AN EVALUATION OF THE EBR-II REACTOR-BUILDING
ISOLATION SYSTEM

by

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ABSTRACT

Since the original construction of the EBR-II plant, several changes and additions have been made that have implications relating to the isolation system. During this same period, the EBR-II reactor has been re-directed toward a mission of an irradiation facility, and several new pertinent standards have been adopted by the industry. The entire isolation system was given a complete review to determine its adequacy and reliability in the light of these changes. The results of this review are given in this report along with recommendations for upgrading the system. Increased coverage by radiation monitors, removal of the temperature trip monitors from the trip circuit, and use of two-out-of-three trip logic are recommended to upgrade the system and reduce the probability of spurious trips.

I. INTRODUCTION

The prime objective of the EBR-II containment, and hence of the isolation system, was stated in 1958 to be "to prevent under all circumstances the escape of a significant amount of fission products."¹ This objective remains compatible and in essential agreement with standards^{2,3,4} that have subsequently been adopted by the industry. The present objective, however, has been expanded to include control over the escape of any harmful radioactive contaminants.

The original philosophy of the isolation system emphasized serious incidents in which it was postulated that the reactor could be seriously damaged or destroyed. Under such postulated conditions, further damage to

reactor systems would be of secondary importance. This philosophy was evident in the isolation system installation, because isolation of the refrigeration system, if sustained, could commit the reactor systems to heat damage.

II. CRITERIA

A search was made of existing EBR-II documents to provide the original basis and criteria for the reactor-building containment and isolation systems. On page 50 of the Addendum to the Hazard Summary Report⁵ for EBR-II, the following statement is made relative to the reactor-building isolation system: "Penetrations are provided (in the reactor building) for access of personnel, equipment, freight, air ducts, and piping. To assure gas-tight integrity of the shell, immediately subsequent to a maximum credible nuclear excursion, it is necessary to render each of these penetrations leakproof." On page 51 of the same report, the statement is made that the isolation valves will close "in the remote event of a significant nuclear incident or primary system sodium fire." Appendix H of that report, which describes the reactor containment systems, includes this statement on page 266: "The function of this system (building containment system) is to localize within the reactor building the effects of a possible sodium-air reaction energy release."

Some guidelines for establishing the upper and lower limits for which the reactor-building isolation system should be effective are found in Section 3.2.2, page 3-2, of RDT Standard C 16-1T² on protection of the public. Two pertinent paragraphs are quoted as follows:

"The PPS (plant protective system), in conjunction with the passive plant structures, shall limit exposures of the general public, including all persons outside the facility area (defined by the Operating Contractor) to less than the guideline values presented in 10CFR100 for all accidents, up to and including the Design Basis Accident.

The PPS, in conjunction with the passive plant structures, shall limit the release of radioactivity resulting from normal plant operations or abnormal conditions, up to and including the severity level of a minor incident, to less than the limits presented in AECM-0524."

The design basis accident is defined in the RDT Standard as "An extremely low probability hypothetical accident involving the primary barrier, which is hypothesized for the purpose of demonstrating the adequacy of the facility design in providing protection for purposes of public safety." A minor incident is defined as "An occurrence in which damage to replaceable components is no greater than some specified loss of effective lifetime."

Neither of the EBR-II documents previously quoted (Refs. 1 and 5) define with any precision the limits for which the isolation system should be effective. Logically, the isolation system should be effective up to and including the event that just exceeds the containment capability of the reactor containment building. This approach appears to be in agreement with the first paragraph quoted from RDT Standard C 16-1T. The original documentation on the EBR-II containment system was composed before common usage of the term "design basis accident" currently used in the RDT Standard. It is clear, however, from H. O. Monson's Geneva paper¹ that the EBR-II containment building "design is based upon a maximum internal pressure of 24 psig."

The early EBR-II documentation did not consider or define the lowest-order event for which the reactor-building isolation system should respond. However, the second paragraph quoted from the RDT Standard provides guidance toward establishing this lower limit.

The RDT Standard makes reference to two other documents for radiation guidelines. These are the Code of Federal Regulations 10 CFR 100³ and AEC Manual 0524⁴. The Code of Federal Regulations 10 CFR 100 specifies that "An individual located at any point on its (exclusion area's) boundary for two hours immediately following onset of the postulated fission-product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure." The AEC Manual 0524 sets tighter limits. A pertinent table and paragraph from that source follows:

<u>Type of Exposure</u>	<u>Condition</u>	<u>Dose (rem) or Dose Commitment</u>
Whole body, head and trunk, active blood-forming organs, gonads, or lens of eye	Accumulated dose Calendar Quarter	5(N-18)* 3
Skin, thyroid, and bone	Year Calendar Quarter	30 10
Hands and forearms, feet and ankles	Year Calendar Quarter	75 25
Other organs	Year Calendar Quarter	15 5

To meet the above dose commitment standards, operations must be conducted in a manner that it will be unlikely that an individual will assimilate in a critical organ, due to exposure received in any calendar quarter (year) by inhalation, ingestion, or absorption, a quantity of a radionuclide that would commit the individual to a lifetime dose greater than is specified for a calendar quarter (year). (See table, above).

*N equals the age in years at next birthday.

III. DESCRIPTION OF ISOLATION SYSTEM

The existing isolation system is essentially as it was described in the Addendum to the Hazard Summary Report⁵, pages 50 and 263-266, except that a xenon-tag sample line installed under plant modification (PM) No. 232 (see Sec. IV) has been added to the list of lines that isolate. There are also plans for adding an argon purge line to be installed under PM 302 (see Sec. IV). These two pipelines are part of a new system for identifying offending subassemblies in event of fuel-cladding penetration. Another building pipeline added subsequent to building construction is the building vacuum-relief line. This line, however, is normally closed and is not interconnected with the building isolation system.

With the addition of the argon purge line, 20 active pipes interconnected with the isolation system will be penetrating the reactor building. These are listed in Table I.

Four sources of signals automatically initiate isolation trip of the 20 lines with the isolation valves: (1) subassembly coolant outlet temperature high as indicated by two-of-four outlet-temperature thermocouples

TABLE I. Isolating and Nonisolating Pipes Penetrating the EBR-II
Reactor Building

<u>System</u>	<u>Number of Pipes</u>
<u>Pipes with Isolation Valves</u>	
Ventilation (including purge exhaust)	3
Shield-cooling refrigeration (one unit)	2
Main-floor refrigeration (two units)	4
Sub-basement and standby refrigeration (two units)	4
Instrument air	1
Plant air	1
Argon supply	1
Xenon tag	1
Argon purge	1
Silicone cooling	<u>2</u>
	20
<u>Pipes without Isolation Valves</u>	
NaK shutdown coolers (two units)	4
Secondary-sodium lines	2
Secondary vent line	<u>1</u>
	7

during reactor operation, (2) building radiation above 450 mR/hr as measured by either of two monitors above the main operating floor of the reactor building, (3) building temperature above 145°F as measured by any one of two resistance thermometers and one thermocouple above the reactor operating floor, and (4) building pressure greater than 8 ± 2 in. wg differential. All the isolation valves are connected to fail closed upon loss of power or air.

Seven pipes that do not isolate enter the reactor building. These are also listed in Table I.

A. Ventilation System

Air is supplied to the reactor building at the rate of approximately 6600 cfm by a fan directly above the main personnel airlock. The air is ducted into the building through air-conditioning and prefiltering equipment outside the building. The cooling systems for the reactor shields and instrument thimbles use air from the reactor building.

A nominal 5000 cfm of air is exhausted from the shield-cooling system, passed through absolute filters, and discharged through ductwork in which it joins the discharge from the thimble-cooling system. The thimble-cooling system discharges an additional 1600 cfm, also through absolute filters. The combined discharge is carried in a common duct through the reactor-building containment shell and underground to the suspect stack located near the FEF, where it is exhausted to the atmosphere. Figure 1 shows the shield-cooling system, and Fig. 2 the thimble-cooling system. A separate exhaust purge fan provided may be used to exhaust air directly from the reactor operating floor to the outside.

Each of the three ventilation penetrations is provided with two 20-in. isolating valves. One, inside the reactor building, is rated for high temperature. The other, outside the building, is provided with a rubber seal for leaktight closure. All these valves were designed and tested to close in not more than 1/2 sec. Energy to close is from a compressed coil spring. Power to open is provided by an air cylinder piloted by two parallel three-way solenoid valves. These systems are more completely described in Volume II, Chapter 7, of the EBR-II System Design Descriptions and in Sections IV D

Fig. 1. Pictorial Flow Diagram of Shield-cooling System

Fig. 2. Simplified Flow Diagram of Thimble-cooling System

and IV E of the EBR-II Operating Instructions.*

If for any reason the building-discharge isolation valves close while the reactor-building supply fan and valves remain open and operative, the reactor building would become pressurized in 2 to 5 min to a maximum pressure of approximately 6 in. wg. Similarly, if the inlet isolation valve closes for any reason while the shield fan and its corresponding isolation valve remain open, the reactor building would be depressurized to a negative pressure of 14.6 in. wg. These two limiting pressures are set by the zero flow head of the respective fans.

The vacuum-relief system of the containment building opens at approximately 7 in. wg negative differential pressure with respect to the outside atmosphere. It recloses when the pressure returns to approximately 2 in. wg negative. This system, however, operates on a 6-in. line, which is too small to limit the negative pressure resulting from operation of the normal building ventilation fans with the inlet isolation valve closed.

To preclude the events just described, the circuitry controlling the isolation systems will be modified by PM 299 (see Sec. IV) to provide interlocks for the building isolation valves. This modification will also provide indication of reactor-building pressure in the control room and an alarm if the pressure exceeds 3 in. wg. In the event that any isolation valve in the ventilation system leaves its open position, all isolation valves in the ventilation system will close. All reactor systems, however, are designed to operate properly and without damage even if the valves malfunction. The ducts, valves, and filters are designed to withstand pressures greater than those of the shutoff heads of the respective fans. The isolation valves are rated for 35 psig. Ductwork embedded in concrete is fabricated of Type 304 stainless steel and has welded construction. The rectangular ducts are 11 gauge, and the round ducts are Schedule 10S pipe. The ducts not embedded in concrete are made of 0.25-in.-thick aluminum and are tested at 21 in. wg pressure. The AEC (absolute) filters are built to withstand 21 in. wg pressure and 14 in. wg vacuum. These

* Formal title of the document commonly called the "EBR-II Operating Manual" is "Operating Instructions for EBR-II." In this report, the title "EBR-II Operating Instructions" is used when referring to the document.

filters as originally installed were constructed of ordinary plywood and kraft-paper separators. This construction was later changed to use fire-resistant materials.

B. Refrigeration Systems

Twelve refrigeration lines, ranging from 1-1/4 to 3 in. in size, enter the reactor building. Ten of the 12 lines contain Freon refrigerant, and the other two contain silicone coolant. The Freon lines are copper except for a short run of stainless steel piping as they pass through the containment penetration. This short run includes the isolation valves on the outside of the containment building. The silicone lines are seamless carbon steel except for a short run of stainless steel piping as they pass through the containment building. This short run includes the isolation valves. Some of the valves are mounted to the power-plant floor, and the other valves are supported from the containment-building wall (see Table II). The mounting arrangement could be significant in the event of an earthquake of sufficient intensity to cause shearing of pipes. Those lines with valves mounted to the power-plant floor could be expected to show a preference for shearing between the valve and the containment-building wall. The others would probably shear beyond the valves. Previous history indicates that an earthquake of this intensity would not occur at the EBR-II site. It should also be noted that the containment might not be breached if there were to be a shearing of the valve lines, since another failure would also be required on the part of the line inside the containment wall.

C. Instrument- and Plant-air Systems

The instrument-air and plant-air systems described in Section X of the EBR-II Operating Instructions each have 1-1/2-in. lines penetrating the reactor building. These lines operate at approximately 80 psig. All system components are designed for conditions of 100 psig and 150°F or greater. The instrument-air system supplies air for vital functions including the control-rod actuators and some other pneumatic instrumentation.

TABLE II. Freon Isolation Valves

<u>System and Refrigerating Capacity</u>	<u>Pipe Size, in.</u>		<u>Penetration No.</u>	<u>Valve Support</u>	
	<u>Inlet</u>	<u>Outlet</u>		<u>Containment- vessel Wall</u>	<u>Power-plant Floor</u>
Shield cooling, 40 tons	1-1/2		15L		X
		3	15L	X	
Sub-basement, 20 tons	1-1/4		13L	X	
		2	13L		X
Sub-basement, 20 tons	1-1/4		15H		X
		2	13L	X	
North space cooler, 20 tons	1		20	X	
		2	20	X	
East space cooler, 20 tons	1		21	X	
		2	21	X	

The plant-air system supplies the air for pneumatic tools when they are used inside the reactor building. It also supplies air to an eductor to purge water condensate from refrigeration coils. The eductor is float-controlled from the water level of an accumulator tank and functions to purge condensate moisture from the refrigeration coils to the suspect exhaust. Both of these air systems have isolation valves outside the reactor building.

There is no functional requirement to maintain the plant-air system operational during periods of reactor-building isolation. The instrument-air system, however, should remain operational to maintain proper operation of the pneumatic instrumentation and the pneumatic controllers of the reactor ventilation system. There is no requirement for supplying air to the control-rod actuators during the period of reactor isolation, because the reactor is shut down under such conditions.

An emergency air compressor was provided in the original installation of EBR-II systems to supply air to the vital functions inside the reactor building during periods of reactor isolation. Tests conducted in October 1968, and again in March 1970,⁶ showed that the emergency air compressor was inadequate for supplying the instrument-air requirements. These tests resulted in the initiation of PM 332 (see Sec. IV) to replace the emergency air compressor with a larger unit. The tests were also used to estimate the actual air requirements in the reactor building during periods of building isolation. Minimum air requirements for the instrumentation were established to be between 9 and 10 cfm. The maximum requirement is 20 cfm. These values for air requirements are in substantial agreement with data given in Figure X a-1 of the EBR-II Operating Instructions. With the installation of the larger emergency air compressor, the reactor building may remain isolated for an indefinite period of time without loss of air supply to vital pneumatic systems. A calculation was made to determine the rate at which pressure inside the reactor building would increase during a period of reactor-building isolation in which the isolation valve on the incoming instrument-air line remains open. Based on the higher value (20 cfm) of air consumption inside the building and on 500,000 cu ft of air volume inside the reactor building, the rate of pressure increase would be 0.83 in. wg/hr.

D. Argon Supply System

The argon requirements of the reactor building are supplied from the FEF through one penetration of the reactor building. The argon is delivered at 57 psi through an isolation valve outside the reactor building. The argon system inside the reactor building is shown schematically in Fig. 3. The system is further described in Section IV F of the EBR-II Operating Instructions and in Volume II, Chapter 7, of the EBR-II System Design Descriptions. The system normally consumes approximately 4 cfm of argon for shaft seals, normal leakage of the primary tank, the fission-gas monitor (FGM), and the chromatograph. Changes of pressure within the reactor building bring the floating-head tank into action to maintain a positive differential pressure in the primary tank with respect to the reactor building. Therefore, changes of pressure in the reactor building or of temperature-induced pressure changes in the primary tank have a corresponding effect on argon demand.

The time limit for which the argon systems inside the reactor building can properly function during conditions of reactor-building isolation is dependent upon the argon supply inventory stored inside the reactor building. The floating-head tank provides an inventory of 8-hr supply⁷ for conditions of normal demand and with the floating-head tank at the low position. An emergency argon supply inside the reactor building uses eight cylinders, each containing 336 cu ft of argon. This supply provides an additional 900 hr of argon inventory for conditions of normal argon consumption. As originally installed, manual action was required to make the emergency argon supply available. This arrangement is being changed by PM 180 (see Sec. IV) to make the emergency supply automatically available without the requirement of operator action.

The pressure-vacuum relief system for the primary tank, installed under PM 120 (see Sec. IV), provides an additional 176 cu ft of argon supply. This system would come into action only if the argon pressure inside the primary tank becomes negative with respect to the reactor building by at least 4 in. wg. It was not provided to supply argon inventory, but rather to provide a backup for the floating-head tank and the capability to compensate for sudden changes in pressure in either the reactor building or the

blanket-gas space of the primary tank. The volume of the gas space in the primary tank is approximately 650 cu ft and normally is at a pressure of 1 in. wg with respect to the reactor building. Loss of the argon supply while all other conditions remain normal would cause the gas pressure in the tank to eventually seek equilibrium pressure with the reactor building. However, if the FGM and the chromatograph were permitted to remain in operation, argon would be pumped out of the gas space at 0.3 cfh, which corresponds to a pressure decrement of 0.3 in. wg for each hour these two systems remain in operation. The pressure-vacuum relief system of the tank would not operate until the pressure in the tank decreased to 4 in. wg negative. With a negative pressure inside the tank, some inward leakage of air may be expected.

During a period of reactor-building isolation, the argon gas consumed from the emergency argon supply may contribute to the pressure of the reactor building. The maximum pressure that can be contributed by this source is limited to the inventory of the emergency argon supply. The maximum inventory is 2688 cu ft, which in a building volume of approximately 500,000 cu ft would cause a pressure change of 0.066 psi or 1.82 in. wg. This assumes that all eight bottles on the emergency argon supply are dumped into the reactor building during leaktight isolation.

The capacity of the emergency argon supply is sized so that it is capable of maintaining proper pressure of the argon-gas blanket in the primary tank with respect to the reactor building if the pressure in the building should increase to its maximum rating of 24 psig. The time duration that this proper balance could be maintained from the emergency argon supply would, of course, be reduced as the building pressure increases.

E. Xenon-tag System

A line for sampling the argon-gas blanket in the primary tank for xenon tags was added in 1969 under PM 232 (see Sec. IV). This line was brought out through a spare penetration and double isolating valves to a sampling station outside the reactor building. The sampling system is used to identify experimental-irradiation subassemblies suspected of releasing fission-product gas to the primary tank. Experimental capsules are tagged

with small quantities of gaseous xenon in unique isotopic combinations, thereby permitting specific identification subsequent to cladding leakage during reactor operation. The sample line has a normally closed shutoff valve at the sampling station. The two isolating valves are outside the reactor building. A key switch allows bypassing of a building-isolation signal, so a xenon-tag sample can be taken with the building isolated.

F. Cover-gas Purge System

A new system for increasing the rate of purging the cover gas in the primary tank will be installed under PM 302 (see Sec. IV). The new system will enhance the functions of the xenon-tag system. The identification of leaking subassemblies in the event of increases in cover-gas activity depends on the determination of the ratio of the isotopes in a particular tag. In turn, this determination requires that concentration of other tags in the cover gas be extremely low to prevent confusion of subassembly identification. This problem will be overcome with the installation of the cover-gas purge system. A spare 1/2-in. tube in reactor-building penetration No. 2 will be used to route the argon purge line through the containment and to the suspect exhaust line. An isolation valve will be on each side of the reactor-building penetration.

G. Silicone Cooling System

The silicone cooling system is used to provide cooling, via a NaK loop, for the cold trap of the primary-sodium purification system. These systems are shown schematically in Figs. 4 and 5. More complete descriptions are in Chapter 7, Volume II, of the EBR-II System Design Descriptions and in Section IV C of the EBR-II Operating Instructions. Two silicone lines penetrate the reactor building, and each has an isolation valve outside the building. The sodium purification system need not be in continuous operation; in fact, it may be shut down for weeks without affecting the operation of the reactor. The silicone isolation valves close whenever an isolation trip occurs, thereby causing loss of cooling to the crystallizer

tank of the cold trap. In this event, standard operating procedure requires that the sodium purification system be shut down, which may be done remotely from the control room. If the operator fails to follow proper procedure, the crystallizer tank will tend to increase in temperature. The normal temperature of the tank is 300°F. An alarm signal is initiated in the control room when this temperature increases to 450°F. Upon receipt of this alarm, the reactor operator would be reminded that the primary purification system should be shut down. If the alarm is ignored and the purification system continues in operation without corrective action, the temperature of the crystallizer tank may increase sufficiently to cause dissolution and return to the reactor of some of the trapped solids in the crystallizer tank.

IV. PLANT CHANGES AFFECTING THE ISOLATION SYSTEM

A list of plant modifications that have had or may have effects on the isolation system follows:

PM No. 18 - Removable covers on isolation valves of reactor-building ventilation system to allow inspection of valve seats.

PM No. 40 - High-pressure annunciator alarm and safety relief valve on reactor-plant argon manifold.

PM No. 52 - Emergency argon supply in reactor building.

PM No. 94 - Partial isolation feature permitting reactor operation with ventilation system isolated to prevent spread of low-level contamination.

PM No. 120 - Pressure-vacuum relief system for primary tank.

PM No. 148 - Construction of the radioactive sodium chemistry loop (RSCL).

PM No. 180 - Making operation of emergency argon supply automatic.

PM No. 232 - Xenon-tag system.

PM No. 299 - Interlocks of the ventilation-system isolation valves so all close if one closes, to prevent pressurization of reactor building. Also provides for indication of reactor-building pressure in control room.

PM No. 302 - Increased rate of purging primary-tank cover gas to make the xenon-tag system more effective.

PM No. 308 - Administrative bypass of control circuitry of isolation valves on the Freon refrigeration lines after isolation occurs.

PM No. 309 - Electrostatic prefiltering for purification-cell ventilation.

PM No. 332 - Larger emergency air compressor in reactor building.

V. REACTOR OPERATING EXPERIENCE BEARING ON THE ISOLATION SYSTEM

A five-channel radiation monitor is located near the base of the suspect exhaust stack. The channels monitor (1) gross particulates, (2) gross fission gas, and, selectively, (3) ^{131}I , ^{41}A , and ^{24}Na . Throughout the history of the EBR-II operation, there have been no incidents in which a detectable quantity of radioactive contamination, as measured by the stack monitor, has been rejected to the suspect exhaust.⁸ There have been two incidents, however, in which low-level contamination could have been rejected to the suspect stack. These were (1) a leak of the cladding of fuel in experimental-irradiation subassembly X011 in May 1967⁹ and (2) a leak and fire in the primary purification system in April 1969.¹⁰

A. Fuel-cladding Leak in X011

On May 24, 1967, a leak in the cladding of oxide fuel in experimental-irradiation subassembly X011 precipitated an increase in cover-gas activity and, subsequently, an increase in the radiation level in the reactor building. This was the first such cladding penetration in the history of operation of EBR-II and the most significant to date. The incident was sufficient to drive radiation levels in the depressed area of the mezzanine floor to a maximum value of 250 mR/hr. While the incident was sufficient to raise background radiation levels throughout the reactor building, no measurable activity was indicated by the stack radiation monitor. Ventilation to the reactor remained normal throughout the period of interest, and the reactor building did not isolate. This incident prompted plans for adding a partial-isolation feature to the circuitry controlling the isolation system.

1. Installation of Partial Isolation

The partial-isolation circuit permits isolation of the ventilation system of the reactor building from the control room by manual control. This feature is used more as a matter of precaution rather than for any situation in which reactor-building isolation is clearly indicated. Throughout any period in which the reactor building is partially isolated, the complete building isolation system continues to be active and subject to trip from any of the controlling monitors. For a low-level incident involving leaks in the cladding in subassemblies inside the reactor, radioactive contamination is limited to the gaseous fission products that escape from the argon-gas blanket. These products are essentially limited to the fission products of xenon and krypton; iodine fission products are retained within the primary tank. A study authorized under Work Authorization Form (WAF) No. 101, dated July 17, 1970,¹¹ evaluated the acceptable leak rate for the reactor building with a hypothetical incident in which a subassembly containing oxide fuel is damaged in the fuel-unloading machine (FUM) or the interbuilding coffin (IBC) and all the iodine, xenon, and krypton is released to the building. This event would cause higher levels of gaseous fission products in the reactor building than those levels actually present with a cladding penetration in the reactor and would add the burden of iodine contamination. The results of the WAF-101 study indicate that a reactor-building leak rate of 10,000 cu ft/day would be acceptable.

B. Leak and Fire in Primary Purification System

On April 3, 1969, a fire occurred in the purification cell. It resulted from a leak in the pump tube of the EM sampling pump. The reactor had been shut down for five days before the incident and, as a result, the radiation levels were relatively low. The fire was restricted to the primary purification system and involved no more than 3-1/2 lb of primary sodium. The building was evacuated approximately 40 min after the first evidence of smoke was observed, when it was noted that the reading of the building-exhaust monitor^{*} had increased approximately 150 cpm above the normal background

* This monitor, one of many general-purpose gamma monitors inside the reactor building, is near the exhaust duct of the building.

of less than 10 cpm. The increased radiation level indicated by the monitor was evidently the result of an increase in background radiation, because no increase in radiation level was detected by the five-channel stack monitor. A gamma particulate monitor near the exit of the suspect exhaust line inside the reactor building also gave no indication of increase in background radiation level.

To maintain a negative pressure in the purification cell with respect to the reactor building, the cell had its own local ventilation system and exhaust blower (later modified), with a nominal flow rate of 300 cpm, operating in conjunction with an absolute filter. Post-fire investigations showed that the filters became at least partially plugged in probably less than 15 min after the start of the fire. With the filters plugged and the air-conditioning system blowing air into the cell, the pressure became positive in the cell, and smoke was blown out of the crevices in the cell. The exhaust blower was shut off about 30 min after the fire started.

PM No. 309 (see Sec. IV) provides for electrostatic prefiltering in the purification-cell exhaust system, which should significantly increase the life of the absolute filter during both normal and abnormal (fire) conditions.

C. Experience with Isolation Trips

The isolation system is interlocked with reactor-shutdown circuitry so that the reactor shuts down whenever the reactor-building isolation is initiated. All isolation trips throughout the history of EBR-II operation have been spurious. Ten such spurious trips occurred in CY 1968-1970 (see Table III).

VI. RESPONSE TIMES OF ISOLATION TRIP MONITORS

The response time of the radiation monitors used in EBR-II to initiate isolation was calculated from information in the instruction manual for the monitor.¹² With an assumed radiation field of 1R/hr, the calculated time constant is 0.8 sec. This time constant represents the time required for the monitor to indicate 63% of a small step change in radiation level.

TABLE III. Trips of Isolation System, 1968-1970

<u>Sensor Initiating Trip^a</u>	<u>Number of Trips</u>		
	<u>1968</u>	<u>1969</u>	<u>1970</u>
Building temperature	0	3	0
Building pressure	1	1	0
Radiation	0	1	0
Subassembly outlet temperature	2	0	0
Other (power failure, failure in circuitry, etc.)	2	0	0

^aAll trips were spurious.

The time constant becomes shorter as the radiation level increases. Appendix A gives the basis for the calculated time constant.

A laboratory test was made¹³ to determine the response time of the pressure trip monitor. For a 4-psi step change in pressure, the sensor output was observed to respond with a 0.2-sec dead time followed by a 2.5-sec time constant (for 63% response). The pressure sensor is set to initiate building isolation at 8 ± 2 in. wg. For the 4-psi input signal used in the test, trip occurred in less than 0.5 sec.

No actual response-time tests have been conducted for either the thermocouple or the resistance thermometers used in the isolation circuit. The thermocouple can be expected to have a time constant of the order of 500 msec, and the resistance thermometer of the order of several seconds.

VII. POTENTIAL SOURCES OF EVENTS REQUIRING ISOLATION

The following sources are considered to be potentially capable of precipitating events requiring building isolation: (1) the reactor; (2) the sodium purification cell; (3) the sodium-sampling cell; (4) the radio-active sodium chemistry loop, or RSCL; (5) the fuel-element rupture detector, or FERD; (6) fuel handling; and (7) equipment handling. Most events that can be hypothesized from these sources would be associated with sodium fire. A possible exception would be a fuel-handling incident above the reactor main floor in which a fuel subassembly may melt or be damaged without an associated fire.

A. Events Originating in the Reactor

The potential events originating in the reactor pertinent to this study are those in which the primary containment is ruptured or radio-active products are permitted to leak from the primary tank. The spectrum of incidents begins with such minor events as fuel-cladding leaks and extends upward to include the worst accident considered on pp. 326-329 of the Hazard Summary Report¹⁴ (energy equivalent to 300 lb of TNT released inside the reactor tank). The cladding-leak type of incident imposes no special burden on the isolation system, because leakage to the reactor

building is limited to gaseous radioactive fission products of xenon and krypton. Iodine fission products are retained inside the primary tank because of the scavenging effect of the sodium environment.

An attempt has been made to compare the response of the isolation system to low-level radiation with the guidelines of Code of Federal Regulations 10 CFR 100³ and AEC Manual 0524.⁴ The radiation monitors used in the isolation trip circuit are set to trip at 450 ± 50 mR/hr. For evaluation, an event was hypothesized in which the radiation level in the reactor building is just below the level that would initiate an automatic trip of the isolation system. Gaseous fission products could begin to migrate to the building exhaust and subsequently to the suspect exhaust stack. The exhaust flow from the reactor building is normally 6600 cfm. The total stack flow is approximately 10 times that amount (66,000 cfm), because other systems at the EBR-II site use the same stack. It may, therefore, be reasonably assumed that the highest airborne radiation level outside the reactor building will be no greater than 10% of the maximum level inside the reactor building if the reactor building is the sole source. A resulting maximum radiation level of 45 mR/hr would fall well within the reference guidelines (see end of Sec. II).

It should be noted that while radiation detection cannot anticipate the event, it can anticipate a release of the fission products to the outside environment. For low-order events in which the reactor building is not pressurized, the transit time through the ventilation system provides a significant time delay between the event and the onset of contamination of the outside environment. The transit time from the shield-cooling outlet manifold to the exit isolation valve was calculated by L. P. Cooper to be 2.4 sec (see Appendix B). This gives a reasonable time for detection of the release and initiation of valve closure before contamination of the outside environment begins.

Considering now the more extreme incidents, the following quotation is taken from Ref. 1.

"The EBR-II primary containment system is capable of withstanding, without breaching, a nuclear energy release within the reactor core equivalent to the detonation of 300 lb of TNT. Although production of missiles of significant energy is improbable, the reinforced concrete missile shielding readily stops the most highly energetic missile possible. Very small amounts

of sodium unquestionably may escape into the building from the "pressure vessel," such escape occurring primarily through the shield air cooling ducts, and possibly, from ruptured sodium pipes within the top closure."

The maximum pressure that can be contained by the primary containment tank is first limited by the maximum stress limits of the vessel bottom.¹⁵ For that reason, most incidents capable of breaching the tank may be expected to cause leakage of sodium coolant to the shield-cooling ducts surrounding the bottom and sides of the tank. Early detection of such an event may, therefore, be obtained by sensors located in the exhaust manifold of the shield-cooling system. The location of sensors is considered in more detail in Sec. VIII.

The reactor-building ventilation system is capable of maintaining exhaust air flow from the building, through the absolute filters, and to the suspect stack in the presence of building pressures up to 6 in. wg. Higher pressures would exceed the maximum driving head of the building supply fan and establish flow in the reverse direction if building isolation has not been initiated. The inlet ventilation lines are short (approximately 50 ft long) and have conventional filters and air-conditioning equipment.

B. Events Originating in the Sodium Purification Cell

The sodium purification cell is a potential source of pool-type sodium fires.¹⁶ The sodium inventory in the cell is approximately 650 gal, and the design flow rate through the cell is slightly less than 100 gpm. Most components of the primary purification loop are below the level of the sodium in the primary tank. In the event of a leak in the purification loop, therefore, large quantities of primary sodium could syphon through the leak. To preclude such an occurrence, a fail-safe syphon-break system installed under PM 169 is connected to the purification-loop tank. This system is described in Section IV C of the EBR-II Operating Instructions and in Volume II, Chapter 7, of the EBR-II System Design Descriptions. The extent of radioactivity associated with this system is strongly influenced by the reactor operating history. After extended shutdown, the radioactivity of the primary system, excluding solidified materials in the crystallizer tank of the purification system, is dominated by ¹³⁷Cs and ²²Na. The total

^{137}Cs activity currently present in the EBR-II primary system is approximately 6.4 Ci. The level of ^{22}Na activity is approximately 16 Ci.¹⁷ The saturation activity of ^{24}Na for 62.5-MWt operation is approximately 4.5×10^5 Ci.

The sodium purification cell is a potential source of a sodium fire involving a few hundred gallons of sodium. The potential events of interest originating in this cell tend to be of the type which permit easy detection and isolation of the reactor building before contaminants can be released to the outside atmosphere.

The absolute filters serving the cell are assumed to be capable of only limited filtration of sodium aerosols or smoke. An electrostatic filter to be installed in the cell ventilation system (PM 309, see Sec. IV) should significantly increase the time before the products of combustion would pressurize the cell and cause contamination or smoke to be released to the reactor-building area. Several improvements were made in the leak-detecting, alarm, and fire-protection equipment serving the cell by PM Nos. 278, 333, and 362 in early 1971.¹⁸

C. Events Originating in the Sodium-sampling Cell

The sodium-sampling cell is also a potential source of radioactive sodium fires. The cell (described in Section IV G of the EBR-II Operating Instructions and in Vol. II, Chapter 7, of the EBR-II System Design Descriptions) contains a limited inventory of sodium (less than 1 gal) that could precipitate potential sodium fires. It is served by a local ventilation system, which includes an absolute filter. The sample lines connect to the sodium-purification cell. Therefore, although the sampling cell has a very small inventory of sodium, fire in the cell could be sustained by flow of sodium from the purification cell. These connecting lines, however, are normally closed except during actual sodium sampling. The absolute filters serving the sampling cell have limited capacity and, like those in the purification cell, are expected to be effective only for small sodium fires. Fires originating in the sampling cell may be expected to be limited in size and to expand at such a low rate that adequate time will be available for initiating reactor isolation if required.

D. Events Originating in the RSCL

The radioactive sodium chemistry loop (RSCL) is designed to contain within its own cells the maximum credible accident relating to the loop. The loop is contained within shielded walls and includes a local ventilation system with a combination of absolute and graded-media filters designed to contain and limit the spread of radioactivity or sodium smoke from the maximum credible fire originating within the system. The system includes a combination of smoke and leak detectors that provide alarm and interlock protection functions. A fire-safety study for this system considered a hypothetical spill of 53 lb of sodium at 910°F uniformly distributed over the floor.¹⁹ The actual system (as built) has a sodium inventory of 50-60 lb of sodium, depending on the experiments being conducted. The sodium purification cell and sodium-sampling cell are close to the RSCL. Any isolation trip monitors provided for detecting escape of radioactivity from those cells will also provide similar protection for the RSCL.

E. Events Originating from the FERD System

The FERD system is described in Volume II, Chapter 6, of the EBR-II System Design Descriptions. According to a recent study,¹⁸ this system has very limited potential for generating incidents of the type in which isolation would be required. The FERD system uses an electromagnetic (EM) pump with a flow rate of approximately 100 gpm to draw primary sodium from near the outlet of the intermediate heat exchanger. The sodium lines route the flow to a bank of FERD-system detectors in the depressed area below the main reactor operating floor. The sodium lines are contained within a shielded channel. The FERD loop operates at a sub-atmospheric pressure throughout the entire system.¹⁸ A contaminating sodium fire is possible only if the system should become pressurized by some unforeseen event. The fire could be expected to be limited in size, and the radiation level would be related to the reactor operating history at the time of the event. Plans have been made¹⁸ to add smoke detectors and fire-fighting equipment (under PM 375) to the area where the FERD system is located.

F. Events Originating from Fuel Handling

Incidents involving damaging or melting of fuel subassemblies during fuel handling could potentially release substantial quantities of fission gases to the reactor-building atmosphere. The maximum activity of fission-product gases in a single oxide fuel pin could be as high as 1550 Ci.²⁰ A ruptured subassembly with no associated fire may permit the escape of iodine as well as krypton and xenon fission-gas products. This type of event was evaluated by R. M. Fryer.¹¹ The evaluation was based on a postulated event in which all the xenon, krypton, and iodine from a high-burnup oxide subassembly is released to the reactor-building atmosphere. The evaluation showed that leakage from the reactor building at the rate of 10,000 cu ft/day under these conditions would not cause the radiation dose rate outside the building to exceed AEC standards (for a major accident). The spectrum of possible incidents of this type may or may not include fire and extends from incidents in which a very minor increase in the reactor-building radiation level is detected up to and including incidents in which all the gaseous fission products of an entire subassembly escape to the reactor-building atmosphere. The most extreme case would not necessarily result in damage to reactor systems, but would present a difficult problem of decontamination in order to permit further reactor operations. Radiation monitoring is the best means of detecting such incidents. Pressure and temperature monitors probably would not detect the event. Speed of response is not critical, because the event is expected to progress on a slow time base.

G. Events Originating from Equipment Handling

Installation of equipment in the primary tank and removal of equipment from the tank offer limited potential for incidents that could require reactor-building isolation. Removal of equipment offers the possibility that sodium adhering to the equipment may ignite as it comes into contact with the air environment of the reactor building. Such operations are conducted only after extended shutdown, when the ²⁴Na activity has decayed to insignificantly low values and the bulk sodium system is at reduced

temperature. Potential incidents are limited to the combustion of a small quantity of adhering sodium containing ^{22}Na and ^{137}Cs . The activities of ^{22}Na and ^{137}Cs in the bulk sodium system currently are approximately 16 and 6.4 Ci, respectively.¹⁷ Traces of ^{210}Po , ^{54}Mn , and ^{60}Co have also been identified in the surface contamination of equipment removed from the primary tank. An event originating from equipment handling that requires total reactor-building isolation has a very low probability because of the limited amount of sodium available for combustion and the low activities of the sodium during such operations. The reactor-building ventilation system may be expected to remove the products of combustion by filtration, thereby preventing the release of radioactive contamination to the outside environment. To preclude the possibility of fire, pulling tubes are normally used during removal of equipment from the primary tank. (Use of these tubes prevents the equipment being removed from coming into immediate contact with air.) As an additional safety measure, partial isolation of the building is usually imposed manually for such operations.

VIII. PROPOSED CHANGES AND RECOMMENDATIONS

This study has revealed areas in which the reliability of the isolation system can be improved. The sensing circuits can also be modified to respond to a wider spectrum of events. Proposals for these improvements follow.

A. Sensors Initiating Isolation and Reactor Shutdown

The sensors currently used to initiate reactor-building isolation were briefly described in Sec. III. They are actuated by high values of (1) subassembly outlet temperature, (2) reactor-building radiation, (3) reactor-building temperature, and (4) reactor-building pressure.

1. Subassembly-outlet-temperature Monitors

It is proposed that the subassembly-outlet-temperature monitors be removed from the circuitry that initiates reactor-building isolation.

These monitors, however, would be retained in the reactor shutdown circuit as they currently exist. They are currently set to trip when the sub-assembly outlet temperature exceeds a set point of 10% higher than the normal outlet temperature for each of the particular subassemblies monitored. These monitors were included in the isolation-system circuitry to anticipate the need for reactor-building isolation. The rationale for the proposal to remove them from the isolation circuitry is as follows:

(1) an increase of subassembly outlet temperature to give a 10% increase in the normal ΔT through the subassembly is neither indicative of the need to isolate nor a reasonable indication that the need for isolation is imminent; and (2) the pressure and radiation monitors provide the needed protection for isolation of the building.

2. Radiation Monitors

The present isolation system includes two radiation monitors set to initiate building isolation when the radiation level exceeds 450 ± 50 mR/hr. One of these is above the control-rod superstructure, and the other above the fuel-handling console. Radiation monitoring is the most direct method of detecting incidents originating from fuel handling. Because of the time delay between the incident and the onset of contamination of the outside environment, the radiation monitors are capable of anticipating the need to isolate the reactor building. These monitors can be expected to respond in less than 1 sec for any event in which the radiation level exceeds twice the radiation level of the set point (see Appendix A). Total reaction time to valve closure would be less than 1-1/2 sec.

It is proposed that radiation monitoring be modified to use two-out-of-three trip logic. The recommended arrangement on the reactor main operating floor is to locate three similar monitors above the fuel-handling console. This arrangement would require relocation of the radiation monitor currently above the control-rod superstructure and addition of a monitor. The monitor above the control-rod superstructure is normally disconnected during fuel handling to permit proper rotation of the

rotating plugs. Relocation of the monitor to the fuel-handling console will permit the monitor to remain active throughout all periods of fuel handling.

Ideally, radiation monitors should be located to provide an "envelope of protection" around each of the systems that are capable of precipitating the need for reactor-building isolation. The tighter the envelope, the greater will be the time available for reaction between the incident and the onset of contamination of the outside environment. An alternative to the "envelope-of-protection" concept is to locate radiation detectors so as to monitor the inlet and outlet of the reactor-building ventilation system. Any contamination migrating to the outside environment would then necessarily pass near the radiation detectors, but there would be less reaction time available before the onset of contaminant leakage to the outside environment.

The three monitors above the fuel-handling console should meet the requirements of both concepts for radiation protection. They would provide an envelope of protection for events originating from the fuel-handling equipment and would also be near the inlet of the reactor-building ventilation system. There they would be able to detect airborne radioactivity from any source about to escape by reverse flow through the reactor-building inlet.

It is proposed that another group of three radiation monitors be located near the shield-cooling outlet manifold in the reactor-building basement (see Fig. 6). This location was chosen for three reasons: (1) any major incident involving the primary tank has a high probability of introducing radioactive contamination into the shield-cooling system; (2) the location is near a three-way junction of the shield-cooling ductwork (where part of the air flow is recirculated), which is the farthest point upstream along the ductwork where total shield-cooling exhaust flow can be monitored; (3) the location is near the sodium purification cell, the sodium-sampling cell, and the RSCL, and in the position most favorable for early detection of contaminating incidents originating within these cells.

The two locations selected for the radiation monitors will give reasonable coverage for early detection of radiation leakage from all

Fig. 6. Plan of Reactor Basement, Showing Proposed Location of Three New Radiation Monitors

potential sources except the thimble-cooling system. The cooling air discharged from the thimbles is routed through two widely separated pipes to the sub-basement, where they join and connect to absolute filters (see Fig. 2). From the filters, the air passes through a turbocompressor and is discharged into the reactor-building exhaust system. To detect radiation leakage from the thimble-cooling system, it is proposed that a third group of three radiation monitors be located near the filter bank of the thimble-cooling system, a few feet upstream of the point where the thimble- and shield-cooling exhausts join before they are discharged to the suspect exhaust stack.

It would be desirable to monitor the combined outlet flow of both the shield- and thimble-cooling systems with a single set of monitors, but to be effective, the monitors must be near or upstream from the absolute filters. Unfortunately, the exhausts of the thimble- and shield-cooling systems do not join until they have passed through separate filter banks. Therefore, there is no single location upstream of the filter banks suitable for monitoring the outlet of the total reactor-building ventilation.

The multiple locations proposed for the radiation monitors will provide upstream monitoring of both the shield- and thimble-cooling systems. These two systems together form essentially the total exhaust of the reactor-building ventilation system. Consideration was given to a possible alternative of locating a set of radiation monitors near the shield-cooling filter bank. Such a location (similar to that proposed for the thimble-cooling system) would take advantage of the enhanced radiation signal resulting from concentration of radioactive contaminants in the filters. The decision to locate the second set of monitors further upstream in the basement area near the RSCL, sodium-purification cell, and sodium-sampling cell is a trade-off compromise. The upstream location gives greater lead time and allows the monitors to do double duty in monitoring the cell complex as opposed to the benefits from the enhanced radiation signal resulting from the filtration process.

Two low-level gamma monitors, one near the shield-cooling filter bank and the other near the thimble-cooling filter bank, now exist as part

of the general radiation-monitoring system of the reactor building. These monitors are two of several such low-level radiation surveillance monitors in the reactor building that function independently of the isolation trip circuit. Although the low-level gamma monitors are independent of the isolation system, they do provide some backup protection. Their primary purpose is detecting and alarming on low-level events. The two units installed near the filter banks are located so that buildup of radiation on the filter will be detected.

3. Temperature Monitors

The isolation system currently uses three temperature monitors in one-out-of-three trip logic. Two are resistance thermometers, and one is a thermocouple. All three are along the periphery of the reactor building, approximately 10 ft above the operating floor. They are set to trip at $145 \pm 5^{\circ}\text{F}$. The purpose of these monitors is to detect significant sodium fires even though the radioactivity of the sodium may be very low. It is proposed that the temperature monitors be removed from the isolation circuit. Justification for their removal is based on the adequate protection that will be provided by the modified radiation-monitoring system.

4. Pressure Monitors

The isolation system is currently served by one pressure monitor set to initiate isolation when the reactor-building pressure exceeds 8 ± 2 in. wg with respect to the outside environment. It is proposed that two more pressure monitors be added to the system and incorporated with the existing one to establish two-out-of-three trip logic. The purpose of pressure monitoring is specifically to detect the extreme incidents in which explosion and/or fire is prevalent and may destroy the radiation monitors in the isolation system. The pressure monitors are outside the reactor building and are served by pressure taps penetrating into the building. They are not expected to be damaged by such incidents.

B. Refrigeration System

The reactor-building isolation system as originally installed was directed toward protecting the outside environment in "event of major nuclear incident."⁵ It was assumed that the incident initiating the need for isolation would have damaged the reactor systems to the extent that further damage was of secondary importance. This philosophy was inherent in the design in that isolation of the reactor included closure of all of the refrigeration lines penetrating the reactor-building shell. Shutdown of the refrigeration system, if sustained, could commit the reactor systems to serious heat damage (see Appendix C).

The evolution of the EBR-II mission from that of an experimental demonstration reactor to an irradiation facility and the addition of special experiments, e.g., instrumented subassemblies, the in-core instrument test facility (INCOT), and the RSCL, have all increased the probability of lower-order events that could require the isolation of the reactor building. These events by themselves may not cause significant damage to the reactor systems. It would be unfortunate if some minor event should indirectly permit heat damage to the reactor plant.

The refrigeration compressors, all inside the reactor building, are interlocked so they are shut down upon receipt of an isolation signal. Upon reopening of the refrigeration isolation valves, the compressors are automatically re-energized. This arrangement prevents over-pressure trips within the refrigeration systems.

On page 265 of Ref. 5, the following statement is made: "The other seven pipes (the secondary-sodium piping, including vent lines and shutdown cooling lines) require no isolation valve because the exterior portions of the piping systems involved are capable of withstanding conditions more severe than those anticipated subsequent to a nuclear incident." This same statement could equally well be applied to the refrigeration piping. All the design ratings of components in the silicone and Freon refrigeration piping systems exceed 140 psi. Each of these systems is a closed loop both inside the reactor building and outside, and thus provide

double containment without the need of isolation valves. It is, therefore, proposed that all isolation valves on refrigeration systems, including both the Freon and silicone systems, be removed from the isolation trip circuit.

The 40-ton compressor serving the shield-cooling system is equipped with an automatic-resetting over-pressure trip. The four other refrigeration systems with Freon refrigerant use manual-resetting over-pressure trips. It is recommended that the manual-resetting over-pressure trips be replaced by the type that resets automatically. This will eliminate the need to enter the reactor building if over-pressure trip should occur during building isolation. Consideration of the consequence of ruptured refrigerant lines inside the reactor building shows that the Freon inventory in the reactor building can have very little effect on a maximum credible type of accident (see Appendix D).

C. Partial-isolation System

The partial-isolation system was added to prevent the spread of low-level contamination while permitting reactor operation. As the system was originally installed, initiation of partial isolation closed the isolation valves of the ventilation system only. With the installation of the xenon-tag system, the xenon-tag sample line was also connected to the partial-isolation circuitry. It is now proposed that the plant- and instrument-air lines also be interconnected to close during partial isolation. (During isolation, the essential needs of the instrument-air system are supplied by an emergency air compressor inside the reactor building.) This will eliminate the tendency for the reactor building to become pressurized by leakage from these two air systems during partial isolation. The rate at which the reactor building would become pressurized for normal operation and consumption of the air inside the building was considered in Sec. III.C of this report. In the present arrangement, the pressurization is limited to 8 ± 2 in. wg, at which pressure complete isolation takes effect to close both air supplies.

IX. SUMMARY

The original criteria for the EBR-II isolation system considered only serious incidents in which damage to the reactor plant would be of secondary importance. In the existing system, tripping of the isolation system imposes in itself a threat of heat damage to the reactor plant, because the tripping results in shutdown of the refrigeration system. The existing isolation system is adequate with respect to the original criteria. However, the operation of the reactor as an irradiation facility has increased the potential need for the isolation system to respond to less serious events than those originally considered. Furthermore, the industry is evolving to a policy of stricter standards that may impose a need for reactor-building isolation for relatively minor incidents in which there is no direct threat of physical damage to the reactor plant.

The indirect threat of heat damage to the reactor plant resulting from shutdown of the refrigeration systems during reactor-building isolation can be overcome by modification of the isolation trip circuit. This modification will not compromise the safety or reliability of the isolation system.

Additional gamma monitors of the type currently in use can provide the increased surveillance necessary to assure detection of low-level events. Incorporating these monitors in two-out-of-three trip logic will reduce the number of spurious trips. The existing trip setting of 450 ± 50 mR/hr appears to be a reasonable setting to avoid false trips while providing protection from incidents capable of damage to the outside environment. In effect, the transit time through the ventilation ducts provides time to act after the onset of a potentially contaminating event before contaminants begin to actually escape the reactor building.

The temperature trip monitors used in the existing system have questionable value, and their removal from the system is recommended in view of the proposed increased coverage by gamma monitors. Similarly, the subassembly-outlet-temperature monitors have questionable value as currently incorporated in the existing isolation trip circuit, and their removal from this circuit is recommended. The reactor-building pressure

monitor provides specific backup for the monitors in an extreme event in which physical damage causes the gamma monitors to become inoperative. It is recommended, however, that two more pressure monitors be added and the three be employed in two-out-of-three logic.

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APPENDIX A

Response Time of Radiation Monitors*

$$\text{Sensitivity} = 10^{-10} \text{ A per R/hr}$$

$$\text{Cable length} = 40 \text{ ft}$$

$$\text{Capacitance} = 20 \text{ } \mu\mu\text{F/ft}$$

$$\text{Total capacitance} = 40 \times 20 = 800 \text{ } \mu\mu\text{F}$$

$$\text{Input resistance} = \frac{0.1}{I}$$

For a radiation field of 1R/hr:

$$\text{Input resistance} = \frac{0.1}{10^{-10}} = 10^9 \text{ } \Omega$$

$$\text{Time constant} = RC = 800 \times 10^{-12} = 0.8 \text{ sec}$$

*Values used in calculations are from Ref. 12.

APPENDIX B

Transit Time from Shield-cooling Outlet Manifold to Exit Isolation Valve

The time for air to travel from the shield-cooling outlet manifold to the exit isolation valve was computed in two parts: (1) the transit time from the manifold to the junction of the recirculating portion of the system, and (2) the transit time from the junction to the exit isolation valve. The transit time for each was calculated from the fundamental relationships of flow rate, cross-sectional area of the ducts, and the length of the ducts. These relationships are

$$Q = VA, L = Vt, \text{ and } t_{\text{total}} = t_1 + t_2,$$

where Q = flow rate (cfm),

V = velocity (fps),

L = duct length (ft),

A = cross-sectional area of duct (ft²)

t_1 = transit time for first duct (min),

t_2 = transit time for second duct (min), and

t_{total} = total transit time from manifold to exit valve (min).

Therefore, the transit time for each duct or for both ducts is $t = \frac{L}{V} = \frac{LA}{Q}$.

The geometry on which the calculations was based is

Total flow rate was 19,600 cfm, recirculation flow rate 14,800 cfm, and exhaust flow rate 4,800 cfm.

From these values,

$$t_1 = (18)(8.4)/19,600 = 0.0077 \text{ min},$$

$$t_2 = (50)(3.1)/4800 = 0.0323 \text{ min, and}$$

$$t_{\text{total}} = 0.0077 + 0.0323 = 0.04 \text{ min} = 2.4 \text{ sec.}$$

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APPENDIX C

Heatup Rate during Building Isolation

The Hazard Summary Report,¹⁴ on p. 40, projected a total reactor-building heat load of 506,000 Btu/hr. Actual heat rejection from the as-built reactor plant is considerably larger. Modifications of and additions to the reactor plant have also tended to increase the heat load. A survey conducted by R. C. Watson²¹ of ANL Plant Engineering in October 1970 indicated that the present reject heat load of the reactor building is approximately 1,500,000 Btu/hr. The capacity of the refrigeration units serving the reactor building is approximately 1,703,500 Btu/hr.

In the original design, trip of the reactor-building isolation system closes the inlet and outlet air ducts to the reactor building and shuts down all the refrigeration systems. All the heat load must then be dissipated within the reactor building. A limiting maximum heatup rate for this condition can be estimated by assuming that all reject heat goes into increasing the air temperature within the building. Actually, much of the initial heat load will go into the massive concrete structures in the building, and some will be dissipated to the outside of the building.

The reactor building contains approximately 500,000 cu ft of air. Based on a specific heat value of 0.240 Btu/lb-°F and a density of 0.062 lb/cu ft, the limiting initial heatup rate would be 202°F/hr. The increase in temperature could result in an initial increase in building pressure of 4.5 psi/hr.

The maximum amount of time that the reactor building can be safely allowed to remain isolated is probably first limited by the maximum permissible temperature in the concrete shielding. The stated maximum permissible temperature for the shielding is 150°F. This limit is set conservatively low to prevent deterioration of the concrete. The shielding is normally cooled by the flow of 20,000 cfm of air, which passes through the cooling coils of a 40-ton refrigeration unit. The normal temperature difference for air entering and leaving the cooling coils is approximately 40°F. With the loss of the refrigeration system, the temperature difference across the coils would vanish, and a rise in shield temperature would ensue.

It is the loss of the refrigeration systems that imposes a time limit on isolation. The safe period of isolation can be extended indefinitely if the refrigeration systems are permitted to remain in operation.

APPENDIX D . .

Effect of Release of Freon Refrigerant on Pressure
in Reactor Building

Five refrigeration units using Freon lines penetrate the reactor building. The operating pressures of the Freon systems all fall between 30 and 125 psi. The highest relief-valve setting is 225 psi, and the lowest 210 psi. The piping is Type K copper tubing. The Freon inventory in the five systems is approximately 1100 lb.

The possibility of interaction between Freon and other systems has been considered. Sodium reacts violently with Freon-12 (the type used at EBR-II). The products of the reaction are primarily sodium fluorides, sodium chlorides, and free carbon. Considerable energy is released in the reaction, although the products of the reaction are generally more dense than the unreacted chemicals.

The vapor pressure of Freon-12 is approximately 100 psi at room temperature. If the Freon piping should become breached, the leaking Freon would quickly vaporize. The atomic weight of Freon-12 is relatively high: 120 as compared to 29 for air. If the entire 1100 lb of Freon were to be released to the building atmosphere during isolation, the building pressure would increase less than 1%. This pressure increase comes from the addition of 9.17 lb-moles of gas to a building containing approximately 1,080 lb-moles of air. The net effect is that the Freon inventory in the reactor building can have very little effect on a maximum credible accident.

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