REDACTED VERSION

See ML24137A312 for Full Version

SAFETY ANALYSIS REPORT

IDAHO STATE UNIVERSITY
AGN-201M RESEARCH REACTOR

LICENSE NO. R-110
DOCKET NO. 50-284

UPDATED April 2024

CONTENTS

SAFETY ANALYSIS REPORT
LIST OF FIGURES

SAFETY ANALYSIS REPORT

LIST OF TABLES

1 Introduction

This report is a periodic review and update to the Safety Analysis Report (SAR) to the United State Nuclear Regulatory Commission (NRC) by Idaho State University (ISU) at Pocatello, Idaho. The AGN-201 Reactor is a research and training reactor, class 104 license, R-110. Because of the extensive operating experience (since 1965) of this reactor and several other AGN-201 reactors, the system has a well established database, and hence no research and development activities were required to evaluate the system, from that covered in the previous edition of the SAR, dated January 2003. The reactor is administered and operated by Idaho State University for education, research, and training of students. However, the reactor is deemed a facility that benefits many interested parties, such as the Idaho National Laboratory, within the regional area.

The primary differences between this report and the last report (dated August 2021, are the following:

a) The description of the Channel 2 scrams based on the 2024 License Amendment Request and the drawings for the Lilibridge Engineering Building were updated.

This report contains information on the Pocatello, Idaho location, and site characteristics such as meteorology, geology, seismology, and demography, etc. It also includes a description of the Lillibridge Engineering Laboratory building where the reactor is housed on the ISU campus and a description of the reactor and listing of its characteristics.

The reactor is licensed to be operated at a steady state power level up to 5 watts. The total operational fuel loading provides for a maximum excess reactivity of 0.65% above a delayed critical condition. (The effective delayed neutron fraction for this very small core is 0.745%). The ultimate safety of the AGN-201 reactor design lies in its large negative temperature coefficient, its nuclear instrumentation safety system, and the thermal safety fuse. However, the maximum credible accident analyzed takes no credit for the effective operation of the nuclear instrumentation safety system. Thus, even if a sudden reactivity insertion were made, the reactor power rise would be attenuated and terminated by the negative temperature coefficient, and the reactor operation would be terminated by the melting of the thermal safety fuse which holds the upper and lower “halves” of the core together.

The principal university officer involved responsible for the administration of the AGN-201 reactor license is the Vice President for Research at ISU. Responsibility for the safety and general operation of the reactor has been delegated to the Nuclear Engineering Department in the College of College of Science and Engineering.

2 Location and Site Characteristics

Information in this section includes maps, building drawings and data on local population characteristics; meteorology, geology, hydrology and seismology of southeastern Idaho.

2.1 Location and Demography

Figure 2.1-1 is a map of southeast Idaho showing county and state demarcations, major rivers, lakes and reservoirs and the principal cities in the region. Pocatello in Bannock County and Idaho Falls in Bonneville County are the largest cities. The main campus of Idaho State University (where the AGN-201 reactor is located) is in Pocatello, and there is a branch campus in Idaho Falls, 50 miles northeast of Pocatello. The U.S. Department of Energy's Idaho National Laboratory is headquartered in Idaho Falls, where many of the research facilities are located, and also has research facilities on a 900 square mile desert site, 30-70 miles east of Idaho Falls (about 40-65 miles north northwest of Pocatello. Hill Air Force Base is about 120 miles south of Pocatello in the vicinity of Salt Lake City, Utah. Idaho State University is located in Pocatello, as shown in Figure 2.1-2. The major employers of this city of approximately 60,000 people is ON (a semi-conductor chip manufacturing company), a phosphate rock processing company Simplot, the Union Pacific Railroad, and the University. The major north-south highway, Interstate 15, passes to the east of the city and the campus. An intersection of I-15 with Interstate 84 which heads west toward the cities of Twin Falls and Boise is located at the north end of the city. The Pocatello Municipal Airport is about 7 miles northwest of the campus on Interstate 84. The Union Pacific Railroad east-west main line tracks go through the center of the city, about 2 miles west of the campus. The city lies in the Portneuf River valley at a general elevation of 4470 ft bordered on the southwest and northeast by hills of the Bannock mountain range which rises about 3,000-4,000 ft above the valley floor. Bannock County population information, including distribution by race, is given in Table

2.1-1 and Pocatello city government financial information if given in Table 2.1-2. Figure 2.1-3 is an ISU campus map showing the location of the Lillibridge Engineering Laboratory as well as the other buildings, etc. on the campus. The University presently has somewhat more than 12,000 students and 1200 faculty. Table 2.1-3 includes demographics of the estimated 10,430 people who reside within a one mile radius of the Lillibridge Engineering Laboratory. The Laboratory building is about 300 ft south of Carter Street. The nearest residential area extends northward from that street. The Lillibridge Engineering Building is located between the Laboratory and the Carter Street residences.

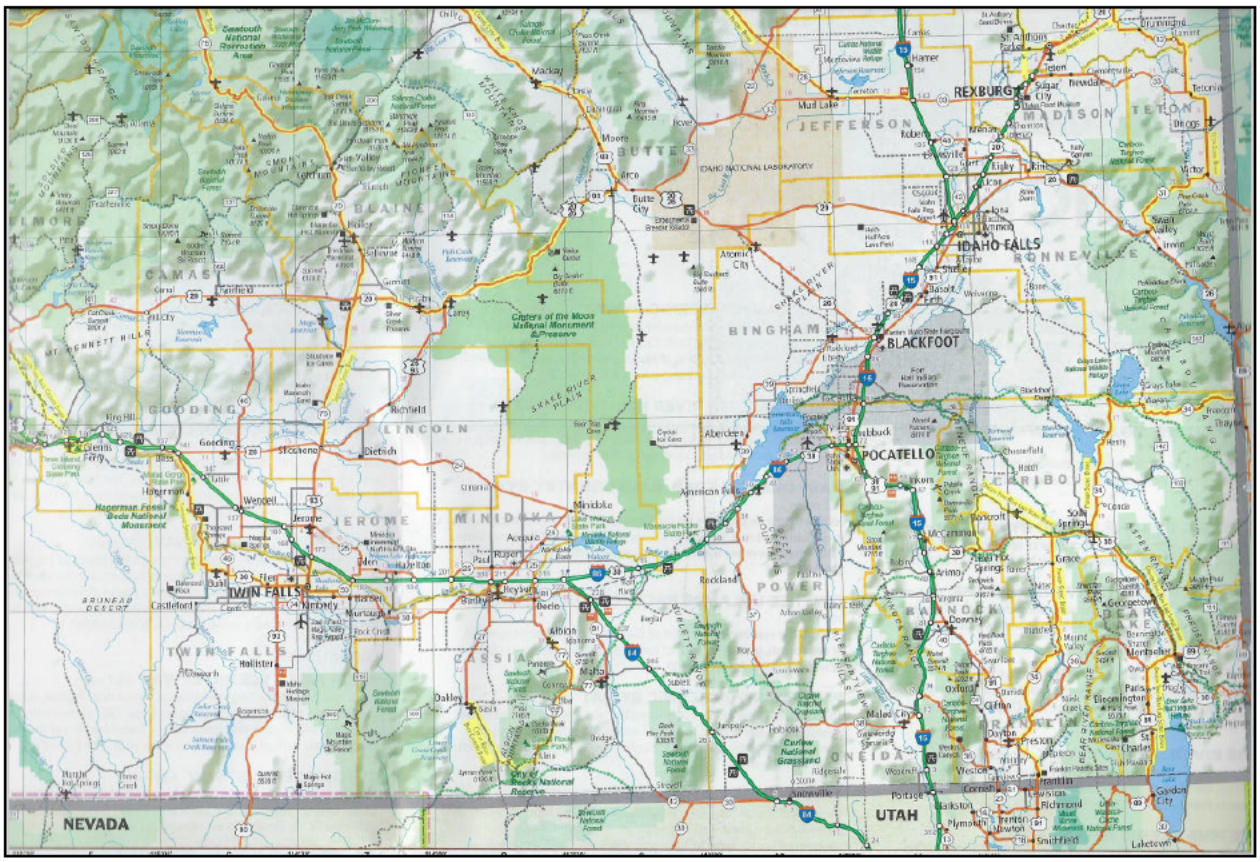


Fig. 2.1-1 Map of Southeast Idaho

Table 2.1-1 Incorporated Cities of more than 1000 population within 75 miles of Pocatello

City Name	Location from Engineering Building	Population 2010 Census	Population Estimate 2018*
American Falls	30 miles west		4,457
Ammon	East side of Idaho Falls	13,816	16,476
Arco	80 miles NW		995
Blackfoot	25 miles N	11,899	11,946
Chubbuck	N side of Pocatello	13,922	15,316
Idaho Falls	49 miles NE	56,813	61,635 (3)
Iona	57 miles NE		1,803
Malad City	60 miles S		2,095
Montpelier	95 miles SE		2,597
Pocatello		54,255	56,268 (1)
Preston	75 miles SE		5,204
Rexburg	74 miles NE	25,484	28,687 (2)
Rigby	66 miles NE		3,945
Ririe	72 miles NE		656
Shelley	39 miles NE		4,409
Soda Springs	55 miles E		3,058
Sugar City	75 miles NE		1,514
Ucon	49 miles NE		1,108

* Source: Idaho Department of Highway and World Almanac (2020)
1) Does not include student population of approximately 11,000 at Idaho State University
2) Does not include student population of approximately 22,000 at BYU-Idaho
3) Does not include on-campus student population, approximately 500 at College of Eastern Idaho

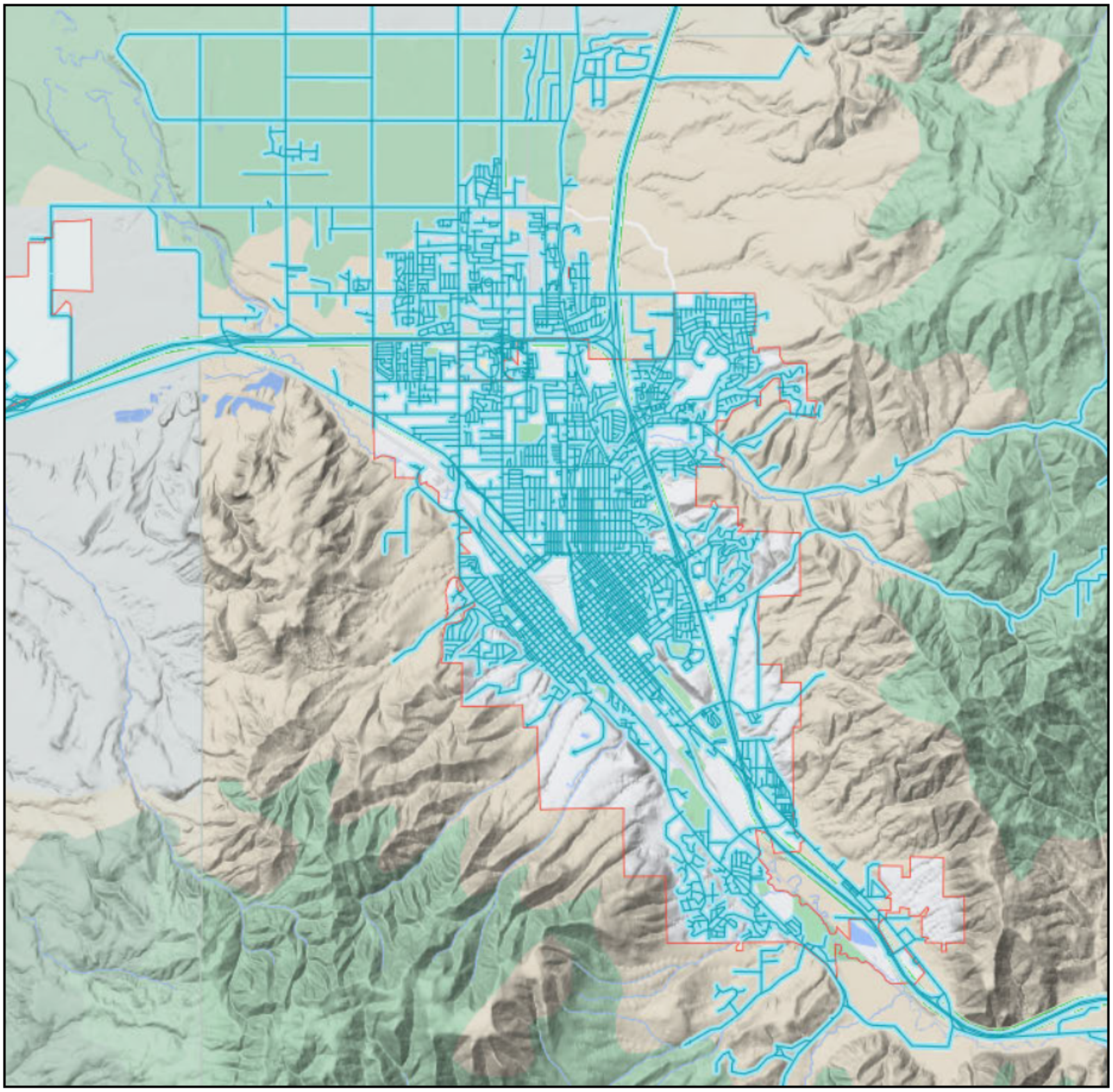


Fig. 2.1-2 Map of Pocatello

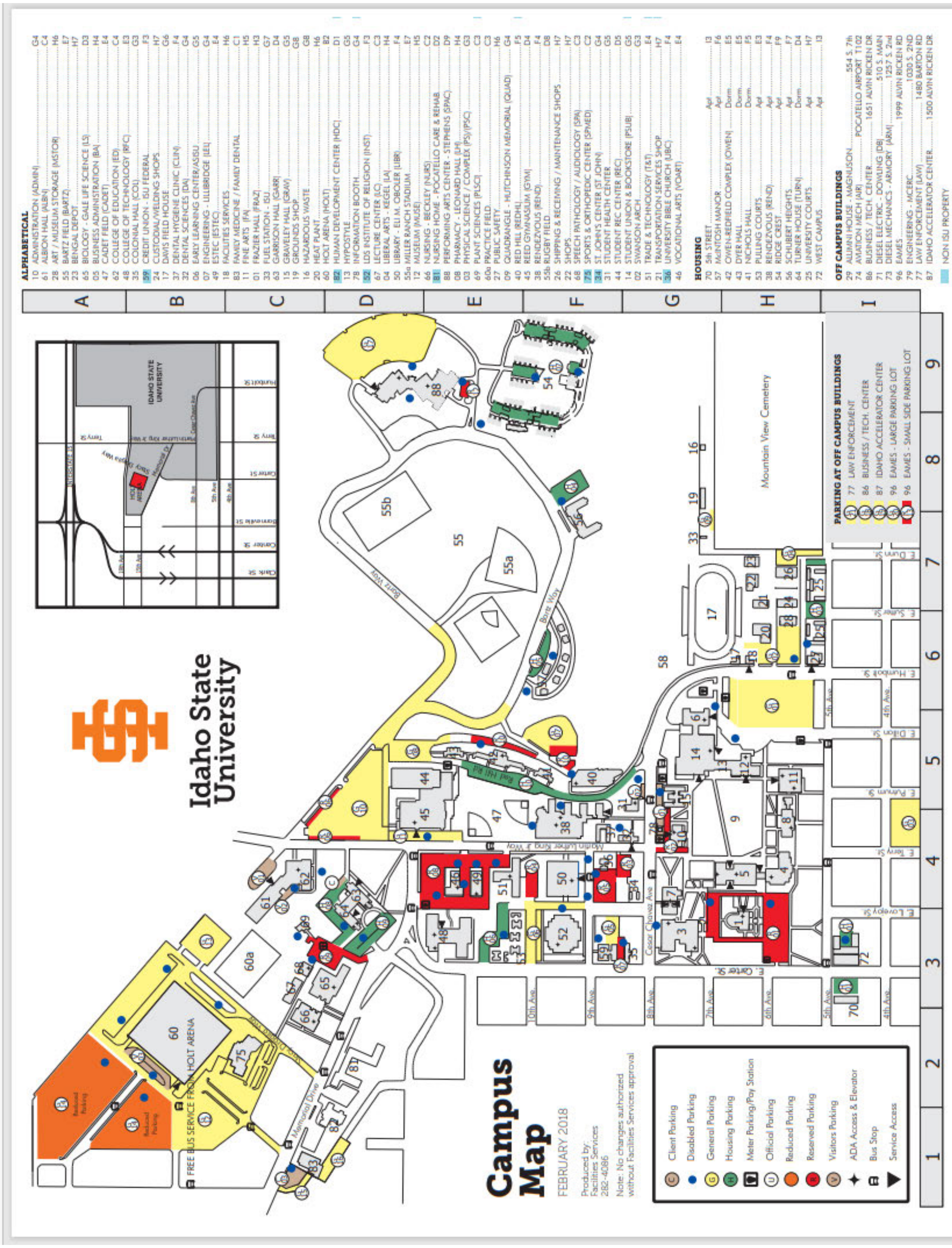


Fig. 2.1-3 Campus of Idaho State University

2.2 Meteorology

2.2.1 Introduction

The climate of Pocatello is semiarid and may be described as a middle latitude steppe climate, where temperatures are relatively high in summer but fall below freezing in the winter and precipitation is sparse and characterized by great variability. Pocatello lies in a valley two to five miles wide (Figure 2.2-1), with mountains on either side rising three thousand to four thousand feet above the valley floor. About three miles north of the city center, the valley broadens and merges into the gently rolling topography of the of the Snake River Plain. The official Weather Bureau station is located at the airport on the southern margin of the Snake River Plain approximately seven miles northwest of town. Because Pocatello is situated in the Portneuf Valley between two spurs of the Bannock Range, weather data obtained at the Weather Bureau may not reflect the actual weather conditions in the city itself.

Records from a second-order weather station (one at which only temperature and rainfall data were recorded) which existed for about 8 years in the northeastern section of the city indicated that temperatures in Pocatello are two to three degrees higher than those at the official weather station except in winter when they may be more than 10 degrees higher. Except in summer, storms bring rain or snow to the whole area, but the location of Pocatello near to the mountains causes some variation in the amount and distribution of precipitation received at the Weather Bureau station. In summer, precipitation is usually produced by thunderstorms which are extremely localized. Average annual precipitation at the weather Bureau Station is 13 inches.

2.2.2 Temperature

The average annual temperature of Pocatello is 47.2°F and monthly temperatures average from 24°F in January to 71.4°F in July. Temperatures in July may reach 100°F or more for short periods, while January temperatures of -30°F have been recorded. The daily range of temperature is also high, reaching 30°F to 40°F in July. This high diurnal range in summer is due to the fact that while daytime temperatures may reach 90°F or more, excessive re-radiation at night under cloudless skies causes the temperature to drop considerably.

In winter, moist maritime air masses from the Pacific bring periods of mild weather when winds blow from the southwest, but otherwise temperatures may stay below freezing for several days and the daily minima fall below zero for approximately 4 months. Frost depth to two or three feet is common during the winter season.

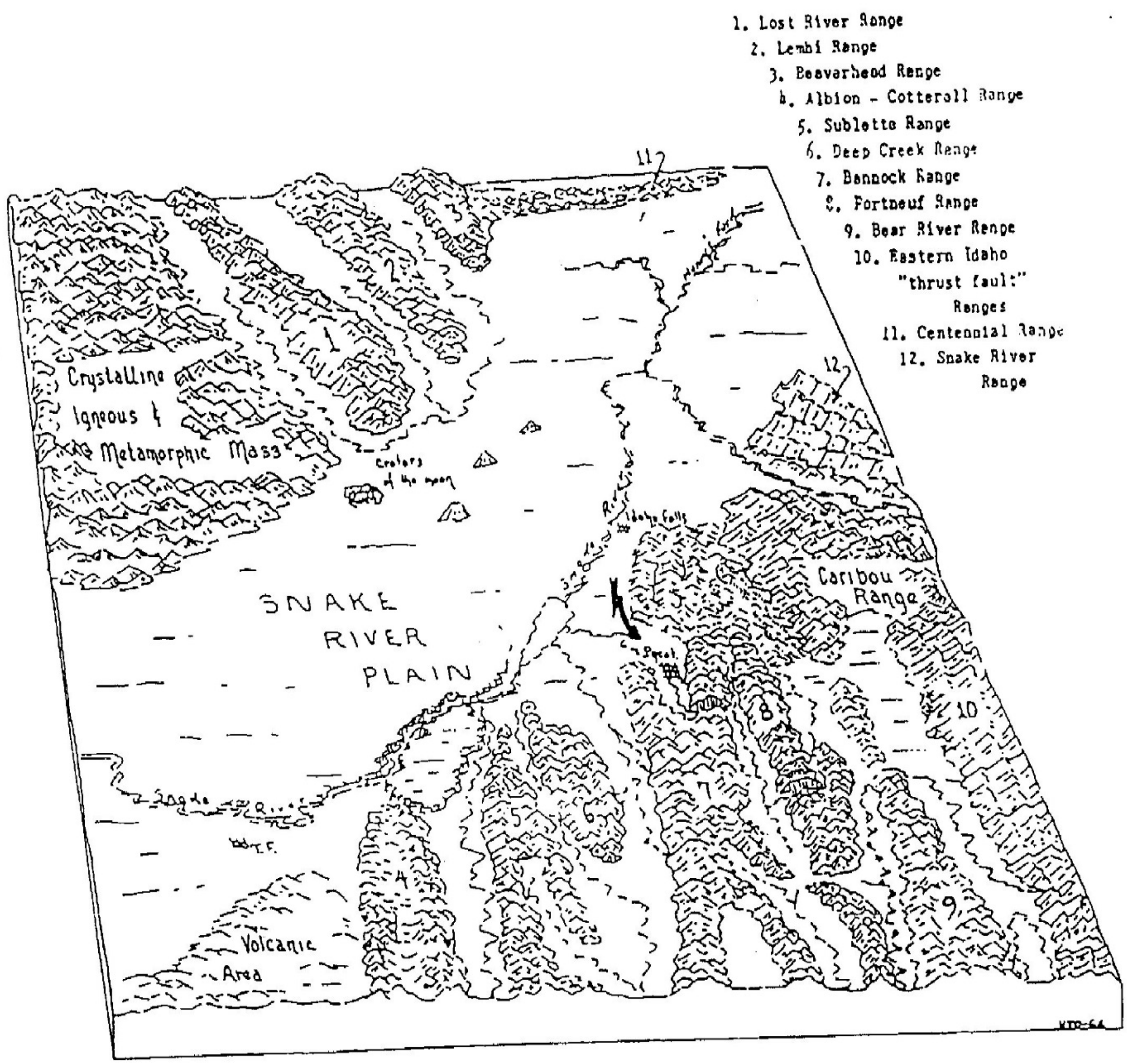


Fig. 2.2-1 Physiographic Location

In the spring, temperatures gradually rise but freezing temperatures at night are general through most of April. The average first occurrence of 32 degrees Fahrenheit in the fall is September 20 and the average last occurrence in the spring is May 20. The first cold wave may appear during late November but usually not until late December.

2.2.3 Precipitation

Average annual precipitation for Pocatello as recorded at the U.S. Weather Bureau is 12.32 inches but the precipitation varies greatly in both amount and distribution. In the thirty year period between 1964 and 1993, seven years had less than 10 inches of precipitation, the minimum for any year being 5.34 inches in 1966. Six out of thirty years had precipitation over 14 inches, the maximum for one year being 20.33 inches in 1983.

During the winter, precipitation falling as snow sometimes accumulates to a depth of a foot or more, but snow depth on the valley floor (elevation about 4470 feet) reaches only 5 or 6 inches at most, and the snow usually melts in a week’s time, during a thaw.

The mountains surrounding Pocatello receive more moisture than the town itself and are covered with snow at higher elevations from November to May. Dry land wheat is raised on the hillsides near Pocatello where slopes are not too steep to prevent cultivation but agriculture can be carried on in or near the valley only by irrigation.

Precipitation is distributed unevenly through the year with 82% of the annual precipitation falling during the period of October through June and only 18% during the months of July, August and September. The fact that precipitation minima occur during the season of high temperatures when evaporation rates are also high, results in dry summers. During the summer, precipitation usually falls as local showers accompanied by light to-moderate thunderstorms, and occasionally by hail. Damage by cloudbursts is rare in Pocatello because contour furrowing done by the Civilian Conservation Corps in the 1930's on the hill slopes above the city has prevented excessive runoff. Cloudy and unsettled weather prevails throughout the winter and spring with measurable amounts of precipitation on about one-third of the days.

2.2.4 Wind

Pocatello lies in the belt of westerlies, consequently, the prevailing wind direction is from the southwest. Average wind speeds are 9 to 10 miles per hour but on rare occasions, during heavy winds, gusts up to 68 miles per hour have been recorded. Winds of 20 to 30 miles per hour may blow continuously for several days in the spring.

Windstorms associated with cyclonic systems and cold fronts do some damage to trees each year, often causing temporary disruption of power and communication facilities; only minor damage is done to structures. Storms of this type may occur from October to June, while during the remaining three months of the year, high winds are almost invariably associated with thunderstorms.

No permanent official wind-recording instruments are located in Pocatello or in the valley adjacent to it. Movement of smoke from the stack of chemical plant located on the northern outskirts of the city indicates that, at times, air may be moving up-valley on the western side of the valley and down-valley during the early morning hours along the eastern side. Winds also appear to blow down-valley during the early morning hours along the eastern side of the valley. These mountain and valley winds are light; velocities are probably not more than 1-3 miles per hour.

2.2.5 Other Climate Factors

- Relative Humidity -Relative humidity is higher in winter and spring and during these seasons is near 70 to 80 per cent. During the summer months, relative humidity is never greater than 50 per cent during the day, not exceeding 65 per cent at night.

- Fog -The Weather Bureau records an average of 10 days of heavy fog per year for Pocatello, and nearly half of these (four) come in January. The valley and mountain winds tend to prevent the formation of fog in the valley and so occurrences of fog in the city itself averages considerably less than 10 days per year.

- Sunshine -Sunny skies prevail over Pocatello during the summer when roughly 50 per cent of the days in July, August, and September are cloudless, and possible sunshine rises to 80 per cent. Cloudiness increases in winter and spring. December and January are the cloudiest months, when the sky is more than eight-tenths covered about two-thirds of the time.

2.2.6 Adverse Weather Effects on the ISU AGN-201 Reactor

2.2.6.1 Flood

In the very unlikely event of a flood, no special precautions are necessary other than those normally taken in the event of a flood at an industrial site. The reactor will be secured and not operated at this time. The radiological hazard problems are not severe as the reactor is built to withstand a minor flood (one foot of water). In the event of a major flood where the reactor might be overturned or carried away, there is again no serious problem since the self-contained reactor has been designed to withstand such an emergency.

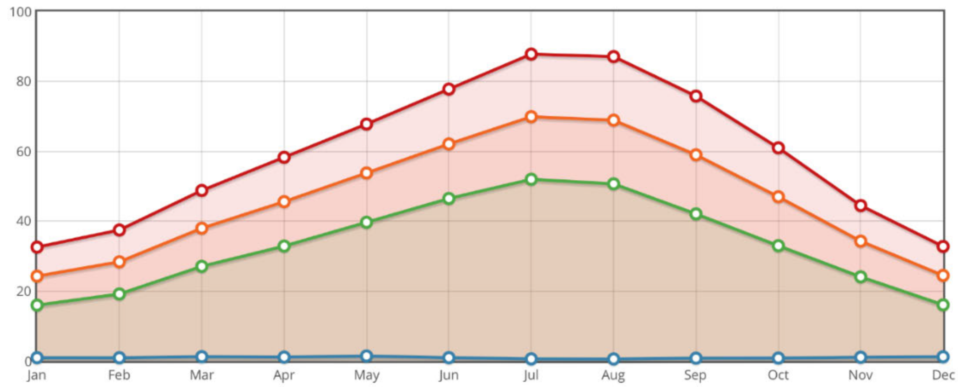
2.2.6.2 Storm

It is highly unlikely that a storm could damage the AGN-201 reactor; however, in the event of a severe storm, the reactor will be shut down and secured. It should also be noted that there is no recorded history of tornadoes in Pocatello, Idaho.

Local climatological and meteorological data from the Pocatello Weather Bureau station is given in Tables 2.2-1 through 2.2-5.

Table 2.2-1 Local and Climatological Data

POCATELLO REGIONAL AIRPORT, ID US



MONTH	PRECIP (IN)	MIN TMP (°F)	AVG TMP (°F)	MAX TMP (°F)
01	0.99	16.0	24.3	32.6
02	0.97	19.2	28.4	37.5
03	1.26	27.1	38.0	48.8
04	1.15	32.9	45.6	58.3
05	1.46	39.7	53.8	67.8
06	0.99	46.5	62.1	77.8
07	0.64	52.0	69.9	87.8
08	0.61	50.7	68.9	87.1
09	0.84	42.1	59.0	75.8
10	0.87	33.0	47.0	61.0
11	1.11	24.1	34.3	44.5
12	1.24	16.1	24.5	32.8

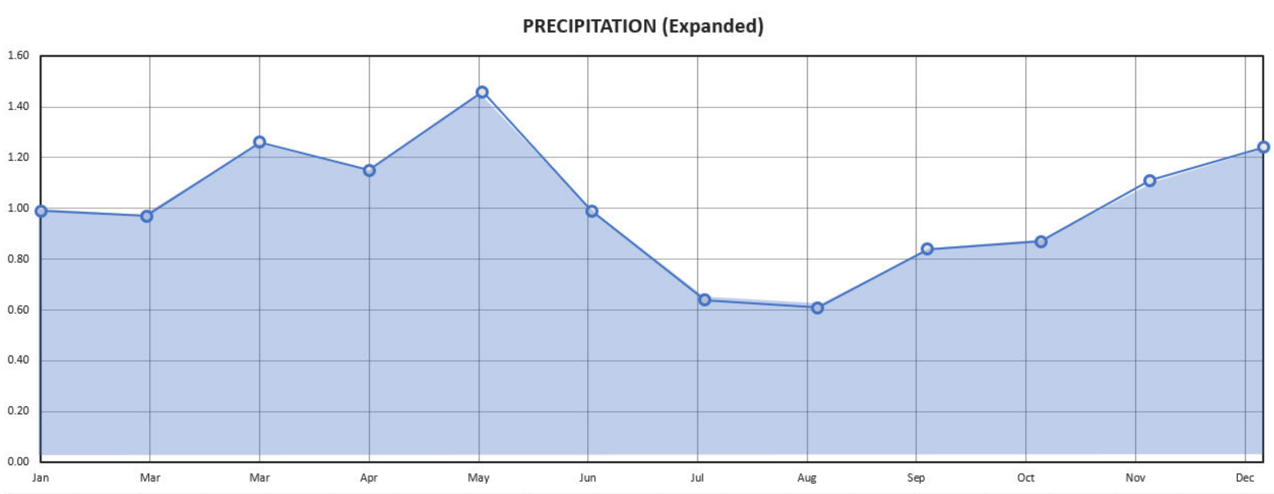


Table 2.2-2 Temperature, Cooling and Heating Days

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data, and Information Service
Current Location: Elev: 4452 ft. Lat: 42.9202° N Lon: -112.5711° W
Station: **POCATELLO REGIONAL AIRPORT, ID US USW00024156**

Summary of Monthly Normals 1981-2010
Generated on 07/26/2021

National Centers for Environmental Information
151 Patton Avenue
Asheville, North Carolina 28801

Temperature (°F)

Mean							Cooling Degree Days Base (above)						Heating Degree Days Base (above)				Mean Number of Days					
Month	Daily Max	Daily Min	Mean	Long Term Max Std Dev	Long Term Min Std Dev	Long Term Avg Std Dev	55	57	60	65	70	72	55	57	60	65	Max >= 100	Max >= 90	Max >= 50	Max <= 32	Min <= 32	Min <= 0
01	32.6	16.0	24.3	5.2	6.5	5.7	0	0	0	0	0	0	952	1014	1107	1262	0.0	0.0	0.7	14.3	29.2	3.6
02	37.5	19.2	28.4	5.9	5.9	5.8	0	0	0	0	0	0	746	802	886	1026	0.0	0.0	3.0	7.9	25.1	1.9
03	48.8	27.1	38.0	4.4	2.9	3.4	-7777	-7777	-7777	0	0	0	529	591	684	839	0.0	0.0	13.6	1.1	23.0	0.3
04	58.3	32.9	45.6	3.9	2.3	2.9	11	6	2	-7777	0	0	293	348	434	582	0.0	0.0	23.2	0.0	12.9	0.0
05	67.8	39.7	53.8	3.9	2.0	2.7	80	55	28	5	-7777	-7777	119	156	222	353	0.0	0.2	29.8	0.0	3.7	0.0
06	77.8	46.5	62.1	3.9	2.0	2.6	234	186	122	46	10	4	20	32	58	131	-7777	2.5	29.9	0.0	0.2	0.0
07	87.8	52.0	69.9	4.3	2.3	3.1	462	401	311	172	61	34	-7777	1	5	20	0.8	13.3	31.0	0.0	0.0	0.0
08	87.1	50.7	68.9	2.6	2.2	2.1	432	371	282	147	47	24	1	2	6	26	0.2	11.9	31.0	0.0	0.1	0.0
09	75.8	42.1	59.0	3.8	2.4	2.6	165	124	74	22	3	1	46	66	106	203	0.0	1.8	29.5	0.0	2.9	0.0
10	61.0	33.0	47.0	3.9	2.3	2.6	18	9	2	-7777	0	0	266	319	405	558	0.0	0.0	25.9	0.1	13.6	0.0
11	44.5	24.1	34.3	5.1	3.9	4.1	-7777	-7777	-7777	0	0	0	621	681	771	921	0.0	0.0	9.3	3.7	24.3	0.6
12	32.8	16.1	24.5	4.5	5.3	4.8	-7777	-7777	0	0	0	0	947	1009	1102	1257	0.0	0.0	0.9	13.4	28.6	3.1
Summary	59.3	33.3	46.3	4.3	3.3	3.5	1402	1152	821	392	121	63	4540	5021	5786	7178	1.0	29.7	227.8	40.5	163.6	9.5

Temperature (°F)

Mean							Cooling Degree Days Base (above)						Heating Degree Days Base (above)				Mean Number of Days					
Month	Daily Max	Daily Min	Mean	Long Term Max Std Dev	Long Term Min Std Dev	Long Term Avg Std Dev	55	57	60	65	70	72	55	57	60	65	Max >= 100	Max >= 90	Max >= 50	Max <= 32	Min <= 32	Min <= 0
01	35.6	19.1	27.4	3.8	4.9	4.2	0	0	0	0	0	0	857	919	1012	1167	0.0	0.0	1.2	9.5	28.9	1.2
02	40.2	21.1	30.6	4.7	5.3	4.8	0	0	0	0	0	0	682	738	822	962	0.0	0.0	3.8	4.3	24.5	0.7
03	49.8	28.5	39.1	4.5	3.3	3.7	1	-7777	-7777	-7777	0	0	493	554	646	801	0.0	0.0	15.6	0.7	20.5	0.0
04	58.6	34.3	46.5	3.7	2.6	3.1	21	14	7	2	-7777	-7777	278	330	413	558	0.0	0.0	23.0	0.0	12.1	0.0
05	68.1	41.7	54.9	4.4	2.4	3.3	110	83	49	15	3	1	113	148	207	328	0.0	0.4	29.6	0.0	2.9	0.0
06	78.3	48.7	63.5	4.4	2.0	3.1	275	227	161	78	26	14	20	31	56	123	0.2	3.7	29.9	0.0	0.2	0.0
07	88.3	56.2	72.3	4.5	2.9	3.6	535	474	385	243	119	81	1	2	5	18	1.5	13.7	31.0	0.0	0.0	0.0
08	87.3	55.1	71.2	3.0	2.9	2.9	503	443	354	216	103	67	1	3	7	24	0.7	13.1	31.0	0.0	0.0	0.0
09	76.5	46.8	61.6	3.6	2.6	2.9	234	189	129	57	18	10	35	49	80	158	0.0	2.4	29.6	0.0	0.8	0.0
10	61.6	37.0	49.3	4.5	2.6	3.4	49	34	17	4	-7777	-7777	225	272	349	491	0.0	0.0	26.1	0.0	8.7	0.0
11	45.4	26.8	36.1	4.9	4.2	4.4	2	1	-7777	0	0	0	569	628	717	867	0.0	0.0	9.9	3.5	21.1	0.2
12	34.9	18.5	26.7	3.4	5.1	4.0	-7777	0	0	0	0	0	877	939	1032	1187	0.0	0.0	1.2	11.5	28.8	1.5
Summary	60.4	36.1	48.3	4.1	3.4	3.6	1730	1465	1102	615	269	173	4151	4613	5346	6684	2.4	33.3	231.9	29.5	148.5	3.6

Temperature (°F)

Mean							Cooling Degree Days Base (above)						Heating Degree Days Base (above)				Mean Number of Days					
Month	Daily Max	Daily Min	Mean	Long Term Max Std Dev	Long Term Min Std Dev	Long Term Avg Std Dev	55	57	60	65	70	72	55	57	60	65	Max >= 100	Max >= 90	Max >= 50	Max <= 32	Min <= 32	Min <= 0
01	32.6	12.9	22.8	4.9	7.0	5.7	0	0	0	0	0	0	1000	1062	1155	1310	0.0	0.0	0.6	13.8	30.3	5.5
02	37.4	15.5	26.4	5.7	6.3	5.8	0	0	0	0	0	0	799	855	939	1079	0.0	0.0	2.6	8.2	26.7	3.3
03	48.9	24.3	36.6	4.7	3.4	3.9	-7777	-7777	-7777	0	0	0	571	633	726	880	0.0	0.0	14.2	1.2	26.0	0.4
04	58.8	30.1	44.5	3.9	2.5	2.9	8	4	1	-7777	0	0	325	381	468	616	0.0	0.0	23.5	0.0	18.2	0.0
05	67.4	37.7	52.6	3.8	2.0	2.7	61	39	17	2	-7777	-7777	137	177	248	388	0.0	0.1	29.7	0.0	6.3	0.0
06	75.7	44.7	60.2	3.8	1.8	2.6	184	139	83	26	5	3	28	43	77	170	0.0	1.5	30.0	0.0	0.7	0.0
07	85.5	49.6	67.5	4.0	2.5	3.0	389	329	241	111	29	14	1	2	7	32	0.2	9.0	31.0	0.0	0.0	0.0
08	85.2	48.0	66.6	2.9	2.3	2.2	362	302	215	92	19	8	2	5	11	42	0.1	8.1	31.0	0.0	0.4	0.0
09	75.0	39.4	57.2	4.0	2.4	2.7	124	89	46	10	1	-7777	58	83	130	244	0.0	1.1	29.5	0.0	4.4	0.0
10	60.9	30.4	45.7	4.0	2.3	2.7	11	4	1	-7777	0	0	300	356	446	600	0.0	0.0	26.1	0.1	17.8	0.0
11	44.2	21.5	32.9	5.1	4.0	4.1	-7777	-7777	0	0	0	0	665	725	814	964	0.0	0.0	8.9	3.6	26.6	0.8
12	32.5	12.7	22.6	4.2	5.6	4.7	0	0	0	0	0	0	1004	1066	1159	1314	0.0	0.0	0.8	14.1	29.8	4.3
Summary	58.7	30.6	44.6	4.3	3.5	3.6	1139	906	604	241	54	25	4890	5388	6180	7639	0.3	19.8	227.9	41.0	187.2	14.3

Temperature (°F)																							
Mean							Cooling Degree Days Base (above)						Heating Degree Days Base (above)				Mean Number of Days						
Month	Daily Max	Daily Min	Mean	Long Term Max Std Dev	Long Term Min Std Dev	Long Term Avg Std Dev	55	57	60	65	70	72	55	57	60	65	Max >= 100	Max >= 90	Max >= 50	Max <= 32	Min <= 32	Min <= 0
01	33.7	17.8	25.8				0	0	0	0	0	0	906	968	1061	1217	0.0	0.0	0.0	11.4	30.2	1.4
02	39.7	20.9	30.3				-7777	0	0	0	0	0	691	747	831	971	0.0	0.0	2.5	4.1	25.7	0.0
03	51.2	28.9	40.1				1	-7777	0	0	0	0	464	526	619	773	0.0	0.0	16.8	0.7	20.5	0.0
04	59.9	36.1	48.0				20	12	5	1	0	0	230	282	365	511	0.0	0.0	24.9	0.0	7.9	0.0
05	69.6	44.2	56.9				114	79	43	11	1	-7777	55	82	140	263	0.0	0.2	30.6	0.0	1.1	0.0
06	79.0	51.1	65.1				308	253	179	84	25	13	6	11	28	82	0.2	2.6	30.0	0.0	0.0	0.0
07	88.5	58.3	73.4				570	508	415	263	127	82	-7777	-7777	-7777	3	0.0	14.0	31.0	0.0	0.0	0.0
08	87.9	56.5	72.2				533	471	380	233	108	68	-7777	-7777	2	10	0.0	12.7	31.0	0.0	0.0	0.0
09	78.0	47.9	63.0				259	210	143	58	15	8	21	32	55	119	0.0	1.9	30.0	0.0	0.0	0.0
10	62.8	37.6	50.2				38	25	11	1	0	0	187	236	315	460	0.0	0.0	27.8	0.0	6.8	0.0
11	45.5	27.2	36.4				1	-7777	0	0	0	0	560	619	709	860	0.0	0.0	10.2	2.4	21.5	0.0
12	35.0	19.0	27.0				0	0	0	0	0	0	867	929	1022	1178	0.0	0.0	1.0	11.9	28.1	1.2
Summary	60.9	37.1	49.0	0.0	0.0	0.0	1844	1558	1176	651	276	171	3987	4432	5147	6447	0.2	31.4	235.8	30.5	141.8	2.6

-7777: a non-zero value that would round to zero

Empty or blank cells indicate data is missing or insufficient occurrences to compute value

Table 2.2-3 Precipitation

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data, and Information Service

Summary of Monthly Normals
1981-2010
Generated on 07/26/2021

National Centers for Environmental Information
151 Patton Avenue
Asheville, North Carolina 28801

Current Location: Elev: 4452 ft. Lat: 42.9202° N Lon: -112.5711° W
Station: **POCATELLO REGIONAL AIRPORT, ID US USW00024156**

Precipitation (in.)								
	Totals	Mean Number of Days				Precipitation Probabilities Probability that precipitation will be equal to or less than the indicated amount		
	Means	Daily Precipitation				Monthly Precipitation vs. Probability Levels		
Month	Mean	>= 0.01	>= 0.10	>= 0.50	>= 1.00	0.25	0.50	0.75
01	0.99	11.4	3.6	0.1	0.0	0.56	1.00	1.40
02	0.97	9.8	3.1	0.2	0.0	0.54	0.78	1.37
03	1.26	9.7	3.9	0.4	0.0	0.63	0.88	1.69
04	1.15	9.0	4.1	0.2	0.0	0.69	1.02	1.43
05	1.46	9.5	4.4	0.6	0.1	0.85	1.32	1.99
06	0.99	6.6	2.6	0.5	-7777	0.37	0.73	1.19
07	0.64	4.4	1.8	0.3	-7777	0.14	0.37	1.00
08	0.61	4.5	1.9	0.4	0.0	0.16	0.44	0.72
09	0.84	4.9	2.5	0.4	0.1	0.27	0.58	1.32
10	0.87	6.2	3.1	0.2	0.0	0.39	0.94	1.31
11	1.11	9.3	3.6	0.3	0.0	0.66	0.98	1.38
12	1.24	11.3	3.8	0.4	0.0	0.57	1.17	1.96
Summary	12.13	96.6	38.4	4.0	0.2	5.83	10.21	16.76

Precipitation (in.)								
	Totals	Mean Number of Days				Precipitation Probabilities Probability that precipitation will be equal to or less than the indicated amount		
	Means	Daily Precipitation				Monthly Precipitation vs. Probability Levels		
Month	Mean	>= 0.01	>= 0.10	>= 0.50	>= 1.00	0.25	0.50	0.75
01	1.09	13.9	3.8	0.1	0.0	0.63	1.10	1.48
02	1.07	9.5	3.7	0.3	0.1	0.70	1.01	1.39
03	1.38	9.6	3.7	0.3	0.0	0.78	1.00	1.80
04	1.35	9.7	4.0	0.6	0.0	0.72	1.25	1.74
05	1.48	9.6	4.6	0.6	0.1	0.90	1.28	1.89
06	1.18	8.2	3.8	0.6	0.2	0.41	0.95	1.46
07	0.70	4.2	1.7	0.3	0.1	0.22	0.42	1.05
08	0.80	5.1	2.6	0.3	0.1	0.28	0.57	0.90
09	1.05	4.9	2.4	0.6	0.1	0.34	0.77	1.56
10	0.98	7.0	3.6	0.4	0.0	0.45	1.18	1.36
11	1.18	8.9	3.1	0.3	0.0	0.74	1.05	1.67
12	1.33	13.7	4.6	0.2	0.0	0.70	1.17	1.74
Summary	13.59	104.3	41.6	4.6	0.7	6.87	11.75	18.04

Precipitation (in.)								
	Totals	Mean Number of Days				Precipitation Probabilities Probability that precipitation will be equal to or less than the indicated amount		
	Means	Daily Precipitation				Monthly Precipitation vs. Probability Levels		
Month	Mean	>= 0.01	>= 0.10	>= 0.50	>= 1.00	0.25	0.50	0.75
01	0.83	9.5	2.8	0.1	0.0	0.50	0.72	1.16
02	0.80	7.7	2.8	-7777	0.0	0.48	0.70	1.10
03	1.07	8.2	3.8	0.3	0.0	0.53	0.73	1.62
04	1.05	7.7	3.2	0.3	-7777	0.63	0.95	1.37
05	1.55	9.3	4.6	0.6	0.1	0.78	1.25	2.24
06	1.08	6.6	2.8	0.6	0.1	0.32	0.85	1.24
07	0.65	3.8	1.4	0.2	-7777	0.01	0.35	1.08
08	0.66	4.4	2.0	0.2	0.1	0.05	0.42	1.02
09	0.76	4.9	2.4	0.3	0.0	0.26	0.53	1.08
10	0.81	6.2	3.1	0.2	0.0	0.47	0.73	1.06
11	0.93	7.7	3.1	0.2	0.0	0.44	0.98	1.18
12	1.08	10.1	3.1	0.2	0.0	0.47	1.03	1.60
Summary	11.27	86.1	35.1	3.2	0.3	4.94	9.24	15.75

Precipitation (in.)								
	Totals	Mean Number of Days				Precipitation Probabilities Probability that precipitation will be equal to or less than the indicated amount		
	Means	Daily Precipitation				Monthly Precipitation vs. Probability Levels		
Month	Mean	>= 0.01	>= 0.10	>= 0.50	>= 1.00	0.25	0.50	0.75
01	0.85							
02	1.06							
03	1.26							
04	1.05							
05	1.67							
06	1.16							
07	0.86							
08	0.89							
09	1.02							
10	1.00							
11	1.16							
12	1.27							
Summary	13.25	0.0	0.0	0.0	0.0	0.00	0.00	0.00

-7777: a non-zero value that would round to zero

Empty or blank cells indicate data is missing or insufficient occurrences to compute value

Table 2.2-4 Growing Degrees

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data, and Information Service
Current Location: Elev: 4452 ft. Lat: 42.9202° N Lon: -112.5711° W
Station: **POCATELLO REGIONAL AIRPORT, ID US USW00024156**

Summary of Monthly Normals
1981-2010
Generated on 07/26/2021

National Centers for Environmental Information
151 Patton Avenue
Asheville, North Carolina 28801

Growing Degree Units (Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	3	8	62	193	427	665	927	895	569	243	40	5
45	-7777	1	19	97	284	515	772	741	423	133	14	1
50	-7777	-7777	3	38	165	368	617	586	286	57	3	-7777
55	0	0	-7777	11	80	234	462	432	165	18	-7777	-7777
60	0	0	-7777	2	28	122	311	282	74	2	-7777	0
Growing Degree Units for Corn (Monthly)												
50/86	1	6	51	140	279	420	578	565	387	186	33	1

Growing Degree Units (Accumulated Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	3	11	73	266	693	1358	2285	3180	3749	3992	4032	4037
45	0	1	20	117	401	916	1688	2429	2852	2985	2999	3000
50	0	0	3	41	206	574	1191	1777	2063	2120	2123	2123
55	0	0	0	11	91	325	787	1219	1384	1402	1402	1402
60	0	0	0	2	30	152	463	745	819	821	821	821
Growing Degree Units for Corn (Monthly Accumulated)												
50/86	1	7	58	198	477	897	1475	2040	2427	2613	2646	2647

Growing Degree Units (Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	1	4	48	167	391	606	854	824	517	207	27	1
45	-7777	1	13	80	250	457	699	670	371	105	8	-7777
50	-7777	-7777	2	30	138	312	544	515	237	41	1	0
55	0	0	-7777	8	61	184	389	362	124	11	-7777	0
60	0	0	-7777	1	17	83	241	215	46	1	0	0
Growing Degree Units for Corn (Monthly)												
50/86	1	6	54	147	272	388	543	534	373	184	30	1

Growing Degree Units (Accumulated Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	1	5	53	220	611	1217	2071	2895	3412	3619	3646	3647
45	0	1	14	94	344	801	1500	2170	2541	2646	2654	2654
50	0	0	2	32	170	482	1026	1541	1778	1819	1820	1820
55	0	0	0	8	69	253	642	1004	1128	1139	1139	1139
60	0	0	0	1	18	101	342	557	603	604	604	604
Growing Degree Units for Corn (Monthly Accumulated)												
50/86	1	7	61	208	480	868	1411	1945	2318	2502	2532	2533

Growing Degree Units (Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	-7777	7	85	250	524	752	1034	998	688	324	48	6
45	0	1	33	140	372	602	880	843	540	196	17	1
50	0	-7777	9	61	230	452	725	688	395	101	4	-7777
55	0	-7777	1	20	114	308	570	533	259	38	1	0
60	0	0	0	5	43	179	415	380	143	11	0	0
Growing Degree Units for Corn (Monthly)												
50/86	-7777	4	62	157	311	466	671	641	440	207	29	1

Growing Degree Units (Accumulated Monthly)												
Base	**Jan**	**Feb**	**Mar**	**Apr**	**May**	**Jun**	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
40	0	7	92	342	866	1618	2652	3650	4338	4662	4710	4716
45	0	1	34	174	546	1148	2028	2871	3411	3607	3624	3625
50	0	0	9	70	300	752	1477	2165	2560	2661	2665	2665
55	0	0	1	21	135	443	1013	1546	1805	1843	1844	1844
60	0	0	0	5	48	227	642	1022	1165	1176	1176	1176
Growing Degree Units for Corn (Monthly Accumulated)												
50/86	0	4	66	223	534	1000	1671	2312	2752	2959	2988	2989

Note: For corn, temperatures below 50 are set to 50, and temperatures above 86 are set to 86.

-7777: a non-zero value that would round to zero.

Empty or blank cells indicate data is missing or insufficient occurrences to compute value.

Table 2.2-5 Snowfall

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data, and Information Service
Current Location: Elev: 4452 ft. Lat: 42.9202° N Lon: -112.5711° W
Station: **POCATELLO REGIONAL AIRPORT, ID US USW00024156**

Summary of Monthly Normals 1981-2010
Generated on 07/26/2021

National Centers for Environmental Information
151 Patton Avenue
Asheville, North Carolina 28801

Snow (in.)												
	Totals	**Mean Number of Days**									**Snow Probabilities** Probability that snow will be equal to or less than the indicated amount		
	Means	**Snowfall >= Thresholds**					**Snow Depth >= Thresholds**				**Monthly Snow vs. Probability Levels** Values derived from the incomplete gamma distribution.		
Month	**Snowfall Mean**	**0.01**	**1.0**	**3.0**	**5.00**	**10.00**	**1**	**3**	**5**	**10**	**.25**	**.50**	**.75**
01	8.8	9.9	3.0	0.6	0.2	0.0	16.2	10.1	6.7	1.3	5.7	7.1	10.5
02	7.1	7.2	2.2	0.5	0.2	-7777	11.3	6.9	3.5	1.1	3.0	4.7	10.1
03	5.4	5.1	1.9	0.4	0.1	0.0	4.5	1.9	1.1	0.1	1.8	4.4	6.3
04	3.2	2.9	1.0	0.3	0.1	0.0	0.9	0.1	0.1	0.0	0.7	1.9	5.0
05	0.8	0.7	0.3	0.1	-7777	0.0	0.2	-7777	0.0	0.0	0.0	0.0	1.0
06	-7777	-7777	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
09	0.1	0.1	-7777	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
10	1.6	1.0	0.5	0.2	0.1	0.0	0.7	0.2	0.1	0.0	0.0	0.2	1.5
11	5.7	5.3	2.0	0.5	0.1	0.0	5.0	2.0	1.1	0.1	1.9	4.3	7.5
12	10.7	10.0	3.2	0.8	0.3	0.1	14.5	7.0	4.3	1.0	5.4	8.5	17.1
Summary	43.4	42.2	14.1	3.4	1.1	0.1	53.3	28.2	16.9	3.6	18.5	31.1	59.0

Snow (in.)												
	Totals	**Mean Number of Days**									**Snow Probabilities** Probability that snow will be equal to or less than the indicated amount		
	Means	**Snowfall >= Thresholds**					**Snow Depth >= Thresholds**				**Monthly Snow vs. Probability Levels** Values derived from the incomplete gamma distribution.		
Month	**Snowfall Mean**	**0.01**	**1.0**	**3.0**	**5.00**	**10.00**	**1**	**3**	**5**	**10**	**.25**	**.50**	**.75**
01	14.5	12.5	5.8	1.4	0.5	0.1	22.3	13.8	9.3	4.1	9.1	12.6	20.7
02	11.2	8.0	4.5	1.1	0.4	0.0	18.6	11.5	5.9	2.6	5.8	9.4	12.5
03	8.4	6.0	3.4	0.7	0.3	0.1	6.5	2.9	1.3	0.4	2.9	7.3	10.3
04	5.1	4.1	1.7	0.4	0.1	0.1	1.1	0.2	0.1	0.0	1.4	4.1	8.0
05	0.9	0.8	0.3	0.2	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.6
06	-7777	0.2	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
09	0.1	0.2	0.1	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.0	0.0	0.0
10	2.8	1.7	1.0	0.4	0.1	0.0	1.2	0.5	0.1	0.0	0.0	2.0	5.4
11	7.2	5.8	2.8	0.9	0.2	0.1	6.0	3.1	1.1	0.2	1.8	7.3	9.7
12	16.2	12.3	5.8	1.5	0.5	0.1	22.1	12.4	6.4	1.6	8.1	12.6	22.6
Summary	66.4	51.6	25.5	6.6	2.2	0.5	77.9	44.4	24.2	8.9	29.1	55.3	89.8

Snow (in.)												
	Totals	**Mean Number of Days**									**Snow Probabilities** Probability that snow will be equal to or less than the indicated amount		
	Means	**Snowfall >= Thresholds**					**Snow Depth >= Thresholds**				**Monthly Snow vs. Probability Levels** Values derived from the incomplete gamma distribution.		
Month	**Snowfall Mean**	**0.01**	**1.0**	**3.0**	**5.00**	**10.00**	**1**	**3**	**5**	**10**	**.25**	**.50**	**.75**
01	7.8	5.6	2.8	0.5	0.2	0.0	20.3	13.2	9.0	4.4	4.2	6.3	11.3
02	5.1	3.9	2.2	0.5	0.1	0.0	14.4	9.3	5.5	2.3	2.4	4.4	7.1
03	2.9	2.6	1.5	0.2	0.1	0.0	4.1	1.2	0.4	0.3	0.6	2.0	3.5
04	1.0	0.7	0.4	0.1	-7777	0.0	0.6	0.1	-7777	0.0	0.0	0.0	1.3
05	0.4	0.2	0.1	-7777	0.0	0.0	0.2	0.1	-7777	0.0	0.0	0.0	0.0
06	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
08	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
09	-7777	-7777	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
10	1.3	0.7	0.4	0.2	0.1	0.0	0.8	0.1	-7777	0.0	0.0	0.0	1.5
11	2.9	2.4	1.4	0.3	0.1	0.0	5.0	2.0	1.0	0.2	0.3	1.5	4.2
12	7.6	6.2	3.4	0.6	0.3	0.0	17.4	9.5	5.8	2.2	2.8	5.6	12.2
Summary	29.0	22.3	12.2	2.4	0.9	0.0	62.8	35.5	21.7	9.4	10.3	19.8	41.1

-7777: a non-zero value that would round to zero

Empty or blank cells indicate data is missing or insufficient occurrences to compute value

2.3 Geology and Hydrology

2.3.1 General Physiographic Setting

Pocatello is located on the boundary between the northeastern corner of the Great Basin Section of the Basin and Range Physiographic Province and the southeastern edge of the Snake River Plain Section of the Columbia River Plateau Province (as shown in Figure 2.2-1). The area thus has stratigraphic, structural, petrologic and geomorphic characteristics of both areas. The Basin and Range Province, particularly in the Great Basin Section, is characterized by alternating basins and mountain ranges; the basins commonly are partially graded up onto the mountain sides by Bogota (filled) and pediment (cut) surfaces. The Snake River Plain is an arcuate, flat-surfaces basalt plateau, dissected by the Snake River and to a lesser extent by some of its tributaries on the western end. Both physiographic areas are relatively young geologically. Seismic and volcanic activity during "Recent" time and the freshness of tectonic forms and surficial extrusive rocks show that the area is still in the process of evolving.

2.3.2 Local Geology and Physiography

The city of Pocatello is in the valley bottom of the Portneuf River, a tributary to the Snake River. The Portneuf drains about 1200 square miles of the Bannock and Portneuf mountain ranges east and southeast of Pocatello. The city is bordered on the southwest and northeast sides by hills generally less than 4000 feet high, with the general terrain becoming more mountainous to the southeast. The river valley is flat floored and at the townsite widens abruptly to the northwest out onto the Snake River Plain.

The bedrock floor of the valley is buried under some 200 feet of alluvium; whether the original shape and depth of the valley was primarily structural (down-faulted) or erosional is unknown. A much larger river (Bear River) probably occupied the valley prior to the diversion of that river to the south 30,000 years or so ago, possibly accounting for the size of the valley relative to that of the current river. Regardless of Its original configuration, the valley has been altered by several episodes of cutting and filling since its inception. Broad benches slope 300-400 feet per mile toward the valley from each side; these are remnants of a Pleistocene valley fill. Much of this sequence has been removed down the axis of the valley, and the surfaces are currently being dissected by small tributaries to the Portneuf. A basalt flow about the same age as the diversion of the

Bear River (about 30,000 years) floors part of the valley upstream from the main part of the city.

Idaho State University buildings are located on terraces 40-60 feet above the valley bottom on the northeast side of the valley. They are built on a combination of the valley-fill alluvium and loess (wind-blown silt deposits), both of which are common in the area. The southeastern corner of the campus is bordered by a bedrock hill composed of Cambrian (Brigham) Quartzite; the block has been elevated along a high-angle fault which may extend to the northwest under part of the campus or which may terminate at the end of the quartzite block. If the fault does extend under the alluvium beneath the campus, the fault is apparently entirely pre-alluvium in age, as no displacement in the alluvium along strike with the fault has been noted.

2.3.3 Subsurface Water of Pocatello and the ISU Campus Area

Gravel lenses and beds near the center of the Portneuf Valley supply water for Pocatello. The non-artesian production is from about 30 to 40 feet in depth for a number of these supply wells in the lower part of the valley, at an elevation of about 4400 ft. The campus is on the eastern slope region, with the campus elevation ranging from about 4450 ft to 4700 ft. The Lillibridge Engineering Building entry doors are at an elevation of about 4500 ft.

Those individuals living on the eastern mountainous slope outside of city limits draw water from deep supply wells, with the producing aquifers typically in the range of 4400 ft or lower. About 1978 the University drilled two wells on campus at extreme ends of the large plot housing the Eli Obler Library (about 150 ft from the Lillibridge Engineering Building). These wells were used to supply heat and cooling through a water source heat pump system, drawing water from one well in the winter and discharging it to the other well. The aquifer depth for water extraction or disposal was in the range of 150 to 200 ft. For cooling in the summer, the direction of flow was reversed, from what was the slightly cooler aquifer area into the slightly warmer aquifer. This system served the library well, until it was abandoned about in the year 2005 because the grid screens through which the water was passed to or from the well to the aquifer become blocked with calcification. (The highly efficient water source heat pump system was replaced with the steam district heating system for the winters, and air-to-air heat pumps for the summer.)

There is a producing well about 0.6 miles south of campus, at an elevation of about 4450 ft., that produced water from about an elevation of 4410 ft., 90 feet below the level of campus. There is also a well about 5 miles north of the campus on the western side of I-15 at an elevation of about 4900 ft that was drilled to a depth of 1100 ft in 1980, and produced 108 F artesian water. (The artesian effect is believed to be due to the temperature/density effect.) If that geothermal aquifer extends to campus, it would be at an elevation of 3800 ft, or at a depth of nominally 700 ft below the campus region where the reactor is sited. Considerations of the elevation (or depth) of the water tables, the elevation of the campus, and the character of the subsurface suggest that no ground water problems will be encountered in the University area

2.3.4 Surface Waters

During the early spring of 1962 and 1963, flooding occurred in the low areas of the valley along the river to elevations of about 4460 feet. This flooding was due to unseasonably warm weather and runoff of meltwater into the river over a frozen subsurface in the mountains southeast of Pocatello. A U.S. Corps of Engineers project to straighten and line the banks of the river with concrete has eliminated flooding; even without this correction there would be little chance of flooding in the future to anywhere near the height of the campus. Local drainage is generally away from the Lillibridge Engineering Laboratory so that no surface drainage problems associated with surface runoff are likely.

2.4 Seismology

Contributions made:
By James E. Zollweg, Northwest Geophysics, Aug 16, 1995,
by Dr. James Mahar, Dept. of Civil and Environmental Engineering, ISU, July 1, 2021,
by Dr. Mustafa Mashal, Dept. of Civil and Environmental Engineering, ISU, June 28 2021

2.4.1 Introduction

Seismic safety is an important engineering consideration in southeastern Idaho because of the proximity of active, late Quaternary/Holocene faults. Careful analysis of potential maximum grown motions caused by earthquakes is required in engineering design and reactor safety. This analysis relies heavily on current expert opinions regarding seismic safety issues.

Pocatello lies along the boundary of two major geologic provinces. The Eastern Snake River Plain (ESRP) is the generally flat plain north of the city. The highland area around the city lies within the Basin and Range province. The Basin and Range province is a region of generally moderate to high seismicity as there are numerous late Cenozoic faults, one of which (Wasatch Fault) is within 30 miles of Pocatello. The Idaho Geologic Survey (IGS 2016) has identified two seismic zones located east and north of the Pocatello area: the north-south Intermountain Seismic Belt and the east-west Central Idaho (Centennial) Seismic Zone (see Figure 2.4-1). Both zones contain active faults.

2.4.2 Methodology

A three-step methodology was used in the assessment of seismic risk at Pocatello. The first of these is consideration of the distribution of historical seismicity. The second is consideration of the proximity of potentially active surface faults. The third element, utilized if the first two steps do not indicate a larger magnitude event controls seismic risk, is consideration of a "floating" earthquake that can occur essentially anywhere within the Pocatello region.

2.4.3 Historical Earthquakes and Earthquake Swarms

<u>Intermountain Seismic Belt:</u>
The Intermountain Seismic Belt is located along the border between Idaho and Wyoming and extends southward into the northern Utah and northward into western Montana (see Figure 2.4-1). The Idaho Geological Survey (2016) reports that the region has experienced numerous earthquakes a few of which had magnitudes of 6.0 and up to 7.5 since 1884. Two earthquakes of note are the Bear Lake/Paris and Hansel Valley

earthquakes south of the campus. The Bear Lake Valley/Paris earthquake (magnitude 6.0) in 1884 caused damage and shaking in the area.

In 1934, the Hansel Valley earthquake (magnitude 6.6) located approximately 20 miles south of the Idaho border produced a 1.6 ft maximum ground surface offset in the 7.1 mile long fault (Hecker 1993). The quake caused structural damage in both southern Idaho and northern Utah. Associate Press (AP) newspaper accounts describe the following 12 March 1934 damage in Pocatello:

- Cracked chimneys in Emerson, Lincoln and Jefferson elementary schools
- Cracked walls in several schools and other buildings
- Loosened beams in the Reed Hall gymnasium balcony
- Large cracks in the administration, auditorium and library on the ISU campus
- And broken residential windows

The most severe recorded earthquake in the Intermountain Belt occurred in 1959 with an epicenter located approximately 6.5 miles northwest of West Yellowstone, Montana. The Hebgen Lake of Yellowstone earthquake registered 7.3 on the Richter scale and caused a massive landslide in the Madison River canyon. Several aftershocks ranging in magnitude 5.8 to 6.3 were recorded after the main earthquake event. There are no reports of the earthquake damage in the Pocatello area.

<u>Central Idaho (Centennial) Seismic Zone:</u>

The 1983 Borah Peak earthquake is the largest recorded seismic event in the Central Idaho (Centennial) Seismic Zone. The earthquake registered 7.3 on the Richter scale and occurred along the Lost River Fault. The quake caused severe damage in towns such as Mackay along the Lost River and in Challis, Idaho close to the epicenter. Even though the earthquake was felt in Pocatello, there are no reports of damage in the area.

Earthquake motions occur in two basic distributions: individual events with aftershocks and swarms. Individual events tend to be infrequent with long recurrence intervals but produce the greatest ground motions because much of the energy is released at one or a limited number of events. In contrast, earthquakes in swarms tend to have lower magnitudes but are much more frequent and occur over longer periods of time. Thus, individual events such as the Hansel and Borah Peak earthquakes release much more energy and cause serious damage, whereas earthquake swarms such east of Soda Spring, Idaho in 2017 release the energy over a series of smaller fault displacements.

Based on the available technical literature, the greatest threat for seismic damage in the Pocatello area is from earthquakes along the Wasatch Fault south of city. Based on geologic studies, the IGS opines that magnitude 7.0 earthquakes occur along the Wasatch Fault every 300 to 400 years. Hecker (1993) predicts the average regional occurrence of large magnitude earthquakes in the Wasatch Front Region during the Holocene period to be on the order of 125 to 300 years or less, although events have been non-uniformly distributed in time.

POCATELLO DAMAGE, March 12 1934-(AP)-Pocatello was severely shaken twice today by two earth tremors which sent citizens, agog with excitement, scurrying into the streets for safety. The shocks occurred at eight-seven and eleven twenty-one a.m. Following the second vibration, which was more intense here than the first, public schools were dismissed until a thorough inspection could be made. Cracked chimneys were reported at the Emersion, Lincoln and Jefferson schools and at the general hospital by Fire Chief A. B. Canfield who made an immediate inspection. The walls in several schools and other buildings were also cracked. The balcony at Reed Hall gymnasium where Pocatello and McCammon High school were to play for the district basketball title this afternoon, was condemned as unsafe after the second shock loosened beams. The game was to go on, however. Several homes and one business house reported broken windows.

March 12 1934 (AP)-Two sharp earth shocks of about 15 seconds duration jarred Pocatello residents at 8:07 a.m. today. The Pocatello Tribune building and others in the downtown section were shaken by the quake being accompanied by a heavy rumbling sound. One home reported a window broken by the tremors

SALT LAKE CITY, March 13 (AP)-Pending a thorough inspection, schools of Salt Lake City will remain closed today but educational institutions will resume normal functions in most other north Utah and south Idaho cities which were rocked by a series of earthquakes yesterday. Damage to buildings in all instances was confined to cracks and toppled chimneys. The most severe disturbance was centered in an area bounded by Boise, Idaho, on the north; Rock Springs, Wyo., on the east; Richfield, Utah, on the south and Elko, Nev., on the west. The quake was less severe in Wyoming and Nevada than in Utah and southern Idaho. On the recommendations of W. L. Payne, Salt Lake City police chief, and Dr. L. E. Viko, city health commissioner, Dr. L. John

Nuttal, Jr., superintendent of public instruction, announced that as a precautionary measure all public institutions under his jurisdiction will remain closed until buildings are closely inspected for hidden flaws. Close to Pocatello, 10 public schools were closed following the disturbance. An inspection was started at once and all structures were found safe. The walls of one building were badly cracked. J. R. Nichols, de an of the University of Idaho, southern branch, at Pocatello, called on the state department of public works at Boise to send an expert to examine building ~~ on the campus. Large cracks appeared in several places in the administration building, in the hall which houses the auditorium and, in the library, considerable damage was done and beams were loosened in the gymnasium.

Earthquake Catalog

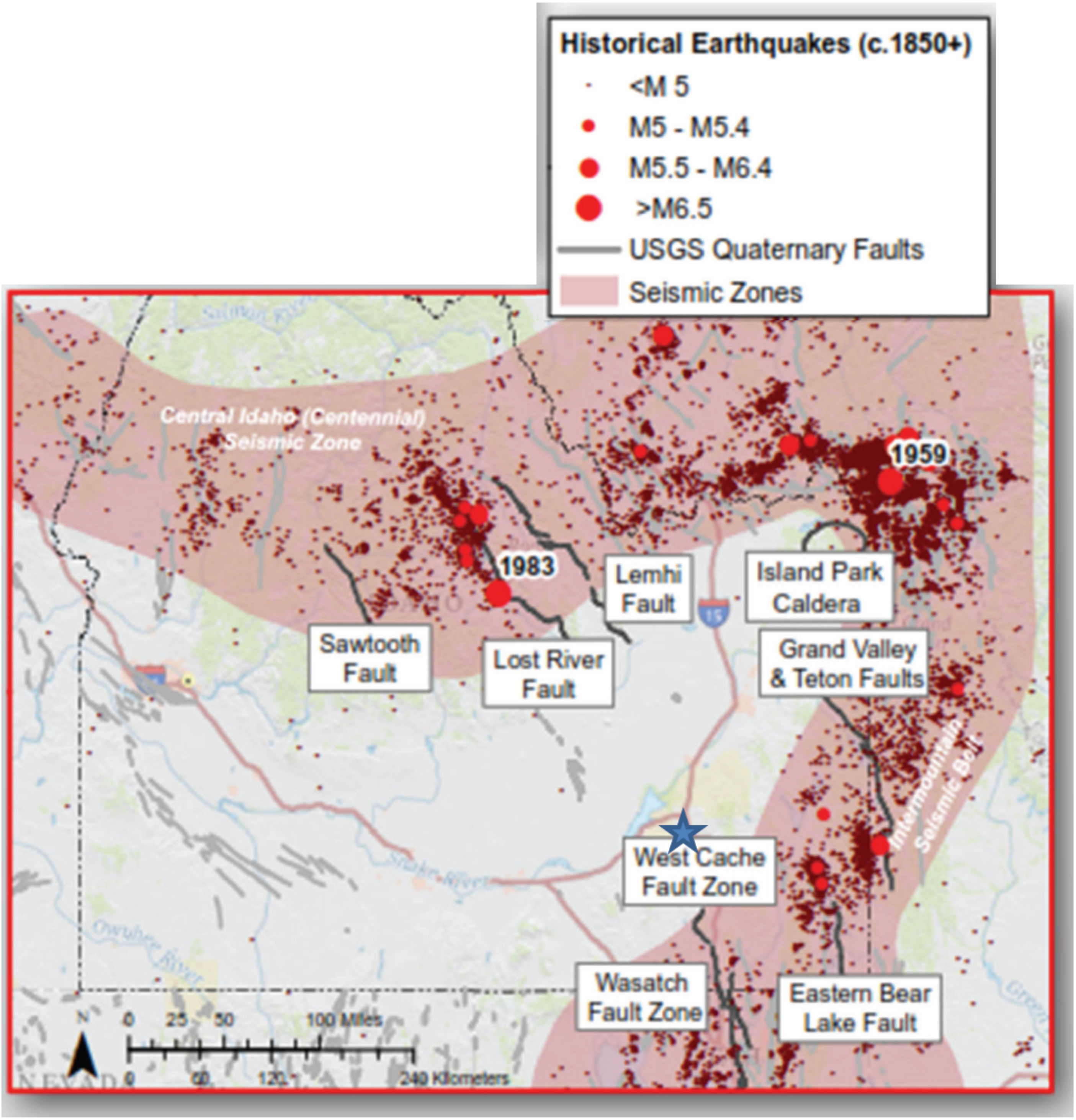
A search was made of available on-line earthquake catalogs to assess the frequency of magnitude 4 or greater events in Eastern Idaho. The search area chosen was 41 to 44.5° north latitude and 110 to 115° west longitude. Seismic monitoring of eastern Idaho is splintered between several organizations and their catalogs are not routinely integrated. Most of the catalogs are not published in any usual sense; they are maintained by their individual organizations as part of routine seismic monitoring efforts. The level of completeness, accuracy, and ending date differ between the catalogs. Those catalogs consulted were developed by Woodward-Clyde Federal Services (see Wong et al., 1992 for a map), the University of Utah, the Idaho National Engineering Laboratory, the Montana Bureau of Mines and Geology, Boise State University, and the U.S. Geological Survey. The Woodward-Clyde catalog was derived from the other catalogs and other sources of historic data, but only runs through late 1989. It was the primary data source for events through its completion date. The composite catalog is believed to be essentially complete through mid-1994.

Figure 2.4-1 shows the resulting epicenter map, and Table 2.4-1 gives epicentral data for the events within 130 km (approximately 80 miles) of 42.868° north, 112.435° west (the approximate location of the Pocatello business district and Idaho State University). Smaller earthquakes have occurred closer to Pocatello, but the distribution of magnitude 4+ events may be a better index to active source areas. No magnitude 4 earthquake has occurred closer than about 60 km (about 35 mi) to Pocatello. The ESRP, which lies to the north of Pocatello, is nearly aseismic at the magnitude 4 level and is not considered to be a likely source of events capable of causing damage in Pocatello. High activity occurs to the south, east, and northeast at greater distances than 35 mi. Many of these earthquakes have been felt in Pocatello, although damaging levels of intensity are not believed to have been achieved historically. The closest magnitude 6+ earthquake was the 1975 Pocatello Valley earthquake (local magnitude 6.0), about 56 mi from Pocatello, and the closest magnitude 7+ earthquake was the 1983 Borah Peak earthquake (surface wave magnitude 7.3), about 105 mi from Pocatello.

It is concluded that the historical earthquake catalog shows Pocatello is in an area of relatively low seismicity, with higher seismicity areas occurring at least 35 mi from the city. This would suggest that Pocatello is at little risk from a nearby source of large earthquakes, but the historic record is short. Geologic evidence discussed in the next section is felt to be of greater importance than the historical seismicity catalog.

2.4.4 Surface Faults and Fault Swarms

Faults in the Basin and Range province usually have long recurrence times between large events, typically 1,000 to 100,000 years on the same fault segment. Therefore, the historical catalog is not the only indication of the potential for damaging earthquakes at Pocatello. Figures 2.4-2 and 2.4-3 (taken from Hilt, 1993) show that while mapped late Cenozoic faults occur within a few miles of Pocatello, the nearest fault with probable late Quaternary movement is located about 100 mi southeast of Pocatello (it is the Bear Lake fault). Late Quaternary movement iS known on faults on the north side of the ESRP at about the same distance from Pocatello. Faults with late Cenozoic movement but no evidence for late Quaternary movement are generally considered inactive for seismic hazard evaluation purposes. Most late Quaternary faults have expectable maximum magnitudes between about surface wave magnitude 6.8 and 7.6.



Approximate location of Pocatello, Idaho

Fig. 2.4-1 Seismic Zones identified by the Idaho Geologic Survey in eastern and central Idaho (IGS 2016)

Table 2.4-1 Earthquakes Near SE Idaho 1909-1994, with Magnitudes Greater than 4

Year	Date and Time (UTC)	Magnitude	Lat(N) deg.	Lon(W) deg.	Location from Pocatello
1909	10/06 02:50:00	6.0	41.75	112.65	77.0 mi S of Pocatello
1909	11/17 06:30:00	4.3	41.75	112.15	78.9 mi S of Pocatello
1915	07/30 18:50:00	4.3	41.73	112.15	78.9 mi S of Pocatello
1930	06/12 09:15:00	5.8	42.60	111.00	75.4 mi ESE of Pocatello
1934	03/12 15:05:48	6.6	41.76	112.66	76.8 mi S of Pocatello
1934	04/14 21:26:32	5.4	41.71	112.60	79.0 mi S of Pocatello
1934	05/06 08:09:42	5.5	41.95	112.81	65.7 mi SSW of Pocatello
1946	08/07 02:30:00	4.3	41.71	112.11	80.1 mi SSE of Pocatello
1960	08/30 16:27:16	4.9	42.40	111.50	57.6 mi SE of Pocatello
1962	09/05 13:35:24	5.7	41.91	111.61	77.3 mi SSE of Pocatello
1962	09/19 03:00:00	4.3	42.00	111.70	70.8 mi SSE of Pocatello
1969	09/19 09:31:45	4.4	43.05	111.41	53.2 mi ENE of Pocatello
1969	09/19 13:33:15	4.9	42.98	111.41	51.7 mi E of Pocatello
1969	09/19 19:57:18	4.4	43.00	111.26	59.9 mi E of Pocatello
1969	09/19 23:58:06	4.1	42.95	111.48	48.4 mi E of Pocatello
1973	04/14 06:45:46	4.4	42.03	112.61	57.9 mi S of Pocatello
1975	03/27 04:48:51	4.2	42.05	112.53	55.7 mi S of Pocatello
1975	03/28 02:31:05	6.0	42.05	112.51	55.8 mi S of Pocatello
1975	03/29 13:01:19	4.7	42.01	112.51	57.8 mi S of Pocatello
1975	03/30 6:56:28	4.1	42.01	112.56	58.1 mi S of Pocatello
1976	11/05 02:48:55	4.0	41.80	112.68	74.4 mi S of Pocatello
1978	10/24 20:30:59	4.1	42.55	111.83	37.6 mi SE of Pocatello
1978	11/30 06:53:40	4.7	42.10	112.50	52.6 mi S of Pocatello
1981	12/09 08:15:04	4.1	42.63	111.41	53.5 mi ESE of Pocatello

Table 2.4-1 (cont'd)

1982	05/30 11:06:42	4.0	42.68	111.23	62.2 mi E of Pocatello
1982	10/14 04:10:23	4.7	42.60	111.41	54.9 mi ESE of Pocatello
1982	10/14 11:09:29	4.1	42.60	111.43	53.7 mi ESE of Pocatello
1983	02/08 10:54:54	4.2	43.30	111.16	70.7 mi ENE of Pocatello
1985	07/02 03:03:56	4.0	43.25	111.15	70.2 mi ENE of Pocatello
1987	03/18 00:00:42	4.1	42.61	111.31	59.2 mi ESE of Pocatello
1988	11/13 11:53:24	4.4	42.61	111.91	78.4 mi ESE of Pocatello
1988	11/19 20:00:53	4.3	42.00	111.46	77.0 mi SE of Pocatello
1992	11/10 10:46:18	4.8	43.08	111.61	43.6 mi ENE of Pocatello
1992	11/10 10:54:50	4.9	43.00	111.45	50.3 mi ENE of Pocatello
1992	11/11 12:08:07	4.0	43.01	111.48	49.4 mi ENE of Pocatello
1994	02/02 11:04:25	4.0	42.75	111.06	69.9 mi E of Pocatello
1994	02/03 07:14:51	4.5	42.75	111.03	71.0 mi E of Pocatello
1994	02/03 09:05:03	5.8	42.75	110.96	74.3 mi E of Pocatello
1994	02/03 09:47:36	4.0	42.71	111.03	71.7 mi E of Pocatello
1994	02/03 09:58:40	4.2	42.75	111.03	70.9 mi E of Pocatello
1994	02/03 10:25:51	4.0	42.78	111.11	66.9 mi E of Pocatello
1994	02/03 11:19:07	4.7	42.76	111.00	72.6 mi E of Pocatello
1994	02/03 11:46:50	4.0	42.78	111.15	64.9 mi E of Pocatello
1994	02/03 12:04:57	4.4	42.71	111.08	68.7 mi E of Pocatello
1994	02/03 02:42:12	5.2	42.70	111.03	71.8 mi E of Pocatello
1994	02/04 03:10:08	4.0	42.83	111.08	68.1 mi E of Pocatello
1994	02/04 21:49:12	4.0	42.61	111.05	71.7 mi ESE of Pocatello
1994	02/07 06:35:47	4.8	42.65	111.03	72.7 mi E of Pocatello
1994	02/07 12:15:45	4.5	42.66	111.01	73.3 mi E of Pocatello
1994	02/10 00:56:11	4.3	42.88	111.06	69.3 mi E of Pocatello

Table 2.4-1 (cont'd)

1994	02/11 04:24:29	4.0	42.81	111.11	66.3 mi E of Pocatello
1994	02/11 14:59:50	5.3	42.76	111.00	72.8 mi E of Pocatello
1994	02/14 16:55:34	4.0	42.80	111.01	71.5 mi E of Pocatello
1994	03/03 07:13:17	4.1	42.78	111.05	69.8 mi E of Pocatello
1994	04/07 16:16:44	4.8	42.53	111.01	75.5 mi ESE of Pocatello
1994	04/08 07:26:21	4.1	42.60	111.08	71.1 mi ESE of Pocatello
1994	04/10 20:04:09	4.6	42.65	111.11	68.4 mi ESE of Pocatello

Table 2.4-2 Earthquakes after 1994, with Magnitudes Greater than 5

Year	Depth	Magnitude	Lat(N) deg.	Lon(W) deg.	Location from Pocatello
1995	1	5.3	41.5	109.6	130 mi ESE of Pocatello
1999	16	5.1	44.8	112.8	160 mi N of Pocatello
2001	-	5.3	42.9	111.4	65 mi E of Pocatello
2004	3	5.0	43.6	110.8	100 mi ENE of Pocatello
2005	12	5.6	43.4	112.6	145 mi N of Pocatello
2008	7	6.2	41.1	114.9	153 mi SW of Pocatello
2008	-	5.1	41.1	114.8	150 mi SW of Pocatello
2015	8	5.0	44.5	114.1	150 mi NW of Pocatello
2017	12	5.8	46.9	112.6	250 mi N of Pocatello
2017	6	5.3	42.6	111.4	50 mi E of Pocatello
2020	-	5.7	40.8	112.0	165 mi S of Pocatello
2020	-	6.6	44.2	115.2	160 mi NW of Pocatello

Anders et al. (1989) believe that the location of active segments of faults around the ESRP is a function of distance from the ESRP axis, and propose a physical model to explain an increase in strain rates with distance from the ESRP. The Anders et al. model would suggest that faults in the immediate vicinity of Pocatello have very low strain rates, and consequently long recurrence times (in excess of 10,000-100,000 years) are probably to be expected. If the Anders et al. model is correct, there is little risk from a large earthquake occurring in the immediate vicinity of Pocatello. Major earthquakes (magnitude 6.8+) are likely to occur no closer to Pocatello than the regions of observed high seismicity, and the Anders et al. model suggests that the more active faults would be located even farther from the city. It is therefore concluded that the known late Quaternary faults are not greater sources of seismic risk to Pocatello than possible blind faults existing near the city, as will be discussed in the next section.

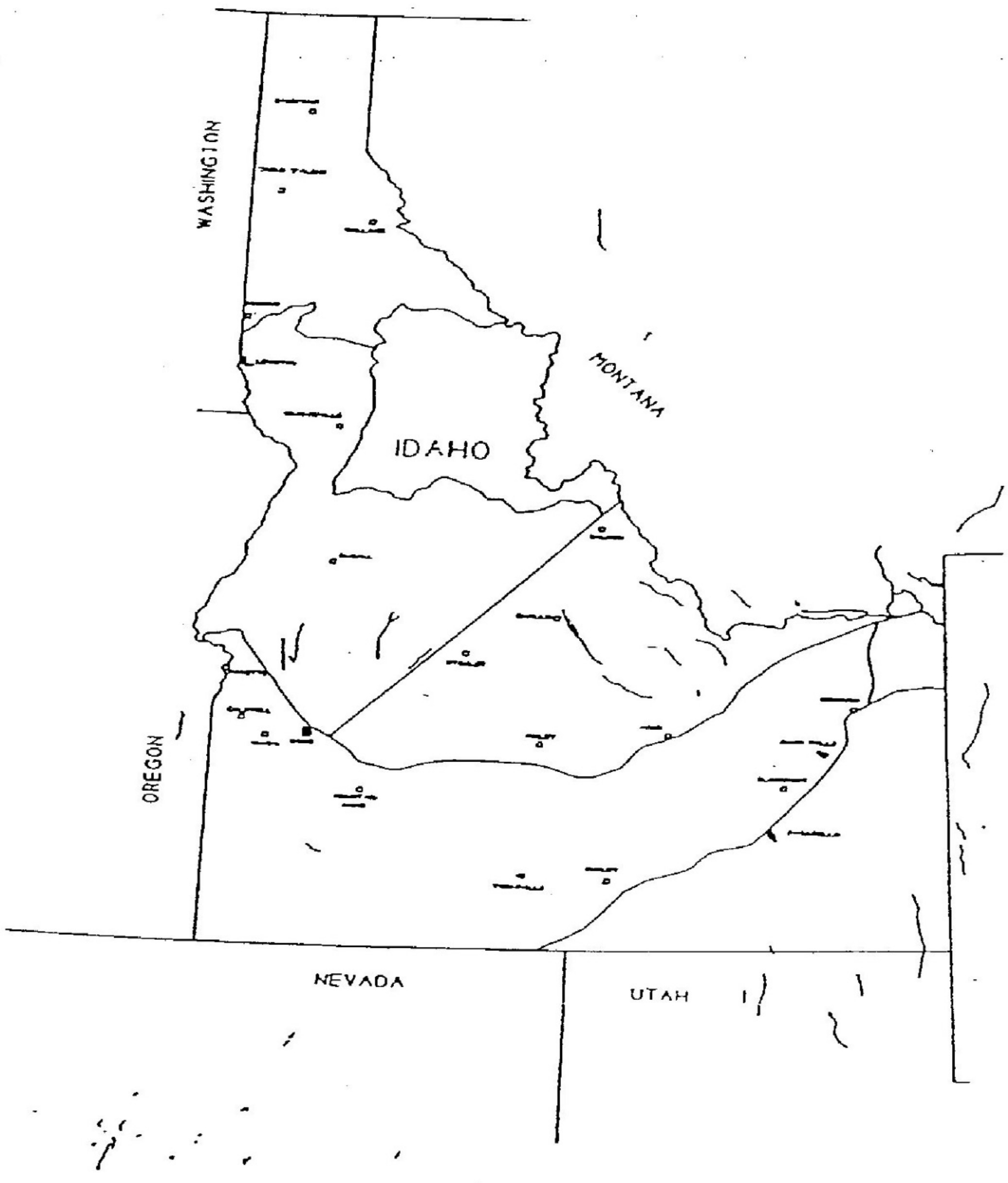


Fig. 2.4-2 Faults with late Quaternary motion in Idaho

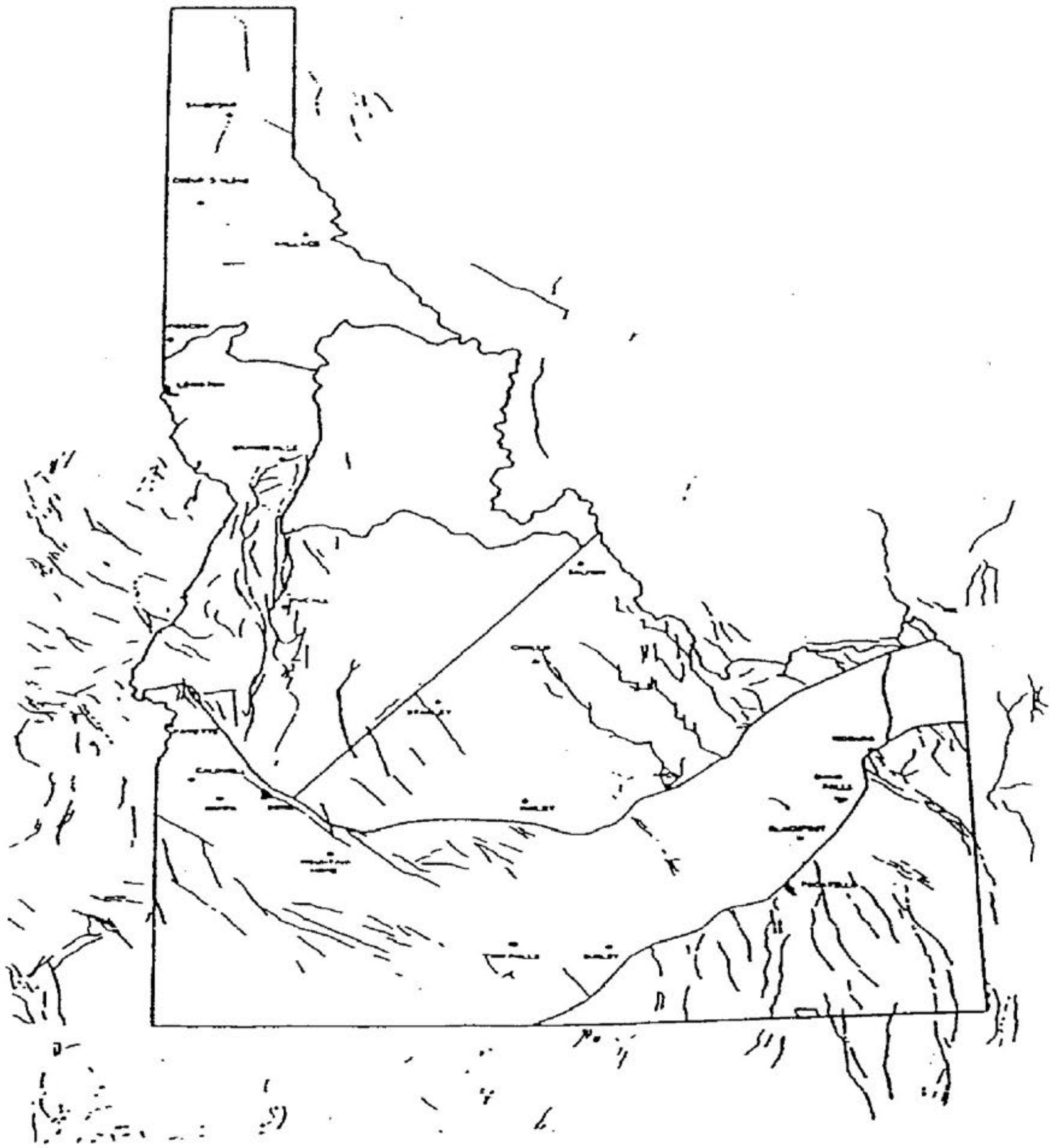


Fig. 2.4-3 Late Cenozoic Faulting in Idaho

2.4.5 Floating Earthquakes

Blind faults (those not recognized at the earth's surface) exist in southeast Idaho, as is proven by the 1994 local magnitude 5.8 Draney Peak earthquake and the 1975 local magnitude 6.0 Pocatello Valley earthquake. These earthquakes did not occur on known faults. To account for such events, seismologists have used the concept of a "floating earthquake" which is customarily chosen as being ½ magnitude unit larger than the largest historical event that is not associated with a known fault. The largest such events are the Draney Peak and the Pocatello Valley earthquakes. Thus, a reasonable estimate of the magnitude of the floating earthquake is 6.5. The floating earthquake is customarily placed at a distance of 25 km from the site in question. Since Pocatello lies at the edge of the Basin and Range province in which the Draney Peak and Pocatello Valley earthquakes occur, the possibility of blind faults near Pocatello cannot be ruled out. Therefore, a magnitude 6.5 earthquake at a distance of 25 km is chosen to be the event controlling seismic hazard at Pocatello. The choice of this event is a state-of-the-art assessment and may change in the future as a better understanding of southeastern Idaho seismotectonics develops.

Design acceleration for floating earthquake. The horizontal peak ground acceleration for a floating earthquake of magnitude 6.5 at a distance of 25 km from Pocatello can be calculated from relationships presented by Joyner and Boore (1981). Because of the lack of significant historical seismicity within 35 mi of Pocatello, it is felt that use of the Joyner and Boore (1981) 50th percentile formula is sufficiently conservative; a higher percentile formula would be justified if significant seismicity were if located near the city. The Joyner and Boore (1981) relationship is:

$$\log A = -1.02 + 0.249\ M_w - \log r - 0.00255\ r$$

where A is horizontal peak ground acceleration in g, M_w is earthquake moment magnitude (roughly equivalent to local magnitude at magnitude 6.5), and r is a distance term defined as:

$$r = (d^2 + 7.3^2)^{0.5}$$

where d is the closest approach of the seismogenic structure in km. For the floating earthquake, d is 25 km. It should be noted that the Joyner and Boore (1981) curve was developed mainly from California area data and application to southeast Idaho can be expected to be only approximate.

The resulting horizontal peak ground acceleration is 0.13 g. Such an acceleration indicates that minor structural damage in poorly-built facilities is about the most that can be expected in the Pocatello area. Since such an event has not occurred historically, this seems to be a conservative estimate suitable for most uses.

<u>Reference:</u>

Anders, M. H., J. W. Geissman, L. A. Piety, and J. T. Sullivan (1989), Parabolic distribution of circumeastern Snake River Plain seismicity and latest Quaternary faulting: Migratory pattern and association of the Yellowstone hotspot, Journal of Geophysical Research 94, 1589-1621.

Hilt, A. P. (1993), Seismic Siting Considerations in Idaho, M. S. Thesis, University of Idaho, Moscow, 136 p.

Joyner, W. B., and D. M. Boore (1981), Peak horizontal acceleration and velocity from strong-motion records including records from the 1979 Imperial Valley, California earth quake, Bulletin of the Seismological Society of America 71, 2011-2038.

Wong, I., K. Coppersmith, W. Silva, R. Youngs, T. Sawyer, M. Hemphill-Haley, C. Stark, P. Knuepfer, R. Castro, F. Makdisi, D. Wells, S. Chiou, R. Bruhn, and W. Daning (1992), Earthquake Ground Motion Evaluations for the Proposed New Production Reactor at the Idaho National Engineering Laboratory, Volume I: Deterministic Evaluation, Informal Report EGG-GEO-10304, Woodward-Clyde Consultants, Geomatrix Consultants, and Pacific Engineering and Analysis, Oakland, CA.

3 LILLIBRIDGE ENGINEERING LABORATORY

This section contains information about the Lillibridge Engineering Laboratory building and the Nuclear Operations Area in [redacted].

3.1 General Description

As shown in Figure 2.1-3, the Laboratory is located [redacted], just south of the [redacted]. The major axis of the rectangular building is parallel to the street and in approximate north-south direction. The building has three (3) floors: a basement (first) level, a ground (second) level and an upper (third) level. The ground level floor is at 4500 ft. elevation above sea level. The building was designed by C. A. Sundberg and Associates, constructed by Taysom Construction and first occupied September 14, 1971. The building conforms with the zone 3 seismic design requirements of the Uniform Building Code, with walls made up of precast concrete slab design. The building has approximately 28,500 sq. ft. divided as follows:

Laboratories - 11,900 sq. ft.

Offices - 2,900

Classrooms - 1,600

Storage & misc. - 12,100

The northern-most portion of the building is the single story initially was a Mechanical Shop and experimental fluids research area, and is currently used for storage. It has a thick floor to support heavy machining equipment, and no basement. The Reactor Laboratory extends [redacted]. The upper portion of this [redacted] of the main building. The floor of the laboratory is [redacted].

The building is supplied with 110, 240 and 480 VAC power and is heated with low pressure steam from the campus system. The building has its own recirculating ventilation vapor compression and cooling system which is interconnected with the Reactor Laboratory ventilation system as described below. Figure 3.1-1 shows the layout of the top floor. The space is devoted faculty offices, the Computer laboratory, conference room and some staff and graduate student offices. The ground floor level shown in Figure 3.1-2 is comprised of the [redacted]

[redacted]

Fig. 3.1-1 LEL Third Level (top) Floor Plan

[redacted]

Fig. 3.1-2 LEL Second Level (ground) Floor Plan

[redacted]

Fig. 3.1-3 LEL First Level (basement) Floor Plan

The Machine Shop is also on this level. As shown in Figure 3.1-3, the basement houses the [redacted]

The Building ventilation and cooling system recirculates about 80% of the flow, discharges 20% from a roof vent and takes in about 20% in fresh (outside) makeup air before being distributed throughout the building. The exhaust from the Reactor Laboratory mixes with air from the rest of the building on its way to the roof vent. The building air and Reactor Laboratory exhaust mix in the ratio 27:1, respectively. In the event of an emergency, the Lillibridge ventilation system fans can be turned off to reduce the emission and recirculation of any radioactive contamination.

3.2 Nuclear Operations Area

This area comprises the main portion of the basement level of the building and includes the Reactor Laboratory, Subcritical Assembly Laboratory, Counting Laboratory, Reactor Supervisor's Office and the student offices.

3.2.1 Reactor laboratory

This [redacted] the AGN-201 reactor, as shown in Figure 3.2-1. [redacted] The 6.5 ft. diameter reactor tank is located in the northwest corner of the room behind an interior L-shaped shield wall. Since the [redacted] Access to the Reactor Laboratory is through [redacted] of Lillibridge. [redacted] An overhead crane with trolley provides large equipment lift coverage [redacted] A covered $4^{1}/_{2}$ ft. diameter cylindrical steel-lined pit extends 16 ft. below the Reactor Laboratory floor just south of the reactor tank. At the bottom of the pit is a [redacted] Ci Am-Be source surrounded by paraffin moderator. Access to the top of the reactor tank is via stairs located along the west wall of the laboratory. The reactor control console is located to the east of the reactor tank in the northeast corner of the room. The console can readily be seen from the Observation room and classroom adjacent to the east wall of the Reactor Laboratory, through full height windows in that wall.

[redacted]

Fig. 3.2-1 Reactor Laboratory Floor Plan [redacted]

The Reactor Laboratory is supplied with 110 and 240 VAC power and fluorescent lighting but no plumbing. A telephone is located at the control console; a fire extinguisher and the fire alarm are located on the east wall adjacent to the double doors. Shielding walls, and radiation fields with the reactor at full power of 5 W, are discussed in Section 5.3. An earthquake detector is located on the support skirt of the reactor tank. Various radiation monitoring devices are located within the Laboratory, and standard locations are monitored on the control console.

3.2.2 Counting laboratory

The Counting Laboratory, [redacted] is located north of the Observation/Conference room, between it and the Subcritical Assembly Laboratory. The Counting Laboratory has a small fume hood and associated compressed air, water, etc. plumbing. Also located in this Laboratory is a locked closet for storage of small, sealed radiation sources. The primary equipment in the Laboratory is a recycled cooling system cooling a germanium detector based computerized spectroscopy system. The Laboratory also has the outboard terminus of a rabbit system connected to the AGN-201 reactor, to an associated counting system.

3.2.3 Subcritical laboratory

This Laboratory, [redacted] contains a subcritical assembly used in the undergraduate and graduate laboratory courses. As shown in Figure 3.2-2, the 19.9% enriched fuel, [redacted] The water tank used to hold the fuel for experiments is mounted on a large set of high purity graphite blocks, the whole assembly constituting a subcritical pile with a k_{eff} of approximately 0.87. The Laboratory is equipped with both neutron and gamma radiation monitors with associated alarms. The [redacted] is connected by piping and a pump to a plastic water storage tank located [redacted]. No other plumbing exists in the room. Beryllium, beryllium oxide, and heavy water are not permitted in this Laboratory without Reactor Supervisor permission.

[redacted]

Fig. 3.2-2 Subcritical Assembly Laboratory Floor Plan [redacted]

3.2.4 Reactor observation room, supervisor's office

As noted above, the student office is adjacent to the Reactor Laboratory and has full height windows in their common wall. This is identified as [redacted] It is locked when not in use, since a radiation field of 6 mr/hr is present in the room when the reactor is operating at a 5W power level. Access to this room, when the reactor is operating, is restricted to authorized personnel. In all other parts of the Nuclear Operations Area, radiation levels are less than 0.1 mr/hr when the reactor is operating at 5 W.

As shown in Figure 3.1-3, the Reactor Supervisor's office, [redacted] is located east of the Observation/Conference room and adjacent to a room containing an industrial shower and a storage area for personnel dosimeters and other radiation monitoring equipment. The ventilation cutout switch, which turns off the Lillibridge ventilation system fans is located on the wall south of the Supervisor's office.

4 AGN-201 Reactor

This section contains a description and detailed technical information about the AGN-201 reactor.

4.1 Introduction

The AGN-201 reactor operating at Idaho State University is essentially identical to all the AGN-201 reactors which have operated around the world since 1957. Currently only five still exist, three in the U.S.A., one in the Republic of Korea and one in Italy. Therefore, the reactor description given in the Safety Analysis Report for the original Aerojet General AGN-201 reactor (Hazard Report and Preliminary Design Report by Aerojet General Nucleonics, Docket F-40) is applicable to this reactor, as are the Hazard and Safety Analysis Reports for other AGN-201 reactors on file with the NRC.

The characteristics and operating parameters of this reactor have been calculated and experimentally determined from data obtained from this and other AGN-201 reactors.

The critical mass of the reactor is approximately [redacted] of ^{235}U. The loading density is [redacted] and the estimated core volume is [redacted] liters. Allowing for control and safety rods, etc., the right circular cylindrical core is [redacted] in diameter by [redacted] high. An additional partial one inch diameter fuel plate sits on top of the cylindrical core (Figure 4.1-1). In order to facilitate critical assembly, the core is fabricated in [redacted] high sections and [redacted] thinner sections. The thinner sections allow variations of the amount of ^{235}U to be inserted in smaller increments in the range from [redacted] to [redacted] total core weight. Each of the [redacted] bottom fuel discs has four holes to permit penetration by the safety and control rods; two holes accommodate the safety rods and the other two holes accommodate the control rods. The glory hole, which is 2.38-cm (15/16") inside diameter, is located [redacted]. The core discs are formed by pressing under high pressure a homogeneous powdered mixture of polyethylene and UO_2 [redacted] in the form of 20 micron diameter particles. The total density of uranium is then [redacted] yielding a weight ratio of [redacted] of U to [redacted] polyethylene (assuming 20%-enriched ^{235}U).

[redacted]

Fig. 4.1-1Top Partial Fuel Disk [redacted]

The core tank has been designed to contain any radiolytic and fission-product gases that might diffuse out of the fuel discs. Sixty-five mil commercial (6061 T6) aluminum is used throughout as the structural material. The core tank may be considered to be made of an upper and lower section, separated by a thin aluminum plate in the same plane as the glory hole that bisects the main core cylinder. This baffle serves to separate the core into two halves, and is part of the fuel system. Detachable top and bottom cover plates as well as control- and safety-rod thimbles from an integral part of the core tank.

The lower section of the core tank contains approximately one-half of the core material as well as part of the graphite reflector are supported by an aluminum rod projecting downward from the thermal safety fuse link. Ample space at the bottom of the cylinder, coupled with a tapered graphite-to-graphite joint, is provided to ensure free-fall of the bottom half of the core plus reflector sections, should the fuse melt from an accidental nuclear excursion. The upper section of the core tank is filled with five fuel discs and part of the reflector. A space for core expansion and gas accumulation is provided in the top section of the core tank.

The reflector consists of 20 cm of high density (1.75 gm/cm^3) graphite on all sides of the core. Appropriate holes are provided for the glory hole, the four safety and control rods, and the four access ports. The top cover of the core tank is removable, as is the top plug of graphite, to permit access to the core. The complete core tank may be removed, also. [redacted] of lead completely surround the core and the reflector. The graphite reflector weighs about 320 kg (700 lb), and the lead shield weighs about [redacted].

The lead shielding, reflector, and core are enclosed in and supported by a thick steel tank [redacted] The outer tank acts as a secondary container for the core tank assembly and is completely fluid-tight. A removable tank is provided over the top of the core to permit access to the core tank. This removable tank may be filled with water for shielding or with graphite when a thermal column is desired. The outer steel tank is completely surrounded on all sides by a water tank for shielding against fast neutrons. The control rods and safety rods enter through the bottom of the tank.

4.2 AGN-201 Characteristics

The AGN-201 is a low-power reactor that has the distinct advantage of small size, which enables it to be readily located in existing buildings. The reactor is employed for use in education and research. Because of the low operating power of the reactor, no external cooling system is required. The main design objectives were to make the reactor as safe and foolproof as possible, to use a minimum critical mass, and to provide a high analytical sensitivity.

Fig. 4.2-1 AGN-201 Nuclear Reactor Control console

The AGN-201 reactor system consists of two basic units, the reactor and the control console (Figure 4.2-1). The reactor unit includes the uranium-polyethylene core, graphite reflector, lead and water shielding, control rod drive mechanisms, and neutron detectors. Fuel-loading control and safety rods are inserted vertically from the bottom of the reactor unit, passing by the instruments which measure the power level and the control mechanisms which provide for the safe and efficient operation of the reactor. The mass (weight) of the reactor unit, with the water shield in place, is 9100 kg (20,000 lb); the mass (weight) of the console unit is 360 kg (800 lb). Power requirements for operation of the control console and associated instrumentation are 2 kW of 110 VAC, single phase, 60-cycle electrical power.

The AGN-201 is a homogeneous, thermal reactor as shown in Figure 4.2-2. The core is made up of a series of [redacted] circular discs, [redacted] diameter and of varying thickness, which consist of UO_2 embedded in radiation-stabilized polyethylene. The core has a critical mass of approximately [redacted] ^{235}U. The cylindrical core is surrounded by a graphite reflector, 20-cm thick, and is shielded by [redacted] lead and 55-cm water. As shown in Figure 4.2-3. the core and part of the reflector are contained in an aluminum tank which has re-entrant thimbles in its base, into which the four control and safety rods are inserted. Each rod contains fuel material sealed in an aluminum capsule, thereby increasing reactivity as the rods are inserted into the core. Both safety rods and one control rod are held in position by electromagnets for scram purposes. Each of the four rods is driven by a lead screw.

The core is divided into half-sections with a plane of separation occurring at the level of the one-inch diameter glory hole that passes completely through the core. The lower section is held in place by a thermal fuse link which is designed to soften at 100ºC, and when it functions as designed, permits the bottom core section to drop 5 cm (2") to the bottom of the core tank. This separation of the core results in a reduction in reactivity of from 5 to 10%. The excess reactivity is limited by the license to less than 0.65% $\frac{\Delta k}{k}$ (note: because the small size of the reactor, the effective delayed neutron fraction is 0.00745).

The two safety rods and the coarse control rod are each worth approximately 1.25% $\Delta k/k$ ($1.69). The fine control rod, which can be loaded with either fuel material or polyethylene, is currently loaded with fuel and worth about 0.31% $\Delta k/k$ ($0.42). The neutron flux is monitored by three independent detectors: two BF_3 ionization chambers and a BF_3 proportional counter, all of which are located in the water tank just outside the lead shield. The detectors are connected respectively to a logarithmic picoammeter, a linear picoammeter, and pulse amplifier and count rate meter located in the reactor console. Each indicator is connected through a relay to a scram circuit. Additional safety interlocks provide for reactor shutdown if the level of the shield water drops, if the reactor temperature falls below 15ºC, or if an earthquake occurs. Sequential interlocks are also present to ensure that the proper operational method is followed. Detailed characteristics of the reactor are given in Table 4.2-1.

[redacted]

Fig. 4.2-2 AGN-201 reactor unit

\

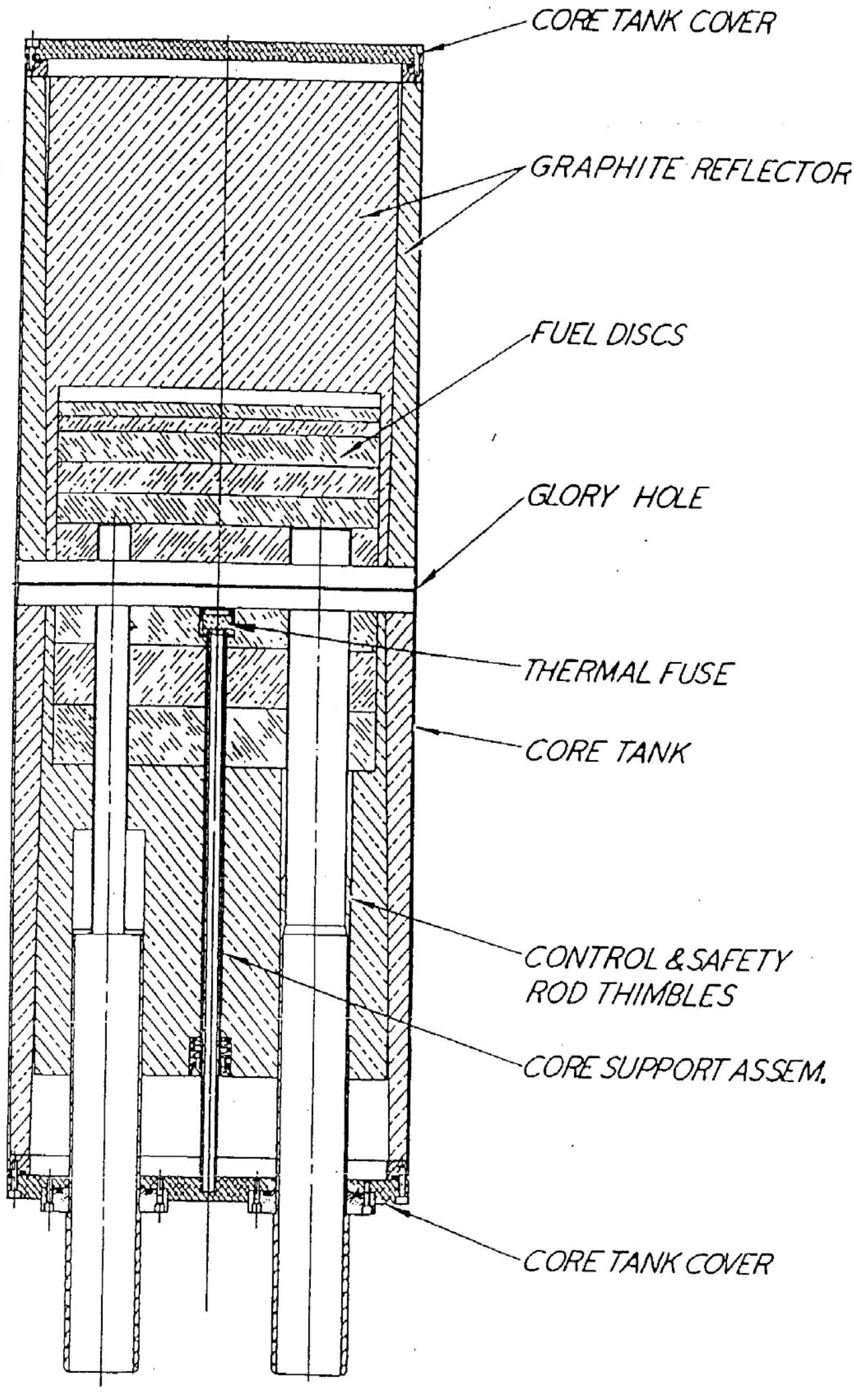


Fig. 4.2-3 AGN-201 core tank and contents

Table 4.2-1 Reactor Characteristics

a. <u>General</u>

Type	Homogeneous Thermal Reactor
Principle Uses	Education and Training
Maximum Operating Power	5 watts
Core	UO_2 (20%-enriched in ^{235}U) particles homogeneously distributed in solid polyethylene moderator, cylindrical geometry
Reflector	Graphite
Shield	Lead and water
Control	Two safety and two control rods, all fuel loaded

b. <u>Fuel</u>

Fuel Material	20%-enriched UO_2
Fuel Disk	Right circular cylinder fabricated from a compressed mixture of fuel material and polyethylene powder Specifications: UO_2 powder ^{235}U enrichment – 19.5 ± 0.5% Particle size – 15 ± 10 µm Polyethylene powder: Particle size – 100 µm Purity – commercial grade
Approximate Core Fuel Disc Sizes and ^{235}U Content	[redacted] number ([redacted] diameter) [redacted]

Table 4.2-1 (cont'd)

Approximate Control and Safety Rod Fuel Disc Sizes and Total ^{235}U Content for All Safety and Control Rods	[redacted]
Total Fuel Loading	[redacted]
^{235}U Density	[redacted]
Normal Lifetime of Fuel Discs	Indefinite
Core Fuse	Small, right circular cylinder, [redacted] & [redacted] high, [redacted] ^{235}U. Fabricated from mixture of fuel material and polystyrene powder.

c. <u>Reactor</u>

Core-Containing Vessel	Gas-tight, 65-mil (6061T6) aluminum cylindrical tank, 32.2-cm diameter and 76-cm high. Vessel has a 2.54-cm glory hole on the horizontal center line and appropriate control rod thimbles. Tank contains core fuel discs and graphite reflector plugs.
Core Description	[redacted] fuel discs, [redacted] diameter, [redacted] high, separated in half at glory-hole mid-plane by a thin aluminum baffle. Lower half of core contains appropriate holes for control and safety rod thimbles and is part of the safety fuse system.
Reflector	Heavy-density (1.75 g/cm^3) graphite, approximately 20 cm on all sides of core. Top and bottom reflector in core tank, side reflector surrounds core tank and tangentially to core tank
Gamma Shield	[redacted] of lead completely surrounding reflector.
Reactor Tank	80-mm-thick (5/16") steel tank, 95-cm diameter, and 148-cm high; contains core tank, reflector and lead shield, and appropriate holes for control and safety rods, glory hole, and access ports. Gas-tight vessel when all seals are made. Upper portion of tank contains a removable thermal column tank.

Table 4.2-1 (cont'd)

Fast Neutron Shield	[redacted] of water surrounds the reactor tank except at its bottom face.
Water Tank	Steel tank serving as the main structural tank, 198-cm (6.5 ft) diameter and contains 3800 l (1,000 gal) of water. Tank is supported at its bottom by the reactor skirt.
Reactor Dimensions	198-cm (6.5 ft) diameter and 290-cm (9.5 ft) high.
Reactor Weight	6800 kg (15,000 lbs.) (less shield water)
Reactor Control	Two safety rods, [redacted] ^{235}U each. One coarse control rod, [redacted] ^{235}U. One fine control rod, [redacted] ^{235}U. All rods, with the exception of the fine control rod, which is mechanically coupled, are magnetically coupled to a carriage that is driven in a vertical direction on a lead screw by a reversible DC motor. Total travel distance is about 25 cm.
Experimental Facilities	2.22-cm (7/8") glory hole passing through the center of core at core median plane. Four, 10-cm-diameter (4") access ports passing though the graphite reflector tangentially to the core. Thermal column tank above the core.
Neutron Source	[redacted] mCi californium source utilizing a spontaneous fission reaction. Neutron yield is approximately [redacted] n/sec.

d. <u>Nuclear Data</u>

(1) Fuel Loading

(a) Approximate Critical Mass [redacted] ^{235}U

(b) Excess Reactivity at 20° C, with current fuel loading of [redacted] of ^{235}U, with Glory Hole Empty 0.18% $\Delta k/k$ ($0.24)

Table 4.2-1 (cont'd)

(2) Neutron Flux

Average Thermal Flux	1.5 x 10^8 n/cm^2-s at 5 W
Peak Thermal Flux	2.5 x 10^8n/cm^2-s at 5 W

(3) Reactivity Worth of Reactor Components

(a) Safety and Coarse Control Rods	
	1.25% Δk/k ($1.68) (each)
(b) Fine Control Rod	
Fuel-loaded	0.310% Δk/k ($0.42)
Polyethylene-loaded	0.155% Δk/k ($0.21)
(c) Standard Core Material in Glory Hole	
At Core Edge	0.042% Δk/k gm^{-1} ($0.06)
At Core Center	0.100% Δk/k gm^{-1} ($0.14)
(d) Polyethylene in Glory Hole (completely filled)	0.29% Δk/k ($0.39)
(e) Access Port Plugs	
1 Wood Plug	0.002% Δk/k ($0.003)
1 Section Pb	0.015% Δk/k ($0.02)
1 Section Graphite	0.194% Δk/k ($0.26)
Total Worth of Plugs in one Access Port	0.422% Δk/k ($0.57)
(f) Temperature Coefficient of Reactivity (Approximate)	-0.035% Δk/k °C^{-1}
(g) Reactor Sensitivity at Core Center, Measured with 1/*v* Absorber	-0.14% Δk/k cm^{-2}

(4) Pertinent Figures

(a) Control Rod Reactivity Shape Curve in Figure 4.2-4

(b) Inhour Equation Given in Figure 4.2-5

(c) Flux Plot vs Radius Given in Figure 4.2-6

Table 4.2-2 Reactor Control and Safety Specifications, and Trips

SAFETY CHANNEL	SET POINT	FUNCTION
Nuclear Safety Channel No. 1 (Startup Count Rate Channel) Low Power	0.5 counts/second	Scram at levels Below the set points
Nuclear Safety Channel No. 2 (Log Power Channel) High Power	6 watts (120% of licensed power)	Scram at power > 6 watts
Reactor Period	5 sec	Scram at periods < 5 sec
Nuclear Safety Channel No. 3 (Linear Power Channel) High Power	6 watts (120% of licensed power)	Scram at power > 6 watts
Nuclear Safety Channel No. 3 (Linear Power Channel) Low Power	5% Full Scale	Scram at levels < 5% of Full Scale
Manual Scram	----	Scram at operator option
Area Radiation Monitor	= 10 mR/hr	Alarm at or below level set to meet requirements of 10 CFR 20

REACTOR TRIPS

Channel No. 1 Trip:	Less than 0.5 counts/second with the source “in” the reactor at startup
Water Tank:	Temperature less than 15° C
Seismic Scram:	Indicator
Shield Tank:	Water level lower than 10 inches from the top
Channel No. 2: (log n)	Period shorter than 5 seconds Power indication more than 6 watts
Channel No. 3: (linear power level)	Power indication more than 6 watts. Linear rotating switch indicator less than 5% or more than 95% of full scale

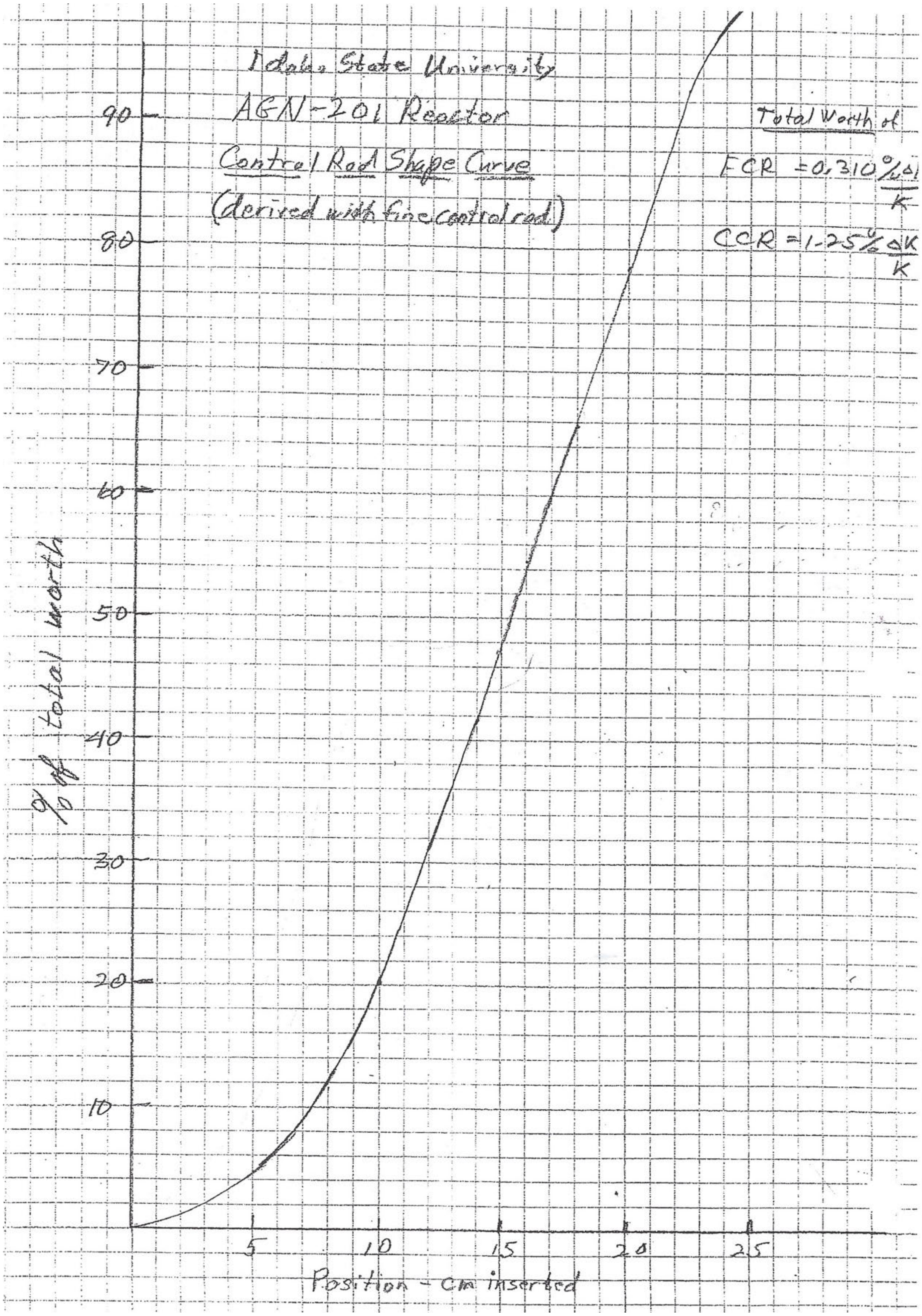


Fig. 4.2-4 Fine Control Rod Calibration Curve

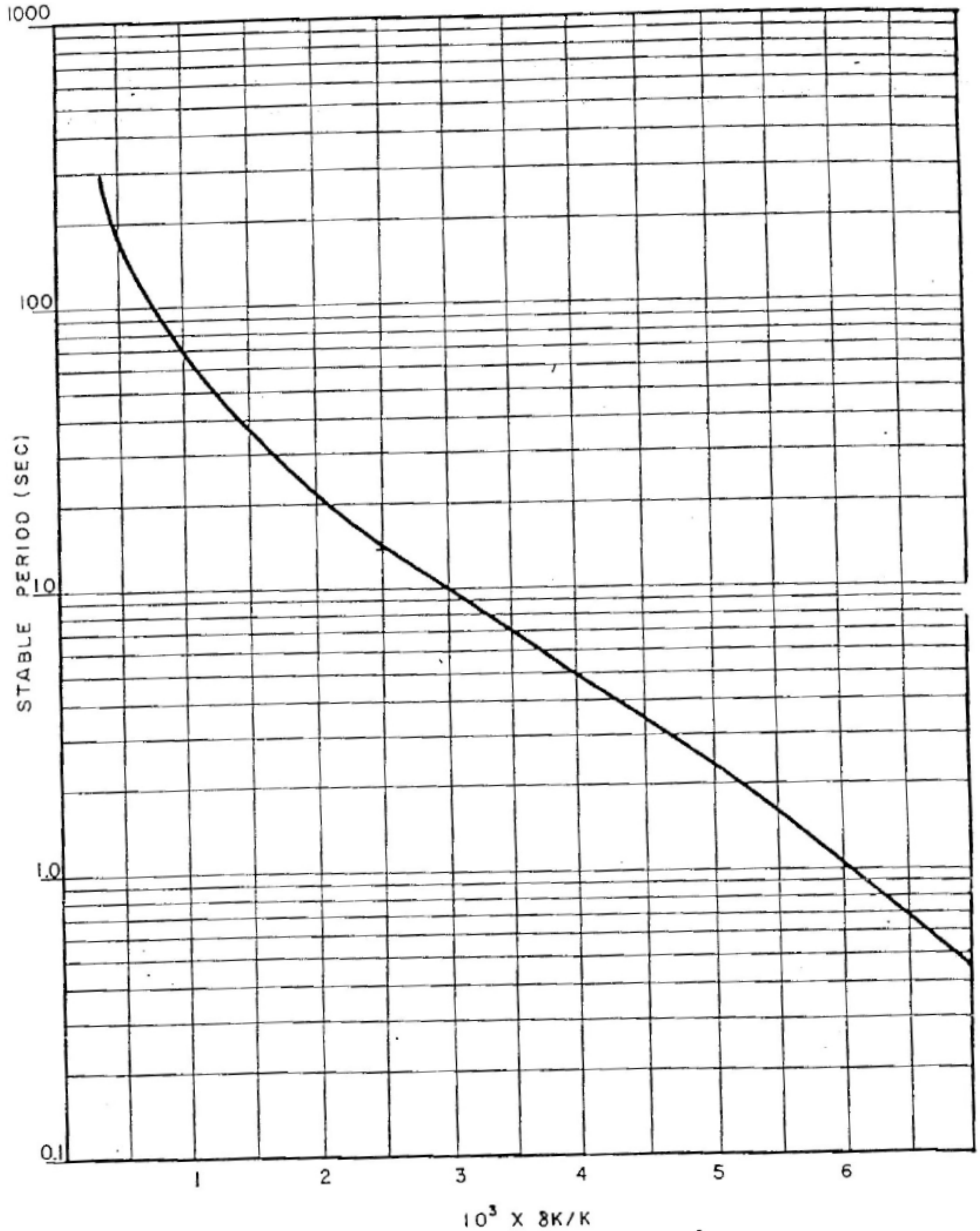


Fig. 4.2-5 AGN-201 Inhour equation

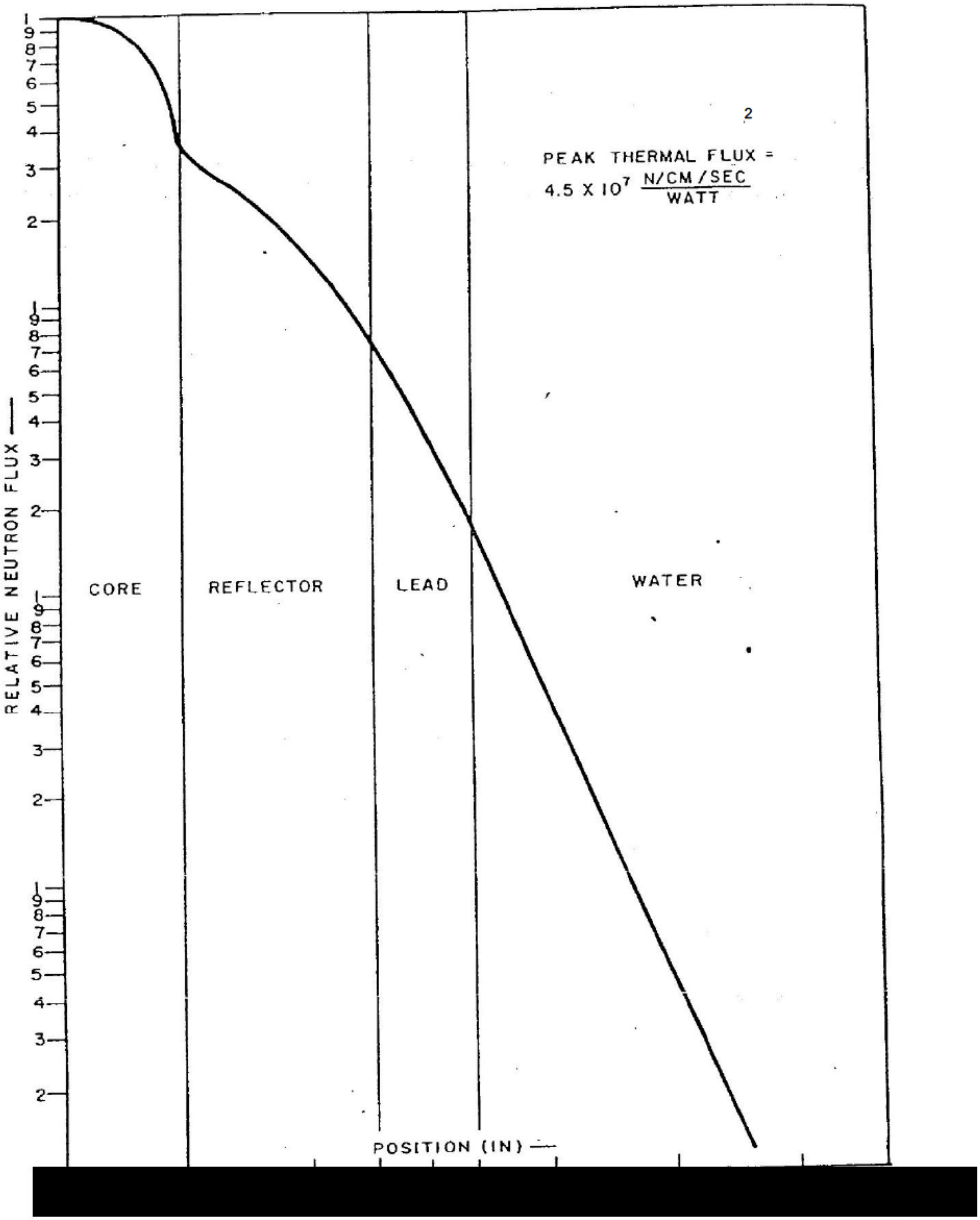


Fig. 4.2-6 Horizontal thermal neutron flux through glory hole at 100 mW

4.3 Control System Upgrade

The major modification made to the reactor and completed in 2020 was to install a newly completed control system that was built using all new components, designed similarly to the original control system using analog logic. The new system was installed in an identical frame to the old console, the frame having been obtained from another university, which had decommissioned its AGN-201 reactor.

The primary modification on the new console was to replace the vacuum tubes, that were in the old console with transistors (or equivalent).

To avoid changes to the main licensing document, the Technical Specifications, it was decided to duplicate all of the current circuitry with equivalent solid-state circuitry. This included replacing the 1950-era relays with modern solid-state relays, in addition to replacing the vacuum tubes.

The new console design changes were vetted with the 50.59 review process in house, and any potential design changes that would affect the 50.59 safety decisions were re-examined and revised accordingly.

It was decided to replace the outdated digital encoders feeding into digital displays for rod position indication, thus replacing the gear system readout for the fine and course control rods. The 50.59 evaluation showed that the electronic light position indicators would be at least as reliable as the original system and would make manual time measurement of steady state period increase more precise. Furthermore, the design basis accident analyzed in this report was unaffected by the operator knowing the rod position indication.

4.4 Control System

4.4.1 Control rods

The AGN-201 reactor has two safety and two adjustable control rods. Three of these, the two safety rods and the coarse control rod, are identical in design although their functions are different. Each contains about [redacted] of ^{235}U and operates in a manner such that reactivity is increased as the rod is inserted. The amount of reactivity each rod controls is nearly proportional to the amount of fuel material contained in the rod. With the same uranium concentration in the rods as in the core, and a total of [redacted] of ^{235}U in each rod, 1.25% of the reactivity is contained in each of the three large diameter rods. The fine control rod is smaller in diameter and is loaded with fuel material for a reactivity worth of 0.31%. Its function is to permit fine adjustment of the power level of the operating reactor.

Figure 4.4-1 shows the design of an active rod which fits into a 5-cm (2-in) diameter hole in the core. The active length of the rod is [redacted] of UO_2 embedded in radiation-stabilized polyethylene identical in composition to the reactor core. This active fuel material is enclosed in two aluminum containers, the outermost providing the fluid seal from the core tank, and the innermost aluminum container sealing the active fuel material in the rod. By this design, a double fluid-tight seal is maintained for the core as well as for the control and safety rods.

For small adjustments of the reactivity of the reactor, the control rods offer a convenient method of adding or removing fuel from the cylindrical volume designed as the "active core". As such, they are designed for use during start-up as well as for small adjustments in available reactivity under normal operating conditions.

The maximum ^{235}U in the rods [redacted] is [redacted] which control 1.25% reactivity. The safety rods contain more than [redacted] of ^{235}U ([redacted]). Thus, the reactor will remain subcritical if any of the large- diameter rods are fully withdrawn from the core.

The safety rods are in the safe or subcritical positions when they are in their outermost, fully-withdrawn position. The total distance of travel is about [redacted] and in the fully-withdrawn position the active fuel in the rod is just inside the lead shield and partially within the graphite reflector. The rods are inserted sequentially by drive mechanism. The release mechanism is constructed in such a manner that if a scram signal is received during insertion, both rods are instantaneously ejected to their outmost positions.

The safety system is a “fail safe” design in that the scram signal deenergizes the holding electromagnets allowing the rods (except for the fine control rod) to be accelerated downward and out of the core by both gravity and spring loading. The spring constant is such that the rods are initially accelerated with a force of about 5 g, resulting in a total ejection time of approximately 120 milliseconds. Each scrammable rod is decelerated by a hydraulic (oil-filled) or pneumatic dashpot during the last [redacted] of travel.

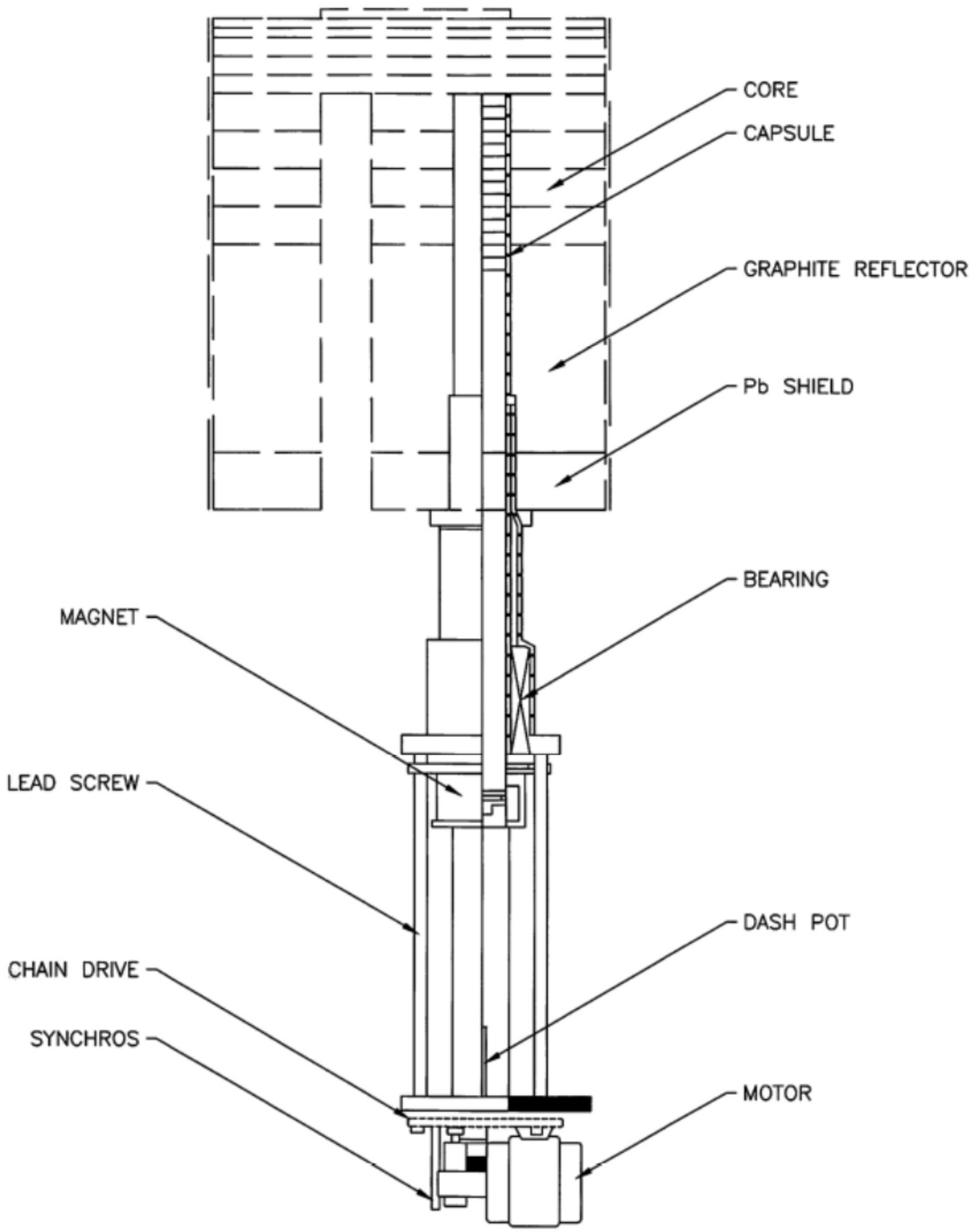


Fig. 4.4-1 AGN-201 control rod and drive mechanism

Both the coarse and fine control rods are driven by reversible DC motors through lead screw assemblies which are controlled by switches at the control console. The maximum speed of travel of the rods is [redacted], yielding a maximum reactivity change of [redacted] for a coarse rod. The positions of both control rods are indicated on the control console. In the event of a scram, the coarse rod and safety rods are automatically ejected to the safe position out of the actual core. At startup, interlocks prevent coarse rod movement unless the safety rods are fully inserted or “cocked”. The safety rods cannot be cocked until the control rods have reached their safe or starting positions. The fine control rod has too little reactivity to be of practical value in the event of a scram. Consequently, on receipt of a scram signal it is driven out of the core by its reversible DC motor at the rate of about [redacted].

4.4.2 AGN-201 Instrumentation System

The instrumentation for the AGN-201 reactor is comprised of three neutron-sensitive monitors: one BF_3 proportional counter and two BF_3 ionization chambers. These neutron detectors are located in the shielding water tank just outside the reactor tank (as shown in Figure 4.4-2). They are placed in water-tight aluminum or plastic cans and can be removed from the reactor through the access cover at the top of the water tank.

The overall control system logic is very similar to that used for other low power reactors (1) and is shown schematically in Figure 4.4-3. The ion chambers are connected through NIM bin amplifier modules to a relay and recorder system. Similarly, the BF_3 proportional chamber is connected to a counting rate meter and, in turn, to the relay system. The relays that form the scram bus may be set to scram the reactor at any predetermined instrument reading. Scrams may also be initiated by the earthquake detector, the water shielding level detector, water temperature detector, and the manual scram button.

A graphic display unit is used to monitor and record #2 and #3 neutron detection device’s outputs.

In the event of an abnormal condition, any one of the three detectors or other safety devices actuates the scram system causing automatic and rapid ejection of both the safety rods and the control rods. Operation of the scram warning light will also occur. The AGN-201 system is designed to be fail safe. In the event of an external power failure, the reactor will also scram. An area radiation monitor will continue to operate in a power failure to provide the reactor operator with continuous radiation readings. The current control console for the AGN is shown in Figure 4.4-4.

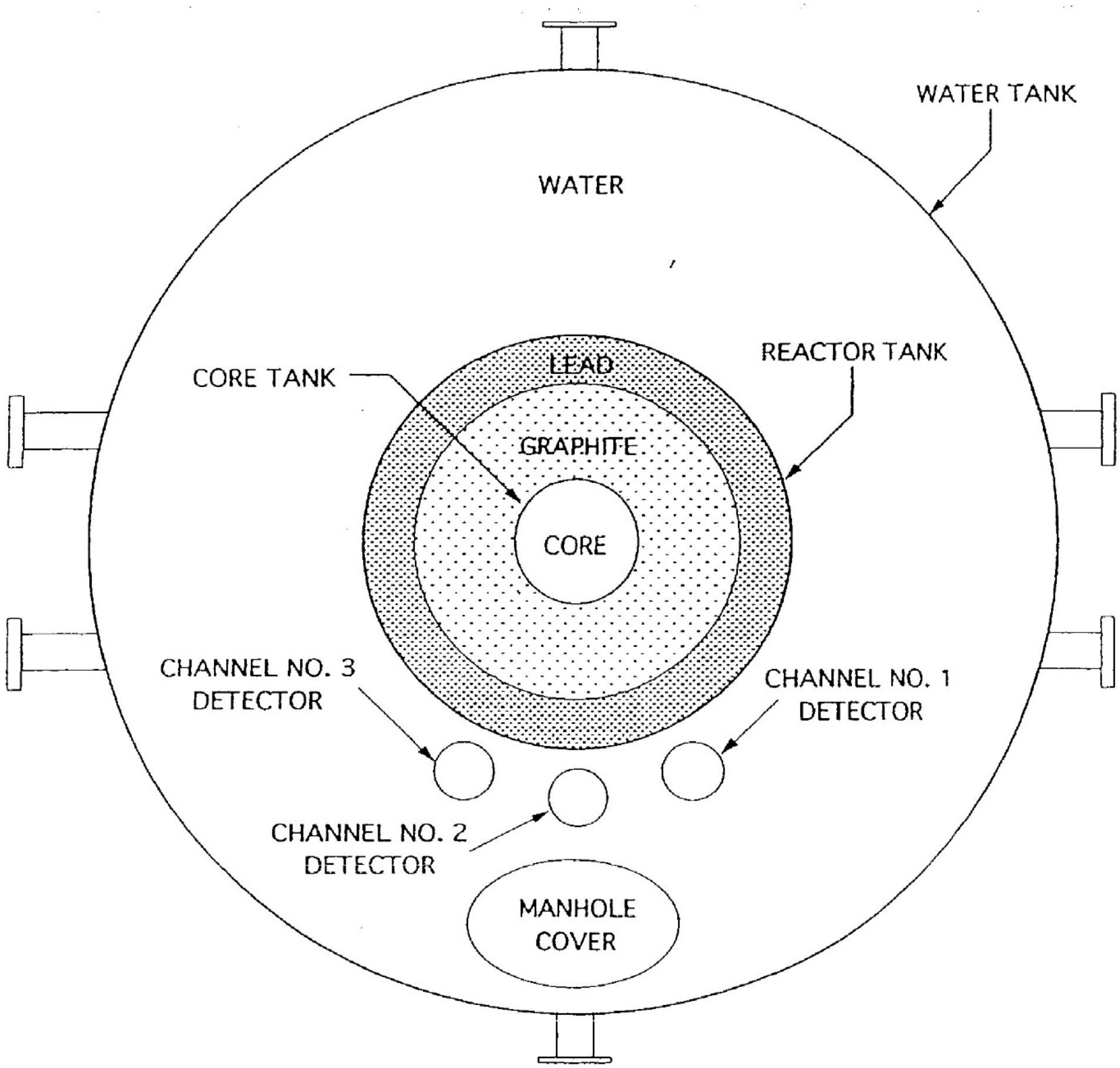


Fig. 4.4-2 Cross section of reactor showing locations of neutron detectors

[redacted]

Fig. 4.4-3 AGN-201 control system

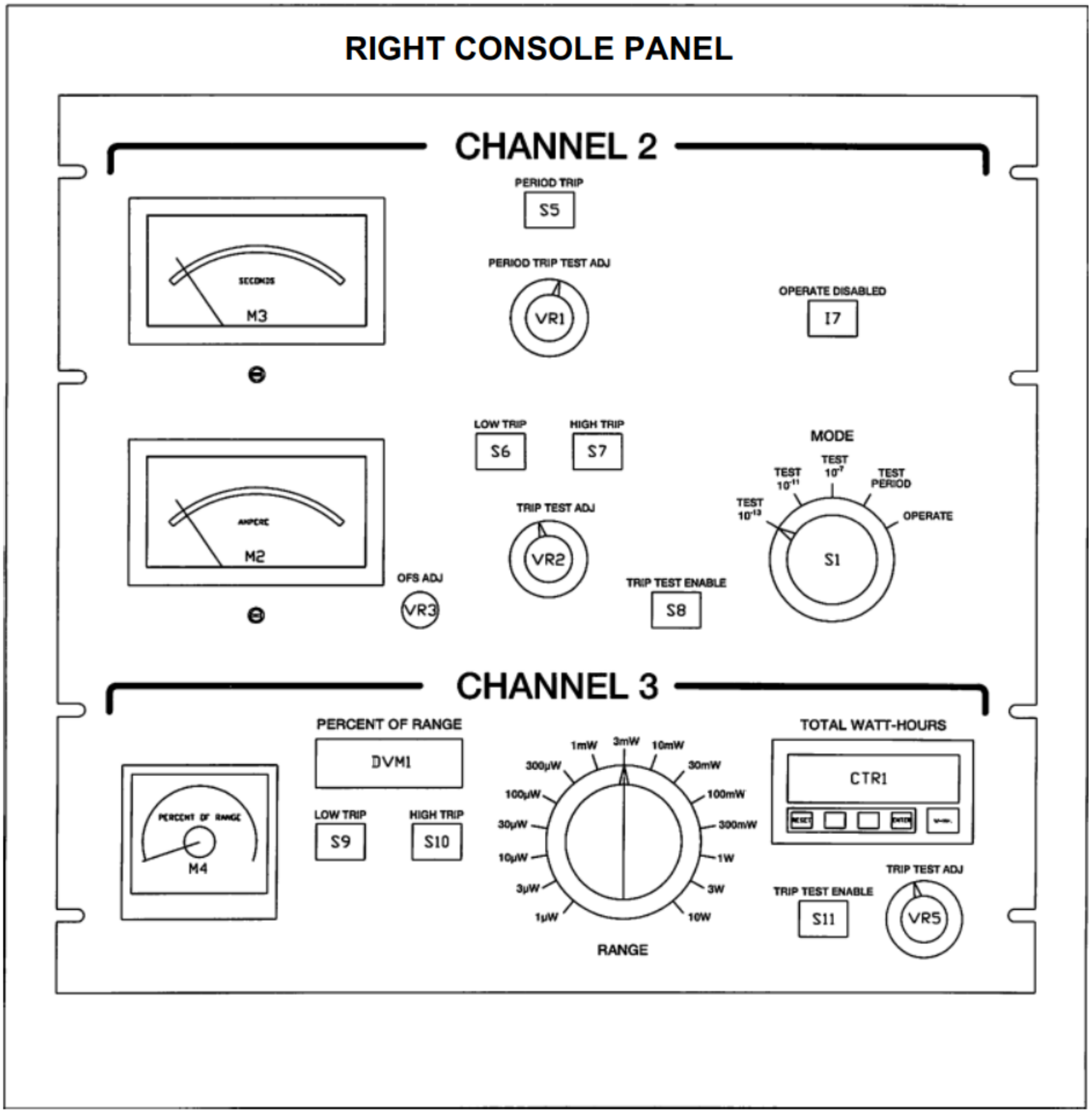


Fig. 4.4-4 Right Console Panel

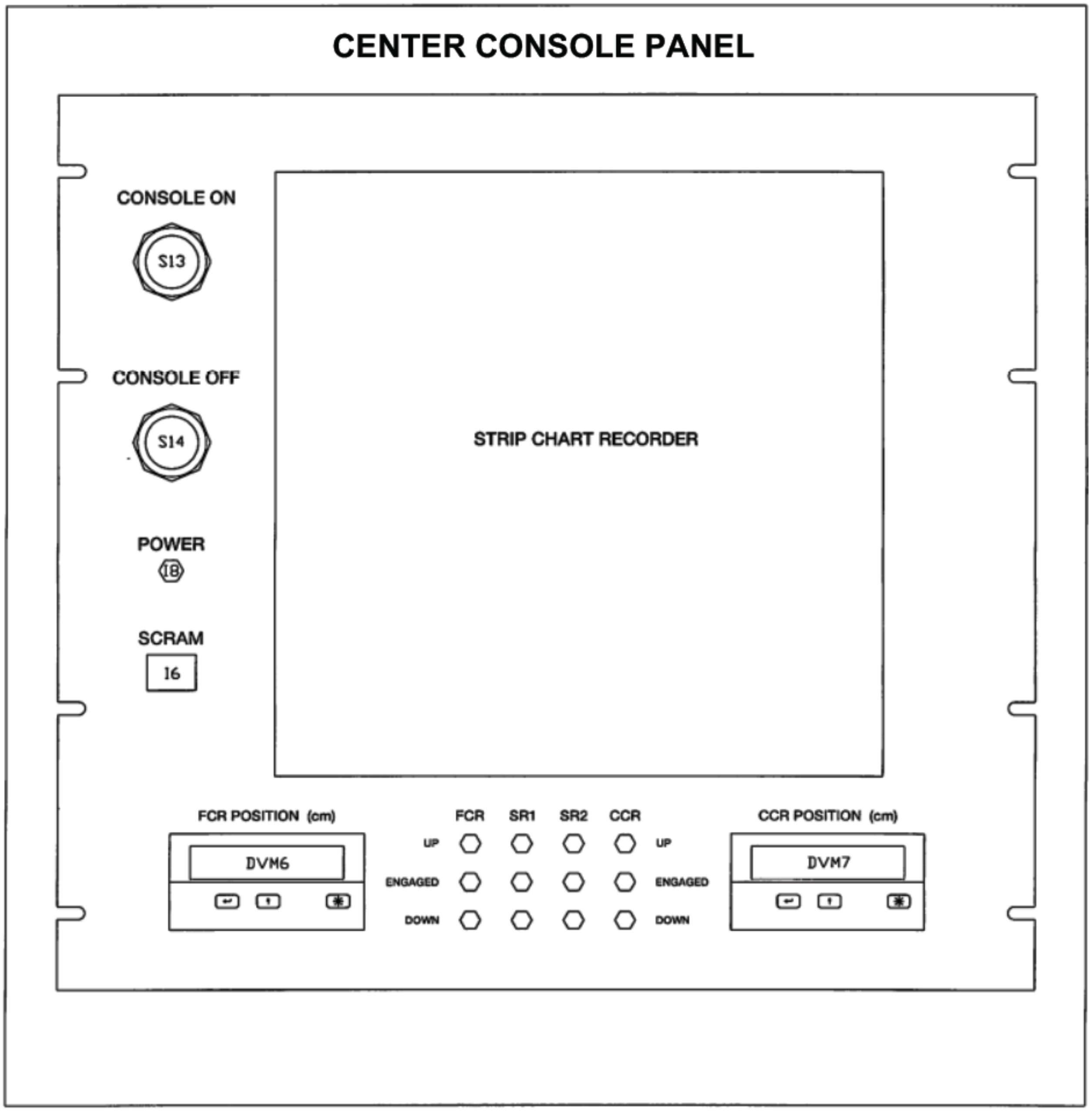


Fig. 4.4-5 Center Console Panel

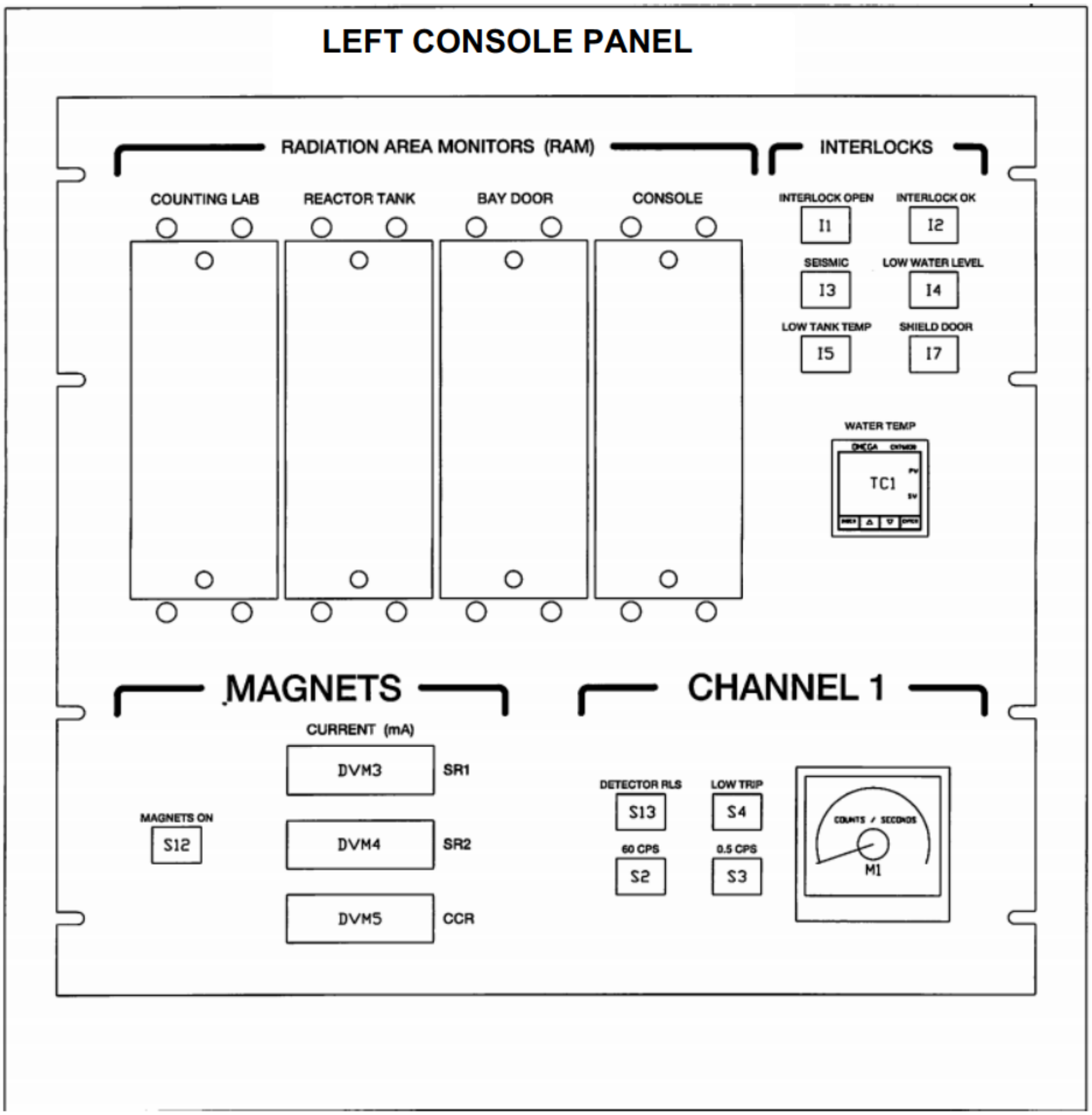


Fig. 4.4-6 Left Console Panel

[redacted]

Fig. 4.4-7 Block diagram for nuclear safety channel No. 1

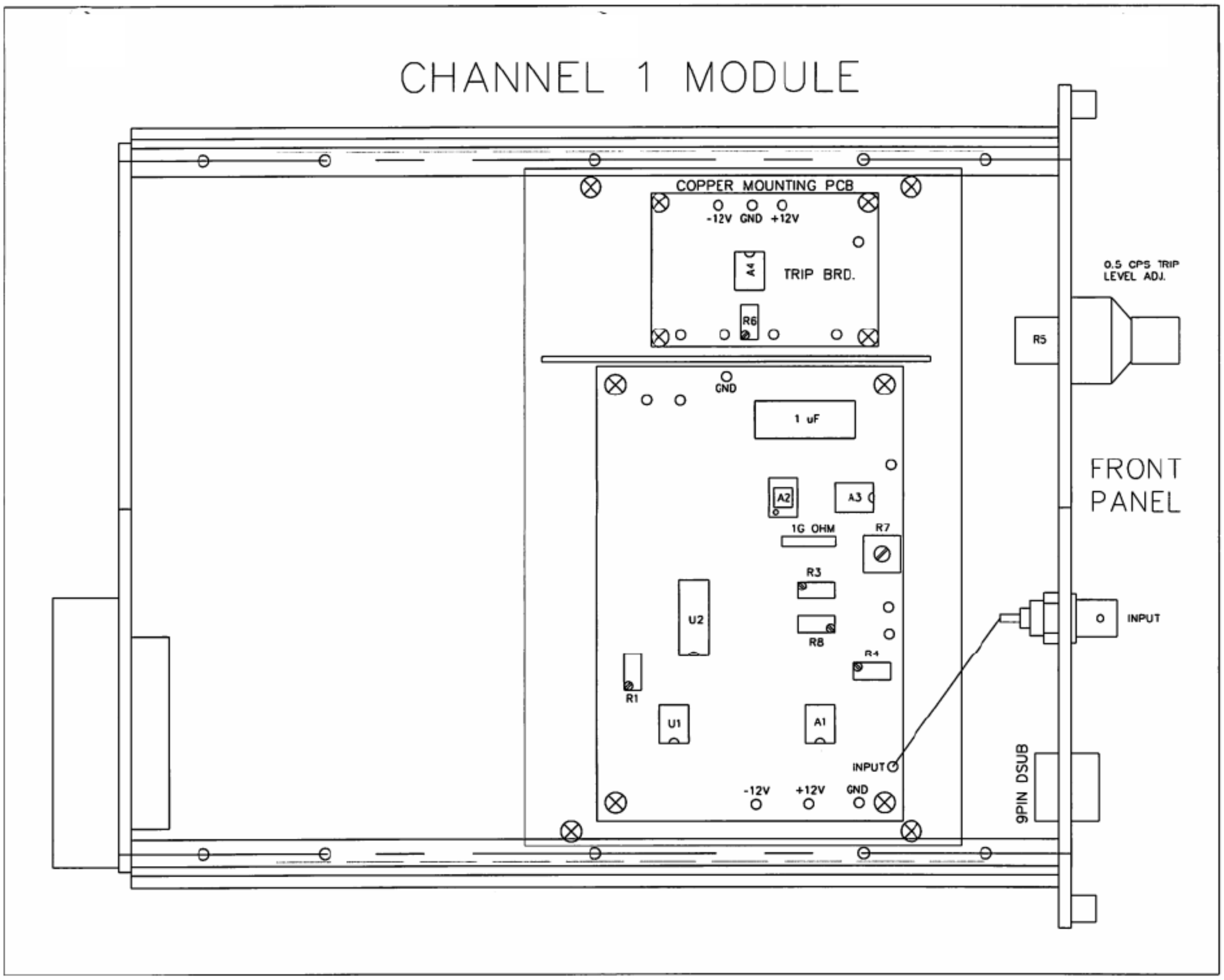


Fig. 4.4-8 Channel 1 Module

<u>CHANNEL 1 - (THE START-UP CHANNEL)</u>

The components for the nuclear safety Channel 1 neutron monitoring system are shown in Figure 4.4-8. The primary purpose of Channel 1 is to provide low level protection so the reactor cannot be started up without a neutron source response being above 0.5 counts/sec.

<u>INTRODUCTION</u>

The CHANNEL 1 block diagram is shown on the print labeled [redacted] in the AGN201 print binder. CHANNEL 1 is made up of the following components: (1) A BF_3 detector, (DET1), located in the water tank outside of the Pb shield at the horizontal midplane of the reactor core. (2) The detectors high voltage power supply, (HV1), [redacted], located in the NIM Bin in the center bay of the reactor operators console, (3) A modified [redacted] pre-amplifier, (PA), located atop the reactor assembly and external to the reactors shielding, and (4), NIM modules located in the NIM bin of the left bay of the reactor operator's console. They are as follows:

(a) The TEST FREQUENCY GENERATOR module, (TFG);

(b) [redacted] Delay Line Pulse Shaping Amplifier, (DLA);

(c) The CHANNEL 1 module, (CH1).

[redacted] PRE-AMPLIFIER

The [redacted] pre-amplifier, (PA), has been modified to allow it to operate at high count rates without saturating. This modification [redacted] The pulse decay time constant of the pre-amplifier was correspondingly reduced from [redacted] The charge gain of the preamplifier was [redacted]

TEST FREQUENCY GENERATOR and CHANNEL 1 MODULE

(1). TEST FREQUENCY GENERATOR module. This module provides 0.5 cps and 60 cps test pulses which are used to test the functionality of the Start-up Channels count rate and trip circuitry located in the CHANNEL 1 module. The 0.5 cps and 60 cps test pulses are selectable via corresponding pushbuttons located on the reactor control console's right front panel. When either pushbutton is depressed the TEST FREQUENCY GENERATOR module opens the path between the output of the preamplifier and input of the OrTec 460 DELAY LINE AND PULSE SHAPING AMPLIFIER and simultaneously injects the selected test signal as an input to the OrTec 460. The 0.5 cps signal provides a reference count rate used when adjusting the 0.5 cps low level trip circuit in the CHANNEL 1 module. Both the 0.5 cps and the 60 cps signals are used to verify the functionality of the CHANNEL 1 module and the calibration of that module's log of count rate circuitry and count rate meter. This module also receives a control input from the mode switch (MODE SW S1), located on right front panel of the reactor control console.

(2). [redacted] DELAY LINE PULSE SHAPING AMPLIFIER, (DLA). This module conditions and processes pulses from the detector's preamplifier and the TEST FREQUENCY GENERATOR module to have the same shape and time constant, then sends them to the input to CHANNEL 1 module.

(3). CHANNEL 1 module. This module provides the log of count rate circuitry that drives the 0.1 to 100k count/sec log scale rate meter located on the left front panel of the reactor control console. The module also provides a low-level trip circuit which is typically set to trip the reactors scram circuitry for count rates equal to or less than 0.5 cps. The 0.5 cps signal from the TEST FREQUENCY GENERATOR module is enabled when calibrating the CHANNEL 1 module's low-level trip. CHANNEL 1 module's Low level trip is sent to the scram module for use in the reactors scram string.

CHANNEL 2 – (THE LOG N AND PERIOD CHANNEL)

The components of CHANNEL 2 are shown in Figure 4.4-9. The primary purpose of CHANNEL 2 is to provide both high and low-level protection on a non-switchable neutron monitoring channel. A secondary purpose is to provide backup power level indication.

INTRODUCTION

The CHANNEL 2 block diagram is shown on the print labeled [redacted] in the AGN201 print binder. CHANNEL 2 is made up of the following components: (1) An ionization chamber, (DET2), located next to (DET1) at the same radial distance from the reactor core. (2) The detectors high voltage power supply, (HV2), [redacted], located in the NIM bin in the center bay rear frame of the reactor operators console, and (3), the CHANNEL 2 NIM module located in the NIM bin in the left bay rear frame of the reactor operator's console.

[redacted]

Fig. 4.4-9 Block diagram for nuclear safety channel No. 2

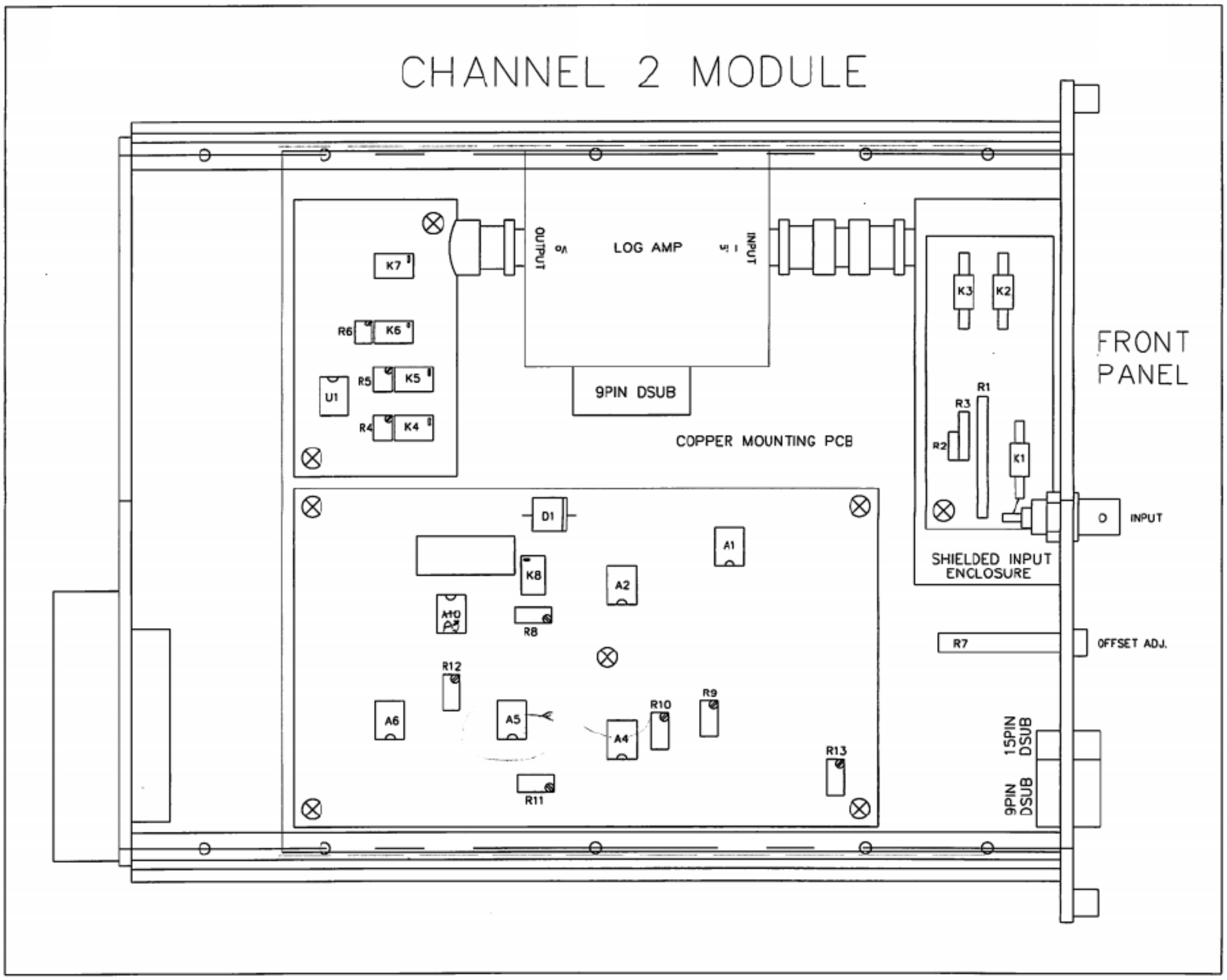


Fig. 4.4-10 Channel 2 Module

FUNCTIONAL DESCRIPTION OF THE CHANNEL 2 MODULE

The CHANNEL 2 module is a unit designed to measure reactor Power Level, based on its proportionality to ionization chamber current; and period, based on the rate of change of that current. The module's log of ionization current and period circuitry drives the CHANNEL 2 log scale current level meter, and reactor Period meter, respectively these meters are located on the upper front panel of the right-hand bay of the reactor's console which provide meter indication of measured ionization current and period over the reactors full operating range.

Reactor power is determined by measuring the magnitude of the ionization chambers output current with the module's logarithmic amplifier, (LOG AMP), which has a logarithmic transfer function, and dynamic range of input current from 10^{-13} to 10^{-6} amperes. The measured Ion chamber current is proportional to power level and this current is displayed on the front panel, 7-decade, log scale meter. This meter, the Hi & Lo level trip circuits, the differentiating amplifier, and graphic display panel are all driven directly by the output of the LOG AMP via a unity gain buffer amplifier.

Period is measured by a differentiating amplifier, whose output, in turn, drives the period meter and the period trip circuit. The CHANNEL 2 module Hi is sent to the scram module for use in the reactors scram string.

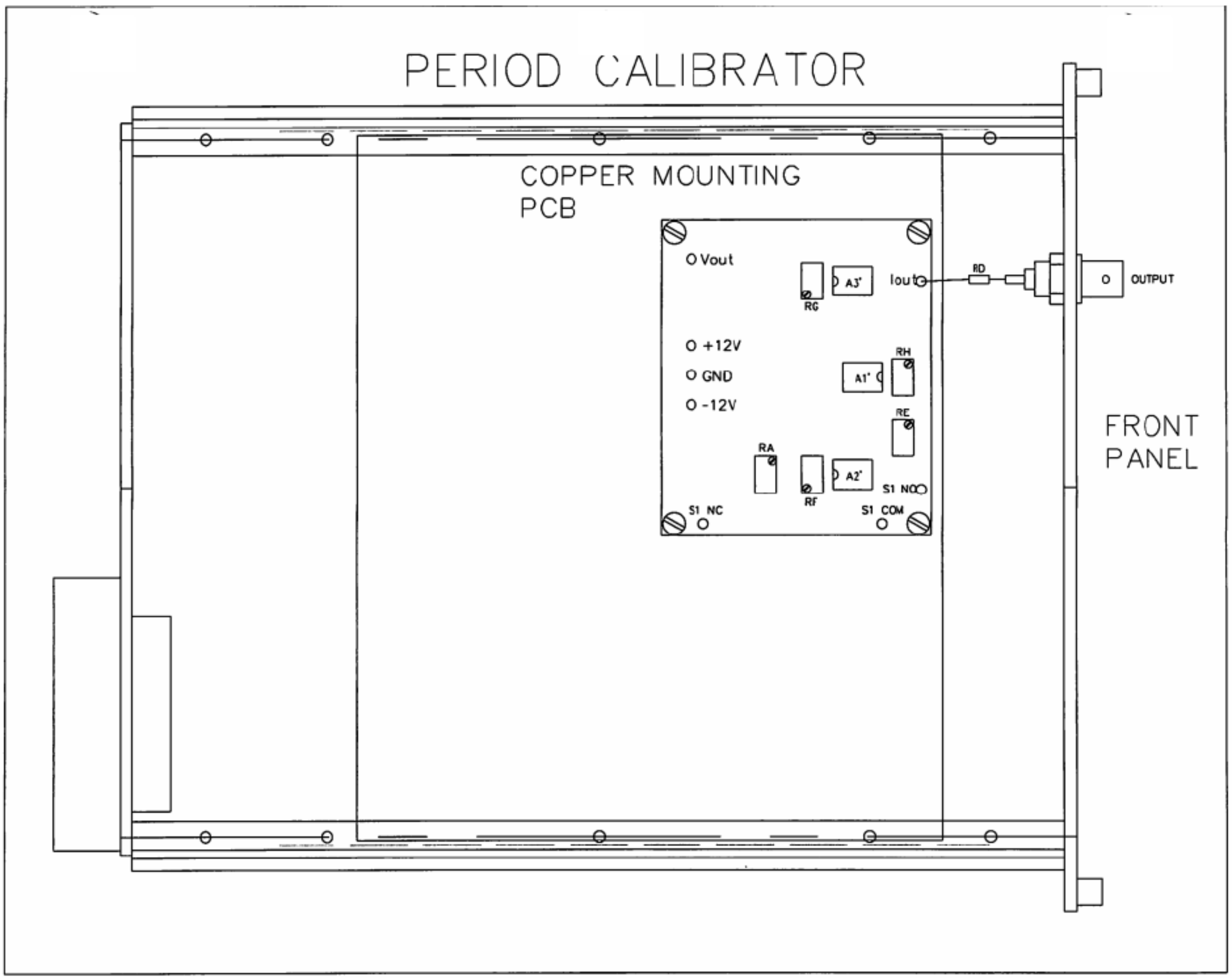


Fig. 4.4-11 Period Calibrator

[redacted]

Fig. 4.4-12 Block Diagram for nuclear safety channel No. 3

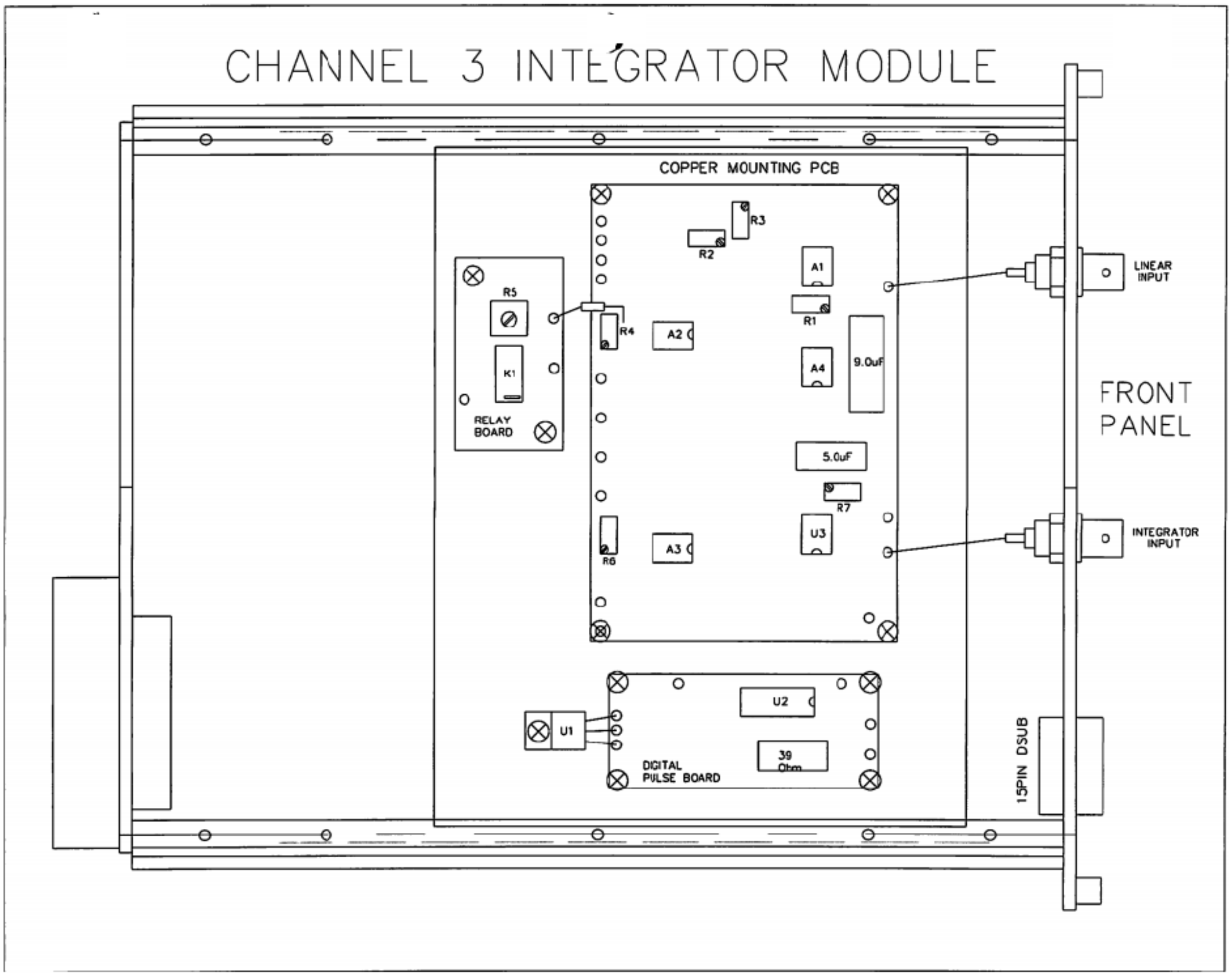


Fig. 4.4-13 Channel 3 Integrator Module

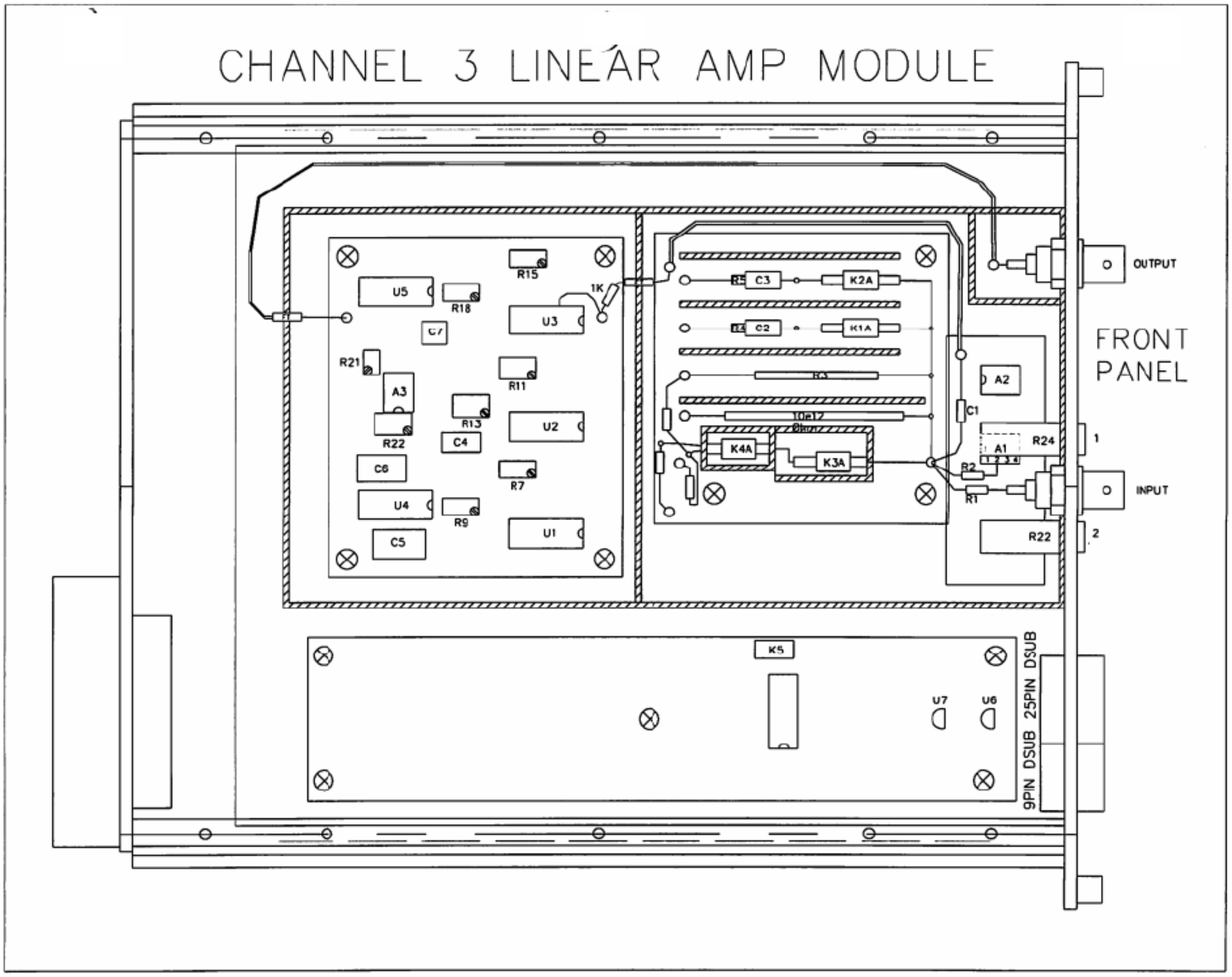


Fig. 4.4-14 Channel 3 Linear Amp Module

CHANNEL 3 – (THE LINEAR CHANNEL)

The components of CHANNEL 3 are shown in Figure 4.4-12. The primary purpose of CHANNEL 3 is to provide both high and low-level protection on a switchable neutron monitoring channel. A secondary purpose is providing power level indication.

INTRODUCTION

CHANNEL 3 is shown in the block diagram of print [redacted], and is made up of the following components: (1) An ionization chamber, (DET3), located next to (DET2) at the same radial distance from the reactor core, (2) An [redacted] high voltage power supply, (HV3), located in the NIM bin in the center bay rear frame of the reactor operators control console, (3) The CHANNEL 3 Linear Amplifier module,(CH3LA), (henceforth termed the LINEAR AMPLIFIER module), mounted within the NIM bin located in the rear of left bay of the control console, (4) The ELECTROMETER RANGE SWITCH module, (ELECT RGE SW), located on the right front operating panel of the control console, and (5), The CHANNEL 3 Power Integrator and Hi & Lo Trip module, (henceforth termed the INTEGRATOR module), (CH3IN), located adjacent to the LINEAR AMPLIFIER module in the reactor control instrumentation NIM bin.

FUNCTIONAL DESCRIPTION OF THE CHANNEL 3 LINEAR AMPLIFIER MODULE

The function of the Linear Channel to provide a precise measurement of Channel 3 ionization detector through 15 calibrated ranges as the reactor goes up in and down in power. An electrometer range switch, (ELECT RGE SW), is located on the right front operating panel of the control console and is housed in an aluminum case mounted on the back side of the control panel. It is a 4 wafer, 15-position rotary switch used to select one of 15 gain ranges from 1micro-watt to 10 watts. Each range has its own adjustment for calibration. Three wafers are used for the linear amplifier module and one wafer is used for the integrator module. All four wafers rotate together on a common shaft to each of the 15 positions. Two mechanical stops, one on each end of the rotation, prevents over rotation. The final amplifier in the linear module has a global gain adjustment to calibrate the output if changes need to be made when adjusting for actual power level obtained from irradiated gold foils. This output is sent to the Integrator module.

FUNCTIONAL DESCRIPTION OF THE CHANNEL 3 INTEGRATOR MODULE

Power measurement is based on the proportionality between ionization chamber current and reactor power level. Voltage from the LINEAR AMPLIFIER is buffered in the INTEGRATOR module. The buffer's output is sent to two front panel meters: (1) a 'Percent of Range" DVM, and (2) a 'Percent of Range' analog meter. It is also sent to graphic display panel (recorder). The buffered output is also sent to Hi & Lo trip circuitry in the module and provides Hi & Lo level trip levels to the reactors scram string. Hi and Lo level trips are adjustable, and are set to trip 95% of selected range for the Hi level trip, and at 5% of (but no higher than 6W) selected range for the Lo level trip. When operation is in the 10W range, a Hi level modifier signal is sent to the Hi level trip circuit to lower the trip to about 5W maximum, for licensing requirements.

The buffered voltage is also sent to voltage-to-frequency, (V/F), circuitry that converts the output voltage from the LINEAR AMPLIFIER module to a pulse rate scaled for proportionality to the reactors power level. Integrated power is determined by counting the output pulses from the V/F converter using a front panel mounted digital counter, (WATT-HRS). This integration is usually used for determining total energy produced over a period of time, such as one year.

RECORDER

A [redacted] digital strip chart recorder (GDP) is located in the center panel of the reactor console. One recorder "pen" (input), is connected to and registers the output signal from the logarithmic amplifier of CHANNEL 2. Another recorder "pen" is connected to and registers the output signal from the linear amplifier of CHANNEL 3. The recorder has a touch screen for programming all parameters pertaining to the display of the waveforms.

[redacted]

Fig. 4.4-15 Simplified circuit diagram of Safety Chassis

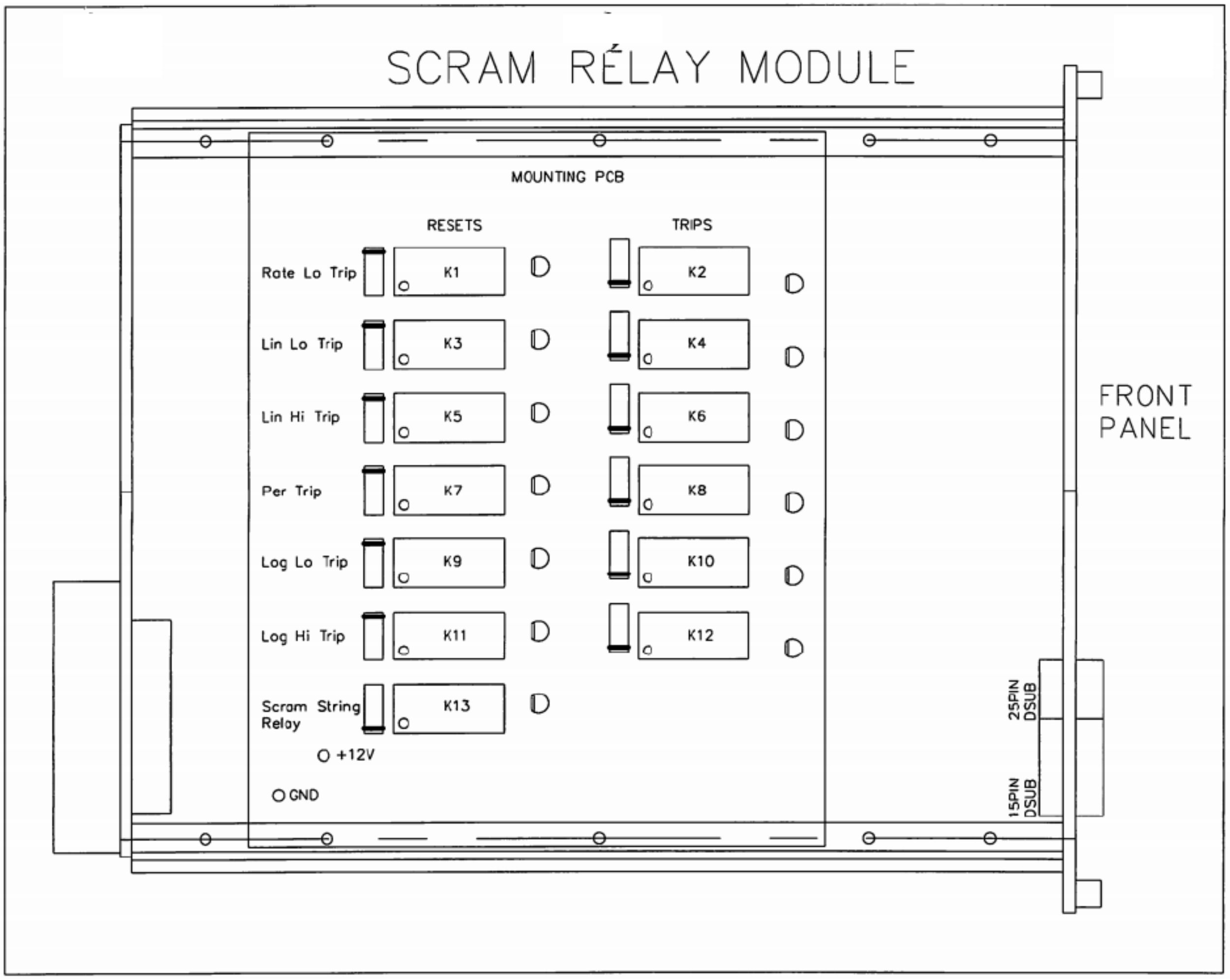


Fig. 4.4-16 Scram Relay Module

4.4.3 The Safety Channel

There are two modules: (1) the SCRAM RELAY MODULE, (2) the INTERLOCK AND MAGNET MODULE, and 2 separate drawer chassis, (1) a motor RELAY CHASSIS, and (2) the 24V DC POWER to power the relays. A simplified circuit diagram is shown in Figure 4.4-15. It is most helpful to follow the operation of these modules by looking at the referenced schematics.

INTRODUCTION

The AGN201 requires very specific conditions to be met for start-up and operation. If these conditions are not met, the reactor will stay in a ‘scram’ condition until all operating parameters are within their normal ranges. Normal operation of the reactor to produce power is to have all external interlocks satisfied, all module scams cleared, a specified source rod inserted, the MAGNETS ON switch depressed and rods sequentially inserted into the reactor vessel by four motorized mechanical drives. The four mechanical drives are commanded from switches on the center console control desktop panel for both travel direction and speed. Three of the four drives incorporate electromagnets that when energized, will maintain a magnetic force that will cause the fuel canisters to rise with the drive carriages. Safety Rod 1 (SR1) is raised to its locked position and when it reaches that position it stops and enables Safety Rod 2 (SR2) to be raised. SR2 is driven to locked position and when it reached that position it stops and enables the Course Control Rod (CCR) to be driven up to its position. Finally, the Fine Control Rod (FCR) can be driven up. Reactor power is then controlled by the movement of the course control rod and the fine control rod by driving those two rods up and down. Since SR1 and SR2 are locked in their inserted position they can no longer be moved down except by a scram signal that will cause the magnets to release the fuel canisters and also drive all four carriages to their initial positions out of the reactor.

A ‘scram’ causes the immediate interruption of magnet holding currents of the two safety control rods and the coarse control rod which will eject the fuel cannisters downward out of the reactor vessel to their initial position. The ejection rate is supplemented by the stored energy of compressed springs in the drive mechanisms during rod insertion. Dashpots are incorporated to soften the landing of the canisters once ejected. A ‘scram’ also initiates an ‘Automatic Carriage Return’ (ACR), condition where the drives are commanded to be driven out of the vessel and returned to their initial position terminating all reactor operations. Depressing the large red SCRAM button on the console’s center control desktop immediately terminates reactor operations.

FUNCTIONAL DESCRIPTION OF THE SCRAM RELAY MODULE

The double wide Scram Relay module shown schematically in drawings [redacted] of the AGN201 Reactor Control Console NIM Bin. The Scram Relay module contains the relays whose contacts signal the reactor control system and reactor operating personnel that a scram due to excessive reactor Period or "out of limit" count rate and/or power level has occurred.

The Scram Relay module circuitry is shown schematically on the drawing titled AGN: SCRAM RELAY MODULE (Figure 4.4-16). The module is comprised of the scram string relays associated with Channels 1, 2 and 3.

Channel 1 Low Count Rate Trip Relays

Following insertion of the start-up source and obtaining a Channel 1 count rate > 0.5 cps, Relays K1 and K2 are energized by depressing the console Channel 1 Low Trip reset pushbutton which closes the path between wires 061 and 062, thus energizing relay K1, given that wire 063 from the output of the channel 1 low count rate trip is at a -11 volt level. (Using wire numbers help clarify the correct signal paths.) Energizing K1 closes the Channel 1 scram bus contacts 5 and 6. K1 contacts 7 and 8 also close, thus energizing relay K2 whose contacts 5 and 6 close the path between wires 061 and 062. Contacts 8 and 9 of relay K2 open, turning off the Channel 1 Low Trip indicator light. If for any reason the count rate drops below 0.5 cps, the voltage on wire 063 from the channel 1 low count rate trip circuit will jump to approximately 0 volts causing relays K1 and K2 to de-energize, thus signaling a scram by opening contacts 5 and 6 of scram relay K1. Contacts 8 and 9 on relay K2 close when K2 de-energizes, thus turning on the lamp within the Console front panel trip reset pushbutton switch, thereby giving indication that a low count rate trip has occurred.

Channels 2 and 3 Trip Relay Circuits

The Channel 2 and 3 scram relays K3 & K4, K5 & K6, K7 & K8, K9 & K10, K11 & K12 have wiring, pushbutton engagement and lamp illumination control identical to that previously described for K1 and K2 of Channel 1. The channel 2 and 3 trip circuits that drive their associated relays also have the same output behavior as that of Channel 1 in that their output levels are approximately 0 volts when tripped and -11 volts when untripped. Contacts 5 and 6 of relays K1, K3, K5, K7, K9 and K11 are series connected, driven by the +12 Vdc NIM bin power supply, and terminate on the coil of the Scram String Relay K13. Relay K13 is energized when all scram conditions have been cleared and their associated trip relays (K1 thru K12) reset. The normally open contacts (5 and 6) of K13 are routed through connector SR2 (pins1&2), to be connected in series with the reactor scram string.

[redacted]

Fig. 4.4-17 Block diagram of reactor interlock system

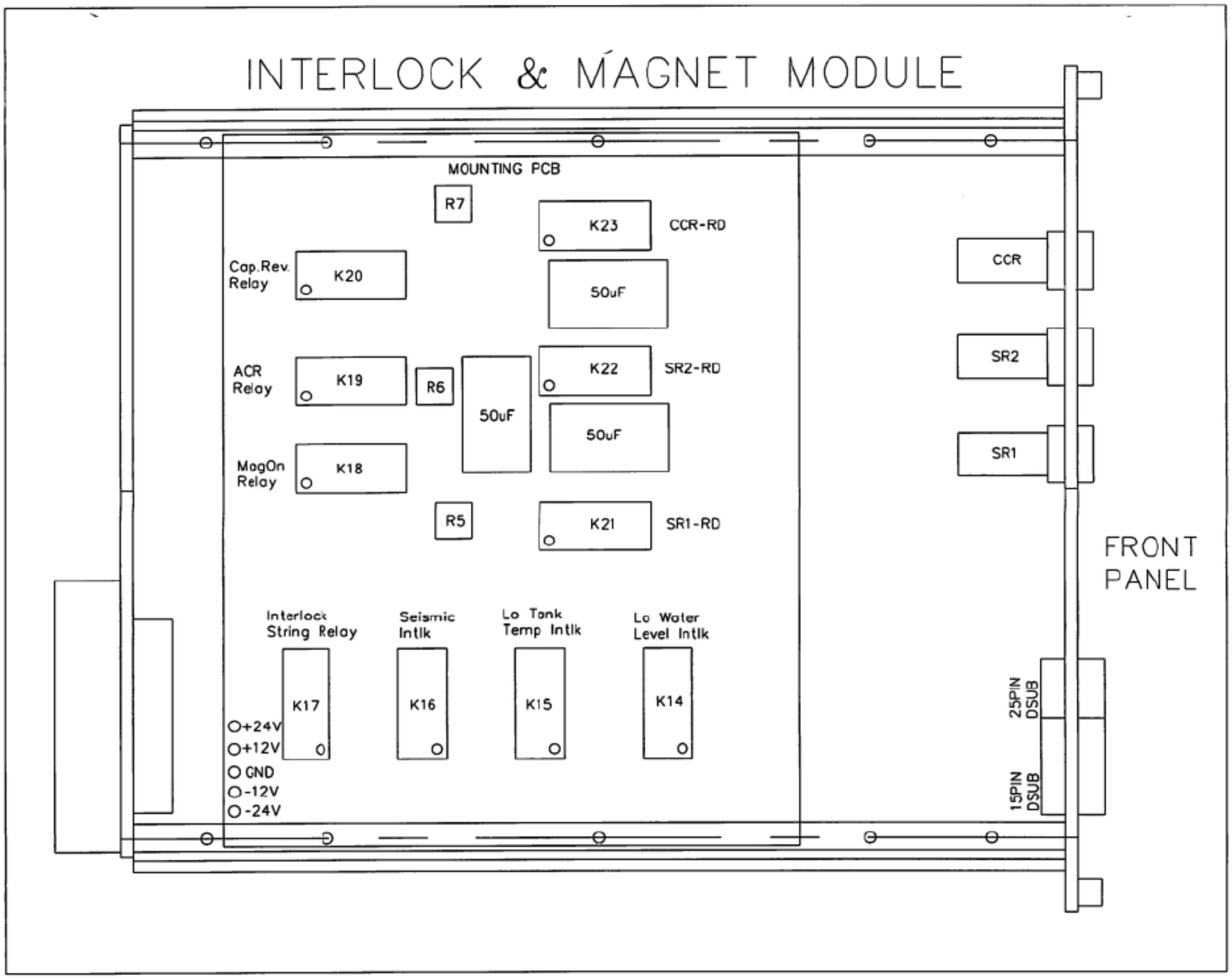


Fig. 4.4-18 Interlock & Magnet Module

4.4.4 The Interlock and Magnet Module

<u>FUNCTIONAL DESCRIPTION</u>

The double wide Interlock module, shown schematically in drawings 4.4-17 and 4.4-18 ([redacted] the AGN201 Reactor Control Console NIM Bin. The Interlock module generates the three INTERLOCK module scram signals: (1) Tank Water Level Lo; (2) Tank Water Temperature Lo; and (3) detected Seismic event, indicated from external sensors mounted in the vessel or close to it.

(1). The shield water level switch, which consists of a water-tight microswitch and an actuator connected to a float bob, opens the interlock if the shielding water level is less than the minimum allowed level.

(2). The low-temperature switch, which has been calibrated to open at 15^{o} C, is a simple bimetal thermal switch. As the temperature of the switch reaches this 15^{o} C level, the bimetal strip which makes up one side of the switch bends, breaking the interlock circuit. Bending action takes place because the two different metals used in the strip have different linear coefficients of thermal expansion.

(3). The earthquake switch consists of a steel ball mounted precariously on two terminal strips to maintain electrical continuity. If the reactor receives a physical shock resulting in a lateral displacement, the ball will move and break the electrical contact being made.

Any event of the above three sensors that generate an open in the interlock chain, will require corrective measures by the operator.

An open interlock lights a panel indicator on the console's left front panel and tells the operator which fault has happened. There are two other indicators: Interlock Open and Interlock OK. Any fault in the interlock string that is not made will cause the Interlock Open indicator to light. When all interlocks are made, the Interlock OK indicator is on. The Interlock module controls the merger of those signals with the SCRAM module signals, (CHANNEL 1 count rate Lo trip, CHANNEL 2 Hi Level and Period trip, and CHANNEL 3 Hi & Lo Level trip), the ROD INTERLOCK switch on the reactor skirt, the MODE switch on the reactor console's right hand front panel, and the Reactor Console SCRAM button. The resulting Scram C signal, generated within the INTERLOCK module, controls the application of power to the electromagnets.

When the Interlock OK indicator is on, the Mode switch is in the Operate position with the Operate indicator on, the SCRAM button is up and all module trips are reset and indicators off, +12V is applied to the Magnets On toggle pushbutton switch which energizes the Magnet On relay that sends power to the three control rod magnets which allow the fuel canisters to rise with the rod drives. Only the Magnet On relay can energize the electromagnets. Any broken interlock or any scram event will cause the immediate de-energizing of the Magnet On relay and the three fuel canisters will drop. All magnets are current limited and fuse protected. Upon de-activation of the relay, a capacitive circuit is switched in to quickly deplete the magnetic field in the magnets to enhance turn off time. At the same time a signal is sent to the relay chassis to cause an Automatic Carriage Return to withdraw the rod drives out of the reactor. Magnet current, in milliamps, is shown on three separate DVMs located on the console's left front panel. These currents are directly measured by a voltage across a precision resistor in each magnet's wiring paths. Current indications from one to another will vary slightly due to wiring differences in the magnet assemblies and the length of time the magnets are on which cause internal heating of the windings till they reach thermal equilibrium.

There are four lighted toggle pushbuttons on the console's center desktop directly above the motor direction switches that are used for additional testing. When the reactor is operational, these switches allow for the dropping of individual rod fuel canisters to permit rod "worth" measurements.

FUNCTIONAL DESCRIPTION OF THE RELAY CHASSIS

The Relay Chassis is shown schematically in drawings [redacted] The Relay Chassis is located in the control console's top right front drawer below the desktop. It houses multiple relays to control the movement of four independent motorized drives, axes or carriages. Four double pole, double throw, center off lever switches (Motor Direction Switches), plus two speed control toggle switches mounted on the console's center desktop panel, are used by the operator to insert the four control rods into the reactor. The four control rods are: Safety Rod 1 (SR1), Safety Rod 2 (SR2), Course Control Rod (CCR) and Fine Control Rod (FCR), and move the attached fuel canisters up and down through a distance of [redacted] The two speed control toggle switches allow for high or low speed movement of the CCR and FCR for fine positioning during reactor operation.

All switch inputs energize relays mounted on the side panels of the relay chassis to control the up/down travel direction of each control rod assemblies, and the speed selection of CCR and FCR. The outputs of these relays energize a combination of power relays, mounted on the relay chassis base plate, to applied +24VDC to the respective drive motor. The motors in use are +24VDC, series connected, meaning the armature coil and the field coil of the motor are in series. Reversal of travel direction is accomplished by reversing the polarity of only the armature. Large power resistors are also installed on the relay chassis mounting plate to limit the maximum current allowed for each motor. Each motor is fuse protected for both armature directions of rotation. A lower limit switch (LLS) and upper limit switch (ULS) is provided for each axis. The limit switches serve three proposes: (1) cuts motor power delivered to activated motor at the end of travel distances, (2) allow interlocking of SR1 and SR2 rods and (3) gives status to the operator via LED indication on the console's center panel. Three colored LED indicators per control rod helps the operator in the rod insertion process.

When the reactor is SCRAMMED, the fuel canisters are automatically ejected from the reactor. A signal is also sent to a power relay in the relay chassis that forces all motors to drive at the highest speed, in the reverse direction, to their initial position out of the vessel. Only when all four axes have returned to their initial positions and have activated their respective lower limit switches can any motor operations return.

A separate subsystem is used to display rod position for CCR and FCR in centimeters. Digital rotary encoders are installed on these two drive carriages to give the operator precise location information of the rods. Encoder display units are located on the console's center panel below the graphic display panel (recorder).

FUNCTIONAL DESCRIPTION OF THE +24VDC POWER SUPPLY

The +24VDC POWER SUPPLY is shown schematically in drawing [redacted]. The 24 Volt DC power supply is located in the Control Console's bottom right front drawer. The power supply has three sections: (1) One used to power the reactor safety rods SR1, SR2, and CCR, (2) a separate adjustable +24VDC power supply used only for the FCR rod, (3) a regulated +24 VDC output to power limit switches and the relays in the chassis relays. All three sections are fuse protected. The two adjustable power supplies utilize variacs for adjustment purposes. Each of three sections have a power transformer and full wave bridge rectifiers to make the dc voltages. Only the regulated +24VDC section utilizes filtering and a three terminal voltage regulator for relays and LEDs voltages in the relay chassis system.

FUNCTIONAL DESCRIPTION OF THE 120VAC POWER SYSTEM

The 120VAC POWER WIRING is shown schematically in drawing [redacted]

Power for the reactor console is derived from the 120VAC Facility power. The incoming power is conditioned by a fused, isolation transformer to limit any line noise coming in on the buildings power system. The output of the isolation transformer is sent to a six-outlet power strip mounted in the console's right bay rear frame. This power strip is unswitched and is on all the time the facility power is on. This strip sends power to a NIM bin rack, located in the console's center bay rear frame, used to house the three high voltage power supplies needed for the three detectors. They need to be powered from 120VAC even when the console power is off. This needs to be remembered when working on the detectors if one thinks the detectors are off when the console power is off. Another outlet sends 120VAC to a circuit to control console power. A key controlled, rotating power pushbutton energizes a 120VAC power relay used to send ac voltages to the various components in the console. Distribution is done by fused DIN rail mounted modules, located on the console's center bay floor. When console power is switched on, another six-outlet power strip, located in console's center bay rear frame, is energized to power the graphic display panel, the NIM bin that holds all the instrumentation NIM modules for the system electronics. When the console power switch is shut off this outlet strip is turned off. If the facility loses ac power the fuel canisters are dropped out of the reactor and the detectors high voltage power supplies are turned off. When the power is restored the control rod drive carriages will return to their initial starting positions. The high voltage power supplies will need to be slowly adjusted to their proper voltages. All scram indicators will need to be reset.

5 SAFETY ANALYSIS

Information in this section includes an analysis of the maximum credible reactivity accident, as well as consideration of radioactive fission product gases, shielding and radioactive effluents, etc.

5.1 General

The following section describes the design features of the ISU AGN-201 reactor and Lillibridge Engineering Laboratory building which ensure that the reactor can be operated under the specified conditions with no hazard to the health and safety of the operators, other occupants of the LEL, or to the general public. Also considered are the effects of conceivable accidents due to component malfunction, human error, or force majeure.

The reactor is located in a medium-sized building on the ISU campus with an estimated daytime occupancy of between 100 to 200 people during the academic year. Thus, an attempt has been made to eliminate or reduce as many of the normal potential nuclear hazards as possible. Emergency procedures for the AGN facility are given in the facility Emergency Plan.

The primary hazard is the possible over-exposure of personnel to radiation. Such over-exposure may occur in any of all of the following ways: chronic exposure to relatively low radiation levels; acute exposure to high radiation levels from sealed sources; acute exposure to elevated radiation levels as a result of an inadvertent power excursion: and exposure to, and possible inhalation and/or ingestion of, uncontained radioactive fission products. The purpose of this chapter is to define and evaluate these hazards and to discuss the various safety features of the AGN-201 reactor. The hazards set forth here have been documented and evaluated by personnel from this facility and other AGN facilities, still licensed and decommissioned. An NRC evaluation of the hazards associated with operation of the AGN-201 reactor is given in Docket F-32.

Under normal operating conditions, the tank and concrete shielding are sufficient to protect personnel from undue exposure to radiation. Precautions have been taken through the scram systems and the administrative limit on excess reactivity to ensure that a nuclear excursion does not occur. However, if this improbable accident does happen, a thermal safety fuse is utilized to minimize consequences and prevent recurrence of the event. In addition, sufficient shielding is provided to protect nearby personnel from serious exposure. The core is designed to minimize the escape of radioactive gas during any accident. The hazards during and following reactor operations are discussed in detail. These are those that occur from the normal operating conditions at a power level of 5 watts and the accident conditions resulting from insertion of all available devices at the lowest operating temperature of 15°C.

5.2 Reactivity Considerations

Reactivities of core components have been given in Table 4.2-1. The insertion of [redacted]

The estimated reactivity worth of each of the control rods is given in Table 4.2-1. The total worth of all rods is about 4.0% $\Delta k/k$ (\$5.41) which gives a shutdown margin of 3.35% $\Delta k/k$ (\$4.53) over the maximum allowable excess reactivity loading of 0.65% $\Delta k/k$ (\$0.74). With the maximum worth rod stuck in the core, the shutdown margin is about 2.2% $\Delta k/k$ (\$2.97).

5.3 Radiation and Shielding

5.3.1 Shielding

The standard radiological tank of the AGN-201 is designed to allow continued access to the vicinity of the reactor at power levels up to 0.1 W. Additional shielding for operation at power levels up to 5 W has been provided by a concrete wall constructed of dense [redacted] concrete blocks and [redacted] barytes concrete blocks. The blocks are held to close dimensional tolerance and stacked in such a manner that voids in the completed wall are at a minimum. Around the beam ports and glory hole, high density blocks are used between [redacted] and [redacted] above the base of the reactor. The use of these blocks further reduces radiation levels near these areas. Overhead shielding is provided by [redacted] that may be opened to permit access to the thermal column when the reactor is shutdown or operating at low power.

As detailed in the amendment for 5-W operation for Aerojet-General Nucleonics, dated 11 February 1957, and on file with the Commission in Docket 50-32, a 45.7-cm (18-in) additional concrete shield wall was sufficient to maintain subtolerance radiation levels (i.e., less than 10 mrem/hr) external to the shield when operating at 5 W. Subsequent analysis by Aerojet-General Nucleonics indicated that [redacted] of ordinary concrete shielding was sufficient. The essential features of the ISU AGN-201 shielding are shown in Figures 5.3-1 through 5.3-3.

The radiation levels associated with 5-W operation of the ISU AGN-201 (peak thermal flux of 2.5 x 10^8 n/cm^2-s) have been extensively measured. Although operating surveys of the ISU AGN-201 show that at 5-W power there are no areas on the reactor floor outside the concrete shield where the total radiation exceeds tolerance levels (i.e., greater than 10 mrem/hr), nevertheless, access to the reactor floor is restricted.

At the maximum operating power, the shielding from [redacted] of lead and [redacted] of water and [redacted] of concrete is sufficient to reduce the equivalent dose rate to personnel next to the reactor concrete shield to less than 10 mrem/hr (0.1 mSv/hr). A nuclear excursion generating a total energy of 5.8 MJ, although impossible to achieve unintentionally without additional fuel, would result in a total radiation dose of 3.2 rem (32 mSv) to a person standing next to the reactor shield. This exposure is considerably below the minimum medically detectable dose which is generally quoted to be about 20 rem (200 mSv).

5.3.2 Operational Radiation Levels at Full-Power Operation

Complete gamma and neutron dose measurements have been made for the ISU AGN-201 reactor at power levels from essentially zero to the maximum licensed power level of 5 W. Measurements inside the concrete block shielding indicate that the average dose rate from neutrons and gammas at [redacted] from the surface of the AGN tank, while operating at a steady-state power level of 5 W, is about 230 mrem/hr (2.3 mSv/hr). However, access to this area during full-power operation is restricted. Outside the concrete shielding, doses averaged over the shield surface are about 8 mrem/hr (80 μSv/hr) total, while average total dose rates at the AGN console are 0.62 mrem/hr (6.2 μSv/hr) at 5 W. Figures 5.3-4 and 5.3-5 provide calculated radiation levels associated with 0.1-W and 5-W operation, respectively.

At maximum power, the equivalent dose rate measured at the entrance to the reactor laboratory is 0.7 mrem/hr (7 μSv/hr). Other areas adjacent to the laboratory have comparable dose rates. The dose rate on the roof of the reactor room immediately above the reactor is 0.15 mrem/hr (1.5 SμSv). The area above the reactor is not normally occupied. Outside of the reactor room, the maximum dose rate occurs in the Observation Classroom, [redacted] at a position on the wall separating the Reactor Room and the Observation Room along the line-of-sight from the AGN glory hole. The dose rate at this location is about 7.5 mrem/hr (75 μSv/hr) from a well-collimated beam of radiation emerging from the glory hole which covers a small area because of limited beam divergence. The Observation Classroom is a restricted access area and is generally not occupied by students during reactor operations.

The recommended limit[1] for uncontrolled university areas is 0.1 rem/year (1 mSv/yr) to any student or member of the general population. If a 5-hour week, 30-week reactor operation schedule is assumed, this results in a 0.1 rem/year (1.0 mSv/yr) dose to a receptor at the entrance to the reactor room. The assumption of a 5-hour exposure per week is conservative in that most reactor operations in which students may be exposed to elevated radiation levels are associated with the teaching of an undergraduate or graduate reactor laboratory course and are limited to one three-hour period per week.

---

[1] Radiation Safety Policy Manual, Rev. 13.1, Radiation Safety Division, Technical Safety Office, Idaho State University, October 2020.

Therefore, the AGN-201 reactor is adequately shielded for 5-W operation under such an expanded operating schedule.

For persons working in the reactor room, under surveillance, the ISU occupational dose limit[2] is the more restrictive of (1) 1,000 mrem/yr (10 mSv/yr) total effective dose equivalent or (2) 10,000 mrem/yr (100 mSv/yr) for the sum of the deep-dose and committed dose equivalent to any individual organ or tissue, excluding the lens of the eye. A student would have to remain at the concrete shield in the reactor room for about 4 hours per week for 30 weeks each year to receive an equivalent dose of this magnitude from the reactor operating at 5 W during the entire time.

[2] Radiation Safety Policy Manual, Rev. 13.1, Radiation Safety Division, Technical Safety Office, Idaho State University, October 2020.

[redacted]

Fig. 5.3-1 External concrete block shielding for AGN-201 reactor. Plan view of shielding. Elevation A is view facing north. Elevation B is view facing west.

[redacted]

Fig. 5.3-2 External concrete block shielding for AGN-201 reactor. Section A-A is a cut-away view facing north. Section B-B is a cut-away view facing west

[redacted]

Fig. 5.3-3 External concrete block shielding for AGN-201 reactor. Elevation C is facing south. Top view of shielding

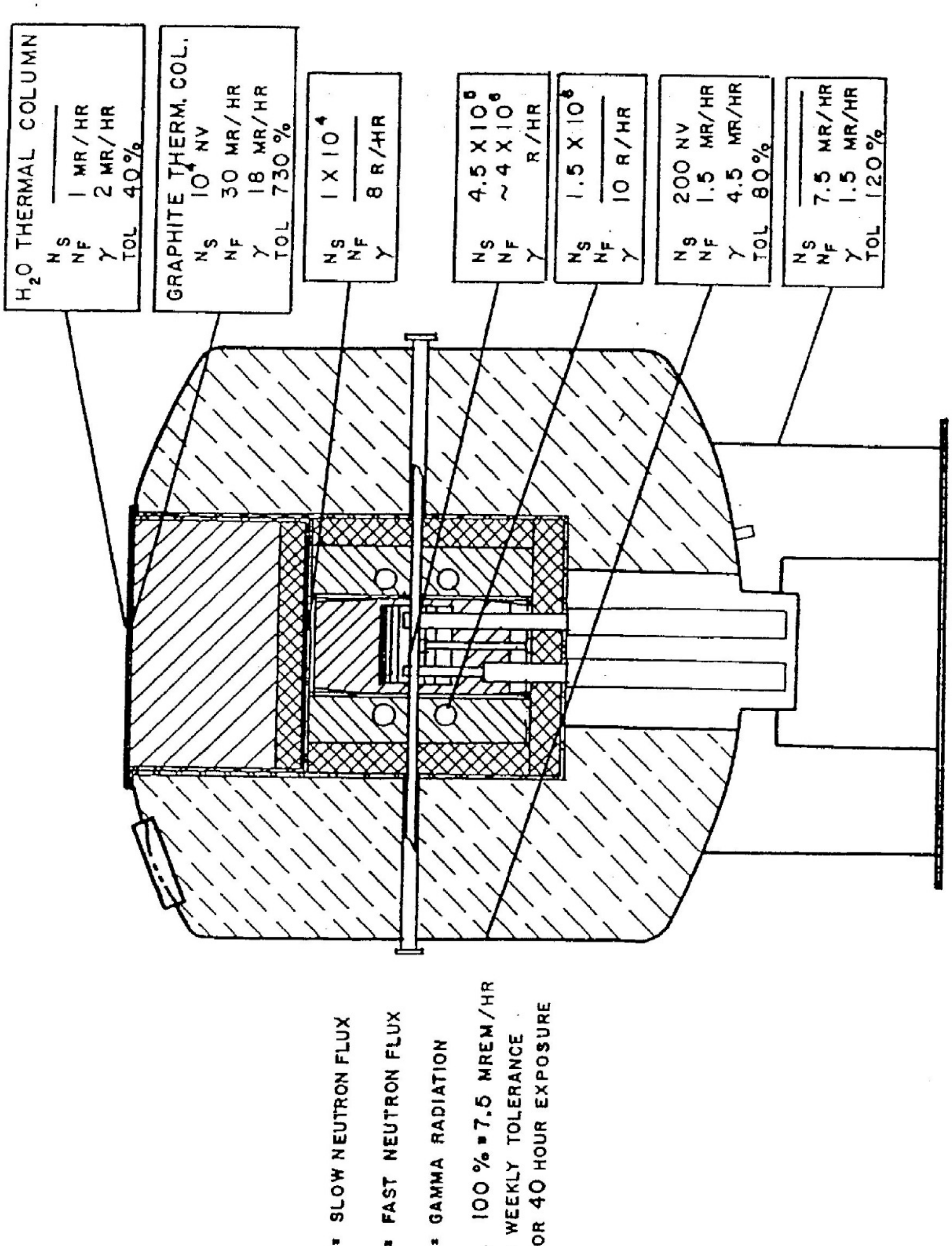


Fig. 5.3-4 Radiation levels of the AGN-201 reactor operating at 100 mW

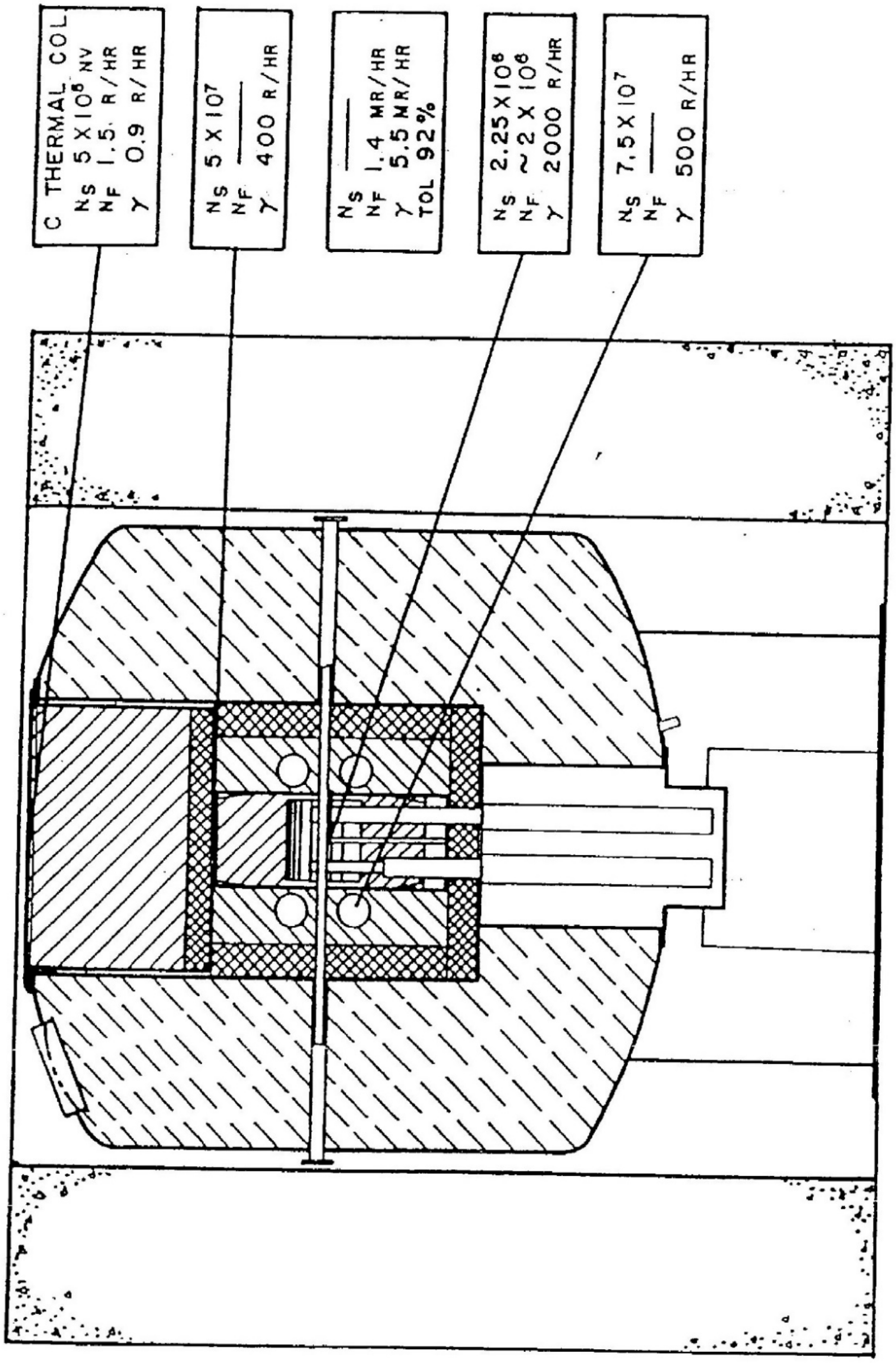


Fig. 5.3-5 Radiation levels of the AGN-201 reactor operating at 5 W

5.3.3 Radiation damage to the fuel matrix

Low-density polyethylene, a polymeric organic material, can sustain radiation damage when exposed to neutron bombardment and the radiation emissions from the decay of fission products. In tests performed by Aerojet-General Nucleonics, more than fifty small samples of core material were exposed in the CP-5 reactor at Argonne National Laboratory in a flux of approximately 10^{12} n/cm^2-s for periods of up to 1 week. An analysis of these samples indicated that radiation damage manifests itself in reduced density and loss of hydrogen from the polyethylene after exposures of approximately 1 week for a fluence of 6 x 10^{17} n/cm^2. By extrapolating these data, on the assumption that the time-integrated flux (fluence or nvt) is responsible for the radiation damage at an average power of 5 W, the core life is approximately 200 years for continuous operation. It is a reasonable assumption that a lower flux for a correspondingly longer time would result in no more radiation damage than occurred in the high flux tests conducted in the CP-5. Therefore, the core should have an adequate lifetime if exposed to no more than an average continuous power level of 5 W.

5.3.4 Production and handling of radioisotopes

Neutron activation in the AGN-201 reactor can produce only very limited quantities of radioisotopes and any induced radioactivity in reactor structures is negligible. Subsequent handling of radionuclides is supervised by individuals experienced in the detection and evaluation of radiological hazards. The reactor staff has been trained in such procedures and supervises all handling of radioactive materials within the reactor area. Outside this area, the use of radionuclides comes under the control of the ISU Radiation Safety Committee.

The maximum amount of activity which can be produced by one irradiation is given by the product of the specific activity and the mass of material irradiated. There are three primary limitations on the amount of mass which can be used. The most obvious of these is the amount of space available in the reactor for irradiation. In the AGN-201 reactor, the useful volume of the glory hole is approximately 50 cm^3. For those cases in which this volume is insufficient, the 10-cm-diameter access ports may be used instead of the glory hole.

The second limitation, pertaining to the total mass of material that may be irradiated is the effect of this material on the criticality of the reactor. Since the material must absorb neutrons to become activated, its insertion into or near the core decreases the reactivity of the system. This effect can be compensated by the use of the control rods up to the limit of the CCR and FCR being fully inserted.

The limitation on the amount of material arises from loss of the neutron absorption efficiency because of self-absorption whereby the outer portion of the material shields the inner portion from the neutrons. This effect is present primarily in strongly absorbing materials, such as indium and gold and involves the resonance integral in particular.

The risk of release of radioactivity by breakage of an irradiated sample is reduced by careful design of handling and encapsulating practices and attention to details of irradiation, such as the effect of radiation on the sample. Such criteria are examined closely with respect to experiments. The use of non-porous paintwork in the Reactor Laboratory is an aid in preventing long-term contamination. Emergency decontamination supplies are on hand at all times, as well as contamination survey instruments.

5.4 Production and release of radioactive gases

Radioactive argon-41 and nitrogen-16 are produced by neutron reactions with air and water in the vicinity of the core of the reactor. Air may be contained in experimental facilities (glory hole & access ports) and is in solution in the tank water.

5.4.1 Production of argon-41

Experience with AGN reactors operating at higher power has shown that no significant release of Ar-41 (half-life = 1.8 hr) occurs from the glory hole during power operations at 5 W or less. This conclusion was the result of a test at the Naval Post Graduate School where Ar-41 activity was measured by irradiating a sample of air at atmospheric pressure in a closed tubular container just filling the AGN glory hole to the boundaries of the core. The irradiated air was transferred to a chamber counter with thin-walled glass G-M tube. Decay was followed over approximately one half-life and was consistent with the decay of Ar-41. The measured activities agreed with those estimated from a calculated efficiency of the counter.

The next most likely location to produce Ar-41 is below the reactor skirt. On the basis of Naval Post Graduate School operating experience, Ar-41 will not be formed in measurable concentrations under the skirt at operation at 5 W. Since the resulting peak Ar-41 activity for the air volume in a sealed empty glory hole is only 45 times greater than the MPC value for Ar-41, release of the Ar-41 in the glory hole into the reactor room will result in natural diffusion and mixing of this irradiated air volume throughout the room will easily reduce the average air activity in the vicinity of the reactor to less than 1% of MPC values for uncontrolled areas. Also, the reactor area is presently, and will continue to be, a control area with limited access. Thus, no hazard from Ar-41 is anticipated, as shown below.

The maximum equilibrium concentration of Ar-41 produced can be easily calculated. At 5 W, the average thermal neutron flux is 1.75×10^8 n/cm^2-s. The saturation reaction rate,$\mathfrak{R}$, for Ar-41 production is given by

$$\mathfrak{R} = \Sigma_\gamma \phi$$

$$= \sigma_\gamma \frac{\zeta m N_A}{A} \phi \tag{1}$$

where
σ_γ = microscopic cross section for $^{40}Ar(n,\gamma)^{41}Ar$[cm^2],
ζ = natural abundance of Ar-40 [dimensionless],
m = the mass of Ar-40 contained within the volume of the glory hole fully contained within the reactor core [grams],
N_A = Avogadro's number [mol^{-1}],
A = atomic mass of Ar-40 [grams mol^{-1}], and
ϕ = average thermal neutron flux [n cm^{-2} s^{-1}].

The mass of Ar-40 is calculated based on the assumption that the air entrapped within the glory hole is a dry, ideal gas at standard temperature and pressure with argon comprising 1.3% of air by mass. Thus, there are 6.2 mg of Ar-40 contained within the

394 cm^2 of air entrapped within the portion of the glory hole fully contained within the reactor core. The resulting Ar-41 production rate is 10,600 atoms/s, or in terms of activity, the production rate is 30.2 pCi/s.

The equilibrium concentration of Ar-41 in the reactor room at a power level of 5 W is given by

$$C_{41} = \frac{L}{V_R r} \tag{2}$$

where

C_{41} = concentration of Ar-41 [μCi/cm³],
L = leakage rate of Ar-41 from the core (assumed to be equal to the production rate) [μCi/s],
V_R = the volume of the reactor room [cm³], and
r = fractional volumetric exchange rate of air in the reactor room equal to the ventilation flow rate, Q [cm³/s], divided by the reactor volume.

The dimensions of the reactor room are [redacted] which give a volume of [redacted] During normal operation, the ventilation flow rate is [redacted] Substitution of these numerical values into Equation (2) yields an equilibrium concentration for Ar-41 of [redacted] μCi/cm³.

From 10 CFR 20 Appendix B, the limiting values for the Derived Air Concentration (DAC) and effluent concentration for Ar-41 are 3×10^{-6} μCi/ml and 1×10^{-8} μCi/ml, respectively. DAC values establish the limiting concentrations of airborne radioactive materials to which occupational workers may be exposed while the effluent concentration provides limiting concentrations for the general public. Thus, the maximum equilibrium concentration of Ar-41 in the reactor room air with the AGN reactor operating at 5 W is 0.0013% of the DAC limit and 0.39% of the effluent concentration limit.

Air flow from the reactor laboratory room is mixed with return air from the rest of the LEL building. Approximately 80 percent of the total air flow is recycled to the building while the remaining 20 percent is discharged from a [redacted] The air flow leaving the reactor room is diluted by a factor of about 27 before being discharged to the environs. Return air is diluted further by mixing with fresh makeup air before being recirculated to the building. The calculated maximum effluent concentration of Ar-41 at the [redacted] is 1.5×10^{-12} μCi/cm^3. The Ar-41 concentration in the recycle air to unrestricted areas in the LEL building is approximately 1.0×10^{-12} μCi/cm^3.

These concentrations are insignificant compared with the maximum permissible concentration in unrestricted areas which is given by the maximum effluent concentration. The maximum concentration of Ar-41 in unrestricted areas of the LEL building from the ventilation system recycle is 0.010 percent of maximum permissible level. The concentration of Ar-14 in air discharged to the environment from the exhaust vent is about 0.015 percent of the maximum permissible level. This value is much larger than the resulting concentration at ground level since the analysis of the ground-level concentration would involve the additional dilution that occurs upon discharge from the vent.

5.4.2 Release of argon-41 from tank water

Argon-40 atoms are present in solution in the tank water and produce Ar-41 on neutron irradiation. However, the Ar-41 produced by the activation of dissolved air in the tank water is negligible for the following reasons: (1) the concentration of argon in the tank water is small, (2) the thermal neutron flux in the shield water is several orders of magnitude lower than in the reactor core, and further, tank access cover retards diffusion of the gas from the shield tank to the atmosphere. An estimate of the number of Ar-41 atoms released into the room air and the concentration in the discharge can be made and results show that this value is small (about 2×10^{-13} μCi/cm^3 air at 5 W) compared to the Ar-41 production from air entrapped in an open glory hole.

5.4.3 Production and release of nitrogen-16

Simple calculations and experimental measurements show that production and release of nitrogen-16 (half-life = 7 seconds) is insignificant for the AGN-201 reactor operating at power levels up to
5 W.

5.5 Maximum Credible Reactivity Accident

The total excess reactivity of the AGN-201 reactor is approximately 0.25% Δk/k ($0.33). However, for the purpose of the AGN-201 Safety Analysis it has been assumed conservatively that the reactivity can be instantaneously increased to 2% Δk/k above delayed critical with the reactor in operation at a power level ≤ 5 W. This situation clearly cannot arise during the normal course of operation of the system, [redacted] This instantaneous 2 % positive reactivity insertion is taken to be the maximum credible accident for the AGN-201 reactor.

Historically, this scenario has been analyzed using one group theory with one group of delayed neutrons. The analysis was first performed by Aerojet-General Nucleonics and published in the Hazards Summary Report for the AGN-201 Reactor in August 1957. This simplified analysis assumed that: (1) at time zero a 2% step increase in reactivity was inserted with the reactor at 100 mW and (2) the energy in the core at time zero was negligible in comparison with the energy liberated during the ensuing excursion. Further, the analysis assumed that there was no energy transfer from the reactor core during the excursion.

A complete description of the analysis may be found in the Hazards Summary Report and will not be repeated here, but the results of their analysis are summarized below. Numerical solution of the governing equations by the finite- difference method yielded a total energy release of 1.7 MJ and raised the average core temperature about 71ºC. The excursion produced a peak power of 54.4 MW at 204 ms after the reactivity insertion. These results were adequate to release any fission products because the average core temperature remained well below the melting point of the polyethylene matrix, which is about 200ºC.

For the purposes of this document, a more detailed analysis is performed using one neutron energy group with six delayed neutron precursor groups. The same assumptions are made regarding the 2% step increase in reactivity at time zero, negligible energy accumulation at the start of the excursion, and the excursion is modeled as an adiabatic process. With these assumptions, the applicable kinetics equations are then

$$\frac{dn}{dt} = \frac{\rho - \alpha T - \beta}{\ell} n + \sum_{i=1}^{6} \lambda_i C_i \tag{3}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\ell} n - \lambda_i C_i, \quad i = 1,2,\ldots,6 \tag{4}$$

$$P = \Sigma_f n v Y V \tag{5}$$

and

$$\frac{dT}{dt} = \frac{P}{\delta V C_P} \tag{6}$$

where

n = space-averaged neutron density in reactor [n/cm3],

ρ = core reactivity[dimensionless],

α = temperature coefficient of reactivity [ºC-1],

ℓ = neutron generation time [s],

β = fraction of delayed neutrons [dimensionless],

βi = delayed neutron fraction for the ith group of delayed neutron precursors [dimensionless],

λi = decay constant for the ith group of delayed neutron precursors [s-1],

Ci = space-averaged density of the ith group of delayed neutron precursors [cm-3],

P = reactor power [W],

Σf = macroscopic fission cross section [cm-1],

v = average thermal neutron speed [cm/s],

Y = recoverable energy per fission [J/fission],

V = volume of reactor [cm3],

δ = density of the core material [gm/cm3],

CP = specific heat at constant pressure [J/gm-ºC], and

T = core temperature [ºC].

Equations (5-3) and (5-4) represent the usual point-reactor kinetics equations; Eq (5-5) gives the thermal power output of the reactor; and Eq. (5-6) gives the temperature change in the reactor core. In solving these equations the following numerical values were used:

ρ = 0.02,

α = -1.66 x 10-4 ºC-1,

ℓ = 7.5 x 10-5 s,

β = 0.0074,

Σf = 0.0074 cm-1,

v = 2.2 x 105 cm/s,

Y = 3.4 x 10-11 J/fission,

V = [redacted] cm3,

VδCP= 27,050 J/ºC].

The value of α given above (-1.66 x 10-4 ºC-1) was calculated by Parker3 and used in an early safety analysis of the ISU AGN-201 reactor. This value is somewhat smaller than the value reported by the manufacturer (-2.5 x 10-4ºC-1) and therefore yields a more conservative solution to equations (5-3)-(5-6), since the calculated energy output is greater than what would actually result from the use of the larger magnitude for the negative temperature coefficient of reactivity. Parker's α was obtained from a DISNEL4 calculation using five core temperature regions in the AGN reactor and accounts for the nonuniform temperature distribution in the core as a result of the flux distribution which is peaked at the center and near the reflector.

Equations (3)-(6) were solved numerically using the TUTSIMTM,5 computer simulation package for three initial power levels at an initial uniform core temperature of 20ºC: P0 = 5 W, 0.1 W, and 0.0001 W. Higher initial power levels do not affect the consequences of the accident. The results of these computations are shown in Figures 5.5-1 through 5.5-3. Figure 5.5-1 gives the power of the reactor as a function of time after the instantaneous insertion of the 2% reactivity. As shown, the power level quickly

[3] Robert Eugene Parker, "Safety Analysis of the AGN-201 Reactor," Idaho State University, Pocatello, Idaho, Master of Science Thesis, 1978.
[4] Acronym for the Diffusion (equation) Iterative Solution, Nineteen Energy Levels computer code.
[5] TUTSIM Products, 200 California Avenue, Palo Alto, CA 94306.

rises to very high values, passes through a maximum of about 170 MW and then rapidly decreases to a power level of about 800 kW which then appears to slowly decay, all within a time window of about 300 milliseconds.

The height of the maxima is observed to be essentially independent of the initial power level, but as might be expected, the time at which the maxima occur increases with decreasing initial power. Following the excursion, the power decreases slowly until after several minutes the thermal power output attains a steady level of about 200 watts in which the reactivity addition is balanced by the compensating reactivity at the consequent elevated core temperature as a result of the large negative temperature coefficient. However, it is reasonable to assume that the thermal fuse has functioned as designed, thereby separating the core so that this equilibrium power is not maintained. The excursion simulation shown in Figures 5.5-1, 5.5-2, and 5.5-3 does not model core separation which

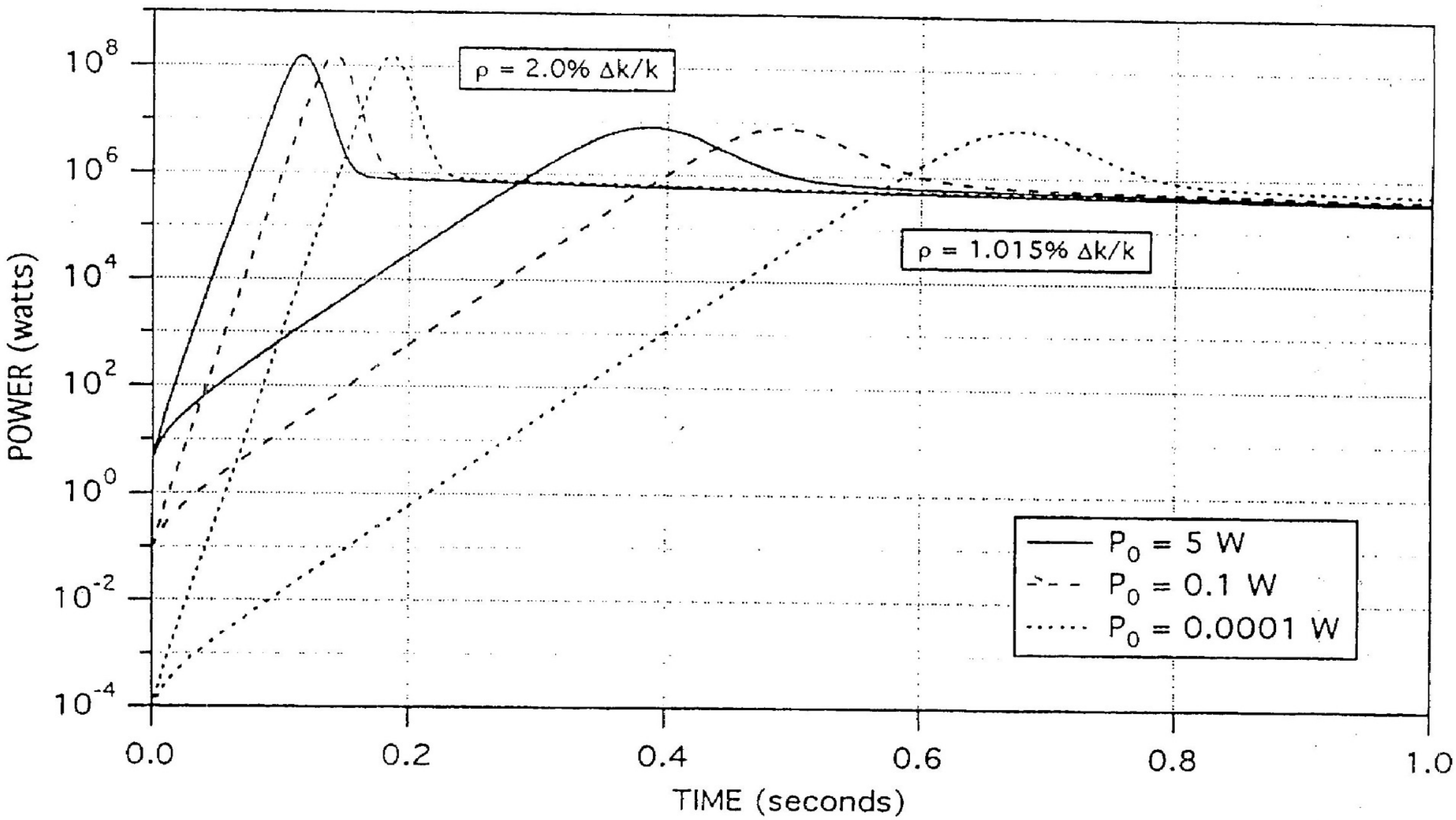


Figure 5.5-1 Reactor power during maximum credible accident.

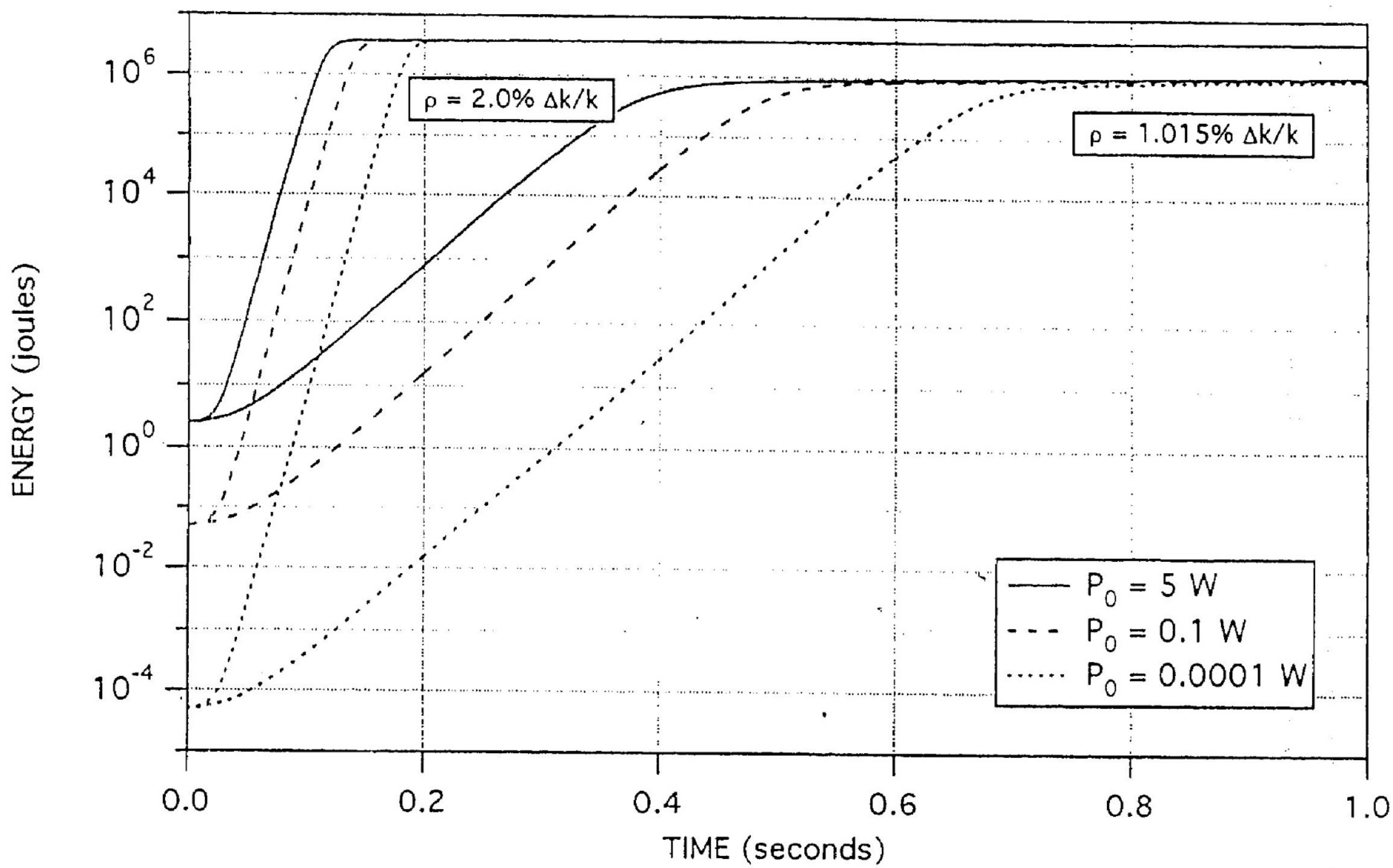


Figure 5.5-2 Energy release during maximum credible accident.

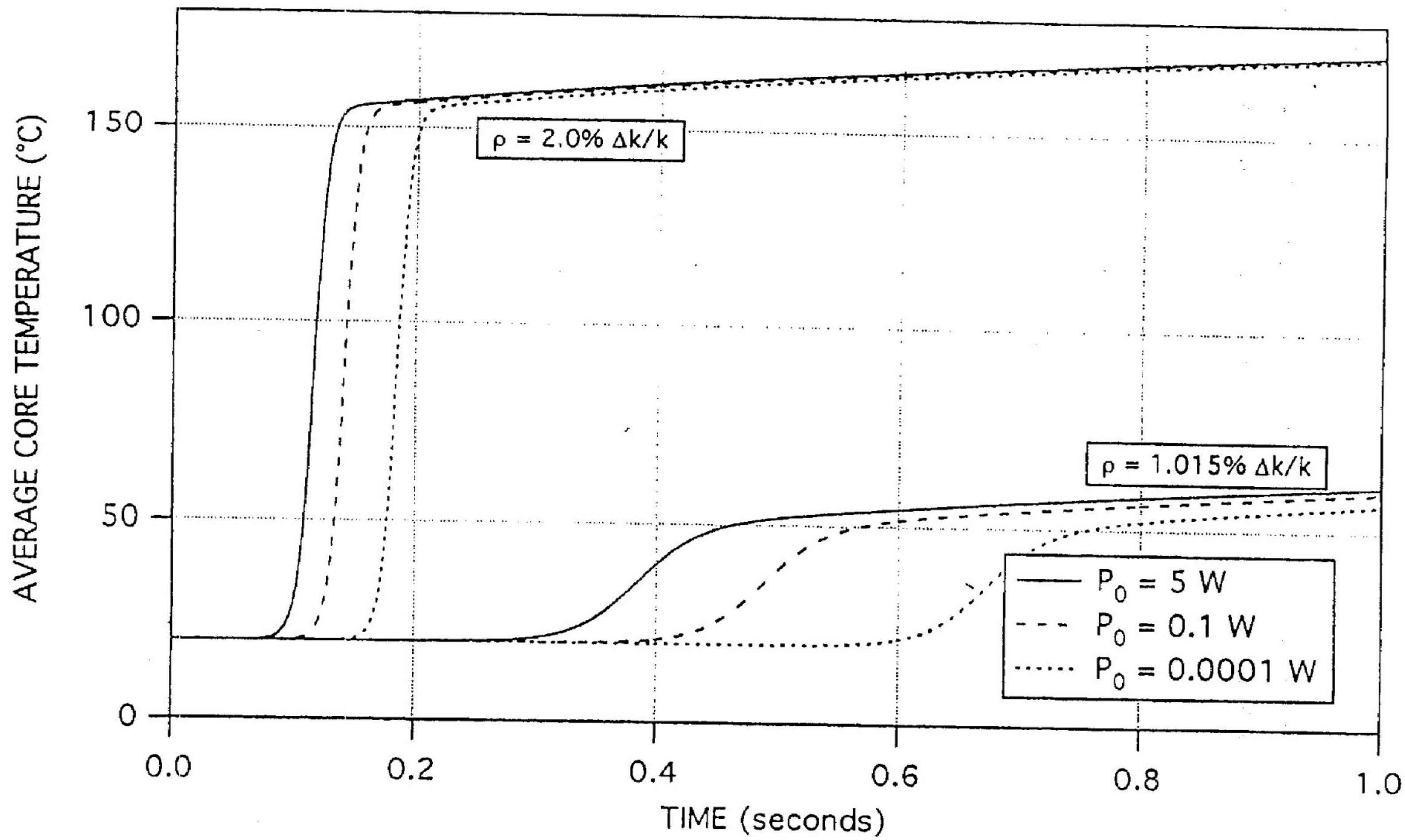


Figure 5.5-3 Core temperature rise during maximum credible accident.

would occur when the temperature of the thermal fuse located near the center of the core exceeds about 100ºC, as discussed in Section 4.2.

The final power level is obtained by modifying Eq. (4) to include Newton's law of cooling and then equating the time derivatives with zero in Eqs. (1), (2) and (4). Equation (4) is then solved for the final power, Pf, which becomes

$$\frac{dT}{dt} = \frac{1}{\delta V C_P}(P - UA[T_f - T_0]) \tag{7}$$

Where
U = overall heat-transfer coefficient [W cm-2}, and
A = external core surface area [cm-2].

The overall heat-transfer coefficient U is assumed not to be a function of temperature. The numerical value of the product UA was obtained from steady-state operation at 0.1 W. According to the manufacturer, the average temperature rise of the core at this power is 0.05ºC, so that

$$UA = \frac{P}{\Delta T} = 2\ W/^0C \tag{8}$$

The value of UA has little influence on the total energy released during the accident, but it does determine the final power level which the reactor ultimately attains. Setting the time derivatives to zero gives

$$P_f = UA\left(\frac{\rho}{\alpha} - T_0\right) \tag{9}$$

where the substitution $T_f = \rho/\alpha$ has been made.

The energy released in the excursion is shown as a function of time in Figure 5.5-2. These curves are simply the time-integrated curves shown in Figure 5.5-1. The average core temperature is shown as a function of time in Figure 5.5-3. Assuming that the initial excursion is an adiabatic process, the maximum temperature is greater than 170ºC for all three initial conditions. The maximum temperature, however, will be less than 150ºC because of core separation once the thermal safety fuse deploys as designed. Core separation will result in an approximately 5% decrease in reactivity, thereby shutting down the reactor and preventing recurrence of the excursion. The final temperature in all cases is about 120ºC.

The radiation dose received by personnel as the result of the maximum credible accident can be computed from the known radiation doses for 5 watt operation and the total energy released during the accident. Assuming that a person is in contact with the surface of the concrete shield [redacted] when the incident occurs, s/he would ordinarily receive a dose which is less than 10 mrem/hr (0.1 mSv/hr) at 5 watt operation. This is equivalent to a dose of about ((10 mrem/hr)/(5 J/s))/(3600 s/hr) = 5.56 x 10-4 mrem/J (5.56 x 10-3 μSv/J). From Fig. 5-7 the total energy released in the accident is about 5.8 MJ. It follows, therefore, that such an individual would receive a dose of (5.8 x 106 J)(5.56 x 10-4 mrem/J) = 3.2 rem (32 mSv) as the result of the accident.

As unlikely as a 2% increase in reactivity is, such a reactivity change with either of the first two initial conditions is extremely remote. This is because, as already noted, the aforementioned reactivity increase can only occur by [redacted] and this would be very difficult to accomplish while the reactor is operating. A step reactivity insertion of 1.015% Δk/k ($1.37) is more reasonable. This value of reactivity was determined by Parker6 as the maximum reactivity that could be inserted into the AGN core through the glory hole based on the volume of the glory hole and the availability of fuel material. The response of the AGN-201 operating at the same initial power levels for this reactivity step has been analyzed and is also shown in Figures 5.5-1 through 5.5-3.

The maximum power, which is attained at about 400 ms for an initial power level of 5 W, is about 80 MW for this reactivity step. The total energy generated is about 2.9 MJ, and the maximum average temperature is about 128ºC, again assuming no energy losses during the excursion. From the results, it may be assumed that the thermal fuse will function as designed and the core will separate. The resulting dose to an individual standing next to the concrete shielding is about one-half of the dose for the 2% reactivity insertion, or about 1.6 rem (16 mSv).

---

[6] Robert Eugene Parker, op cit.

5.6 Loss of Water Shield from AGN Tank

If the reactor is operated without water in the shielding tank at 0.1 watt power, the radiation level just outside the reactor tank will be about 10 mrem/hr (0.1 mSv/hr) of gamma rays, and about 50 mrem/hr (0.5 mSv/hr) of fast neutrons. These levels are six and eleven times as much, respectively, as they would be through the water shield. At the outside surface of the concrete shield, radiation levels would be about 0.14 mrem/hr (1.4 µSv/hr) of gamma rays and 3.4 mrem/hr (34 µSv/hr) of fast neutrons at 0.1 watt without the water shield. At 5 watts the radiation levels would be about 7 mrem/hr (0.07 mSv/hr) for gamma rays and 170 mrem/hr (1.7 mSv/hr) for fast neutrons. This radiation level would trip the high-level radiation alarm mounted on the reactor console and initiate laboratory evacuation.

While it is extremely unlikely for an excursion to occur without the water shield in place, the maximum acute dosage a person might receive at the surface of the concrete shield would be 18 rem (0.18 Sv) of gamma rays and fast neutrons. This would of course be a high amount of radiation but is within the guidelines for emergency doses.

Another potentially hazardous condition which can be envisioned is the case where the control and safety rods are fully inserted and the scram mechanisms are made to be inoperative. Under such circumstances the reactor power would continue to rise until the negative temperature coefficient reduced the reactor to a delayed-critical state at some high power. Under this circumstance, equilibrium is determined by the condition that the rate of energy conducted away from the core be exactly equal to the fission energy generation rate. Since the temperature coefficient of reactivity is approximately -1.66 x 10-4 ºC-1, and the heat conductivity rate from the core may be estimated, it may be readily calculated that with 0.2% excess reactivity the equilibrium temperature is approximately 10ºC above ambient. This corresponds to a fission rate of approximately 10 watts. Postulating these conditions of the reactor operating at a continuous power level of 10 watts, the radiation received by a person outside the concrete shield would be approximately 12 mrem/hr (0.12 mSv/hr).

The above postulated exposure, although constituting a slight hazard, is considered improbable since it is doubtful that anyone would stay in such a position under reasonable administrative control for more than a few hours. It is interesting to try to predict whether or not operation at this high power level would cause the fuse to melt, and, accordingly, shut down the reactor. Unfortunately this is a very difficult heat transfer calculation, due to the complicated geometries and, although it is believed there is a reasonable expectation that the fuse would function, no claim is made to this effect.

5.7 Energy Released

At 5 watts, the total fission rate is about 1.6 x 1011 fissions/sec. Each fission produces approximately 0.6 neutrons that may leak out, 5 MeV of prompt gamma rays, 6 MeV of delayed gamma rays, and a small number of delayed neutrons. By far the largest source of radiation is due to the radioactive fission products. If the reactor has been operated at this level for a long time, the activity in MeV-curies equivalent at a time t seconds after shutdown is given by 0.4t -0.2, 7: the activity which produces 3.7 x 1010 MeV of ionizing radiation is defined as one MeV-curie equivalent. Table 5.7-1 shows these values for various times after shutdown. In the event that the reactor is operated on an eight-hour-per-day schedule, the figures for one day and one month may be reduced by a factor of one-fourth. Five-watt operation leads to 50 times the radiation fields generated at a power level of 100 mW, as shown in Table 5.7-1.

Table 5.7-1
Activity Contained in the Reactor Core for Various Times After Shutdown

	Activity in MeV-curies Equivalent	
t	5-W Operation	100-mW Operation
0 (reactor operating)	50.	1.0
1 second	20.	0.40
1 hour	3.75	0.075
1 day	2.0	0.040
1 month	1.0	0.020

5.7.1 Operational Containment of Fission Products

[7] A.T. Biehl, R.P. Geckler, S. Kahn, and R. Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, CA, October 1957.

The one significant difference resulting from 5 watt operation is the increased fission product inventory. Concerning the levels of activity in the core following 5 watt operation, reference is made to Biehl, Geckler, Kahn, and Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, Calif., October 1957, pp. 19-21. Further, it is noted that operation of Aerojet-General Nucleonics AGN-201 at 5 watt levels resulted in no detectable release of radioactive effluents due to the retention of fission products by the fuel matrix material. Even if there were gaseous effluents released they would be contained in the core tank. If, following recent 5-watt operation, it becomes necessary to open the core tank, samples of the gas within the core tank will first be taken and analyzed to assure that there has been no hazardous release of radioactive effluents from the fuel material. If any significant levels do exist, appropriate radiological safety procedures will be followed prior to and during subsequent opening of the core tank. These procedures are under the direct supervision of the Reactor Supervisor as authorized by the Idaho State University Radiation Safety Officer.

The core tank is vented to purge radiolytic hydrogen and noble gas fission products as part of a biennial surveillance procedure. Only small quantities of radioxenon and radiokrypton have been detected in the performance of these surveillances. Thus, significant fission product leakage from the fuel core is assumed not to occur. At power levels of 5 watts or less, leakage is insignificant and measurable amounts have not been found in other similar facilities. If leakage is experienced under any conditions of operation within the scope or the authorization requested, procedures for the safe and authorized disposal of fission gas that may be formed together with procedures to assure containment will be formulated.

Two assumptions are used as a basis for calculating the power generated in an accident:

a. At time zero, a 2% step increase in reactivity is inserted with the reactor at low power (≤ 5 watts).

b. At time zero, the thermal energy in the core is negligible in the comparison with the energy liberated during the accident and there is no heat removed from the core during the excursion.

The time-dependent behavior of the neutron density, including one average group of delayed neutrons is considered. A numerical finite-difference solution of the three coupled nonlinear differential equations yields a value of 54.4MW for the peak power at t=204 milliseconds and a total energy release of 2.61 MJ, resulting in a temperature rise of 71.3ºC. There will be about 1.45 x 10^{17} Mev of prompt gamma radiation produced in this excursion. Table 5.2 presents the residual activity formed in the core as a function of time after this excursion.

Table 5.7-2

Activity Contained in the Reactor Core following 2% Step Increase in Reactivity

t	Activity MeV-curies Equivalent
1 second	10^6
1 hour	50
1 day	1
1 month	0.02

5.8 Gaseous Radioactive Product Release

For the purpose of analysis, the gaseous fission products have been divided into two groups as shown in Table 5.8-1. The first group comprises those radionuclides that will remain in the tank water should the release occur when the tank is filled with water. This group includes the bromines and iodines. In the incredible event that no tank water is present, those isotopes would be added to the radioactive cloud and add to the hazard.

The second group comprises the insoluble volatiles, the krypton and xenon isotopes. They are the major source of potential radioactivity in the room (and outside) if tank water is present.

Table 5.8-1 Gaseous Fission Products in AGN Fuel at 5-W Operation for 30 days

GASEOUS FISSION PRODUCTS IN AGN FUEL AT 5-W OPERATION FOR 30 DAYS

Nuclide	Decay Constant (hr^{-1})	Inventory (mCi)
Group I (soluble volatiles)		
Br-83	3.02×10^{-1}	20.0
Br-84	1.31×10^{0}	46.1
Br-84m	6.95×10^{0}	0.95
Br-85	1.39×10^{1}	62.5
Br-87	4.49×10^{1}	112
I-129	4.60×10^{-12}	41.8
I-131	3.58×10^{-3}	121
I-132	3.07×10^{-1}	184
I-133	3.34×10^{-2}	271
I-134	7.93×10^{-1}	318
I-135	1.04×10^{-1}	247
I-136	2.90×10^{1}	130
	Total Iodines	1320 mCi
	Total Group I	1560 mCi
Group II (insoluble volatiles)		
Kr-83m	3.66×10^{-1}	20.1
Kr-85m	1.59×10^{-1}	62.5
Kr-85	7.67×10^{-6}	12.6
Kr-87	5.35×10^{-1}	113
Kr-88	2.50×10^{-1}	155
Kr-89	1.31×10^{1}	193
Kr-90	7.55×10^{1}	216
Kr-91	2.54×10^{2}	129
Xe-131m	2.41×10^{-3}	1.20
Xe-133m	1.26×10^{-2}	6.55
Xe-133	5.50×10^{-3}	2.71
Xe-135m	2.67×10^{0}	74.0
Xe-135	7.60×10^{-2}	200
Xe-137	1.07×10^{1}	24.7
Xe-138	2.45×10^{0}	229
Xe-139	6.08×10^{1}	237
Xe-140	1.56×10^{2}	249
	Total Group II	2190 mCi

5.8.1 Water activity

In the fission product release accident in which the tank water remains in situ, the fraction of the total soluble fission products which are released from the element is distributed in the water. Thus about 1.5 x 10^{-5} times the Group I fission products or 2.34 x 10^{-5} Ci remains in the tank. Since the volume of water in the reactor tank is 1000 gallons or 3.57 x 10 cm^3, the activity concentration is 6.55 x 10^{-6} μCi/cm^3. In 24 hours the activity would decrease to 1.17 x 10^{-6} μCi/cm^3. The activity then remains moderately constant because of the small decay constants for I-129, I-131, and I-133. However, under normal conditions the polyethylene moderator will prevent release of the gas, even to the inside of the core tank. It is therefore expected that the radioactive gas will not constitute a hazard to personnel.

The 5.8 MJ excursion will form 34 kCi of radioactive gas in the fuel particles. Gas diffusing out of these particles should be contained by the polyethylene fuel matrix. However, if any leakage from the core does occur, the presence of the two additional fluid-tight metal tanks insures that the gas will be contained. In the implausible event that the core leaks 0.1% of the activity and both tanks rupture, allowing 1.0% of this free radioactive gas to leak out of each tank, there would be 82 mCi of free gas outside the reactor 1 minute after the accident.

5.8.2 Exposure inside the reactor room

The maximum exposure from fission products to a person in the Reactor Laboratory will occur if all fission product gases from the core tank without water are distributed instantaneously within the room and no air change occurs after 30 days of continuous operation at 5 watts.

The total release of radioactive gases is 82 mCi and the volume of the reactor room is [redacted] so that the concentration is 1.6 x 10^{-4} μCi/cm^3, or 5.9 Bq/cm^3. If the room is assumed to be equivalent to a hemisphere of radius R = [redacted], the exposure rate at the center is given by

$$D = \frac{S_V}{2K\mu}\left(1 - e^{-\mu R}\right) \tag{10}$$

where S_V = volumetric source strength [Bq/cm³],
K = conversion factor for flux-dose rate
= 4.2×10^4 γ/cm²-sec per mR/min for photons of 0.7 MeV (average energy of fission product decay)
μ = attenuation coefficient for air
= 3.5×10^{-5} cm⁻¹ for E_γ = 0.7 MeV

Thus, the maximum exposure rate is 0.03 mR/min or 2.1 mR/hr (20.5 μSv/hr). Thus, an individual could remain in the room for about 47 hours before exceeding a dose of 100 mrem (1 mSv). The controlling factor, however, will be the dose to the thyroid from radioiodine. In any event the exposure rate from the plume added to the normal exposure due to reactor operation is sufficient actuate the radiation alarms so that the reactor room will be promptly evacuated. In addition, with the ventilation system in operation during such an accident, venting air at the rate of [redacted], the dose rate to any person in the reactor room will be significantly reduced. Also, diffusion and escape of the gases through the concrete shield would decay and reduced the release.

The dose to the thyroid can be calculated at any time t from the following equation[8]

$$D_t = \frac{(5.92 \times 10^2) A_\tau f_a \bar{E}\left(1 - e^{-\lambda_e t}\right)}{m\lambda_e} \text{ (rads)} \tag{11}$$

where A_T = inhaled iodine [Ci];
f_a = fraction which is deposited in critical organ;
$\bar{E}$ = effective energy absorbed by thyroid per disintegration [MeV];
λ_e = effective decay constant, including both radioactive decay and biological elimination [s]; and
m = mass of the thyroid [gm].

[8] J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

For a long time, $t \rightarrow \infty$, and using $T_e = 0.693/\lambda_e$, Eq (11) becomes:

$$D_\infty = \frac{(8.54 \times 10^2) A_\tau f_a \bar{E} T_e}{m} \text{ (rads)} \tag{12}$$

$$= constant \times \sum_{i=1}^{n} A_{\tau i} \bar{E}_i T_{ei}$$

where the parameters are summed over the i isotopes the amount of iodine inhaled is given by

$$A_{\tau i} = R\tau\left(\frac{A_i}{V}\right) \tag{13}$$

where A_i/V is the activity concentration and R the mean breathing rate over interval τ.

If τ is made equal to 1 second, then

$$D_\infty = \frac{(8.54 \times 10^2) f_a R}{m} \sum_i^n \frac{A_i \bar{E}_i T_{ei}}{V} \text{ (rads/sec)} \tag{14}$$

For the standard man,[9]

$f_a = 0.23$,
m= 20 gm,
R= 10 m^3/8 hr = 3.47 x 10^{-4} m^3/s,

and the value of the constant is thus 3.41 x 10^{-3}.

The data necessary for the summation are contained in Health Physics, 3, June 1960, and the necessary activity concentrations are calculated from data given in Table 5.8-1. The summation for I-131 through 1-136 yields a value of about D_∞ = 1.4 mrem/sec (14.4 μSv/sec).

Assuming that 300 rem (3 Sv) to the thyroid is a limiting dose,[10] a person will have approximately 60 hours to evacuate from the reactor room. Actually the time will be longer than this since the room exhaust will be in operation and will reduce the dose. Further the leakage of the soluble iodine through the moderator and two containment

[9] "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics 3, June 1960.
[10] J. J. DiNunno et al., loc. cit.

tanks and the assumption no water in the shield tank is an extremely improbable scenario.

5.8.3 Exposure outside the building

On release of radioactive fission products into the reactor room, the facility Emergency Plan requires the reactor operator to initiate the evacuation of the Lillibridge Engineering Building by the actuation of the building fire alarm. The reactor operator will also shut the doors to the reactor room and trip the ventilation cut-out switch to prevent air exchange with the remainder of the building. Reentry to the laboratory will be allowed by the Reactor Supervisor in consultation with the ISU Radiation Safety Officer only after thorough radiation monitoring and when radiation levels permit entry. The room air may then be exhausted by normal operation of the ventilation system.

At [redacted] of the Lillibridge Engineering Laboratory Building, the ventilation system exhausts with a discharge flow rate of about [redacted] [redacted]. It is reasonable to assume a dilution of the reactor room effluent by a minimum factor of 27. Using this, and assuming no further dilution outside the building the maximum dose to the thyroid may be calculated.

The calculation assumes: (1) complete mixing in the reactor room at all times, and (2) the person is immersed in the effluent from the building in which the concentration of radioactivity, X_t, at any time t is equal to $C_t/70$ where C_t is the concentration in the exhaust and in the room.

With assumption (1), C_t is given by a simple first order rate law:

$$C_t = C_0 exp(-\frac{a}{V}t) \tag{15}$$

where

Q = exhaust rate [cm^3/sec],
V = room volume [cm^3],
C_0= initial concentration, at t = 0

the inhaled activity $A_{dt} = RX_t$ dt as above and thus the total amount inhaled up to time t is

$$A_t = \frac{RC_0}{70}\int_0^t exp\left(-\frac{a}{V}t\right)dt \tag{16}$$
$$= \frac{RC_0V}{70a}\left(1 - exp\left(-\frac{a}{v}t\right)\right)$$

If the exposure is for an infinite time,

$$A_t = \frac{RC_0V}{70a} \tag{17}$$

and, comparing with the previous calculation, the total dose

$$D_{total} = \frac{D(rads/sec)\cdot V(m^3)}{70a(m^3/sec)} \tag{18}$$
$$= 180\, rads\ (1.8\, mGy)$$

Thus we may conclude that for any location outside of the building, the maximum possible dose to the thyroid from a fission product release will be considerably less than 0.07% of a maximum permissible dose, even when no credit is taken for radioactive decay.

5.9 Emergency Procedures

The postulated Maximum Credible Accident assumes that the maximum available reactivity in the laboratory is inserted into the core, that as the control rods are inserted, the high-level trips on channels two and three do not work, and the operator is oblivious of the fact that the trips are not working. The excursion turns around after 14 seconds, and at about 20 seconds the temperature of the core no long rises. By that time the health physics alarms would have activated and the operator should then be alerted to the fact that an incident had occurred.

If the operator decided that an accident might have occurred, the operator would evacuate the reactor area and initiate building evacuation. The operator would then inform the appropriate administrative personnel, who would see that reentry to the area was not made until conditions were tolerable in regard to both radioactivity from reactor as well as any airborne radioactivity that might have escaped. Complete emergency procedures are given in the facility Emergency Plan.

Any person close to, or exposed to radiation from the reactor excursion would be placed under observation at the Bannock Regional Medical Center which is prepared to handle and treat such persons. Similarly, an early attempt would be made to evaluate exposures by pocket dosimeters, film badges, etc., and to determine if any fission product leakage has occurred so that appropriate action could be taken.

After reentry, an estimation of the energy release would be made by gamma-ray flux measurements. When the radioactive hazards had been sufficiently reduced to permit working near the reactor another effort would be made to detect any radioactive gas leakage and the reactor would be secured by inserting cadmium poison rods into the glory hole and access ports. After a period of about a week, the control and safety rods would be permanently removed. At this time the core activity should be down to the millicurie range where the normal radiological handling procedures could be used. To place the reactor back in operation, the core would be assembled as a new critical assembly with a new safety fuse.

5.10 Safety Devices

Every effort has been made to make the reactor safe against all foreseeable circumstances. In the event of an electric power failure, the control system is designed to “fail safe” and scram the reactor. The terminology of “fail safe” for a power failure means that energy from the power system is used to hold the control rods in a critical position in the reactor, i.e., spring and gravity forces acting on the safety rods are held in check with an electromagnet. Loss of power de-energizes the holding magnet and the rods containing fuel are accelerated out of the reactor to their safe, stable positions.

Although every effort has been made to make the standard instruments used in the reactor “fail safe”, the very nature of their electronic operation makes this quality intrinsically imperfect. However, instrument failure as a potential danger in the reactor operation is decreased by having three independent systems.

One of the scram systems in the reactor is connected to a water-level indicator so that the reactor shuts down if the water level decreases by a small amount. This safety device protects personnel from inadvertently operating the reactor with a low water level, as well as stopping operation is a significant leak develops.

A thermal safety fuse is provided which, in the event of a 2% reactivity excursion, will shut down the reactor in a manner such that the accident cannot recur. The basic concept of the fuse involves having a safety device that prevents large overloads, is foolproof (not dependent on the electronic circuits), and is not easily circumvented. The core fuse is a polystyrene plug containing [redacted] of ^{235}U which supports the bottom half of the core and a section of the reflector.

The higher uranium density loading results in heat being generated at a higher rate in the fuse than in the core. The temperature in the fuse rises about twice as fast as it does in the core. At a core temperature of about 120ºC, the fuse melts and the core separates, thereby completely shutting down the reactor. This temperature is not reached except in cases of large energy releases, such as an excursion. Polystyrene is used as the fuse material because of its resistance to changes in physical properties induced by radiation. The melting point is unaffected below doses of 100 megarad, and therefore the properties of the fuse are not affected by normal operation for many years. Even if the safety and control rods remain in the core, the separation of the two halves of the core would cause the reactor to go subcritical.

In the event of an earthquake of horizontal amplitude greater than 0.16 cm (1/16 inch), the earthquake switch is designed to shut down the reactor.[11] The safety rods are spring loaded and operate very nearly independently of the gravitational field orientation. Thus this safety system makes the reactor safe in a major tremor.

The reactor itself is designed to be structurally earthquake proof: continual horizontal accelerations of 0.6 g will not overturn or otherwise damage the system. Vertical oscillations of comparable magnitude are allowed for in the design safety factor. If the support does break and the reactor overturns, the control and safety rods, which are

---

[11] A.T. Biehl, R.P. Geckler, S. Kahn, and R. Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, CA, October 1957.

housed within a compartment that is enclosed by the main water shield tank, will not be pushed into the core. Finally the concrete shielding walls should prevent severe tilting of the reactor.

Some additional accidents or events that might conceivably result in the release of radioactive materials from the reactor are considered below. In the event of fire, damage to the reactor is not considered likely, since the AGN-201 reactor superstructure is intrinsically fireproof, as is the reactor room. The reactor may be sprayed with water, CO_2, or any other fire- quenching material without damage to the reactor tank or fear of inadvertent criticality. The reactor will be shut down and the reactor room locked in the case of fire, and the Reactor Supervisor or designated alternate will be notified. If a fire involving the reactor or laboratory does occur, the reactor will be thoroughly inspected for damage before operation resumes.

In the extremely improbable event of a flood, no special precautions are necessary other than those normally taken in the event of a flood at an industrial site. The reactor will be secured and not operated at this time. The radiological hazards are not severe as the reactor is built to withstand minor flooding (one-foot of water). In the event of a major flood where the reactor might be overturned or carried away, there is no serious problem since the self-contained reactor has been designed to withstand such an emergency. Further, because the core resides within a water-tight shield tank, there is no risk of inadvertent criticality in the event of complete immersion of the reactor.

It is extremely unlikely that a storm could damage the AGN-201 reactor. However, in the event of a severe storm, the reactor will be shut down and secured. In the event of civil disturbance such as a strike or riot, the reactor will be shut down and secured and guards will be posted at the entrance of the reactor laboratory to prevent unauthorized entry.

In addition to all the above safety features and administrative controls, there exists a negative temperature coefficient in the reactor core. The temperature of the reactor will vary during normal operating conditions as well as during an excursion. In both cases, the change in temperature will cause a change in reactivity. The temperature equilibrium rise of the core can be shown to be on the order of 2ºC when the reactor is operated at 5 watts. Thus, under this condition, the steady state temperature is essentially ambient temperature.