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OAK RIDGE NATIONAL LABORATORY

MARTIN MARIETTA

The Oak Ridge Research Reactor - Safety Analysis

Volume 2
Supplement 2

S. S. Hurt

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NOTICE: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

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Martin Marietta Energy Systems, Inc.
for the
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1. INTRODUCTION

The Oak Ridge Research Reactor Safety Analysis was last updated via ORNL-4169, Vol. 2, Supplement 1,[1] in May of 1978. Since that date, several changes have been effected through the change-memo system described below. While these changes have involved the cooling system, the electrical system, and the reactor instrumentation and controls, they have not, for the most part, presented new or unreviewed safety questions. However, some of the changes have been based on questions or recommendations stemming from safety reviews or from reactor events at other sites.

Those changes which do not pose new or unreviewed safety questions or which have not resulted from safety reviews or activities related to reactor incidents should be documented in ORNL-4169, Vol. 1, entitled, The Oak Ridge Research Reactor -- Functional Description;[2] and a supplement containing appropriate descriptive material will be published.

It remains, then, to discuss those changes which were judged to be safety related and which include revisions to the syphon-break system and changes related to seismic considerations which were very recently completed. The maximum hypothetical accident postulated in the original safety analysis[3] requires dynamic containment and filtered flow for compliance with 10CFR100 limits at the site boundary.

It will be shown that the changes described herein provide assurance of natural circulation cooling of the shutdown reactor and therefore make it incredible that a seismic event will result in damage to the fuel. It follows then that the credible seismic event, which will probably defeat the confinement system, will not result in a violation of 10CFR100 limits, as no radioactivity will be released to the environment.

2. ORR CHANGES, METHOD FOR EFFECTING AND DOCUMENTING

Changes to ORNL reactors are effected through the change-memo system which has been used by the Operations Division for several years* (Operations Division Special Administrative Requirement No. 7-1) and has recently been adapted for general use by ORNL (SPP-29, Attachment III). This system involves the use of two forms, i.e., the Instrumentation and Controls Design Change Memo and the Mechanical Design Change Memo. These forms are used as certification that the proposed changes are reviewed and approved prior to being effected, that the changes are properly completed, and that drawings and procedures are revised as needed. Review by the DOE (earlier ERDA or AEC) is required only if the change involves an unreviewed safety question or for other reasons is judged (by ORNL's Office of Operational Safety) to require DOE approval.

Those changes to the ORR which were completed prior to the publication of the functional description, dated September 1968, were included in that descriptive document. This includes reviews documented by I&C Change Memos Nos. 1 through 85. Of the remaining changes, as shown in Table 1, only six required DOE approval. Mechanical and electrical changes prior to September 1968, while not documented by change memos, are also included in the description. None of the mechanical change memos, Table 2, required DOE approval.

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*Instrument and controls change memos date back to March 19, 1959, mechanical change memos to October 29, 1968.

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required Division	RORC	DOE	Date Initiated	Completed
86	BLDG. VENT FLOW	To replace sail-type switches with pressure switches.	Yes	No	No	8-20-68	9-18-68
87	ORR REA. DEM. MODIFICATIONS	To install a second transmitter for conductivity readout and replace a single-pen recorder with a two-pen recorder.	Yes	No	No	7-19-68	9-18-68
88	COOLING TOWER FAN SPEED CONTROLLER	To install a new and redesigned fan speed controller.	Yes	No	No	9-17-68	10-8-68
89	STEAM TURBINE SUPPLY PRESSURE	To provide reactor setback and reverse actions should steam supply reduce to 56 psi.	Yes	No	No	11-1-68	11-13-68
90	ORR EXPERIMENT TIE-IN SYSTEM	To provide two completely independent channels of reactor setback protection for each experiment.	Yes	No	No	2-26-69	3-5-69
91	PROCESS RADIATION MONITORS	To replace vacuum tube type with a solid-state device.	Yes	No	No	6-17-69	8-27-69
92	TEST MEASUREMENT OF ORR CONTROL ROD WORTH	To connect spare flux signal from output of micro-microammeter for transmission to HFIR computer.	Yes	No	No	6-23-69	6-27-69
93	SAFETY CHAMBER SUPPLY VOLTAGE	To add a 30,000-ohm resistor and a 20-μfd capacitor to chamber voltage supply to prolong life of uncompensated sections of PCP-11 and 111 chambers in use.	Yes	No	No	6-26-69	6-27-69
94	REUTER STOKES RSN-76A CHAMBERS	Safety chambers removed from LITR were canned for underwater use at ORR.	Yes	No	No	10-10-69	10-10-69
95	LOG N AMPLIFIER	To install a zener and selenium rectiier to prevent variation with ambient temperature in output of Log N amplifiers.	Yes	No	No	10-24-69	10-31-69 No.1 Log N 11-5-69 No.2 Log N
96	EXPANDING THE MONITORING OF DC UNITS	To provide a separate monitoring station in control room for each dc unit.	Yes	No	No	10-30-70	4-14-71
97	ACTIVATING NEW INSTRUMENTATION FOR NOG SYSTEM	The installation of a new filtering system.	Yes	No	No	4-26-71	8-5-71

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
98	ORR FISSION CHAMBER ASSEMBLY	To replace the preamplifier located in reactor flux with an out-of-flux preamplifier.	Yes	No	No	4-26-71	4-28-71
98 Addendum 1	ORR FISSION CHAMBER ASSEMBLY	To install a 0.1 μf 400-V condenser across relay coils R-67 and R-90 to reduce noise in the counting channels.	Yes	No	No	4-28-71	4-28-71
99	SAFETY CHAMBER VOLTAGE SUPPLY	To reduce B(+) supply voltage from 385 V dc to 275 V dc to lengthen service life of chambers.	Yes	No	No	6-17-71	8-24-71
100	MONITORING OF PERIOD CHANNEL CHAMBER LEADS	To assure integrity of Log N system by providing a means for checking that all cables are connected.	Yes	No	No	8-30-71	10-12-71
101	GAMMA CHAMBER NO. 63-3 MONITORING	To provide monitoring B(+) and signal braids on chamber leads.	Yes	No	No	9-7-71	9-13-71
102	SHIELD FOR HIGH FREQUENCY PULSES IN MODULES OF PROCESS RADIATION INSTRUMENTATION	To replace aluminum box which houses filter chokes with a steel box to prevent interaction between the modules.	Yes	No	No	2-10-72	2-29-72
103	REMOVAL OF SETBACK ACTION FROM LOW STEAM SUPPLY TO CELL VENT SYSTEM	To remove setback action which is adequately covered by other means.	Yes	Yes	No	3-3-72	4-3-72
104	NORTH AND SOUTH GAMMA CHANNELS	To provide a scram test for both channels.	Yes	No	No	4-11-72	8-28-72
105	INSTALLATION OF ACID PUMP FOR ORR POOL TOWER pH CONTROL	To improve operation of tower pH control.	Yes	No	No	5-5-72	8-25-72
105 Addenum 1	ACID PUMP FOR ORR POOL TOWER pH CONTROL	To provide required dead-band in pH control circuit.	Yes	No	No	8-21-72	8-25-72
106	CURRENT ADDER FOR LOG N AMPLIFIER	To prevent false period signals during reactor startup.	Yes	No	No	1-9-73	4-10-73

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
107	FISSION CHAMBER PREAMPLIFIER	Changed capacitor No. C7, 0.1 μf to a 1.0 μf 25 V ceramic to stabilize gain on amplifier.	Yes	No	No	2-20-73	4-10-73
108	AC POWER FAILURE MONITORING FOR TEMPERATURE CHANNELS	To provide annunciation for loss of ac power to outlet temperature and ΔT recorders.	Yes	No	No	1-14-74	1-29-74
109	MODIFICATION OF SAFETY TROUBLE MONITOR	To replace 6-V relays (no longer in production by manufacturer) with 24-V ac relays.	Yes	No	No	6-14-74	10-4-74
110	SERVO DRIVE RELAYS R13X AND R14X	To replace the two electro-mechanical relays with two solid-state relays.	Yes	Yes	Yes	3-10-75	4-7-75
111	TESTING ALL ANNUNCIATORS	To permit testing all annunciators at once with a single pushbutton.	Yes	No	No	6-13-75	10-8-75
112	MODIFICATION OF EXPERIMENT TIE-IN SYSTEM	To limit reactor power reductions from certain experiment initiated setbacks to a level of not less than 0.6 N_F.	Yes	No	No	3-18-76	3-31-76
112 Addendum 1	MODIFICATION OF EXPERIMENT TIE-IN SYSTEM	To provide automated reduction in power level when operating in manual mode by insertion of No. 6 shim rod until all level safety recorders indicate less than 0.6 N_F.	Yes	No	No	3-26-76	3-31-76
112 Addendum 2	MODIFICATION OF EXPERIMENT TIE-IN SYSTEM	To provide a second experiment with limited setback action.	Yes	No	No	2-20-80	3-24-80
113	N^{16} INSTRUMENTATION	Replacement of vacuum tube picoammeters with solid-state versions and relocate instrumentation.	Yes	Yes	Yes	1-3-77	4-29-77
114	LOG N CHANNELS	To install a 10,000-ohm filter resistor in series with input to Log N amplifier to reduce electrical noise in channels.	Yes	No	No	4-10-79	8-31-79
115	REACTOR TEMPERATURE CHANNELS	To replace Type 4FH batteries in ΔT and reactor outlet temperature recorders with units designed especially for this purpose.	Yes	No	No	7-23-79	10-1-80

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
116	10 KILOWATT SERVO CONTROL	To permit temporary servo control of reactor power at 10 kilowatts.	Yes	No	No	10-18-79	11-1-79 (Restored to normal)
117	COMPENSATED ION CHAMBER POWER SUPPLY	To replace vacuum tube power supply with a solid-state unit with zener diode regulation.	Yes	No	No	1-28-81	2-11-81
118	SERVO CHAMBER POWER SUPPLY	To replace the three zener reference diodes with five zener reference diodes to permit adjustment of the output voltage to match the chamber operating characteristics.	Yes	No	No	2-13-81	2-23-81
119	SIGMA AMPLIFIER AC LINE MONITOR	To install an under-voltage monitor to prevent possible misoperation of safety sigma amplifiers due to low ac line voltage.	Yes	No	No	3-30-81	11-10-83
120	PONY MOTOR VOLT-AMP ALARMS	To install and upgrade new units for monitoring high or low voltage and high or low current conditions.	Yes	No	No	4-20-81	4-24-81
121	REACTOR PRIMARY MAKEUP MONITORING	To permit measurement of makeup flow by use of an orifice-standpipe arrangement and provide pool level information.	Yes	Yes	Yes	9-10-81	1-25-83
122	SERVO AMPLIFIER REFERENCE VOLTAGE	This Change Memo cancelled. Issued for number continuity only.					
123R	LOG N PERIOD SCRAM INHIBIT	To prevent false reactor shutdowns from electrical noise spikes when reactor power is >1.5 MW.	Yes	Yes	Yes	10-6-82	1-27-83
123 Addendum 1	LOG N PERIOD SCRAM INHIBIT	To prevent false reactor scrams from noise signals originating in the period inhibit circuitry.	Yes	No	No	4-25-83	4-27-83
124	ANNUNCIATOR ACKNOWLEDGE AND RESET PUSH BUTTONS	To provide a set of push buttons on right side of reactor console in series with existing set on left side.	Yes	No	No	8-30-82	9-15-82
125	PRIMARY COOLANT STRAINER ΔP	To provide a high/low alarm and control room readout of strainer ΔP.	Yes	No	No	9-27-82	6-28-83

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
126	FISSION COUNTING CHANNELS	To upgrade fission counting channel by replacing binary scaler with a modular NIM solid state decade scaler.	Yes	No	No	2-15-83	10-22-84
126 Addendum 1	FISSION COUNTING CHANNELS	To replace a commercial vendor pulse amplifier with an in-house amplifier that has already been approved for use at the HPRR.	Yes	No	No	11-1-83	1-4-84
127	ANNUNCIATOR ADDITION AND REARRANGEMENT	To provide additional control room annunciators for slow scram, setback and reverse and rearrange existing annunciators to be more compatible.	Yes	No	No	7-20-83	8-1-83
128	REPLACEMENT OF CONTROL ROD POSITION INDICATORS	To replace pointer-gage type indicators with a more reliable digital display.	Yes	No	No	1-9-84	2-4-84
129	GAMMA SAFETY CHANNELS	To remove the gamma channels from the slow scram bus.	Yes	Yes	No	7-17-84	9-5-84
130	COMPUTER TO CONTROL ROOM INTERFACING	To permit monitoring of various reactor operating, measuring, and testing parameters.	Yes	No	No	1-30-85	3-15-85
131	ORR HEAT POWER MEASURING CHANNEL	To permit installation of new temperature probes in existing vacant pipe penetrations to provide additional functional capabilities that are not included in present equipment.	Yes	No	No	3-1-85	3-13-85
132	ORR SERVO REPLACEMENT	To permit installation of solid-state equipment to replace obsolete and difficult-to-maintain vacuum tube equipment.	Yes	Yes	Yes	3-22-85	8-19-85
132 Addendum 1	ORR SERVO SYSTEM INSTRUMENTATION	To permit changing the method of operator demand control, add a new feature for selection of flux calibration, and add switches to provide servo flux signal conditioner sensitivity adjustment.	Yes	No	No	4-7-86	5-6-86
133	REACTOR PROTECTION SYSTEM UPGRADE	To replace obsolete and difficult-to-maintain vacuum tube electronics with solid-state.	Yes	Yes	Yes	3-11-85	10-4-85
134	CELL VENTILATION ELECTRIC BLOWER MONITORING	To prevent reactor operation above 300 kW when the cell vent electric blower is not operating.	Yes	No	No	4-19-85	5-16-85

Table 1. ORR - I&C Design Change Memos from August 20, 1968, through February 27, 1986

Change No.	Title	Reason for change	Review required Division	RORC	DOE	Date Initiated	Completed
135	REACTOR SECONDARY TOWER BASIN WATER LEVEL MONITOR	To provide a control room indication of tower basin water level.	Yes	No	No	5-30-85	6-6-85
136	SEISMIC CHANNEL ANNUNCIATION	To provide a control room alarm from a strong motion accelerograph recently installed in basement adjacent to the south facility instrument rack.	Yes	No	No	2-18-86	3-6-86
136 Addendum 1	SEISMIC CHANNEL ANNUNCIATION AND ADDITION OF TWO CHANNELS OF SEISMIC SCRAM	To provide an automatic reactor scram when strong motion accelerograph exceeds set point.	Yes	No	No	6-12-86	Not complete
137	RERTR PROGRAM APPROACH TO CRITICAL MEASUREMENTS	To increase sensitivity of each neutron current amplifier and Log N amplifier. This changes the low flow mode from present 667 kW to 5 kW and increases the sensitivity of Log N amplifiers by a factor or 100. This is a temporary change to permit approach to critical measurements.	Yes	No	No	2-18-86	2-20-86
138	SET POINT CHANGES ON REACTOR PRIMARY FLOW, ΔT AND OUTLET TEMPERATURE	Reactor scram set points changed as an added safety margin for the LEU whole-core demonstration (set points lowered).	Yes	No	No	2-27-86	3-6-86
139	REACTOR STATUS REMOTE MONITOR	To provide signals to the new Laboratory Emergency Response Center.	Yes	No	No	2-27-86	5-9-86

Table 2. ORR - Mechanical Change Memos from October 29, 1968, through June 24, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
1	BLANKING CROSSOVER VALVES AT POG FILTER PIT	To prevent air leaking around filters through the crossover valves.	Yes	Mo	No	10-29-68	11-12-68
2	REPLACEMENT OF EXISTING ACCESS COVER WITH A MODIFIED ACCESS COVER	To provide an access cover which can be used to monitor water leakage between the reactor system and pool system by use of a double O-ring seal.	Yes	No	No	2-13-69	This Change Memo on "Hold" for an indefinite period
3	MODIFICATION OF LUBRICATION SYSTEM FOR THE ORR DIESEL GENERATOR TO INCLUDE A HAND-OPERATED PRIMING PUMP	To permit manual priming of the lubrication system prior to test runs.	Yes	No	No	3-4-69	4-29-69
4	CHANGES TO POG PIPING	To provide a better auxiliary piping scheme for conducting filter efficiency test.	Yes	No	No	5-19-69	6-27-69
5	INSTALLATION OF THE WASTE SPOOL PIECE EMERGENCY BYPASS LINES	To provide a means of decontaminating the demineralizer units (primary) should high radiation levels exist in the demineralizer cells.	Yes	No	No	8-1-69	10-17-69
6	REVISIONS TO ORR PIPING	To provide a means to discharge effluent from underwater saw box and west pool underwater vacuum cleaning device directly to LLW system.	Yes	No	No	8-28-69	10-12-69
7	DEACTIVATING THE QUICK OPENING VALVE ON THE ORR PRIMARY WATER SYSTEM	To install blank flange over outlet opening to prevent inadvertent opening of valve and dumping water from primary system to the ground and surrounding area.	Yes	No	No	2-11-70	2-19-70
8	ADDITION OF SOLENOID VALVE TO FUEL LINE ON GASOLINE ENGINE WHICH POWERS EMERGENCY COOLING PUMP	To permit flow of gasoline to the engine only when there is a demand for the engine to run.	Yes	No	No	2-24-70	3-10-70
9	STRAINER INSTALLATION ON REACTOR-POOL EQUALIZER LINE.	To provide better protection against flow blockage into the equalizer line.	Yes	No	No	5-21-70	6-4-70

Table 2. ORR - Mechanical Change Memos from October 29, 1968, through June 24, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
10	MODIFICATION TO WATER SUPPLY TO SHIM ROD DRIVES	To permit using facility cooling water for shim rod shock absorbers, wipers and annulus of bottom plug and minimize amount of demineralized water added to reactor system.	Yes	No	No	8-20-70	9-18-70
11	AIR CONDITIONER FAILURE MONITOR-EMERGENCY POWER MONITOR	To provide a visual and audible warning upon failure of the building air-conditioning compressor and to monitor for failure in electrical circuit of the fan in Building 3003.	Yes	No	No	10-30-70	11-13-70
12	MONITORING OUTPUT VOLTAGE OF THE BATTERY CHARGER FOR THE EMERGENCY DIESEL GENERATOR	To provide direct monitoring of the battery charger.	Yes	No	No	5-10-71	6-24-71
12 Addendum 1	MONITORING OUTPUT VOLTAGE OF THE BATTERY CHARGER FOR THE EMERGENCY DIESEL GENERATOR	Separate the monitoring of "diesel low oil pressure" and "high water temperature" from "battery breaker open."	Yes	No	No	5-13-71	6-24-71
13	FILTERING SYSTEM FOR NORMAL OFF-GAS	To install a filtering system which will provide removal of both particulate matter and radioactive nuclides.	Yes	No	No	5-24-71	8-25-71
14	REMOVAL OF THE "FLOW ELBOW" INSTRUMENT LINES TO BYPASS LINE 115	Flow through bypass line is no longer required. Trane coolers for 20-MW operation removed from service.	Yes	No	No	6-15-71	6-24-71
15	MODIFICATION TO ALARM SET POINT FOR LOW AMP TO MOTOR DC NO. 3	To provide a new potentiometer which permits a more precise adjustment of set point.	Yes	No	No	11-8-71	11-8-71
16	ADDING AN ISOLATING TRANSFORMER ON THE AC LINE SUPPLY TO NO. 3 DC MONITORING SYSTEM	To prevent spurious alarms due to noise spikes on the ac supply.	Yes	No	No	12-28-71	1-3-72
17	MONITORING STATUS OF 3042 BASEMENT SUMP PUMP IN THE PIPE TUNNEL VIA CONTROL ROOM ANNUNCIATOR	To provide annunciation in control room from high level of the north pump which is an abnormal condition.	Yes	No	No	1-24-72	2-15-72

Table 2. ORR - Mechanical Change Memos from October 29, 1968, through June 24, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
18	REMOVAL OF GCR 1 AND 2 AND MSR EMERGENCY POWER SIGNAL FROM ORR DIESEL	Emergency power is no longer required by these experiments as they have been removed from reactor.	Yes	No	No	9-20-72	11-7-72
19	REROUTING DRAIN LINE OF REACTOR COOLING TOWER BASIN	To separate the basin drain from the floor drains inside pumphouse to prevent water backing up in pumphouse when basin is being drained.	Yes	No	No	10-6-72	11-1-72
20	INSTALLING A NEW 208-VOLT SUMP PUMP AND MOTOR FOR REMOVAL OF STEAM CONDENSATE AND GROUND WATER FROM ORR BASEMENT	To replace existing pump with one of adequate capacity.	Yes	No	No	2-21-73	3-7-73
21	REPLACING BATTERY CHARGER ON EMERGENCY DIESEL	To provide adequate capacity to maintain batteries in the fully charged state.	Yes	No	No	4-23-73	5-31-73
22	IMPROVEMENT OF ELECTRICAL GROUND CONNECTION FOR BUILDING 3042	To move the present building ground system from the 6-in. fire main to the existing experimental ground system located at the southwest corner of Building 3042.	Yes	No	No	5-23-73	8-30-73
23	REMOVE TRANE COOLER BYPASS LINE AND RELOCATE BYPASS FILTERS	To permit removal of obsolete bypass line and relocate bypass filters to facilitate handling and minimize piping.	Yes	No	No	3-6-79	4-27-79
24	REROUTE DEGASIFIER EFFLUENT FROM HOT DRAIN (RETURN TO SYSTEM)	To permit routing entrained reactor primary water and steam condensate to the reactor system rather than to the low level waste system.	Yes	No	No	3-7-79	5-16-79
25	ALTER DRAIN SYSTEM FOR PRESSURIZABLE OFF-GAS FILTER PIT	The existing drain line is connected to the low level waste system. The drain line will be altered by connecting to both the low level waste system and the process waste system with appropriate cut-off and switching valves.	Yes	No	No	3-7-79	This Change Memo on "Hold" for an indefinite period
26	15-PLATE SHIM RODS	To permit changing from 14-plate to 15-plate in fuel section of shim rods to decrease cost of manufacturing.	Yes	Yes	No	10-23-80	10-27-80

Table 2. ORR - Mechanical Change Memos from October 29, 1968, through June 24, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
27	REVISION TO SYPHON BREAK SYSTEM-ADDITION OF CHECK VALVES AT TOP OF REACTOR TANK	To provide cooling water from the reactor pool to reactor core should syphon break action occur.	Yes	Yes	No	3-3-81	11-20-81
28	PLACING THE EMERGENCY GASOLINE ENGINE DRIVEN PRIMARY COOLING PUMP IN STANDBY	The emergency gasoline engine became obsolete with the installation of battery operated dc pony motors.	Yes	No	No	12-9-81	6-24-83
29	ACID MIX SYSTEM FOR DEMINERAL-IZER CATION COLUMN REGENERA-TION	To permit installation of a 750-gallon acid mix tank east of Building 3004.	Yes	No	No	2-8-82	6-24-83
30	DEMINERALIZER RECYCLE PUMPS	To permit installation of two recycle pumps, one for reactor coolant demineralizers and the other for the pool demineralizer.	Yes	No	No	7-9-82	6-24-83
31	ACCESS PORTS IN PRIMARY COOLANT LINES	To provide spool pieces in that portion of reactor coolant inlet and exit lines in the pipe chase to enable these lines to be inspected and repaired internally.	Yes	No	No	12-16-82	6-24-83
32	INSTALLATION OF STRAINERS IN PRIMARY COOLANT LINES	To prevent any large debris from recently installed mechanical patches between the basket strainer and reactor vessel reaching the fuel elements in reactor core.	Yes	No	No	2-22-83	6-24-83
33	INSTALLATION OF POOL COOLING TOWER TWO-SPEED REVERSIBLE FAN STARTER AND CONTROLS	To permit installation of new equipment to replace original equipment that manufacturer no longer manufactures components for.	Yes	No	No	1-14-83	3-24-83
34	ADDITION OF "INSERT CATCHER" TO HOLLOW BERYLLIUM CORE PIECES	To provide a 1/2-in. diam rod across the bottom of all the new hollow beryllium pieces to prevent loss of inserts.	Yes	No	No	5-20-83	12-12-83
35	PLANT DEMINERALIZER	To prohibit nitrate discharge to creek by replacing present demineralizers with vendor supplied cartridge type demineralizers.	Yes	No	No	4-10-86	8-6-86

Table 2. ORR - Mechanical Change Memos from October 29, 1968, through June 24, 1986

Change No.	Title	Reason for change	Review required			Date	
			Division	RORC	DOE	Initiated	Completed
36	ADDED SUPPORT FOR VERTICAL SECTION OF POOL EQUALIZATION SYSTEM	To permit installation of additional seismic restraints both horizontally and vertically to the pool equalization system as determined by a seismic engineering analysis.	Yes	No	No	6-24-86	8-28-86

3. CHANGES TO THE SYPHON-BREAK SYSTEM

The loss of adequate cooling for the ORR is discussed in the safety analysis.[3] In that treatment, it is shown that a primary coolant system break of sufficient magnitude to uncouple the reactor vessel from the pumps is highly unlikely. Such a break, due to the nature of the system, could normally occur only in the short spans of primary piping which are not buried, i.e., near the pumps or near the reactor vessel.

Routine inspection (in 1979) of these exposed portions of the piping indicated a slight degree of degradation due to external pitting, a condition which, considered separately, is of no serious concern. However, due to the age of the reactor, it was decided that a review of the capability of the cooling system, particularly in regard to leaks, should be performed. This review as reported in ORNL/CF-80/230[4] indicated that revisions to the ORR syphon-break system were in order.

The ensuing modifications, as documented by Mechanical Change Memo No. 27 and as described in the operating manual for the ORR,[5] ensure that the reactor vessel will not be drained (core uncovered) by any perceivable break in the primary cooling system, regardless of location. Further, it is ensured (through pump coastdown or makeup from the pool) that flow through the core will continue for several seconds subsequent to a shutdown triggered by such a loss-of-coolant event.

The instrumentation changes associated with the syphon-break system revision are documented through I&C Change Memo No. 121. This instrumentation provides for an alarm, reactor shutdown, and primary pump shutdown* at appropriate make-up water flow rates (effectively, leaks from the primary system). The instrumentation was made a part of the reactor protection system.

*Shutdown of the ac motors only -- the battery powered motors continue to operate.

4. SEISMIC CONSIDERATIONS

4.1 INTRODUCTION

The ORR and its ancillary systems were designed and assembled in accordance with building codes and other specifications applicable to nuclear facilities in the early and mid 1950s. However, as far as is known, none of the various components were seismically qualified, and there was no great concern regarding seismic events because of the low probability of occurrence, coupled with the equally low probability that a massive release of radioactivity would result from a tremor of the magnitude which at that time was considered applicable to the Oak Ridge area. The probability of a massive release was considered low because such an event would require that the fuel cladding integrity be lost, that the primary coolant system be breached, that the pool be partially drained, and that the confinement be rendered completely inoperative. Nonetheless, concern regarding seismic activity and seismic qualification of research reactors has increased over the years.

As a result of this increased concern regarding seismic events, limited seismic evaluations of the ORR were conducted by review teams from Idaho Nuclear Engineering Laboratory (INEL)[6] and the Central Engineering Division of Martin Marietta Energy Systems, Inc. (MMES).[7] These evaluations were based on a postulated earthquake which would result in a 0.15 g acceleration level. The selection of this level of acceleration is described by Beavers.[8]

4.2 THE INEL ASSESSMENT OF THE POSSIBLE RESULTS OF A MAJOR SEISMIC EVENT

Section 2 of the INEL evaluation[6] describes a possible accident sequence which would result from the postulated major seismic event (hereafter referred to as the event). Consideration is given to the various ORR components which are requisite to achieving and maintaining reactor shutdown, providing adequate decay heat removal, preventing a criticality event in the pool fuel-storage area, and mitigating the consequences of an accident. These effects are considered in the order in which they are presented in the INEL evaluation.

4.2.1 Primary and Secondary Pumps

It is postulated that the event would possibly result in loss of electrical power to the primary and secondary pumps with the reactor undergoing a flow coastdown. This condition, by itself, presents no particular problems, and the reactor has undergone many such flow coastdowns during power outages due to various causes. However, it is further postulated that the reactor protection system might be lost, and thus, a reactor shutdown would not be assured. This is not probable, as loss of the protection system itself results in a reactor shutdown; however, the point is well taken in that the exact sequence of events would be difficult to predict. The possible loss of forced shutdown cooling is also postulated, as the battery racks for the pony motors and the concrete block walls of the pump house, which flank the batteries and motors, are not seismically qualified.

4.2.2 Diesel Generator

It is considered possible that the instrumentation needed to monitor the reactor status could be lost, along with building lighting. These conditions would result due to loss of the diesel generator because its ancillary systems, i.e., starting batteries, cooling system, and fuel system are not seismically qualified. Supports and anchors for the diesel generator itself were considered to be adequate by the INEL team.

4.2.3 Primary Coolant System Piping

The primary piping system, it is postulated, will not suffer a large double-ended pipe break, but rather will suffer isolated cracks and small leaks at flanges. It is pointed out that an adequate supply of make-up water is dependent upon the proper operation of the syphon-break system.

4.2.4 Reactor Tank and Internals

The event will probably leave the reactor tank and its contents intact and the core in a coolable geometry. There is some threat of damage due to the possibility that the crane or working bridge might fall onto the tank.

4.2.5 The Reactor Pool

The pool is expected to remain intact following the event, even though the 3039 stack poses a minor threat.

4.2.6 The Reactor Building

Confinement integrity is not assured even though the building will probably withstand the event.

4.2.7 The 3039 Stack

According to a previous analysis,[9] the stack is expected to fail during a seismic event which would produce an acceleration of 0.06 g. The failure of the stack poses a threat to the exhaust fans, the filter banks, and to the confinement shell of the reactor. The exhaust stack for the graphite reactor also poses a less significant threat to the ORR confinement.

4.2.8 Plant Emergency Procedures

Emergency procedures which would be placed in effect during the event were judged to be adequate. The plant evacuation system (which includes the area evacuation system) is not seismically qualified, and there is no assured tie between that system and the local area evacuation system.

4.2.9 Fuel Racks in the Pool

Calculations indicate that fuel rack restraints are not needed to prevent tipping or spilling during the event. Thus criticality accidents due to stored fuel will not result from the event.

4.2.10 Fire Protection System

This system was not evaluated; however, the fire suppression system in the ORR control room was noted to lack seismic supports, which indicates that seismic loading was not considered in the design of the fire protection system.

4.2.11 Conclusions and Immediate Decisions Resulting from the INEL Evaluation

Immediately following the INEL evaluation, it was concluded that the probability of damage to the ORR core as a result of the event could be minimized by: (1) scramming the reactor automatically upon detection of preshock ground movement or waves which precede the strong forces of a seismic event by several seconds, (2) seismically qualifying the syphon-break system, and (3) administratively controlling the position of the crane and working bridge when not in use.

These three actions provide assurance that the fuel will remain intact throughout and after the event, and therefore, that the health and safety of the general public is not endangered.

It was further concluded that reactor operation could be safely continued while other recommendations presented in the INEL evaluation are considered or acted upon as discussed in Appendix A.

Accordingly, the physical changes outlined above were effected and documented via I&C change memo No. 136, Addendum No. 1, and by mechanical change memo No. 36. The administrative changes were effected and documented through the use of applicable standard procedures.

4.3 MARTIN MARIETTA ENERGY SYSTEMS, INC., EVALUATION

A somewhat different approach is taken in the assessment of possible results of a major seismic event as presented in the Martin Marietta Energy Systems, Inc., evaluation,[7] in that it is assumed that a safe shutdown condition will be assured if and when the reactor is shut down by the control or protection system and the core remains covered by water. The ORR complex is divided into six subsystems for the purpose of the Martin Marietta evaluation. These subsystems agree roughly with the subsystems previously discussed, and the conclusions reached are in general agreement; however, to provide clarity, the following summary is offered.

4.3.1 The Cooling Water System External to the Pool, Including Piping, Supports, Pumps, Buildings, and Heat Exchangers

This subsystem is essentially the reactor primary cooling system, outside the reactor pool, and is not considered essential in the Martin Marietta evaluation since the syphon-break feature would prevent loss of water from the reactor tank through this piping.

4.3.2 Penetrations of the Reactor Pool Wall Including Experimental Ports

The penetrations are judged to be of adequate strength to resist damage from the event. This is particularly true of most of the experiment access ports which are capped inside and outside the pool, thus requiring a double failure to result in water loss.

4.3.3 The Reactor Pool and Pool Supports

The Martin Marietta analysis of the pool structure and supports indicates that the pool walls will suffer only small deflections during the event, and that the pool support structure is adequate to resist the forces expected. Loss of water is not predicted.

4.3.4 The Reactor Vessel

The Martin Marietta review team found the vessel and its supports adequate to maintain integrity during the event. There remains, however, the threat of damage from falling objects such as the crane and working bridge as previously discussed.

4.3.5 The Reactor Building, Crane Supports, and Overhead Crane

It is possible that deformation of the rails would allow the crane to fall. While damage to the pool structure produced by the impact of the falling crane would be minor, it is possible that the syphon-break system could be rendered inoperative. The Martin Marietta team also judged that the building walls might be damaged by the falling crane.

4.3.6 The Syphon-Break Piping System

The syphon-break system, as existing at the beginning of the Martin Marietta evaluation,[7] was found by calculation to be marginally adequate to withstand the forces expected during the event. Satisfactory supports for the system, however, were designed, built, and installed during the course of the evaluation. As a result of this effort, the modified syphon-break system is considered seismically qualified to resist the forces developed during the event.

4.4 ANALYSIS OF CREDIBLE CONSEQUENCES OF A MAJOR SEISMIC EVENT

As stated in the Safety Analysis Report,[3] overheating of the fuel is the only mechanism which can cause serious damage to the ORR core. Overheating, in turn, could only result from a power increase, a coolant loss, or a combination of the two. It is important that the consequences of a major seismic event be considered in this light.

4.4.1 Accidental Reactivity Accidents as Related to a Major Seismic Event

Power increases can be triggered only by reactivity increases due to credible events. The two types of reactivity accidents considered possible are: (1) a startup accident and (2) a rapid insertion of reactivity due to failure or malfunction of a reactor or experiment component or because of misoperation of the reactor.

It was concluded in the safety analysis[3] that a startup accident cannot cause damage to the ORR core. The possible results of a major seismic event do not change this conclusion because the reactor is protected against the predicted events such as loss of primary flow, loss of electrical power, or loss of reactor instrumentation, all of which result in the termination of rod withdrawal and/or a scram. The addition of a scram as a result of the seismic event itself, as described earlier, reinforces this conclusion.

There is no apparent manner in which a seismic event could result in a rapid insertion of reactivity sufficient to cause damage to the core, since the reactor vessel and its internals will remain intact as shown by both the INEL and the Martin Marietta Energy Systems, Inc., evaluations.[6,7] This conclusion is supported by the argument presented in the safety analysis.[3]

4.4.2 Loss-of-Coolant Accidents as Related to a Major Seismic Event

Loss of coolant caused by a major seismic event would result from a break in the piping or from loss of all pumping power, either of which would result in a reactor shutdown and a pump coastdown (reactor shutdown would also be effected by the seismic event). Loss-of-coolant accidents previously considered in the safety analysis,[3] however, were essentially developed on the assumption that at least one of the three battery-powered motors would maintain flow at a rate greater than 500 gpm. The probability that at least one of the battery-powered motors will remain operable would be lessened by the possible effects of the event. If all pony motors are lost, then natural circulation cooling of the core, following an appropriate pump-coastdown period, becomes a very important consideration.

It is stated in the safety analysis[3] that the maximum heat flux to be expected from a shutdown ORR core is 75,000 Btu ft^{-2} h^{-1}. That value is based on an assumed 45-MW operation and no decay, and therefore, one finds that the value for 30-MW operation and an appropriate decay time is much lower.

Recent conservative calculations[10] of maximum shutdown heat flux following pump coastdown after 30-MW operation (30-s decay), result in an expected maximum of 27,000 Btu ft^{-2} h^{-1}. This compares with a maximum of 24,400 Btu ft^{-2} h^{-1} for the shutdown HFBR (40-MW and 45-s decay) as reported by Tichler.[11]

Gambill and Bundy[12] determined experimentally that the burnout heat flux for unrestricted natural circulation cooling for ORR conditions following a shutdown is approximately 125,000 Btu $ft^{-2}h^{-1}$. More importantly, they also determined that a relatively small burnout penalty is incurred by substantial restriction of the return flow path, a condition which exists in the ORR. Specifically, with a return flow area restricted to 11.3% of the test section area, the average burnout heat flux was determined to be 69,100 Btu ft^{-2} hr^{-1}. Tichler and Hill[13] found that for similar conditions, the burnout heat flux during the flow reversal process is approximately 46,000 Btu $ft^{-2}h^{-1}$.

It is readily determined from the information presented above that burnout conditions will not be encountered during flow reversal and natural circulation cooling of the shutdown ORR, providing that the return flow area available is greater than 11.3% of the area through which upflow will occur. To ensure that this condition prevails, administrative controls have been established requiring that the lattice contain four "dummy" fuel elements in low gamma heat positions as a prerequisite for operation at 30 MW. With this requirement, more than adequate return flow area is assured, and well-understood flow conditions for forced shutdown cooling are retained. The administrative requirement for the redundant, maximum-reliability system for forced cooling to remove afterheat remains in place. This requirement ensures a more orderly cooldown of the ORR core following shutdown except for the very improbable loss of all electrical power (including battery power) to the primary pump motors.

In summary, then, it may be stated that changes to the reactor and its administrative control have:

1. alleviated concern relative to loss of all primary and secondary pumps, since natural circulation cooling will prevent damage to the fuel following reactor shutdown,
2. reduced the need for the diesel generator, as its function related to instrumentation which monitors the reactor status following a shutdown will be replaced by a battery-powered system (see Appendix A),

3. lessened the concern regarding leaks in the primary coolant system piping because the syphon-break system will prevent loss of water from the reactor vessel,
4. decreased the probability of damage to the reactor vessel or pool through administrative control of the position of the crane and working bridge when parked, and
5. moderated the concern with respect to the probability of damage to the building (loss of confinement) or the 3039 stack as there will be no release of radioactivity if fuel damage is prevented.

4.4.3 Other Accidents as Related to a Seismic Event

Accidents involving an experiment containing radioactive material, accidents involving other radioactive material such as contaminated waste, and accidents related to criticality are considered in the safety analysis.[3] The effect of a major seismic event on these types of accidents is discussed below.

The radioactive content of experiments which are planned during the remaining life of the ORR is limited to induced activity, principally in metals such as stainless steel, and in heat transfer media such as a few grams of NaK or a small volume of inert gas. This material would not present a significant hazard to the public or the environment, even upon loss of containment which might result from the event.

Remarks regarding waste-handling accidents, as presented in the safety analysis,[3] remain applicable regardless of the cause of the accident, i.e., no new hazard is presented by the event.

Concern regarding the only threat to criticality safety presented by a seismic event, specifically the spilling of fuel from the storage racks, was alleviated by calculations as reported in the INEL evaluation.[6] Otherwise, the remarks in the safety analysis[3] remain fully applicable.

5. CONCLUSIONS

It may be concluded that the probability of damage to the ORR core has been reduced considerably by the changes to the syphon-break system and the addition of a seismically induced scram. These changes/additions and provision of a return flow path for natural circulation cooling ensure that the fuel will remain submerged in the coolant and that several seconds of forced cooling will be provided following reactor shutdown. Under these conditions, tests indicate that fuel-cladding integrity will be maintained, and therefore, there will be no major threat to the health and safety of the general public.

We also find no credible situation in which the changes entailed herein can create an accident or malfunction of a type different from those previously considered.

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APPENDIX A

RECOMMENDATIONS FROM THE INEL SEISMIC EVALUATION OF THE ORR - RESPONSES AND REACTIONS

Eleven recommendations are presented in the INEL evaluation. Those changes considered necessary prior to continuing operation of the ORR were immediately accomplished and reported.[1] The current status regarding all eleven recommendations is outlined below.

1. A thorough systematic seismic assessment should be performed for ORR on an as reasonably practicable basis.

   Detailed seismic assessments[2] were performed for the ORR pool structure, and for the syphon-break system, which are necessary to minimize the probability of damage to the fuel. The remaining ORR systems are being examined carefully to determine if, on a reasonably practicable basis, thorough seismic assessment is warranted prior to the anticipated shutdown of the facility.

2. An acceleration of 0.09 to 0.12 g should be used for seismic assessments of the ORR.

   This recommendation was considered, but it was decided that an acceleration of 0.15 g would be used. The basis for this decision is given by Beavers.[3]

3. The control-rod system should be seismically qualified by test.

   Calculations by INEL indicate that the control-rod system will undergo only slight deflections during the postulated seismic event. The static seismic qualification tests initially proposed by INEL were found not feasible, and therefore current efforts are directed toward determining the type of test which would result in qualification of the shim rod drive system. Both in-situ testing and test stand work are being considered.

   The Engineering Analysis Section of the Engineering Division of Martin Marietta Energy Systems, Inc., is currently extending the seismic analysis of the reactor tank and supports to determine the movement expected at possible test input points.

   Information resulting from this analysis will be used to help determine the type of qualification test to be used.

4. An automatic seismic shutdown system should be installed.

   This recommendation has been met.

5. The safety analysis for loss of flow should be redone to assure the reactor can be cooled by natural convection and to assure that a steam bubble either does not form or can be managed if it does form.

   A supplement to the safety analysis is being prepared.

6. The diesel electric power generation and distribution system should be seismically qualified.

   The diesel is not considered necessary for safe reactor shutdown, and only provides control room power for shim-rod seat lights and for one exit temperature recorder. A battery powered system is being designed to provide for these functions. This will lessen the need for seismic qualification of the diesel generator.

7. Piping restraints should be installed on the syphon-break system.

   Support members, seismically qualified for the credible event, were designed, built, and installed.

8. The four syphon-break check valves should be seismically qualified.

   Work toward seismically qualifying these valves is in progress.

9. Administrative restrictions should be placed on the parked location of the crane and working bridge.

   The suggested administrative restrictions have been placed in effect through the use of standard operating procedures provided for handling such administrative matters.

10. The requirements and need for an assured plant and area evacuation system should be evaluated and modifications made as appropriate.

    The primary method of communications for a plant or area evacuation is by means of signals sounded over the laboratory-wide Bell Telephone public-address system followed by verbal instructions. The laboratory-wide system operates on normal or emergency power. The system is divided into fifteen zones, and each zone has emergency diesel generators which automatically start on the loss of normal electrical power. Throughout the system, there are more than sixty-four public-address amplifiers and more than two thousand speakers. In addition, there are battery-powered bull horns available in emergency vehicles and at all portals. It is inconceivable that the majority of plant personnel would not begin evacuation on their own volition during a major seismic event.

11. The requirements and need for an assured fire protection system should be evaluated and modifications made as appropriate.

The fire protection at ORNL conforms to DOE Order 5480.1, Chapter VII, entitled "Fire Protection." The sprinkler system in the ORR was installed according to standards as recommended by the National Fire Protection Association NEPA 13, Standard for the Installation of Sprinkler Systems, with the exception of Section 3-103, "Protection of Piping Against Damage Where Subject to Earthquakes." According to "Factory Mutual System Loss Prevention" data sheets, it can be expected that severe damage to sprinkler systems will result if the ground motion is severe enough to cause partial building collapse or breakage of underground water mains. However, by the time the sprinkler system could be damaged, there would no longer be a need for the sprinkler for safety system purposes since the reactor would be shut down; thus reactor safety is not compromised.

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