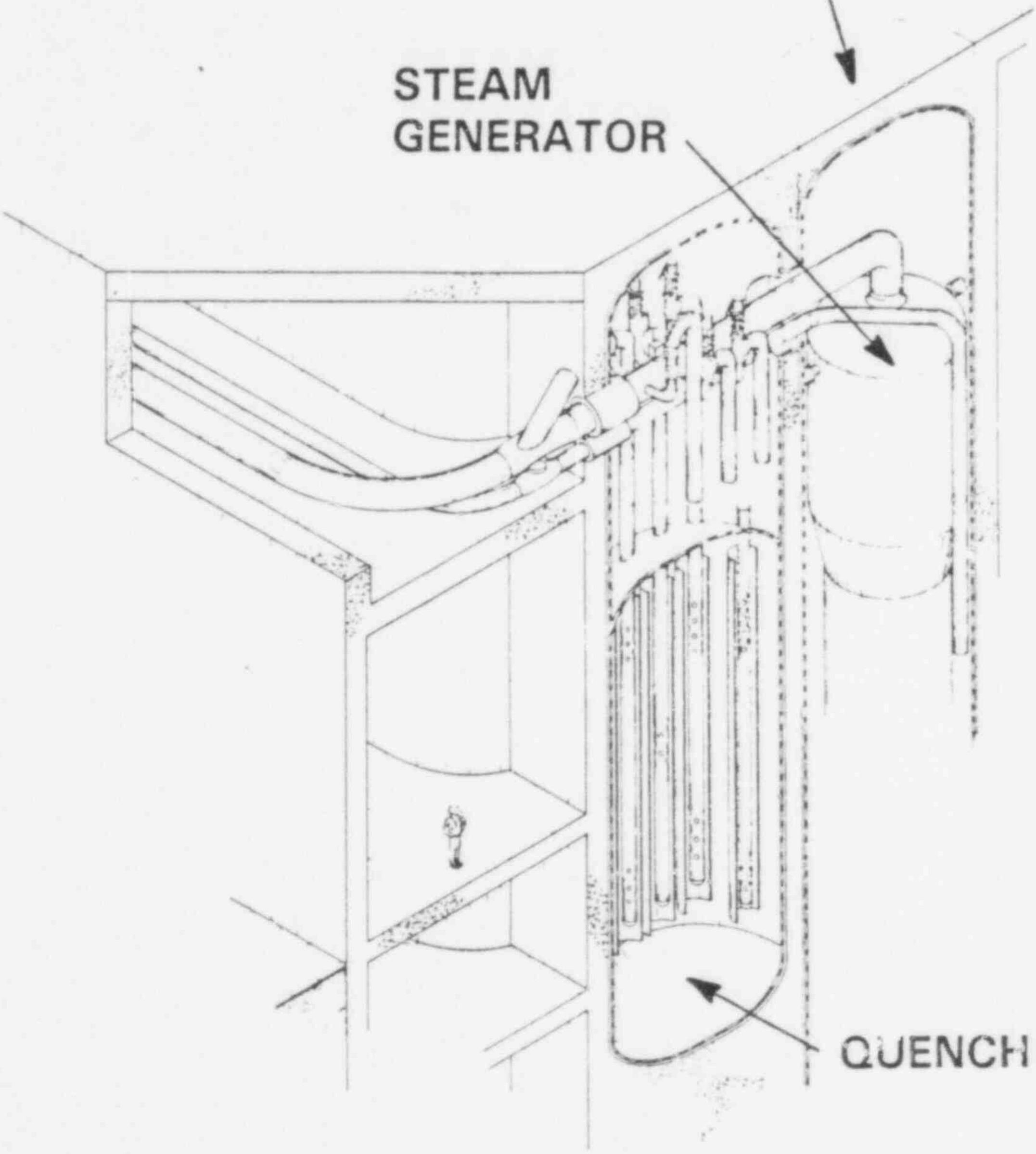

FIGURE 8
QUENCH TANK CELL

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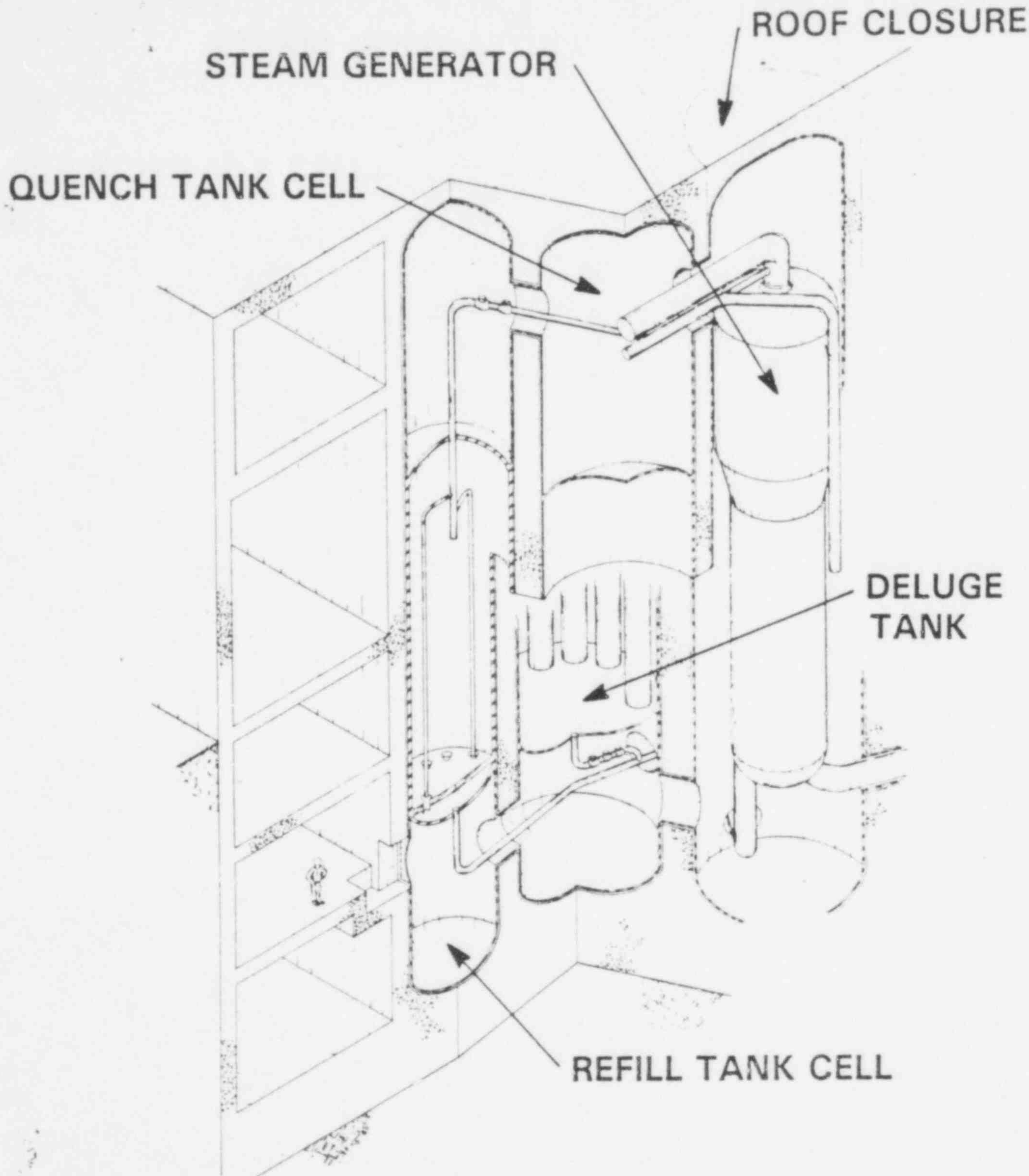


FIGURE 9
REFILL TANK CELL

646 208

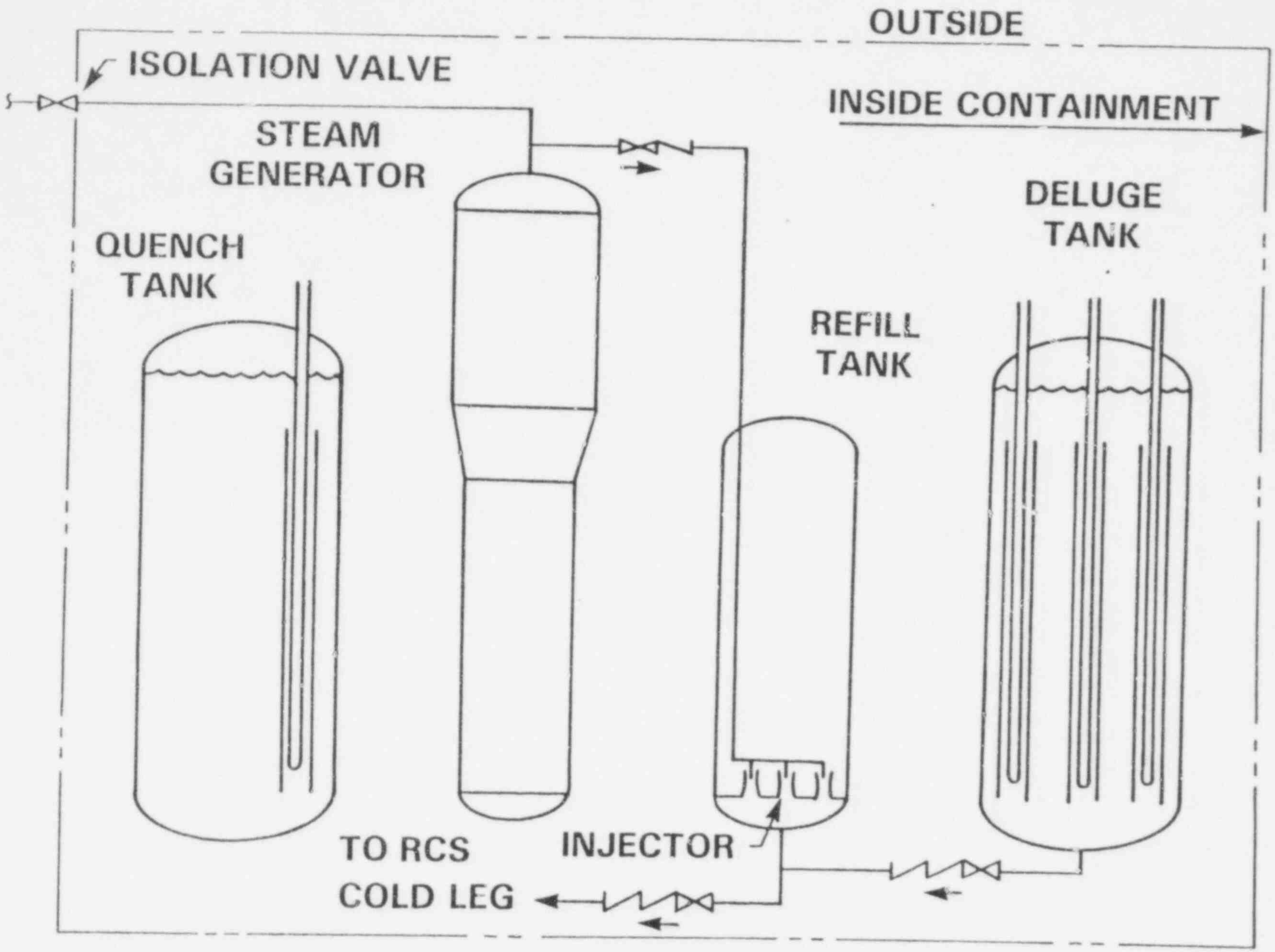
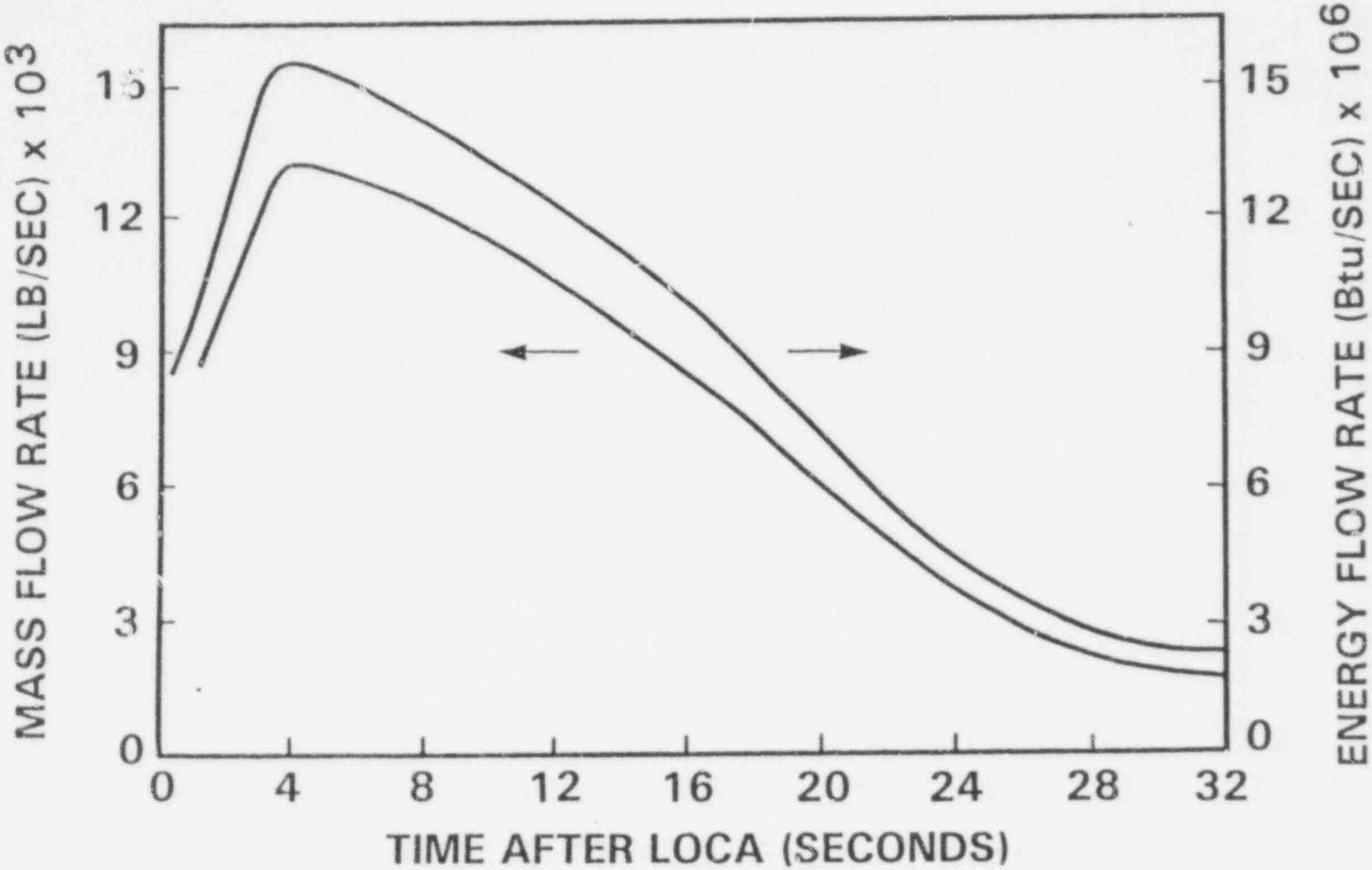


FIGURE 10

ENGINEERED SAFETY SYSTEMS FOR LOCA

646 209



646 210

FIGURE 11

STEAM FLOW INTO DELUGE TANKS

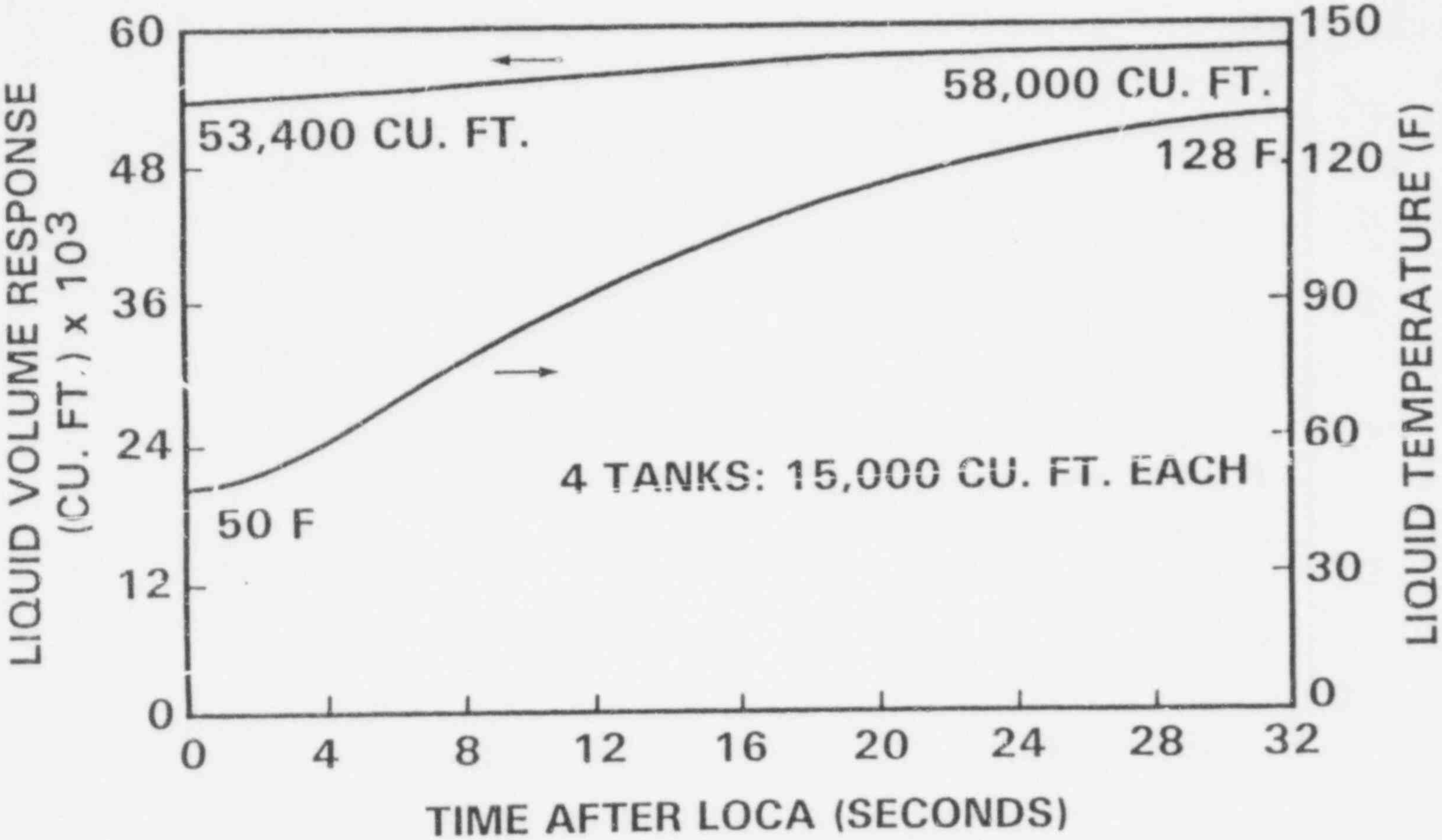


FIGURE 12

STEAM CARRYOVER INTO DELUGE TANKS

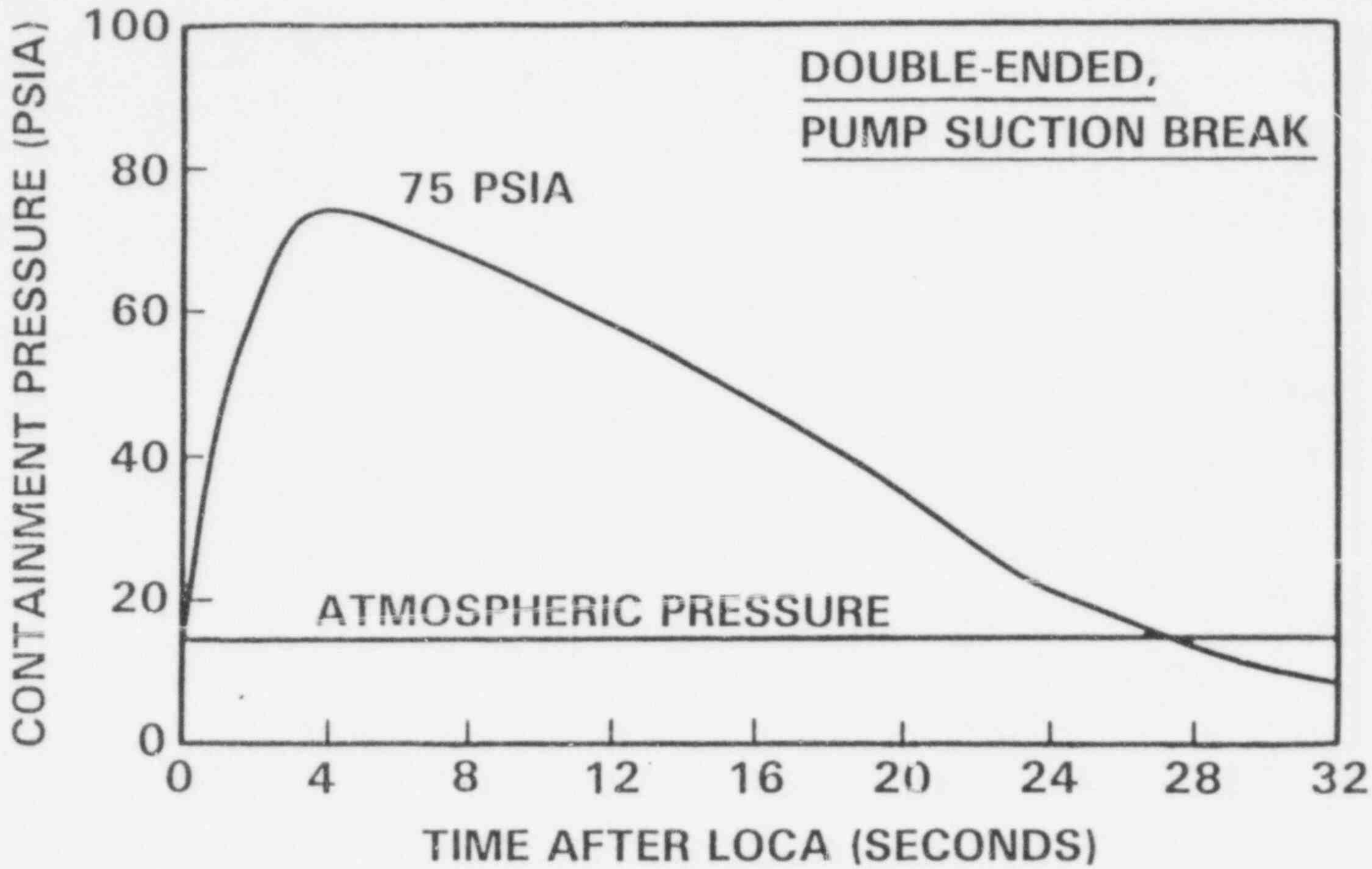
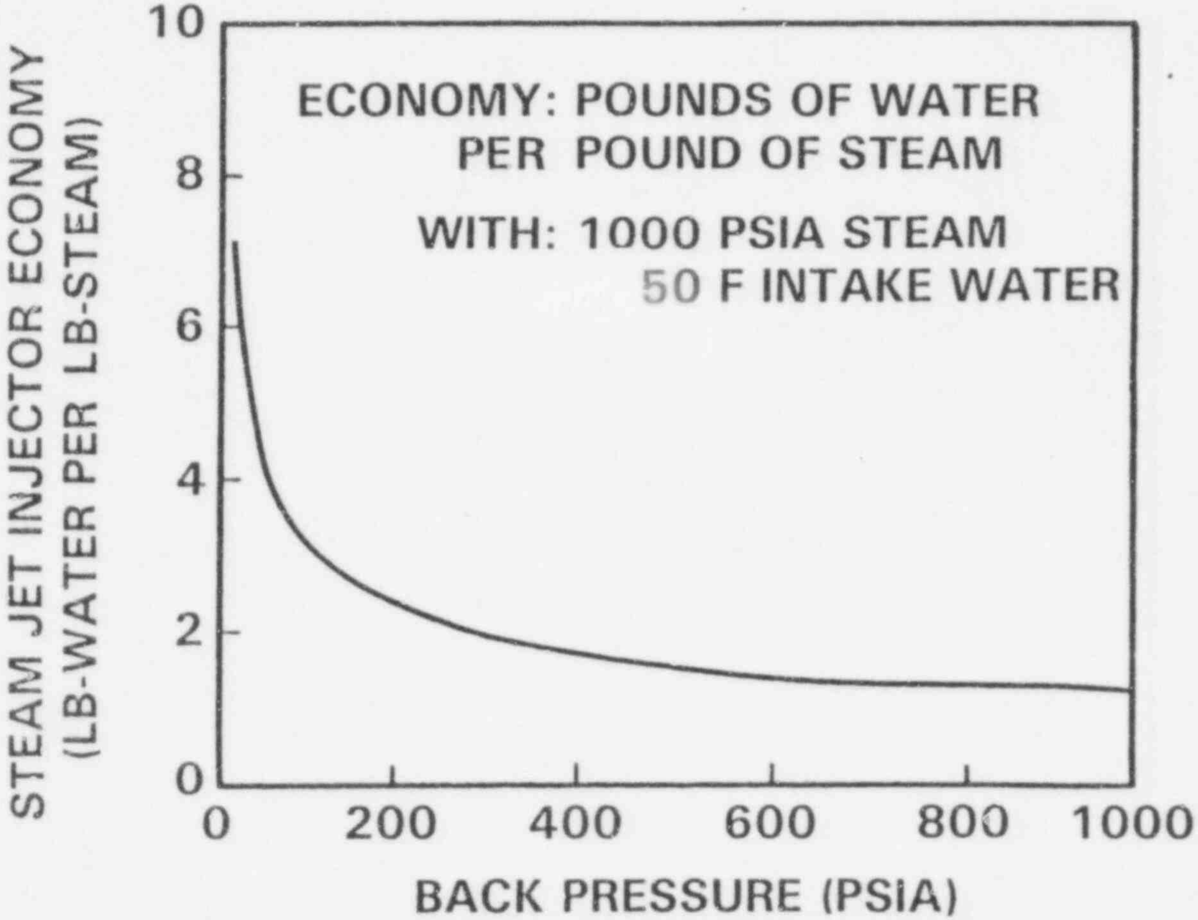


FIGURE 13

CONTAINMENT PRESSURE RESPONSE



646 213

FIGURE 14

STEAM JET INJECTOR ECONOMY

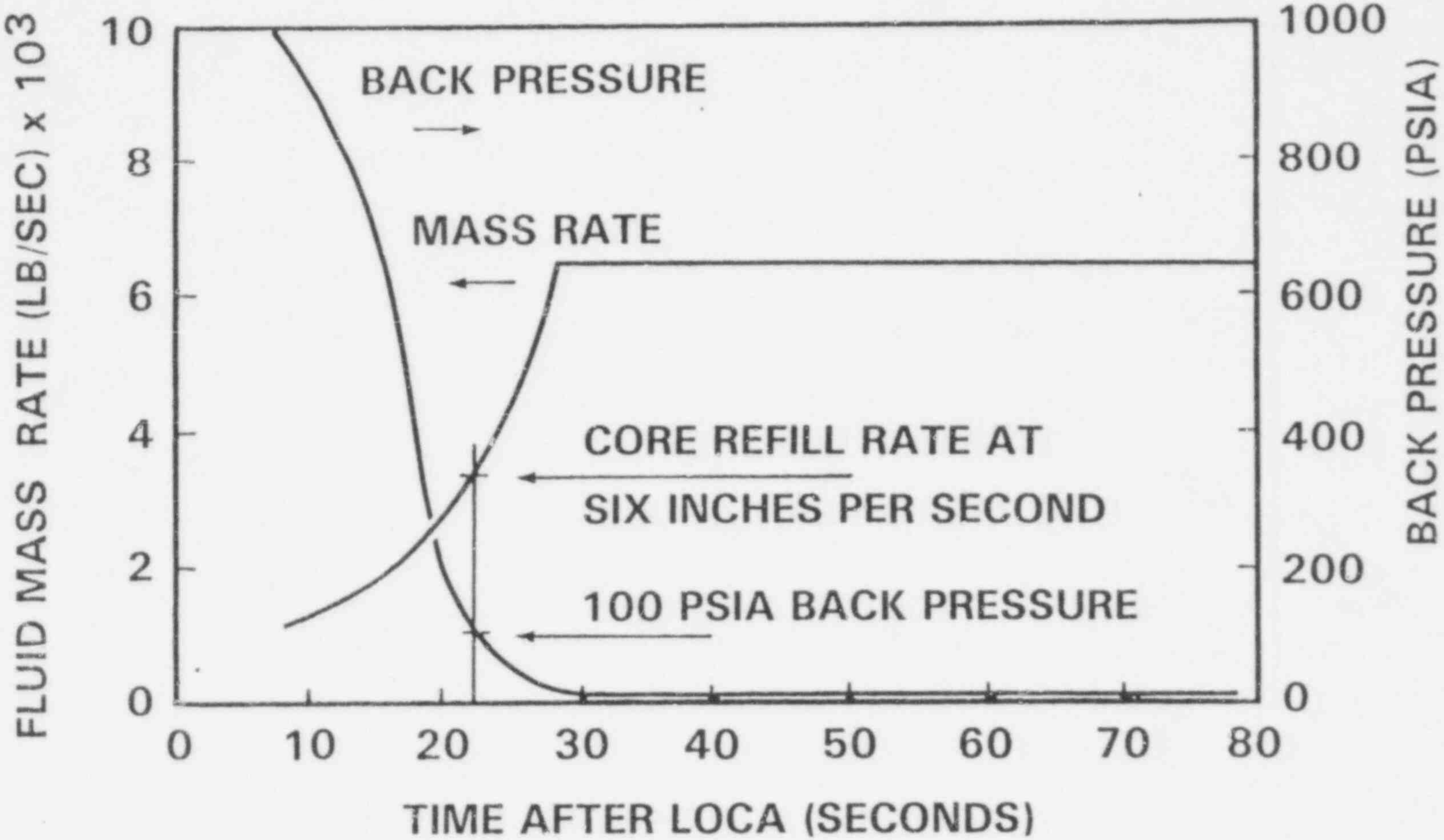


FIGURE 15

SAFETY INJECTION MASS FLOW RATES

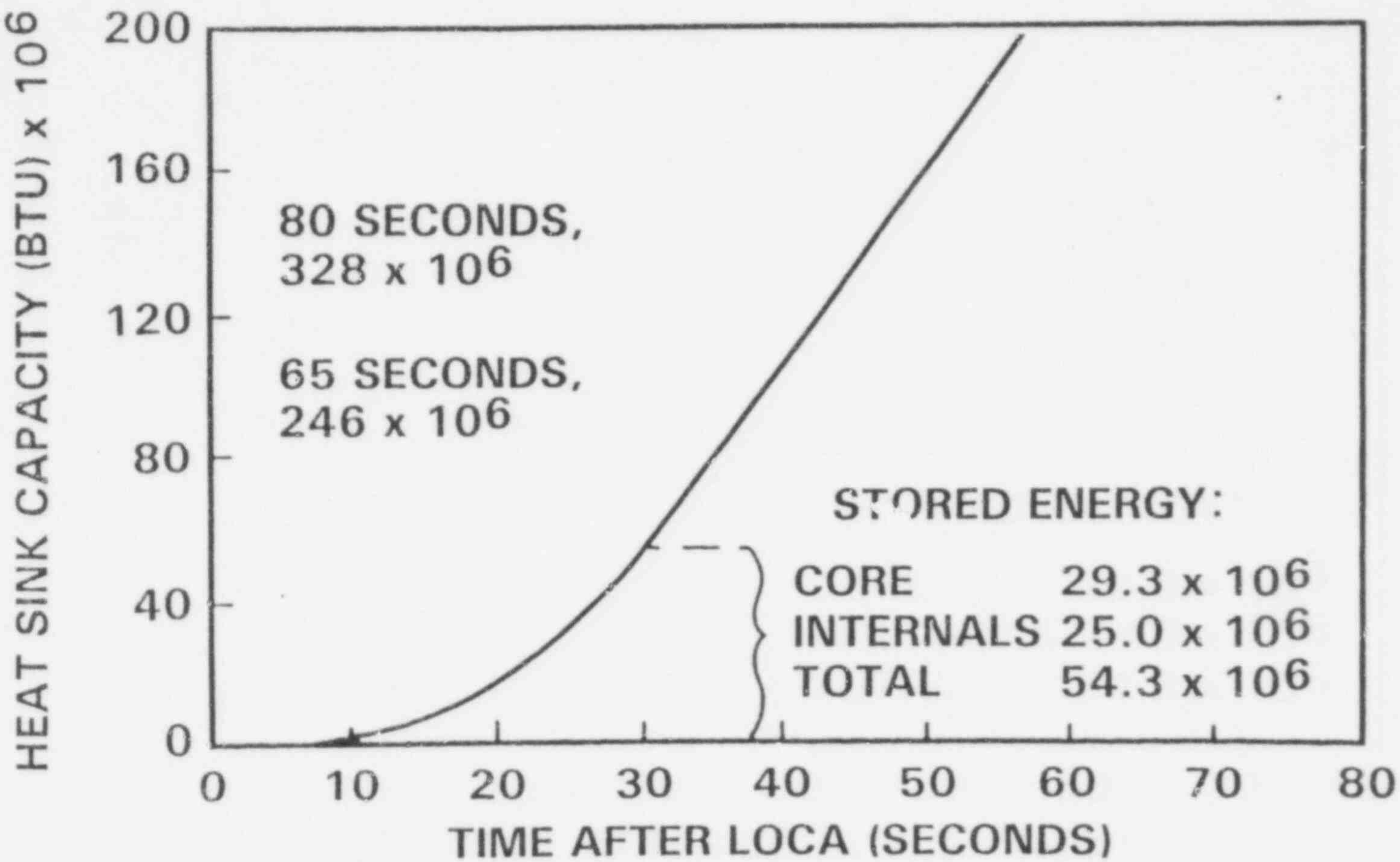
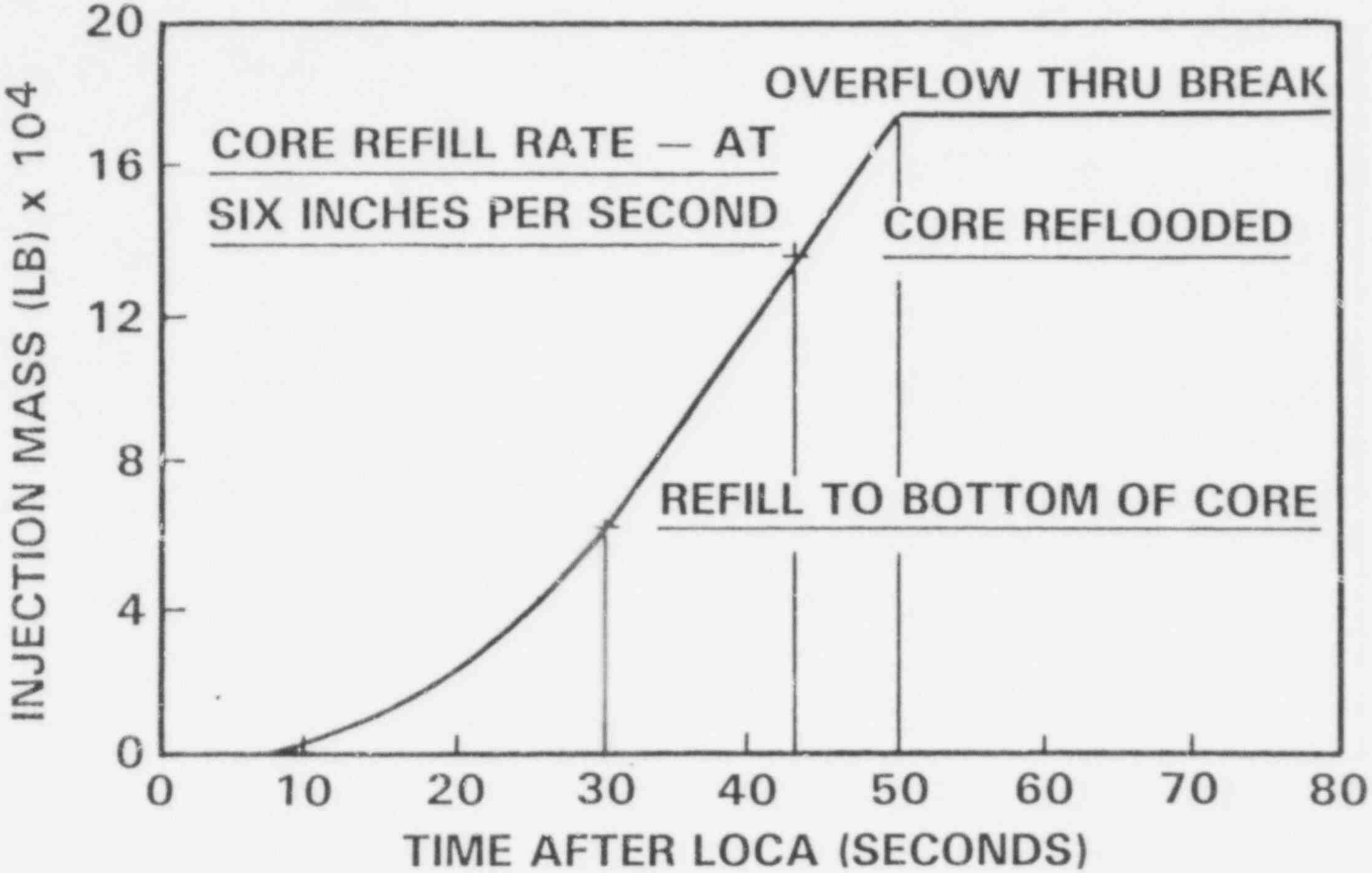


FIGURE 16

SAFETY INJECTION FLUID ENERGY CAPACITY

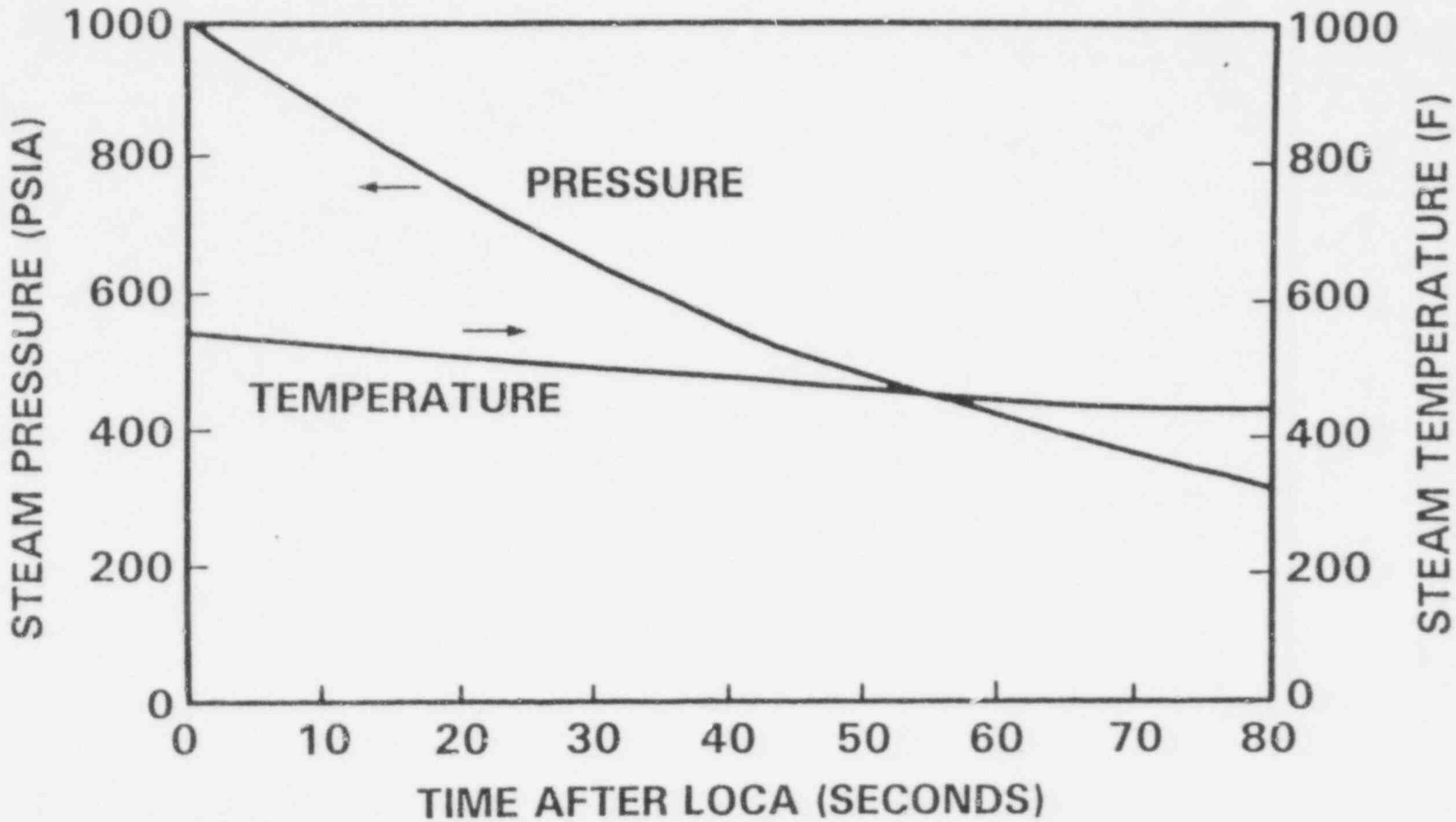
646 215



646 216

FIGURE 17

SAFETY INJECTION REFILL CAPABILITY



646 217

FIGURE 18

SECONDARY SYSTEM PRESSURE RESPONSE

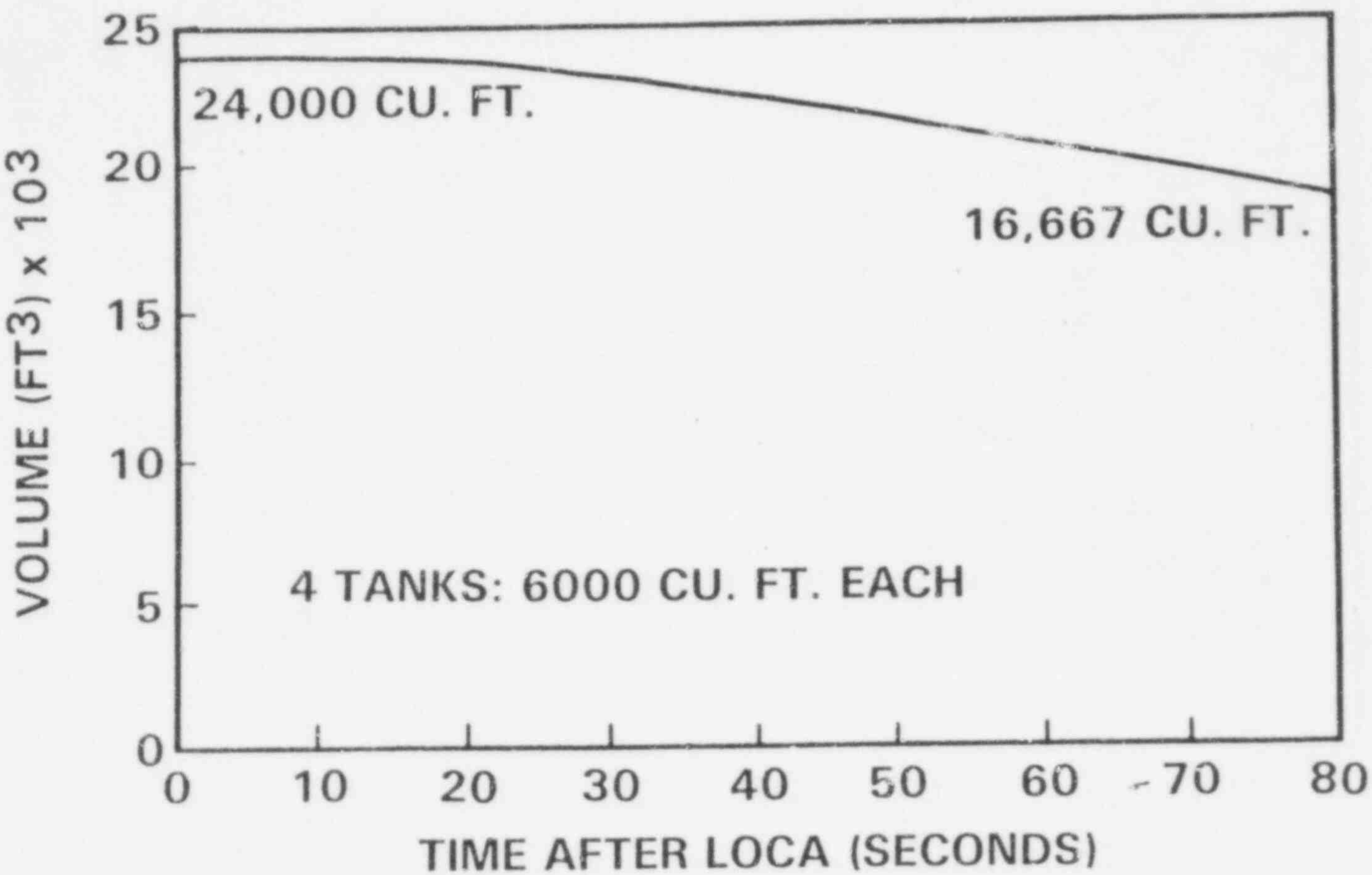


FIGURE 19

REFILL TANK DEPLETION

646 218

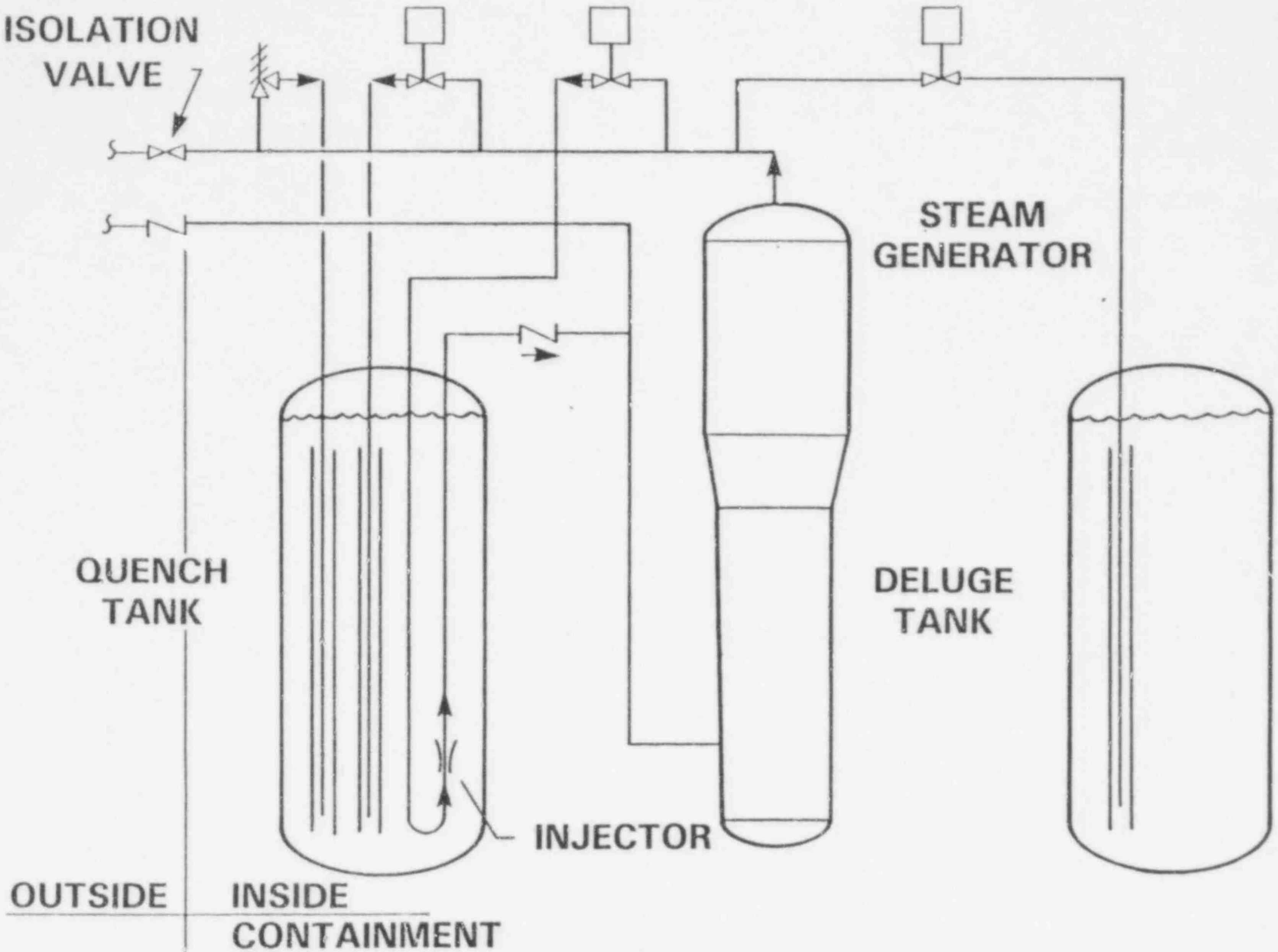


FIGURE 20

ALTERNATE DECAY HEAT REMOVAL SYSTEM

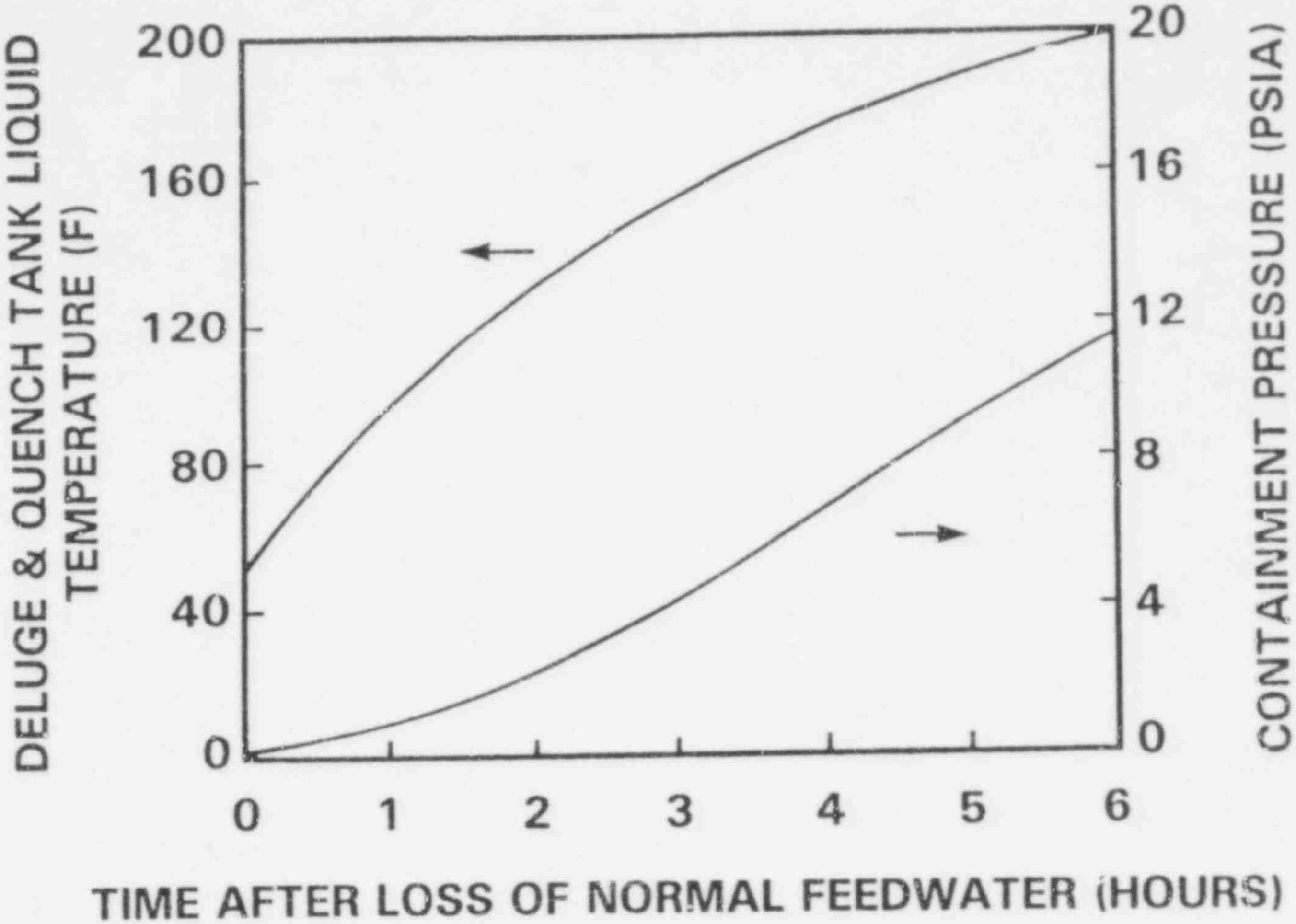


FIGURE 21

RESPONSE FOR ALTERNATE DECAY HEAT REMOVAL SYSTEM

646 220