SAFETY ANALYSIS REPORT

for the

PURDUE UNIVERSITY

PUR-1 REACTOR

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1 THE FACILITY

1.1 Introduction

This report is submitted in support of the application for renewal of the operating license (R-87) for Purdue University Reactor (PUR-I) for a period of 20 years, and a power increase from 1 kW, to 10 kW continuous steady state operation, with a maximum allowed power of 12.5 kW.

The reactor is located in the Nuclear Laboratories in the Duncan Annex of the Electrical Engineering Building on the eastern edge of the campus in West Lafayette, Indiana. The Duncan Annex is of brick and concrete block construction and was originally built as a high voltage laboratory. In 1962 the reactor was built in half of the existing high voltage laboratory, which was a high bay area. Offices, classrooms, and laboratories had been built in the remainder of the original building.

1.2 Summary and Conclusions on Principal Safety Considerations

The design power level for PUR-1 is 10 kW, and the reactor has operated safely since its construction in 1962. The analyses presented in this SAR support the continued operation of PUR-1, and also support the case for a power uprate to continuous operation at 10 kW, and maximum short term power of 12.5 kW.

The limiting safety system settings then of 110% and 120% power (11 kW and 12 kW, respectively) are then adequate to protect the reactor and fuel. Even in the unlikely case of a failure of the reactor protective system, the reactor is self-protecting, with a calculated maximum power level of 2.38 MW, and a maximum clad temperature of 133°C, which is still well below the safety limit of 530°C.

1.3 General Description

The PUR-I is a 10 kW pool type reactor, previously licensed for operation at 1 kW, utilizing MTR type enriched fuel plates, which are graphite reflected, and light water moderated and cooled. It was designed and built by Lockheed Nuclear Products of Lockheed Aircraft Corp., Marietta, Georgia.

The reactor is controlled by three blade-type control rods located in the core region of the reactor. There are two shim-safety rods made of solid borated stainless steel, utilizing a magnetic clutch between the blades and the lead screw operated drive mechanisms, and a regulating rod which is a screw operated direct drive and made of hollow stainless steel. Each control blade is protected by an aluminum guide plate on each side within the fuel assembly.

Fuel movement is only by a fuel handling took, [redacted].

1.4 Shared Facilities and Equipment

The reactor facility is located within the former high voltage laboratory (Duncan Annex) in the Electrical Engineering Building. This space was converted prior to the construction of PUR-1 to also house classrooms and laboratories.

1.5 Comparison with Similar Facilities

Similar research reactors are in use at the University of Missouri at Rolla, and the Ohio State University. The safe operating histories of these reactors, and PUR-1 demonstrate the reliability and safety of these systems. Both Missouri-Rolla and Ohio State are licensed for operation at much higher powers (200 kW for Rolla, and 600 kW for Ohio State), with similar reactor systems. The safe operation of these reactors at their respective higher powers also supports the case for an uprate for PUR-1.

1.6 Summary of Operations

The PUR-1 reactor has been in operation since 1962. It is used for teaching and research to support the mission of Purdue University Nuclear Engineering, and the university as a whole. The reactor operates about 90 times per year on average, and presently has three licensed senior operators. There have been periods, however, when there has only been one operator on staff.

1.7 Facility Modifications and History

Table 1-1 summarizes the facility modifications and history.

Table 1-1: Summary of amendments and changes to the PUR-1 reactor facility.

May 1964	Amendment 1	Permit 10 kW Operation
December 1965		Installation of pool traversing mechanism completed
July 1966	Amendment 2	License Renewal
October 1968	Change 1	Installation of stainless steel liner
September 1969		Installation of air conditioner
January 1972	Change 2	Change of pH of pool water
February 1974	Change 3	Regeneration of demineralizer procedure change
November 1978	Amendment 3	Technical Specifications
August 1980	Amendment 4	Physical security plan
February 1981		Installation of catwalk around air conditioner
March 1981	Amendment 5	Physical security plan
September 1982	Amendment 6	Technical specifications—revision 1: Surveillance intervals, scram initiation
October 1982	Amendment 7	Technical specifications—revision 2: Minor typographical modifications
April 1983	Amendment 8	Technical specifications—revision 3: Surveillance intervals, RCO qualifications
August 1988	Amendment 9	License Renewal
February 2000	Amendment 10	Technical specifications—revision 4: CORO members
June 2007	Amendment 11	Possession limit increase
August 2007	Amendment 12	HEU to LEU conversion order

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The PUR-I reactor is located in the Duncan Annex of the Electrical Engineering Building on the campus of Purdue University in West Lafayette, Tippecanoe County, in the State of Indiana, as shown in Figures 2.1, 2.2, and 2.3. The Lafayette-West Lafayette area is about 60 miles northwest of Indianapolis, the State Capitol, and about 140 miles south-southeast of Chicago, Illinois.

2.1.2 Population Distribution

According to the 2000 census summary for Tippecanoe County, the total population was 148,955. The estimate of the county population for 2006 was 156,169[1]. Figure 2-4 shows the population within 1, 2, 4 and 8 km from the reactor location (summarized in Table 2-1) and Figure 2-5 shows the projected population in 2030. It should be noted that the population projections, given to Purdue by the Tippecanoe Area Planning Commission, use a different system for calculation, and are not as accurate as the 2000 population report.

Table 2-1: Population Data for Reactor Vicinity, centered on reactor location.

Circle radius	Population (2000)	Projected 2030 Population
1 km	17,156	18,325
2 km	31,352	31,992
4 km	60,828	68,628
6 km	93,761	117,271
8 km	117,285	165,374

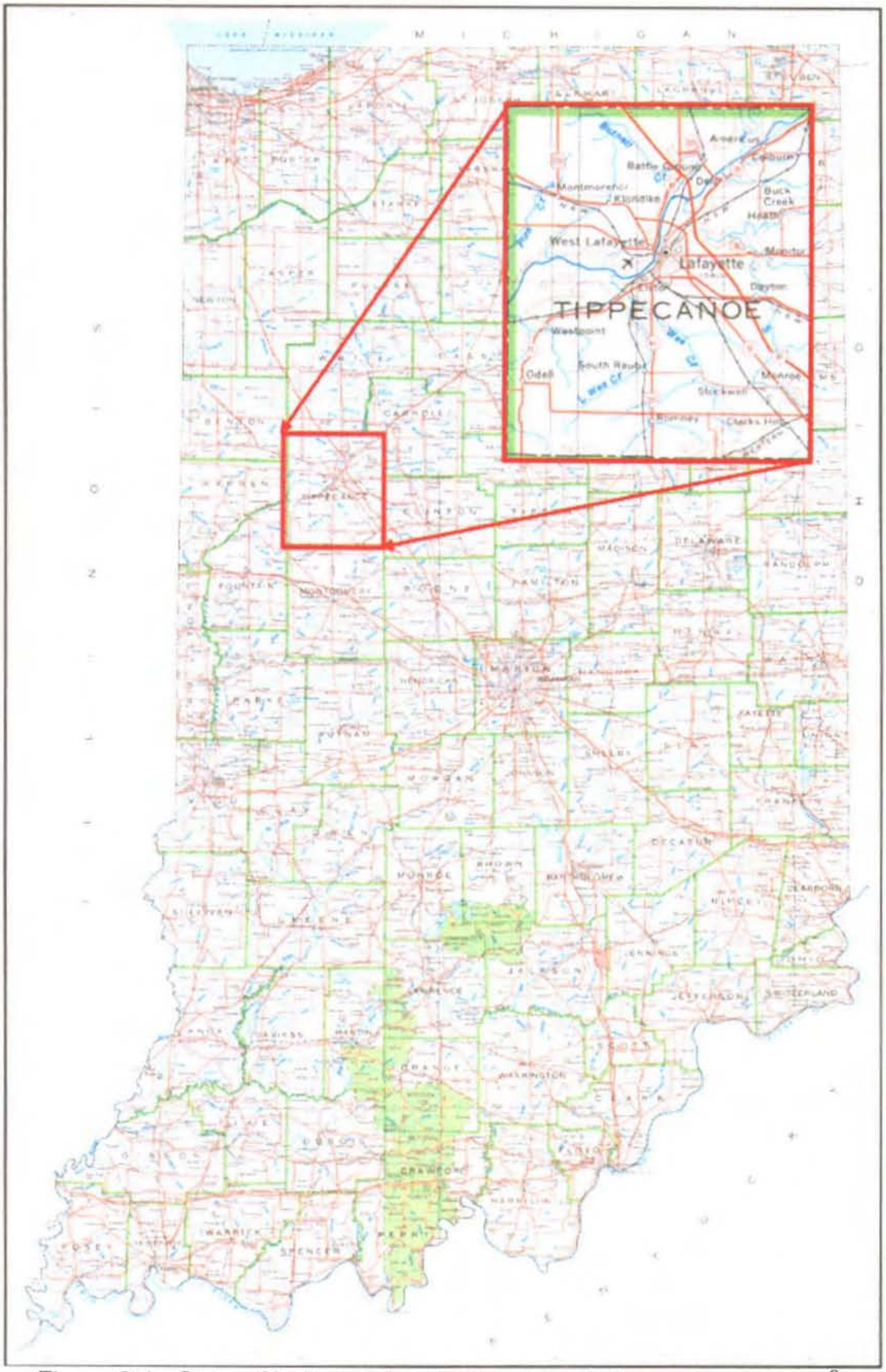


Figure 2-1: State of Indiana, showing location of Tippecanoe County.[2]

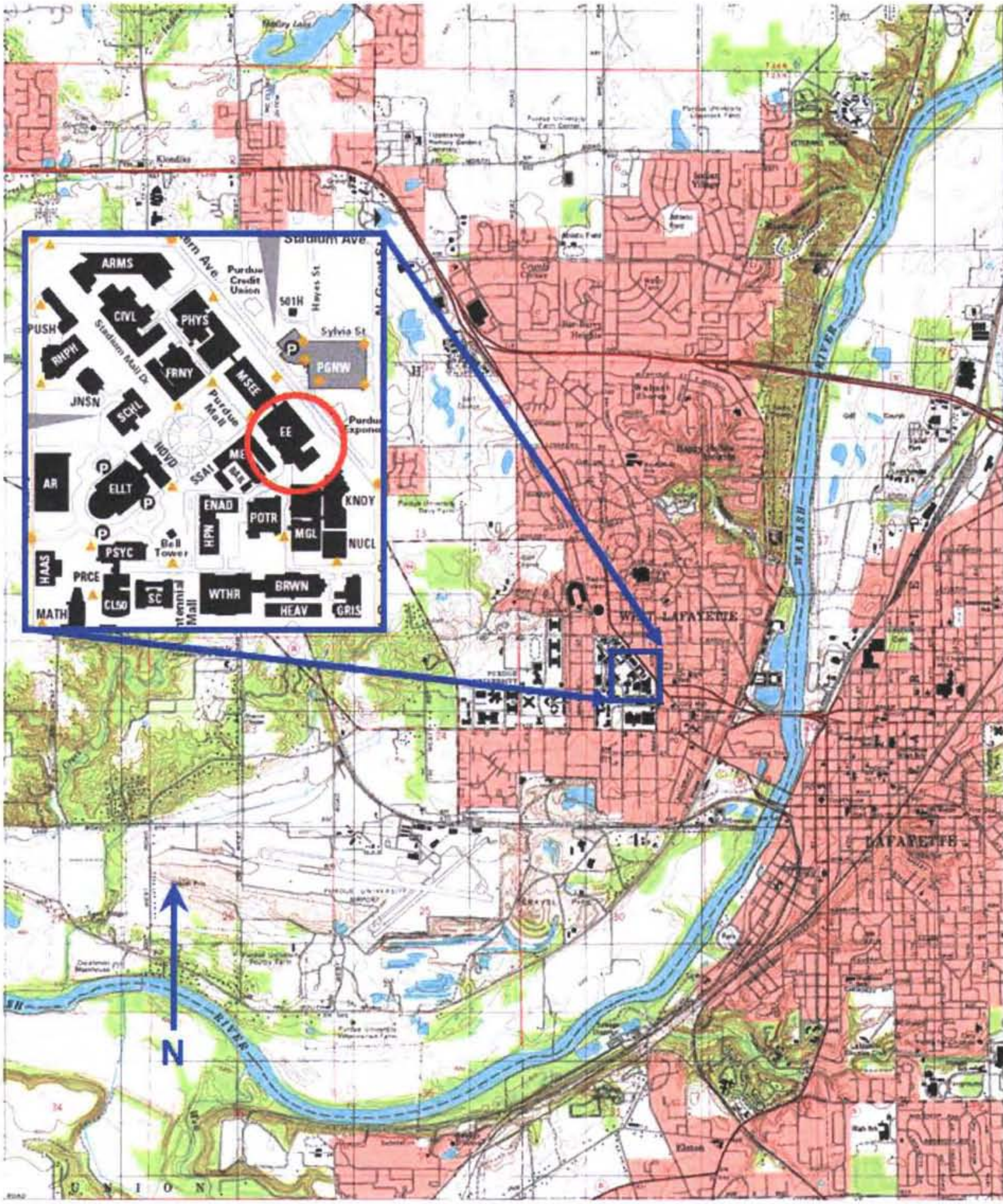


Figure 2-2: Map of West Lafayette, Indiana, showing inset picture of the location of the Electrical Engineering Building on the Purdue Campus.[2]

[redacted]

Figure 2-3: Map of the Duncan Annex of the Electrical Engineering Building

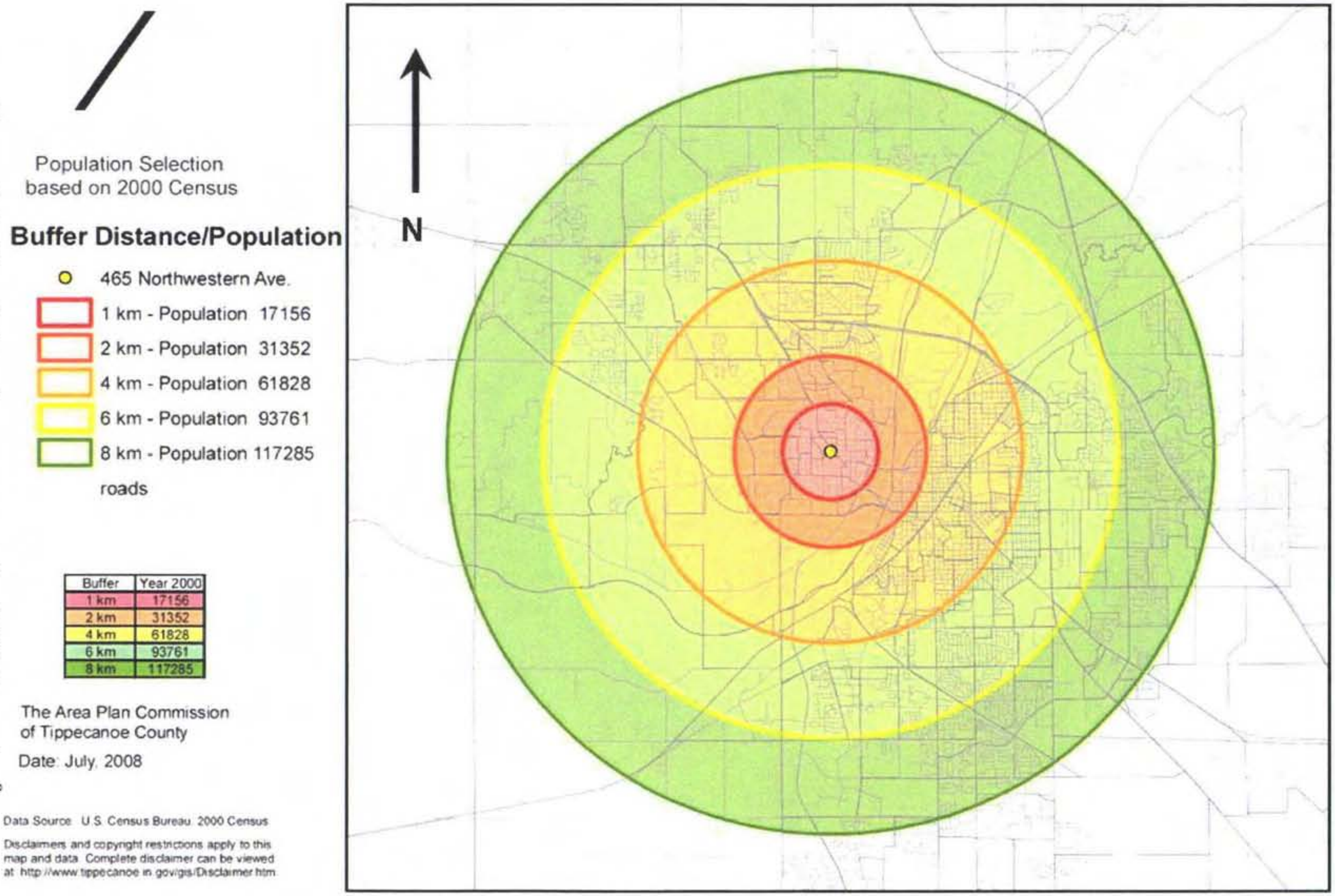
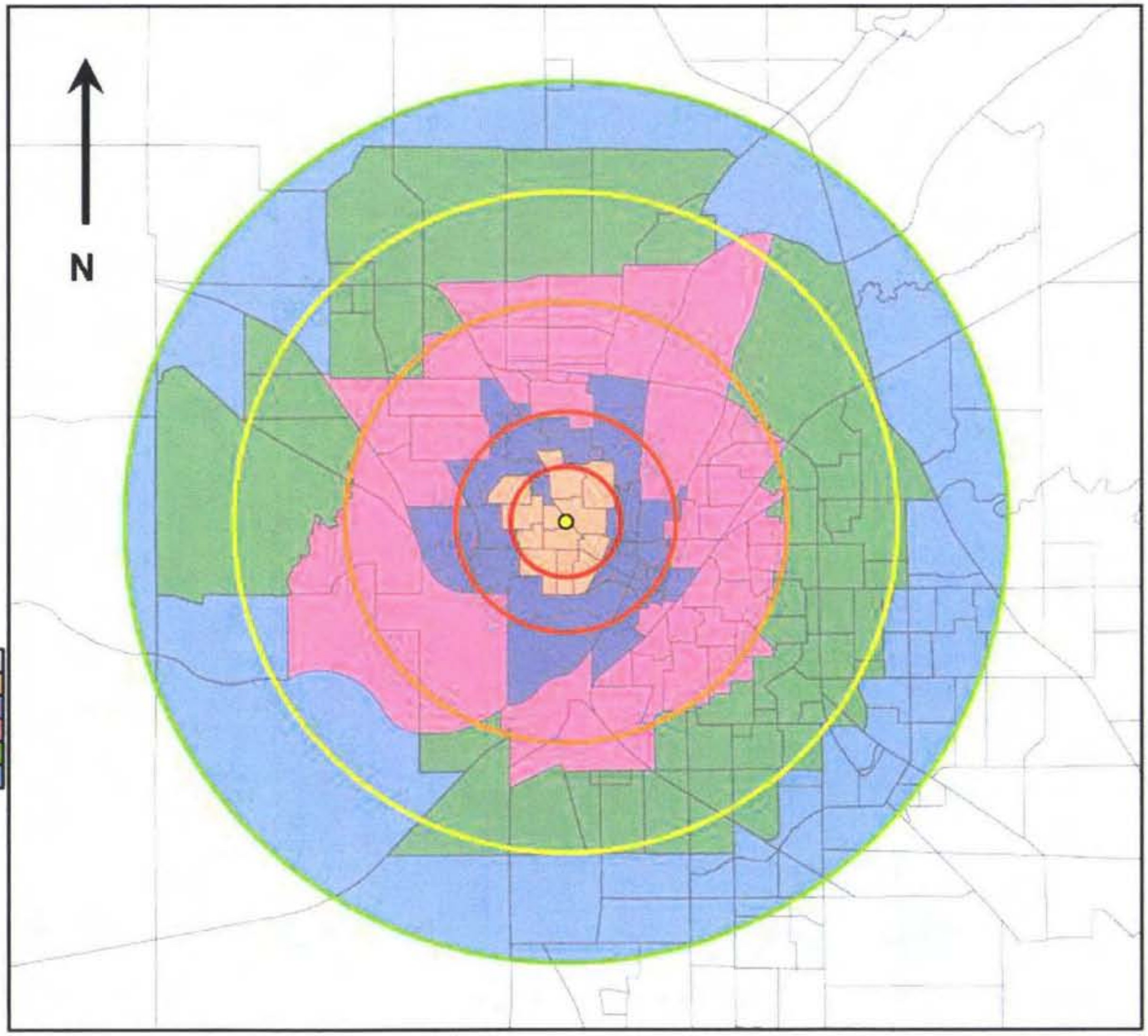


Figure 2-4: Population of area surrounding PUR-1 reactor.[3]



Population Selection
based on Traffic Zones

Buffer Distance

- 465 Northwestern Ave.
- 1 km
- 2 km
- 4 km
- 6 km
- 8 km
- Traffic Area Zones

Population

Buffer	Year 2003	Year 2030	Change
~1 km	18630	18325	-305
~2 km	32421	31992	-429
~4 km	67971	68628	657
~6 km	101380	117271	15891
~8 km	129587	165374	35787

The Area Plan Commission
of Tippecanoe County

Date: July, 2008

Data Source: APC Transportation Plan for 2030 (Socioeconomic Data Table for 2003 and 2030)

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Figure 2-5: Population projection for 2030 in area surrounding PUR-1 reactor.[3]

2.2 Nearby Industrial, Transportation and Military Facilities

The Purdue University enrollment for 2007-08 was 39,102 full and part-time students, and Purdue University employs approximately 15,304 faculty and staff members (2007-08). So, an approximate total campus population is approximately 54,400 at peak times. The data for the last ten years is detailed in

Table 2-2: Purdue University campus population detail for 1998-2008.[4]

	98-99	99-00	00-01	01-02	02-03	03-04	04-05	05-06	06-07	07-08
Full-Time Students	32,788	33,725	33,907	34,442	34,563	34,867	34,745	34,968	35,497	35,549
Part-Time Students	4,090	4,037	3,964	3,766	4,001	3,980	3,908	3,744	3,731	3,553
Total Students	36,878	37,762	37,871	38,208	38,564	38,847	38,653	38,712	39,228	39,102
Total Faculty/Staff	12,888	13,144	13,411	13,831	14,052	14,329	14,636	14,966	15,217	15,304
Total Campus Population	**49,766**	**50,906**	**51,282**	**52,039**	**52,616**	**53,176**	**53,289**	**53,678**	**54,445**	**54,406**

Purdue University owns an airport (LAF) at the southwest edge of the West Lafayette Campus, as shown in Figure 2-6 below. It has two runways, the longest of which is Runway 10/28, which is 6600'x150', and the other is Runway 5/23, which is 4230'x100'. The Purdue airport averages about 130,000 aircraft operations annually (115,000 for the calendar year 2007), and it is the second busiest airport in Indiana.

The university has a thriving Aviation Technology program, the flight instruction department of which constitutes the majority of the air operations at the Purdue Airport. During the fall and spring semester, the airport is the busiest, with an average of 750 takeoffs and landings per day. The maximum daily takeoffs and landings is 1000, and the minimum is zero (0), on Christmas day. During the summer, the airport sees and average of 30 flight operations per day.[5]

The students primarily use Runway 5/23 (outlined in red on the figure), the nearest end of which is 6344 feet (1 93 km) to the southwest (228°) from the Duncan Annex of EE. Only light aircraft can use this runway, due to its length. The approach to this runway passes near the EE building. Larger aircraft must use the longer runway (10/28), which does not direct aircraft either on approach or takeoff near the EE building.

Due to the size restrictions of aircraft that use the 5/23 runway, and the heavy concrete block, concrete and brick, windowless construction of the Duncan Annex, the location of th[redacted], and the fact that no aircraft have crashed on this section of campus in the history of the program, damage due to a crashed light aircraft does not pose a significant threat to the safe operation of the reactor. Should an aircraft impact the building, the location of the reactor [redacted] in a tank of water will mitigate any potential immediate damage, and the close proximity of the Purdue Fire Department will ensure that the public safety is maintained in the event of extensive fire damage or explosions.

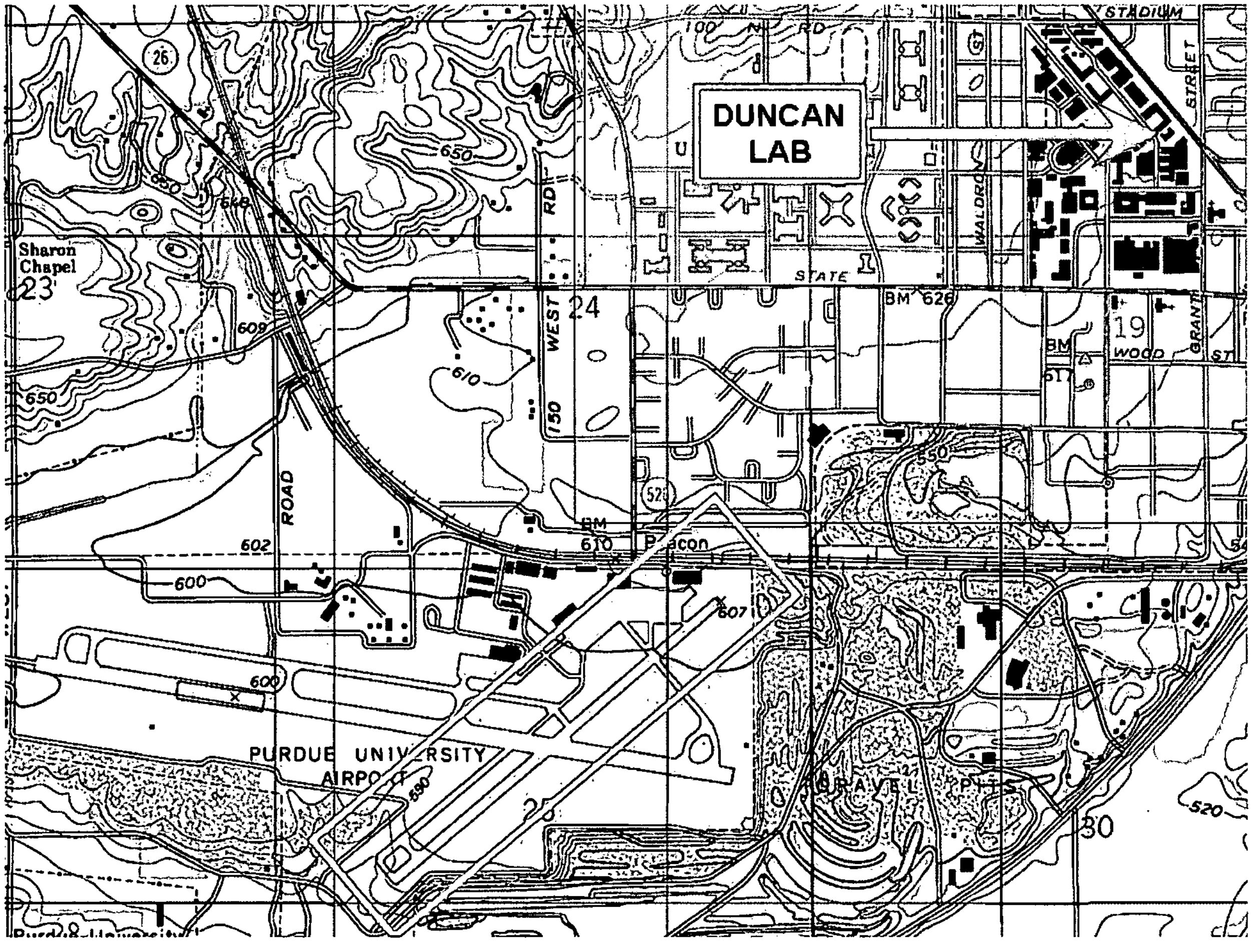

Figure 2-6: Map showing the location of the Purdue University airport in relation to the location of PUR-1.[2]

2.3 Climatology and Meteorology

2.3.1 General and Local Climate

The climate of the county is continental with hot summers and cold winters. The seasons are strongly marked, and the weather is frequently changeable. Climatological data available from NOAA for West Lafayette are summarized in

Table 2-3, Table 2-4, and Table 2-5. The tables show the conditions are measured at the West Lafayette station (West Lafayette 6NW), where the latitude is 40°28', the longitude is 87°00', and the ground elevation is 705 feet.

Table 2-3: Climatography Data for West Lafayette, Indiana; Mean and Extreme Temperatures[6].

Mean* (°F)				Extremes (°F)										
Month	Daily Max	Daily Min	Mean	Highest Daily†	Year	Day	Highest Month Mean	Year	Lowest Daily†	Year	Day	Lowest Month Mean*	Year
Jan	31.5	15.0	23.3	68	1906	20	35.9	1990	-24+	1985	20	7.7	1977
Feb	36.8	19.0	27.9	73	2000	26	38.9	1998	-23	1963	26	12.4	1978
Mar	48.4	29.1	38.8	87	1910	24	46.5	1973	-12	1960	1	29.6	1984
Apr	60.9	39.2	50.1	90	1930	11	55.7	1985	7	1982	7	44.6	1982
May	72.5	50.3	61.4	96+	1911	27	68.9	1977	26+	1966	10	56	1997
Jun	81.4	59.6	70.5	104	1934	1	75.0	1991	35	1992	22	65.8	1972
Jul	84.5	63.0	73.8	111	1936	14	77.9	1983	42+	1972	5	70	1996
Aug	82.5	60.6	71.6	103	1918	5	77.7	1995	35	1965	29	66.7	1992
Sep	77.0	52.9	65.0	100	1933	9	69.4	1978	25	1995	23	59.4	1993
Oct	64.8	41.6	53.2	90+	1922	2	61.3	1971	18	1925	30	47.3	1988
Nov	50.0	32.2	41.1	78+	1930	19	46.8	1999	-3	1930	28	32.9	1976
Dec	37.0	21.1	29.1	71+	1982	3	38.6	1982	-22	1989	22	16.5	2000
Annual	**60.6**	**40.3**	**50.5**	**111**	**Jul 1936**	**14**	**77.9**	**Jul 1983**	**-24+**	**Jan 1985**	**20**	**7.7**	**Jan 1977**

* Derived from 1971-2000 serially complete data
† Derived from station's available digital record: 1901-2001
+ Also occurred on an earlier date(s)

Table 2-4: Climatography Data for West Lafayette, Indiana; Mean Days Information[6]

Month	Mean Number of Days*					
	Days Max ≥ 100°F	Days Max ≥ 90°F	Days Max ≥ 50°F	Days Max ≤ 32°F	Days Min ≤ 32°F	Days Min ≤ 0°F
Jan	0.0	0.0	2.4	15.9	28.7	5.6
Feb	0.0	0.0	4.6	10.7	24.5	3.5
Mar	0.0	0.0	13.1	3.3	20.5	0.1
Apr	0.0	0.0	24.3	0.2	7.8	0.0
May	0.0	0.4	30.7	0.0	0.6	0.0
Jun	@	4	30	0.0	0.0	0.0
Jul	0.1	5.7	31	0.0	0.0	0.0
Aug	0.0	3.2	31	0.0	0.0	0.0
Sep	0.0	1.7	30	0.0	0.3	0.0
Oct	0.0	@	28.6	0.0	5.5	0.0
Nov	0.0	0.0	14.5	1.7	16.6	0.0
Dec	0.0	0.0	4.4	10	26.7	2.4
Annual	**0.1**	**15**	**244.6**	**41.8**	**131.2**	**11.6**

* Derived from 1971-2000 serially complete data
@ Denotes mean number of days greater than 0 but less than 0.05

The average annual temperature is about 50°F. The mean temperature in January, the coldest month, is 23°F, and in July, the warmest month, is 73.8°F. About nine days per year the temperature falls below zero, and about 137 days per year the temperature goes below freezing (32°F).

Table 2-5: Precipitation Normals for West Lafayette, Indiana
(Station: West Lafayette 6 NW)

PRECIPITATION NORMALS(Total in Inches)												
JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	ANNUAL
1.79	1.57	2.84	3.57	4.35	4.24	4.00	3.68	2.98	2.73	3.08	2.43	37.26

The average annual precipitation is 35.68 inches. July is the wettest month with 4.74 inches, and February is the driest month with 1.41 inches of precipitation. Prevailing winds are from the west or southwest during the winter and from the south during the summer. Wind velocity is highest in February and lowest in August.[1]

2.3.2 Weather

Wind conditions as measured at the Purdue University Airport (West Lafayette 6NW) are summarized over the period 1977 to 2006, and are detailed in Table 2-6. These data indicate an annual mean wind speed of 8.78 miles per hour and a maximum wind speed of 72 miles per hour. According to the Unified Building Code, 1985 edition, the Purdue University lies in the maximum wind zone of 80 miles per hour, which translates to a wind load of 17 pounds per square foot. Buildings at Purdue University are designed to withstand this wind load.

Table 2-6: Average and Maximum Wind Data Measured at Purdue University Airport for 1977-2006[7]

Month	Avg. Wind Direction* (degrees)	Avg. Wind Speed (mph)	Max. Wind Speed† (mph)
January	216.91	10.27	64
February	205.39	9.84	62
March	198.20	10.51	62
April	195.53	10.36	63
May	193.11	8.73	72
June	191.52	7.55	62
July	200.37	6.90	57
August	191.07	6.44	60
September	196.03	7.09	57
October	198.89	8.35	63
November	206.99	9.58	63

* Denotes the compass direction the wind is blowing from (0° = North).

† Max. Wind Speed is not the average maximum. It is the actual maximum that occurred during the 30 years of data.

December	215.05	9.71	52
Annual	**200.75**	**8.78**	**72**

2.3.3 Severe Weather

This region of the United States is subjected to tornado activity, primarily during the late spring and early summer months. Table 2.4 shows the tornados occurring in Tippecanoe County for the period from 1950 through February 2008. Thirty-eight tornados occurred over this period, which averages to less than one per year. The probability of damage due to a tornado is minimal.

Table 2-7: Tornados Reported in Tippecanoe County, Indiana between 01/01/1950 and 02/28/2008[8]

#	Location or COUNTY	Date	Time	Fujita Scale	Deaths	Injuries	Property Damage ($)
1	TIPPECANOE	6/13/1953	21:00	F1	0	0	0K
2	TIPPECANOE	4/3/1956	17:00	F2	0	0	25K
3	TIPPECANOE	3/6/1961	6:05	F1	0	0	0K
4	TIPPECANOE	4/22/1963	20:15	F2	0	0	3K
5	TIPPECANOE	6/10/1963	12:00	F1	0	0	3K
6	TIPPECANOE	4/11/1965	18:07	F4	0	10	0K
7	TIPPECANOE	9/14/1965	20:15	F2	0	0	250K
8	TIPPECANOE	6/24/1967	12:30	F2	0	0	3K
9	TIPPECANOE	5/15/1968	21:51	F2	0	0	3K
10	TIPPECANOE	3/19/1971	2:03	F2	0	0	25K
11	TIPPECANOE	5/29/1973	14:20	F0	0	0	0K
12	TIPPECANOE	6/12/1973	9:45	F1	0	0	25K
13	TIPPECANOE	6/12/1973	10:29	F1	0	0	0K
14	TIPPECANOE	4/1/1974	16:32	F2	0	0	25K
15	TIPPECANOE	3/12/1976	14:25	F1	0	0	0K
16	TIPPECANOE	3/20/1976	15:20	F4	0	6	2.5M
17	TIPPECANOE	4/10/1978	12:15	F2	0	0	25K
18	TIPPECANOE	4/23/1978	18:00	F1	0	0	250K
19	TIPPECANOE	6/25/1978	15:55	F0	0	0	0K
20	TIPPECANOE	6/25/1978	17:00	F3	0	0	0K
21	TIPPECANOE	7/2/1978	12:15	F1	0	0	25K
22	TIPPECANOE	6/7/1980	16:15	F2	0	0	0K
23	TIPPECANOE	6/24/1981	19:47	F1	0	0	250K
24	TIPPECANOE	3/27/1991	18:10	F0	0	0	0K
25	Lafayette	4/26/1994	23:58	F4	3	70	5.0M
26	Lafayette	1/18/1996	14:30	F0	0	0	0
27	West Lafayette	7/4/1998	1:30	F1	0	0	200K
28	Battle Ground	9/28/1999	18:54	F1	0	1	300K
29	Lafayette	6/11/2003	18:47	F0	0	0	0
30	West Lafayette	6/11/2003	19:00	F0	0	0	0
31	Lafayette	7/21/2003	4:00	F0	0	0	0
32	Romney	5/30/2004	19:26	F0	0	0	0
33	Dayton	5/30/2004	19:29	F2	0	0	1.0M
34	Dayton	7/26/2005	20:00	F0	0	0	10K
35	Romney	4/2/2006	20:25	F1	0	0	50K
36	Cairo	4/14/2006	18:58	F0	0	0	0
37	Americus	4/14/2006	19:18	F1	0	0	30K
38	Odell	6/25/2006	14:10	F0	0	0	3K
	TOTALS:				**3**	**87**	**10.004M**

2.4 Hydrology

Most of Tippecanoe County is covered by glacial drift. The drift ranges in thickness from a thin veneer to about 425 feet and was deposited upon a bedrock surface that was eroded by a preglacial drainage system. Much of the surface drift consists of glacial till. Water-laid cross bedded sand and gravel are associated with the till. The subsurface glacial deposits also include much till with interbedded sand and gravel. Locally, clay deposits are as much as 106 feet thick. Within the drift, five sheetlike water bearing units are differentiated in parts of the county. Ground water within these units occurs under artesian and water-table conditions. Locally these may occur within the same unit.[9]

This area was repeatedly glaciated during the Pleistocene epoch. Before glacial times, a giant drainage way, now know as the Teays River, flowed from the Appalachian Mountains across Ohio, and passed northwestward through the present site of Lafayette-West Lafayette.[10] Illinoian ice dammed the preglacial Teays River channel and ponded the relative small Glacial Lake Lafayette. An outlet channel, developed to drain this proglacial lake, was subsequently perpetuated as the present Wabash River drainage line southwestward from the Lafayette-West Lafayette area.[11]

The elevation of the Purdue University campus is approximately 706 feet and the level of the Wabash River is approximately 510 feet. With this difference of over 100 feet the flow of both surface water and ground water is in a generally easterly and southerly direction toward the Wash River, which flows around two sides of the campus.

Any leakage of contaminated water from the PUR-I represents no potential hazard to either the West Lafayette or Purdue University water supply, since these flows are away from the well fields of both. The Wabash River represents a natural barrier between the reactor and the Lafayette well fields, so no potential hazard exists there.

2.5 Geology and Seismology

2.5.1 Regional Geology

The county lies within the Tipton Till Plain of Indian and is a section of the Till Plains subprovince of the U. S. central Lowlands physiographic province. Most of the soils in this area are derived from the glacially deposited material. Extensive upland areas are covered with a thin mantle of loose deposits. A few areas are covered with soils of alluvial, colluvial or organic origin.[11] Glacial drift covers the bedrock to a depth ranging from a few feet to more than 300 feet. The underlying bedrock consisting of flint, shale, sandstone, and limestone of the Mississippian period, is exposed as rock terraces in the Wabsh Valley and on the upland in the western part of the county.[10] Purdue University is located above an extensive glacial deposit of sand and gravel.

The land surfaces of Tippecanoe County are flat to rolling, except where the major streams have cut deeply into the surface. The entire county lies within the drainage basin of the Wabash River and its tributaries.[9] The land slopes generally southwestward with the streams flowing westward. Two main tributaries, the Tippecanoe River and the Wild Car Creek enter the Wabash upstream from the campus. Minor tributaries include Little Pine Creek, Indian Creek, Burnetts Creek, Mott's Creek, Sugar Creek, Buck Creek, Wea Creek, and Flint Creek.

2.5.2 Seismology

The three most significant seismic source zones which are closest to West Lafayette are:

1. The New Madrid area of southeastern Missouri;
2. The Wabash Valley Fault system of southwestern Indiana and southeastern Illinois;
3. The Anna, Ohio area.

Reasonable estimates of the maximum magnitude events which could occur in those areas give values of 7.4, 6.6 and 6.3 (body wave motion) for the seismic zones, respectively. Based on the distance from these zones (400, 200 and 200 km respectively) and attenuation curves, estimates for peak horizontal acceleration at West Lafayette for maximum magnitude events which could occur at these three seismic zones are approximately 5-15% G.[12] The figures that follow (Figure 2-7 through Figure 2-11) show the earthquake probabilities for the area.

The way, in which the reactor facility was constructed by modifying an existing building with no reinforcing bars tied into the original structure, the reactor pool can be considered a free standing unit in the event of any seismic activity. The reactor pool consists of steel cylinders containing compacted magnetite sand between the cylinders and the 1/3 inch carbon steel tank. The inside of this tank was later lined with 1/16 inch stainless steel. With these barriers to contain the reactor pool water and considering the reactor pool as a free standing unit it is highly unlikely that any reactor water would be lost during any severe seismic activity.

SA 1s 2%50yr PE, 2008

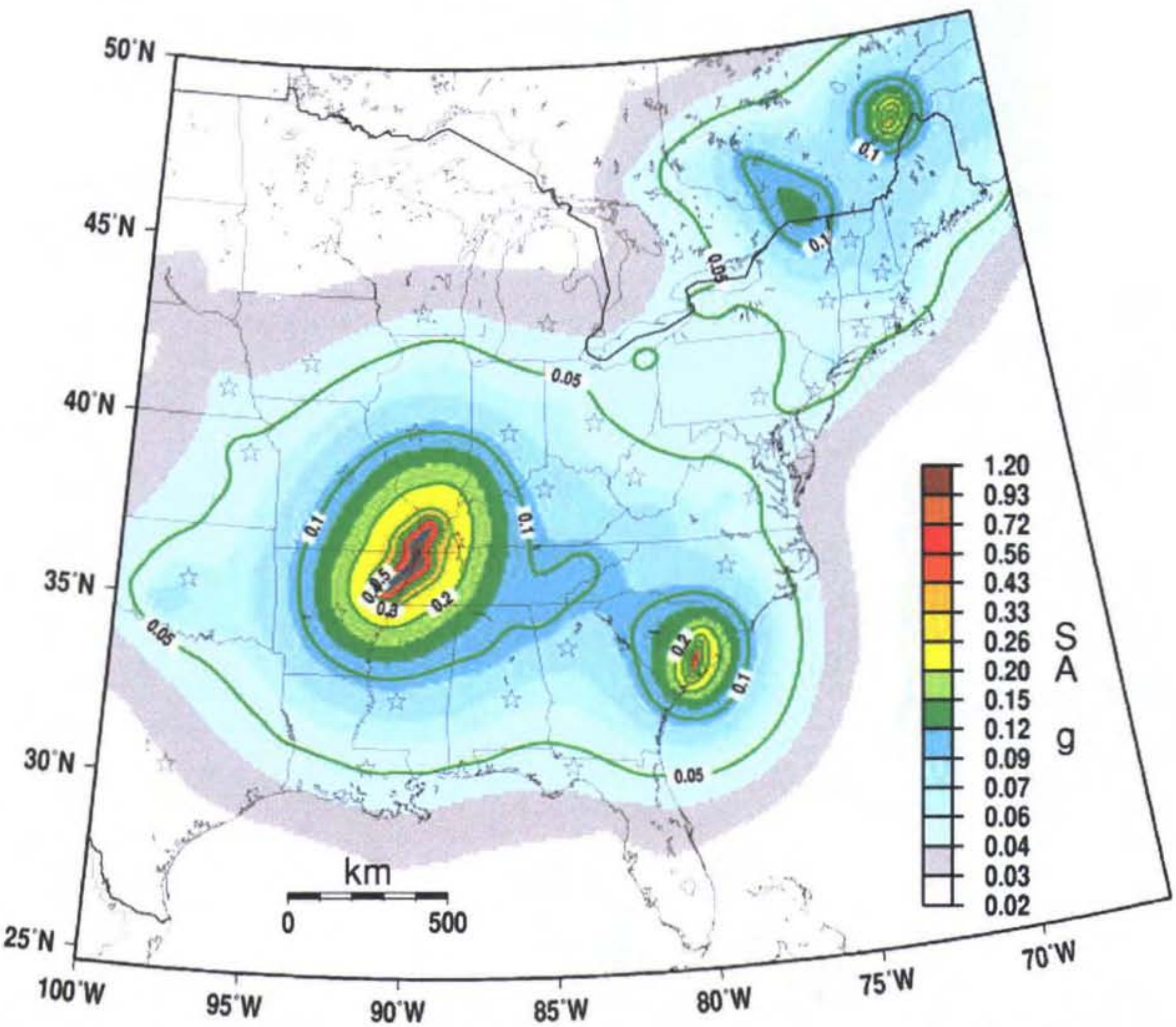

Figure 2-7: Map of the 1-Hz spectral acceleration for 2% probability of exceedance in 50 years for the Central and Eastern United States in standard gravity (g).[13]

SA 0.2-s 2%/50year PE, 2008

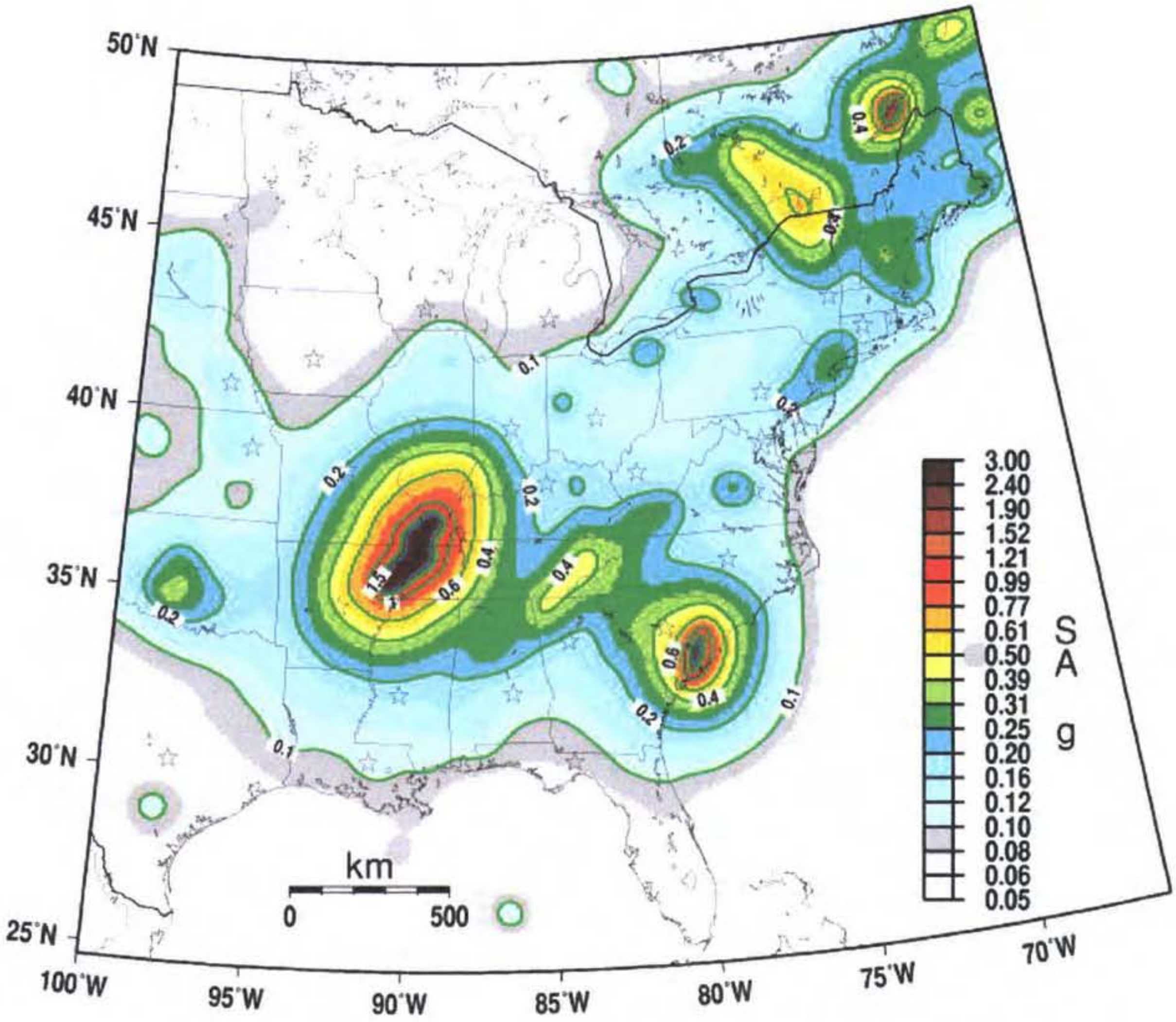

Figure 2-8: Map of the 5-hertz spectral acceleration (SA) for 2% probability of exceedance in 50 years in the Central and Eastern United States in standard gravity (g).[13]

PGA with 2%/50 yr PE, 2008

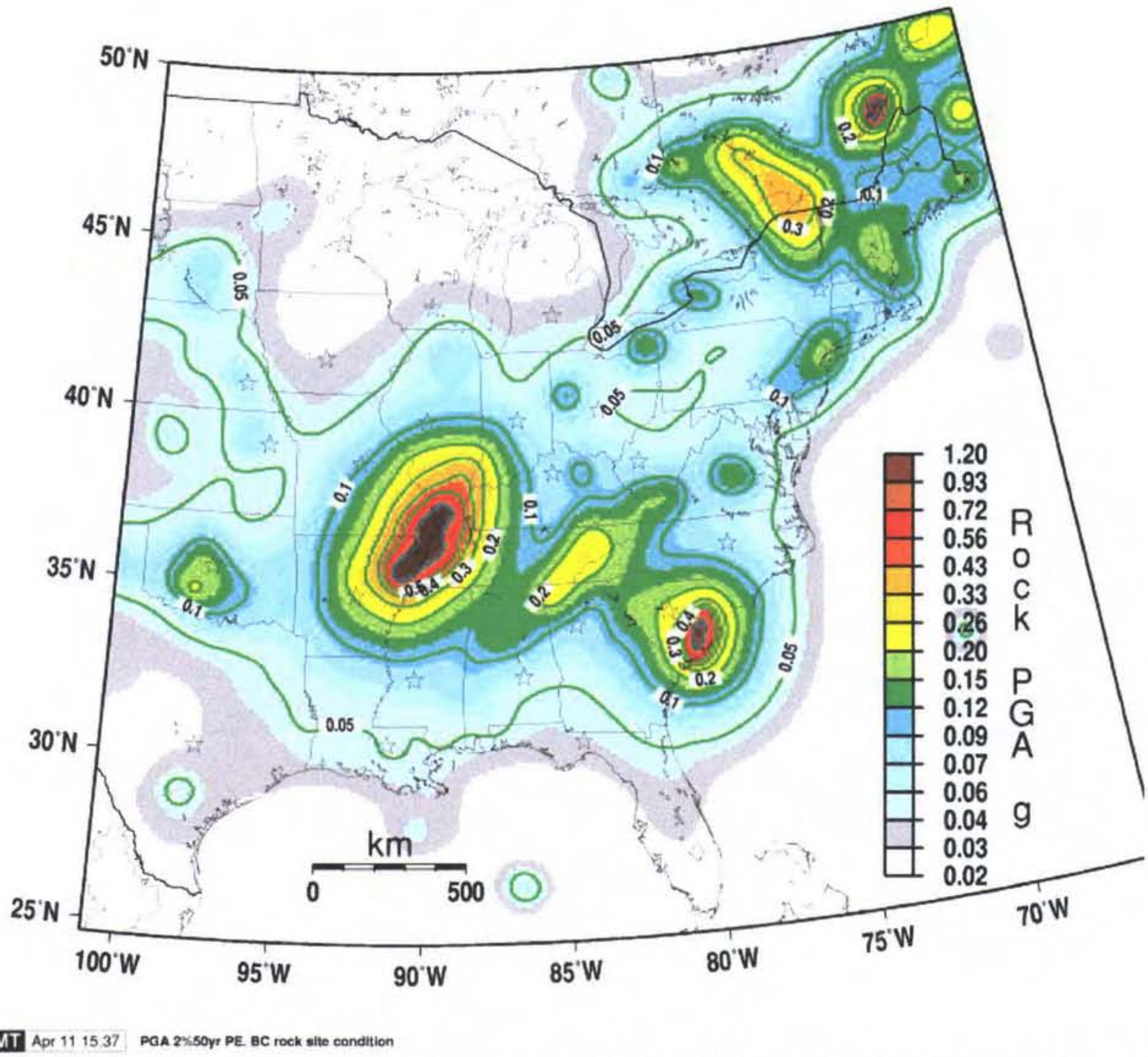

Figure 2-9: Map of peak ground acceleration (PGA) for 2% probability of exceedance in 50 years in the Central and Eastern united States in standard gravity (g).[13]

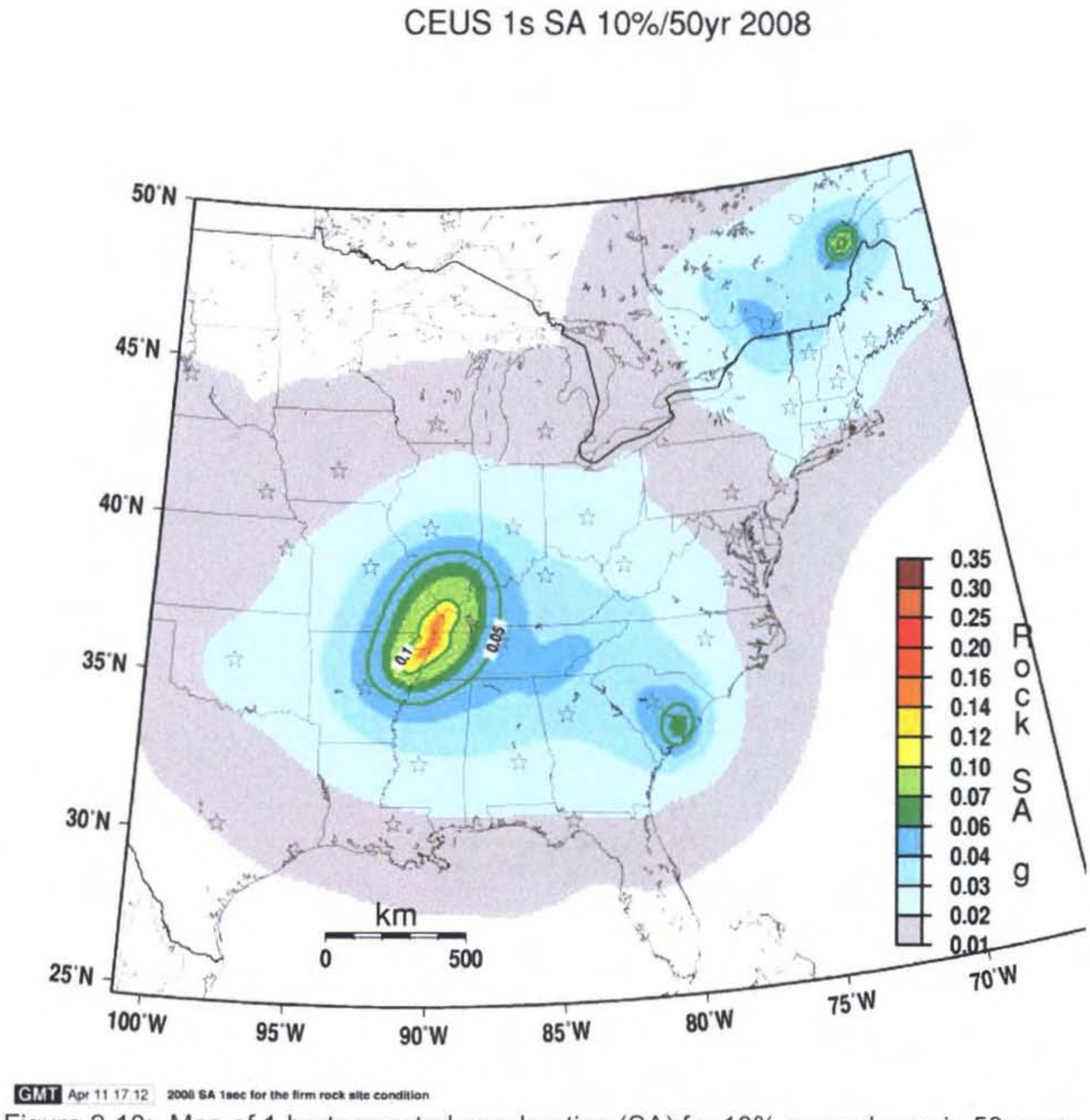

Figure 2-10: Map of 1-hertz spectral acceleration (SA) for 10% exceedance in 50 years in the Central and Eastern United States in standard gravity (g).[13]

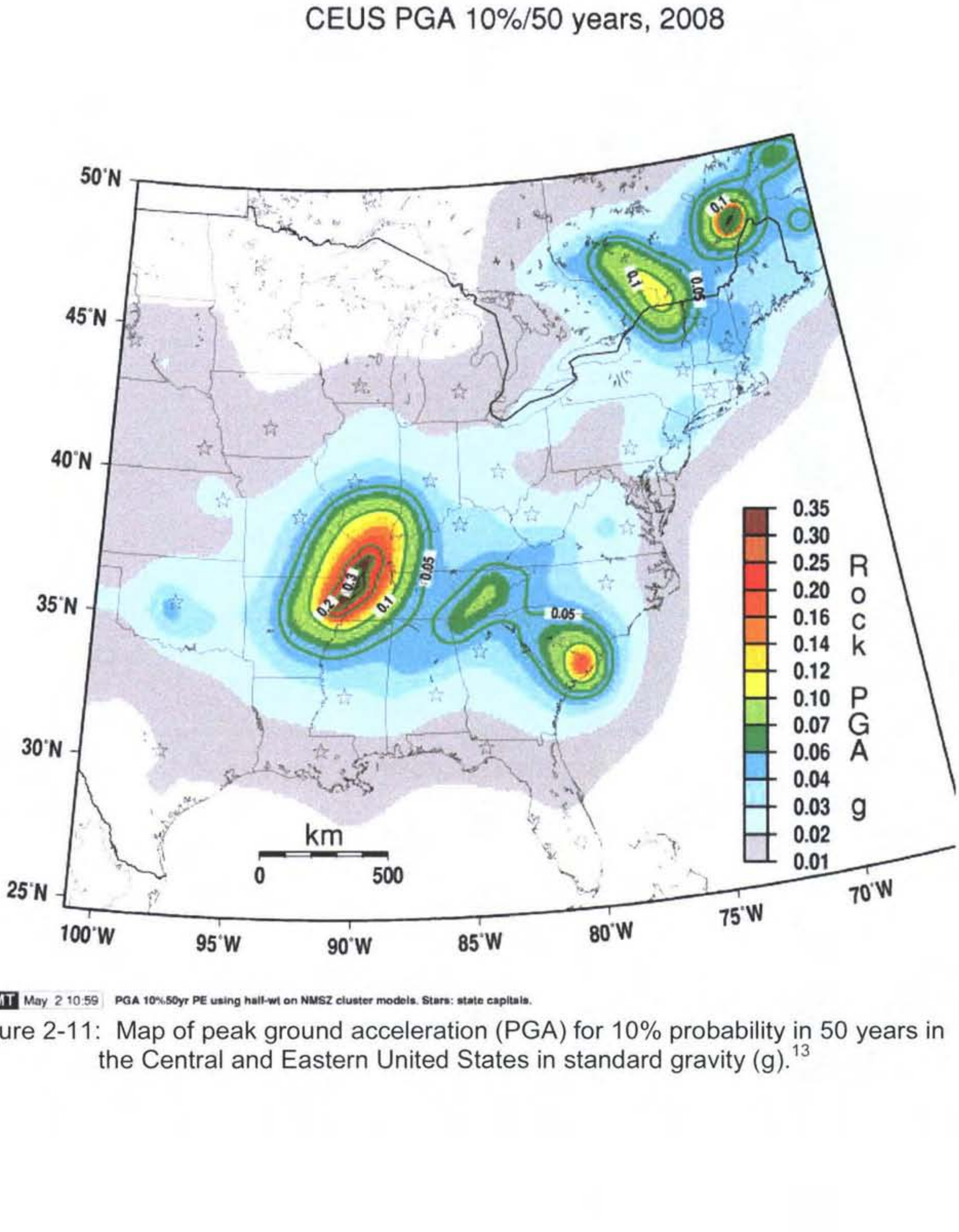

Figure 2-11: Map of peak ground acceleration (PGA) for 10% probability in 50 years in the Central and Eastern United States in standard gravity (g).[13]

2.6 References

[1] U.S. Census Bureau, http://www.census.gov

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[3] Tippecanoe County Area Planning Commission, (2008)

[4] Purdue Data Digest, http://www.purdue.edu/datadigest/

[5] Private Communications with Betty Stansbury, Airport Administrator, Purdue University, (2008).

[6] National Climatic Data Center, NOAA-NESDIS, http://www.ncdc.noaa.gov/oa/climate/normals/usnormals.html, (2004)

[7] Indiana State Climate Office, Purdue University, http://www.iclimate.org, (2008).

[8] National Climatic Data Center, NOAA-NESDIS, http://www.ncdc.noaa.gov/oa/ncdc.html, (2008)

[9] Resenshein, Joseph S., "Ground-Water Resources of Tippecanoe County, Indiana', State of Indiana, Indiana Department of Conservation, Division of Water Resources Bulletin No. 8, 1958.

[10] Ulrich, H.P., Barnes, T.E., and Krantz, B.A., "Soil Survery, Tippecanoe County, Indiana Series 1940, No.22',1959.

[11] Maarouf, Abdelraham M., and Melhorn, Wilton N., "Technical Report No. 61', Purdue University Water Resources Research Center, June, 1975.

[12] Braile, L.W. Professor, Purdue University, private communications.

[13] Petersen, Mark D., et al., "Documentation for the 2008 Update of the United States National Seismic Hazard Maps", Open File Report 2008-1128, U.S. Geological Survey, (2008).

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The Duncan Annex of the Electrical Engineering Building is of brick, concrete block and reinforced concrete construction which was originally designed as a large high voltage laboratory. It was subsequently subdivided into offices, classrooms and laboratories. The reactor is located in the southwest corner on the ground floor in a high bay area of the building. [redacted]

The outside air supply and exhaust are both passed through HEPA filters. The reactor room is maintained at negative air pressure (minimum 0.05 inches of water). All doors to the reactor room have foam rubber seals.

The only floor drain to the sewers is sealed except for a vent opening. This vent is raised about two feel above the floor and has a filtered inverted opening. Condensate from the air conditioner is released to this drain through an opening 12.0 feet above the floor.

3.2 Meteorological Damage

According to the Unified Building Code, 1985 edition, the Purdue University lies in the maximum wind zone of 80 miles per hour, which translates to a wind load of 17 pounds per square foot. Buildings at Purdue University are designed to withstand this wind load.

With this information, and the low incidence of tornado activity in the campus area, tornado damage to the building is very unlikely. The Nuclear Engineering Laboratory is the shelter-in-place location for tornado warnings. [redacted]

3.3 Water Damage

The Electrical Engineering Building, and all of the Purdue Campus, lie well above any flood plain. In 46 years of operation, there has been no standing water in the reactor room. The top [redacted] of the reactor pool stand above floor level, and the sides of the pool are made of 15 inches of reinforced concrete In the unlikely event that this wall were to break, it would only result in 3 inches of standing water on the reactor room floor. None of the reactor instrumentation would be harmed by such an unlikely occurrence.

3.4 Seismic Damage

The way, in which the reactor facility was constructed by modifying an existing building with no reinforcing bars tied into the original structure, the reactor pool can be considered a free standing unit in the event of any seismic activity. The reactor pool consists of steel cylinders containing compacted magnetite sand between the cylinders and the 1/3 inch carbon steel tank. The inside of this tank was later lined with 1/16 inch stainless steel. With these barriers to contain the reactor pool water and considering the reactor pool as a free standing unit it is highly unlikely that any reactor water would be lost during any severe seismic activity.

3.5 Systems and Components

Should the need arise for an emergency shutdown of the reactor, the 2 shim-safety control rods can be scrammed automatically by instrumentation, or manually via two scram buttons: one located within easy reach of the operator on the control panel, the other outside the main personnel access door to the reactor room.

During emergency conditions, the ventilation systems can be shut off by a console switch, and the sealed room will prevent the rapid spread of contamination. During an emergency the air conditioner and the valve on the drain from the condensate holdup tank are shut off with the same switch that shuts off the ventilation system. The condensate would then be held until it is tested by Radiological Control before it is released to the sewer. If contamination is found, it would be disposed of as radioactive liquid waste.

4 REACTOR DESCRIPTION

4.1 Summary Description

The 10 kW PUR-1 reactor described herein was designed and constructed by Lockheed Nuclear Products of Lockheed Aircraft Corp. of Marietta, Georgia. It was designed for continuous steady-state operation at 10 kW, but previously licensed for operation at 1 kW.

PUR-1 is a heterogeneous, pool-type non-power reactor. The core is cooled by natural convection of light water, moderated by light water, and reflected by water and graphite. The reactor is located near the bottom of a water-filled tank surrounded and supported by a concrete shielding structure as shown in Figure 4-1. An aluminum grid plate structure supports the reactor and control mechanisms at the bottom of the pool, with additional support of the control mechanisms provided by a fixture at the top of the pool. Three detectors used for monitoring reactor conditions are located in fixed positions next to the reactor core. And the startup detector is located in a tube affixed to a fuel element in the core, which allows the detector to be removed from the neutron flux when the reactor is at power.

[redacted]

Figure 4-1: PUR-1 Pool Layout

The reactor core is composed of sixteen fuel elements positioned in holes in the aluminum grid plate. The grid plate contains a rectangular matrix of holes to allow the changing of fuel element locations and the insertion of graphite reflector elements to displace reflector water. [redacted] Fuel elements of this general configuration were first designed for and used in the Materials Testing Reactor (MTR) and thus are referred to as MTR-type fuel elements. Three of the fuel elements are fabricated without the four middle plates, providing space for the insertion and movement of the reactor control rods.

Reactivity of the reactor core is changed by the operator moving the control rods that are suspended from fail-safe electromagnets. The ionization chambers used for sensing neutron and gamma-ray fluxes are located near the core. The control console, from which the operator can observe the reactor pool and top structures, is located adjacent to the reactor, and consists of typical read-out and control instrumentation.

Heat removal is achieved by natural convection, with a general flow up through the nozzle at the bottom of the fuel assemblies. The reactor is located within a 6400 gallon cylindrical water tank, 17 feet deep and 8 feet in diameter.

Safety and other operational characteristics of this reactor system are similar to other reactors using the MTR type fuel assembly. The power and flux level of the PUR-I are of adequate range and the experimental facilities are sufficiently flexible to encompass a wide variety of training and research experiments. The reactor is designed so that a minimum of restrictions are imposed on the experimenter, and the console can be readily operated by one person.

Safety is an overriding requirement in a training reactor. Self-limiting features of the PUR-1 core, coupled with carefully designed control instrumentation, assure the highest degree of safety. The safety record of this facility, demonstrated over the past 46 years give proof that the design, construction, and installation of the reactor system, coupled with the administrative control over operation, maintenance, and utilization, are more than adequate to provide protection for the public health and safely.

Table 4-1 and Table 4-2 summarize the key design parameters for the PUR-1.

Table 4-1: Summary of Design Parameters for PUR-1

DESIGN DATA	Value
Design Power Level	10 kW
Fuel Type	MTR Plate
Fuel "Meat" Composition	U_3Si_2-Al
Fuel Enrichment U-235 (nominal)	19.75%
Mass of U-235 per plate (g, nominal)	[redacted]
Fuel Meat Dimensions Width (mm) Thickness (mm) Height (mm)	
Fuel Plate Dimensions Width (mm) Thickness (mm) Height (mm)	
Cladding Composition	6061 Al
Cladding Thickness (mm)	0.381
Dummy Plate Composition	6061 Al
Dummy Plate Dimensions	Same as Fuel
Standard Fuel Assemblies Number of standard assemblies Number of plates per standard assembly	 13 14
Control Fuel Assemblies Number of control assemblies Number of plates per control assembly	 3 8
Total plates in core (fuel and dummy)	[redacted]
Fuel plates in core (current)	
Dummy plates in core (current, expected)	16
Plate spacing in standard assemblies (mm)	3.66
Plate spacing in control assemblies (mm)	4.60

Table 4-2: Summary of key reactor parameters for PUR-1.

REACTOR PARAMETERS	Calculated
Fresh core excess reactivity (%Δk/k)	0.42[1]
Shutdown margin (%Δk/k)	-1.80[1]
Control rod worth (%Δk/k)[1] Shim-safety 1 Shim-safety 2 Regulating Rod	 3.93 2.22 0.27
Maximum reactivity insertion rate $\left(\frac{\%\Delta k}{k\cdot s}\right)$ Shim-safety 1 Shim-safety 2 Regulating Rod	 1.75E-02 8.75E-03 4.66E-03
Avg. coolant void coefficient $\left(\frac{\%\Delta k}{k\cdot\%void}\right)$[2]	-1.93E-1±7%
Coolant temperature coefficient $\left(\frac{\%\Delta k}{k\cdot{}^{\circ}C}\right)$[3]	-9.05E-3±9%
Fuel temperature coefficient $\left(\frac{\%\Delta k}{k\cdot{}^{\circ}C}\right)$[4]	-8.05E-4±10%
Effective delayed neutron fraction (%)	0.784
Neutron lifetime (μs)	81.3

4.2 Reactor Core

The PUR-1 core layout is a sixteen assembly (4x4 array), heterogeneous, light-water moderated, graphite reflected, water cooled reactor fueled with LEU plate-type fuel. The core layout is shown in Figure 4-2. Each of the thirteen standard fuel assemblies in the core can hold up to 14 fuel plates, or a mixture of fuel and dummy plates. The three control elements each hold up to eight fueled plates.

Twenty graphite reflector assemblies surround the core, 6 of which contain a cylindrical aluminum tube normally filled with graphite. These 6 elements comprise the irradiation facility. The graphite can be removed from these tubes and replaced with experiment capsules which can then be irradiated with normal reactor operation.

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[1] From the 2008 Annual Report.
[2] Calculated for the representative range of 0-0.6% void.
[3] Calculated for the representative range of 20-30°C.
[4] Calculated for the representative range of 20-127°C.

[redacted]

Figure 4-2: PUR-1 Grid Plate

The reactor is controlled by three control rods located in the core region of the reactor. There are two shim-safety rods made of solid borated 304 stainless steel, utilizing a magnet clutch between the blades and the lead screw operated drive mechanisms, and a regulating rod, which is a screw operated direct drive and made of hollow stainless steel. Each control blade is protected by an aluminum guide plate on each side within the control fuel assemblies.

Each of the standard assemblies and the control assemblies are contained in a 6061 aluminum container. The standard graphite assemblies and the irradiation facility graphite assemblies are contained in similar 6061 aluminum containers. The startup neutron source is located outside the core in a similar 6061 aluminum container.

4.2.1 Reactor Fuel

The reactor is fueled by standard MTR LEU plates installed in 2007 during the conversion of PUR-1 from HEU to LEU fuel. The LEU fuel is silicide dispersion standard MTR plates manufactured by BWX-Technologies (BWXT) of Virginia. Dummy aluminum plates identical in size to the fuel plates were also manufactured by BWXT, and these are used in place of fuel in the reactor in some locations. The assembly cans that contain the plates were manufactured by General Atomics, of California.

The fuel and dummy plates are inserted into both the standard and control assembly cans individually, each one contained within its own slot. These slots control the plate spacing to close tolerances over the length of the plates, which is a significant improvement over the old design where the plates were pinned together at the four corners. The nominal fuel plate information is detailed in Table 4-3.

Table 4-3: Characteristics of the PUR-1 Fuel Plates

	Design Detail
Fuel Type	U_3Si_2-Al
Fuel "Meat" Composition Enrichment Mass ^{235}U per fuel plate	 U_3Si_2-Al 19.75% [redacted]
Fuel Plate Dimensions Width (mm) Thickness (mm) Height (mm)	[redacted]
Fuel Meat Dimensions Width (mm) Thickness (mm) Height (mm)	[redacted]
Cladding Type	6061 Al
Cladding: Along width (mm) Along thickness (mm)	 3.63 (min) 0.381

Figure 4-3 and Figure 4-4 show the design of the wall spacers that control the plate locations and channel thicknesses for the standard and control elements.

[redacted]

Figure 4-3: Standard assembly can detail, showing wall spacers.

[redacted]

Figure 4-4: Control assembly can detail, showing wall spacers.

Figure 4-5 and Figure 4-6 show the cross-sectional view of the standard and control assemblies for the PUR-1 reactor. Figure 4-5 shows the handle assembly at the top of the assembly can, which can be removed for insertion and removal of the fuel plates. The Control assemblies have no such handle.

Figure 4-5: Standard fuel assembly.

Figure 4-6: Control fuel assembly.

The nominal plate-to-plate spacing, and the nominal plate-to-wall spacing is shown in Table 4-4.

Table 4-4: Channel Types and Thickness in PUR-1 Assemblies

	Plate-to-plate (mils)		Plate-to-wall (mils)	
	Standard	**Control**	**Standard**	**Control**
Dimension	144±15	181±15	127±8	127±8

Fuel and dummy plates are uniquely identified by serial numbers, and the dummy plates are differentiated by [redacted], as shown in Figure 4-7 and Figure 4-8.

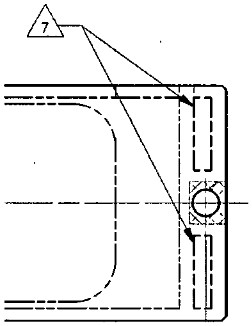


Figure 4-7: Plates are identified by unique serial numbers as shown here.

[redacted]

4.2.2 Control Rods

Table 4-5: Summary of control rod characteristics as listed in the PUR-1 Operations Manual.

CONTROL RODS	
Number of Regulating Rods	1 - 304 stainless steel, hollow
Number of Shim Safety Rods	2 - Boron-stainless steel, solid
Operating Rates Regulating Rod Shim Safety Rod Scram	 17.7 in/min 4.4 in/min Less than 1 second (from signal to complete insertion).
Size Regulating rod (inches) Shim safety rods (inches)	 ½ x 2 ¼ ½ x 2 ¼ x 25 ½
Maximum rate of reactivity change Regulating rod Shim safety rods	 $1.1\text{x}10^{-4}\ \frac{\Delta k}{k \cdot s}$ $9.9\text{x}10^{-3}\ \frac{\Delta k}{k \cdot s}$
Average rate of reactivity charge Regulating rod Shim safety rods	 $5.8\text{x}10^{-4}\ \frac{\Delta k}{k \cdot s}$ $5.4\text{x}10^{-3}\ \frac{\Delta k}{k \cdot s}$

4.2.3 Neutron Moderator and Reflector

PUR-1 is moderated by the light water of the reactor pool. A graphite reflector surrounds the four sides of the reactor, and light water reflects the top and bottom. The graphite in the reflector assemblies is nuclear grade, manufactured by Union Carbide, and contains less than 1 part per million boron.

4.2.4 Neutron Startup Source

A 5 Curie Plutonium-Beryllium neutron startup source is located next to the core, and is removable when the reactor reaches criticality. A drive mechanism controlled from the operator's console raises and lowers the source as needed.

4.2.5 Core Support Structure

The reactor is supported on an aluminum grid plate typical of all Lockheed MTR reactors. This grid plate controls the placement of fuel within the core, and can be used to locate experiments that would be placed outside the core. A drawing of the grid plate is located in the Appendix 1.

The grid plate is manufactured with 6061 aluminum, and has not shown any degradation over its greater than 46 years of service. It is expected that the grid plate will continue to function as designed.

4.3 Reactor Pool

[redacted] The pool is contained in a cylindrical tank 17 feet, 4 inches deep and 8 feet in diameter. The core is located to one side to give additional experimental space. The pool has a welded stainless steel liner.

The supports for the drive mechanisms for the control rods, the fission chamber and the source, and the neutron detectors are fastened to the support plate at the top of the tank. A traversing mechanism was mounted on the top of the reactor pool wall after the reactor was built. A light weight, portable aluminum bridge can be placed across the pool for maintenance and fuel handling operations.

The average pool temperature in recent PUR-1 operating history is 26°C. The figure below shows measured pool temperatures from 1993 to 2006.

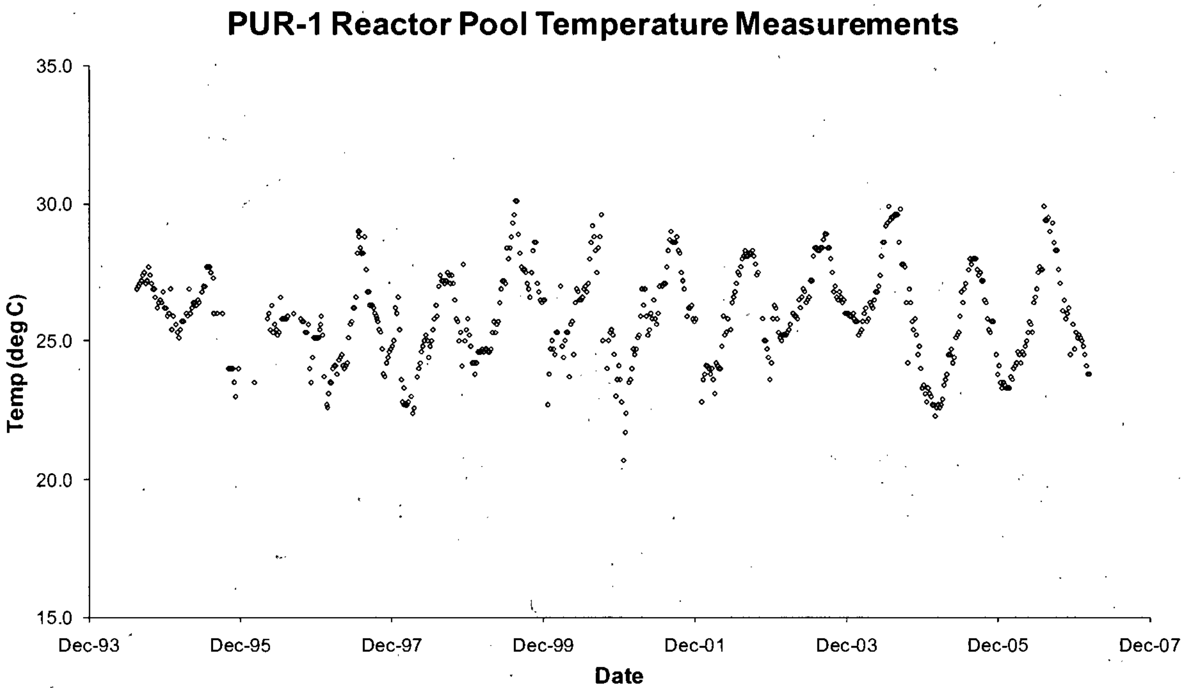


Figure 4-9: PUR-1 Reactor pool temperature measurements from 1994-2007.

4.4 Biological Shield

The biological shield consists of light water, magnetite sand, earth, and structural concrete. [redacted] The water level is normally 4 inches below the top of the tank.

Shielding over the core is provided by 13 feet of light water. This reduces the radiation level to a calculated less than 1 mrem/hr when the core is operating at 1 kW. The concrete biological shield is designed for a maximum radiation level at any point along its outer lateral surface of 0.1 mr/hr at 1 kW.

4.5 Nuclear Design

Neutronics analysis was performed with the Monte Carlo N-Particle Code (MCNP5), published by Los Alamos National Laboratory. Thermal-Hydraulic analysis was performed with NATCON, a natural convection analysis code written by Argonne National Laboratory (ANL). Transient analysis was performed using PARET, also written by ANL.

4.5.1 MCNP Model

The MCNP model of the fuel plates along various dimensions is shown in Table 4-6.

The materials used in the model are standard for the LEU MTR plates, with the addition of 20 parts per million of a boron-equivalent to the 6061 cladding material to account for impurities in the alloy.

Table 4-6: Representation of fuel plates.			
	 Figure 4-10: plate in X-Z plane	 Figure 4-11: plate in Y-Z plane. (magnified)	
 Figure 4-12: plate in X-Y plane, cutaway view.			

Figure 4-13: Model representation of standard assembly plate spacing detail showing wall spacers.

The representations of the complete standard and shim safety control fuel assemblies for the core are shown in Figure 4-14 and Figure 4-15 respectively. There are one or two dummy plates in the standard assemblies, and no dummy plates in the control assemblies, which are the plates without the center fuel material in the figures. The assembly cans are identical in size and composition.

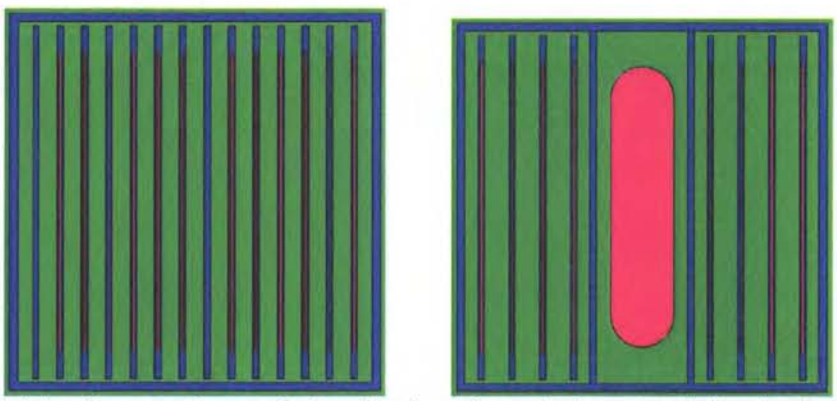


Figure 4-14: Comparison of standard and control assemblies in the model.

As stated before, the control assemblies contain no dummy plates, but do contain a guard plate made of 6061 aluminum between the fuel elements and the control rods to prevent any mechanical damage to the fuel from insertion of the control rods. The shim safety control rods themselves, made of borated stainless steel, are oblong shaped plates inserted down the center of the assemblies. The regulating rod is similar to the shim-safety control assemble shown in the figure above, but reflects the fact that the rod is hollow, and filled with water.

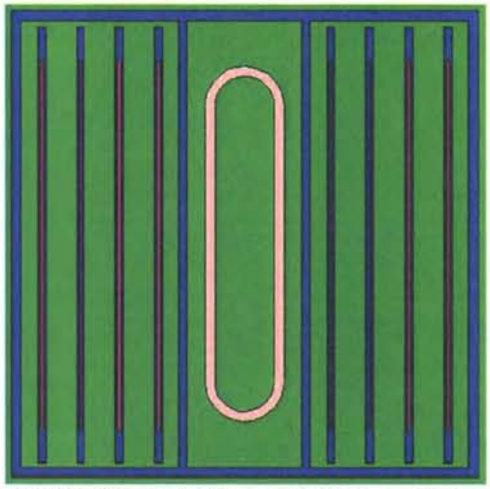


Figure 4-15: Model of regulating rod assembly.

The entire core model is shown in Figure 4-16. This image shows the present configuration of the core, the orientation of the plates, the location of the irradiation facilities, and the startup neutron source [redacted].

[redacted]

Figure 4-16: Representation of LEU core load

4.5.2 Normal Operating Conditions

4.5.2.1. <u>PUR-1 Core Power Distribution</u>

The PUR-1 LEU fuel plates are fabricated with U_3Si_2 dispersion fuel. Each fuel plate has a nominal loading of [redacted] U-235. A fresh core with [redacted] "dummy" aluminum plates was evaluated in the models presented herein. Figure 4-16 shows the core layout modeled in MCNP, and drawings of control and 14-plate fuel assemblies. The numbers in parenthesis in the core layout drawing indicate either the number of fuel plates in the 14-plate fuel assemblies (the remainder are dummy plates of aluminum) or the label of the control assembly (shim or regulating rods).

Table 4-7 compares the heating by assembly and in non-fueled components for the LEU core with the rods banked at 53.5 cm. The banked rod critical configuration was found to have the highest peak power density in the analysis of the HEU core. Elements 3-4, 4-3, and 4-4 are noted as having the highest average plate powers, and are therefore of interest for thermal-hydraulics analysis. Figure Q27-1 provides a schematic of the core layout and the plate orientations.

Table 4-8 compares the power in individual plates in elements 4-4, 3-4, and 4-3 for the reactor with the banked rods critical configuration. The tallies were summed over the fuel meat in each fuel plate, all clad, coolant, and the bundle can. The plates are numbered from left-to-right in these elements (see the element drawings in Figure 4-17). It can be seen that plate 1348 in assembly or bundle 4-4 has the highest power (8.07 W at 1 kW). This plate is adjacent to the large water hole that the SS1 rod falls into, and nearer the center of the reactor than plate 1355 on the other side of the water hole. Plates 1228 (bundle 3-4) and 1315 (bundle 4-3) face the center-line of the core, and have roughly equal power (6.51 and 6.41 W at 1 kW).

Figure 4-18 and Figure 4-19 compare the local-to-average axial power density profiles for fuel plates 1348, 1228, and 1315 for the banked rods critical configuration. Plate 1348 has the highest peak power density of all the plates in the LEU core and was evaluated in the thermal-hydraulics analyses. The slight "pinching" of the axial power profile due to the insertion of the SS1 shim rod is evident.

The local-to-average axial power density profiles in plates 1228 and 1315 are nearly identical due to their symmetric positioning across the core center-line. For these two plates, the highest peak/average density (1.712) occurs in plate 1228. The peak power density in plate 1228 of standard element 3-4 is about 22% lower than that of plate 1348. Furthermore, the coolant channel thickness in the standard fuel assembly is narrower than that in the control assembly (144 vs. 181 mils). This reduction in channel thickness brings the channel thickness closer to the optimum value (≈ 100 mil), which will result in a higher ONB power with natural convection cooling. Therefore, plate 1348 is the most limiting of all fuel plates.

A higher water-to-fuel ratio at the edges of the fuel plates induces a power density profile along the width of the fuel plates. This is shown in Figure 4-19 for plates 1348, 1228, and 1315 with the rods in the banked critical position. The radial segments are numbered from bottom-to-top in the MCNP model (see the plate orientations indicated in Figure 4-17). It is expected that the power density will be higher at the edge of the plate closest to the core center-line.

Lastly, Table 4-9, Table 4-10, and Table 4-11 provide the axial and radial power profiles for plates 1348, 1228, and 1315, respectively. These data were used for thermal-hydraulic analysis of the LEU-fueled PUR-1.

[redacted]

Purdue LEU Core Layout ([redacted] plates)

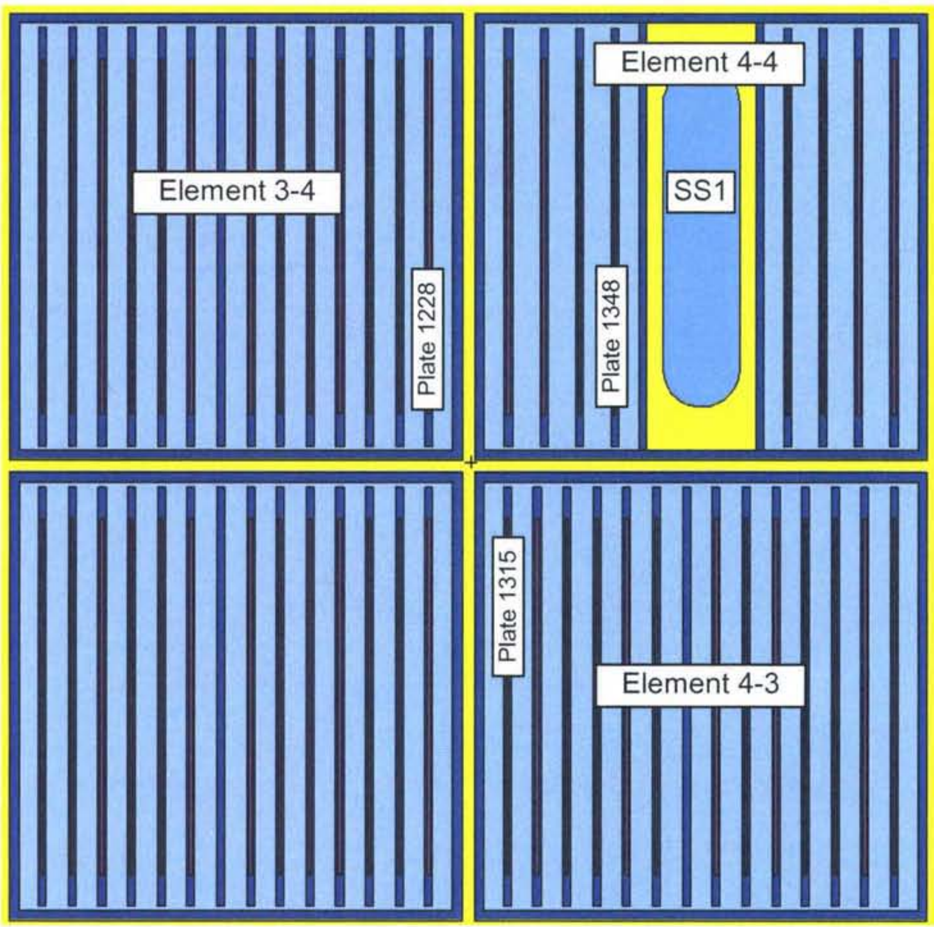


Figure 4-17: PUR-1 Core Layout and Bundle Drawings.

Table 4-7: Bundle Powers Predicted by f7 and f6 Tallies in MCNP.

	Power (W) 1000 W	Average Plate Power (W)	Power (W) 10 kW	Average Plate Power (W)
2-2 (RR)	38.19	4.77	381.9	47.7
2-3	61.85	4.76	618.5	47.6
2-4 (SS2)	45.65	5.71	456.5	57.1
2-5	45.93	3.83	459.3	38.3
3-2	62.56	4.81	625.6	48.1
3-3	78.09	6.01	780.9	60.1
3-4	78.42	6.03	784.2	60.3
3-5	61.20	4.71	612	47.1
4-2	63.52	4.89	635.2	48.9
4-3	80.78	6.21	807.8	62.1
4-4 (SS1)	60.09	7.51	600.9	75.1
4-5	62.29	4.79	622.9	47.9
5-2	52.32	4.03	523.2	40.3
5-3	63.97	4.92	639.7	49.2
5-4	63.49	4.88	634.9	48.8
5-5	47.64	3.97	476.4	39.7
Inter-assembly water	3.78		37.8	
Graphite reflector	9.52		95.2	
Grid plate	2.27		22.7	
Water reflector (pool)	18.00		180	
SS1	0.23		2.3	
SS2	0.17		1.7	
RR	0.03		0.3	
Total	1000.0		10000	

Table 4-8: Plate Power (W) Computed from Heating Tallies in Bundles 4-4, 3-3 and 3-4 in PUR-1 Core with [redacted] Fuel Plates.

Bundle 4-4	**Power (W at 1kW)**	**Power (W at 10 kW)**
Plate 1345 Meat	7.00	70
Plate 1346 Meat	7.09	70.9
Plate 1347 Meat	7.40	74
Plate 1348 Meat	**8.07**	80.7
Plate 1355 Meat	7.74	77.4
Plate 1356 Meat	6.98	69.8
Plate 1357 Meat	6.59	65.9
Plate 1358 Meat	6.43	64.3
Clad	0.36	3.6
Water	2.19	21.9
Can	0.24	2.4
Total	**60.09**	**600.9**
Bundle 3-4		
Plate 1215 Meat	5.80	58
Plate 1216 Meat	5.56	55.6
Plate 1217 Meat	5.49	54.9
Plate 1218 Meat	5.50	55
Plate 1219 Meat	5.61	56.1
Plate 1220 Meat	5.84	58.4
Plate 1222 Meat	5.91	59.1
Plate 1223 Meat	5.76	57.6
Plate 1224 Meat	5.73	57.3
Plate 1225 Meat	5.76	57.6
Plate 1226 Meat	5.92	59.2
Plate 1227 Meat	6.12	61.2
Plate 1228 Meat	**6.51**	65.1
Clad	0.48	4.8
Water	2.19	21.9
Can	0.24	2.4
Total	**78.42**	**784.2**
Bundle 4-3		
Plate 1315 Meat	**6.41**	64.1
Plate 1316 Meat	6.16	61.6
Plate 1317 Meat	6.07	60.7
Plate 1318 Meat	6.04	60.4
Plate 1319 Meat	6.09	60.9
Plate 1320 Meat	6.28	62.8
Plate 1322 Meat	6.21	62.1
Plate 1323 Meat	5.92	59.2
Plate 1324 Meat	5.76	57.6
Plate 1325 Meat	5.68	56.8
Plate 1326 Meat	5.65	56.5
Plate 1327 Meat	5.68	56.8
Plate 1328 Meat	5.84	58.4
Clad	0.49	4.9
Water	2.19	21.9
Can	0.24	2.4
Total	**80.71**	**807.1**

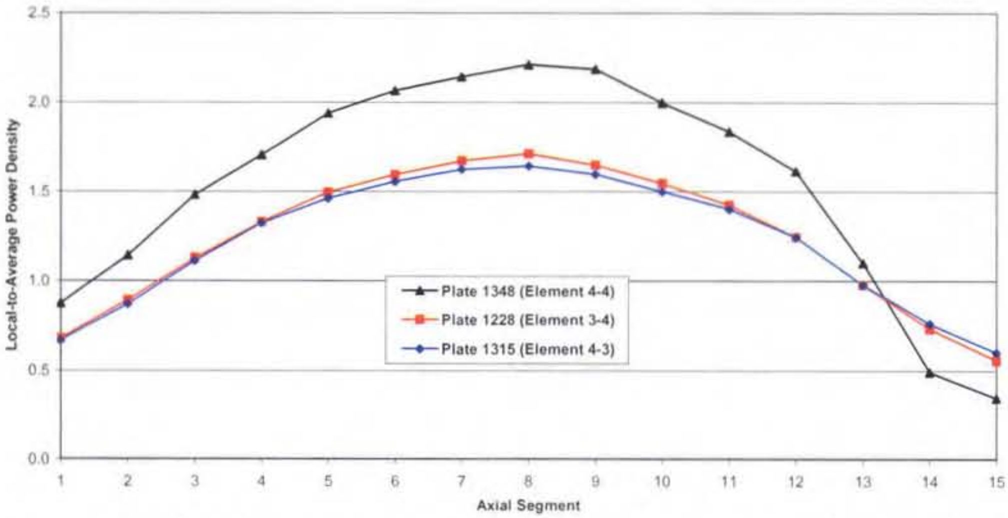


Figure 4-18: Axial Power Profiles in Plates 1348, 1228 and 1315 for Banked Rod Critical Configuration in PUR-1

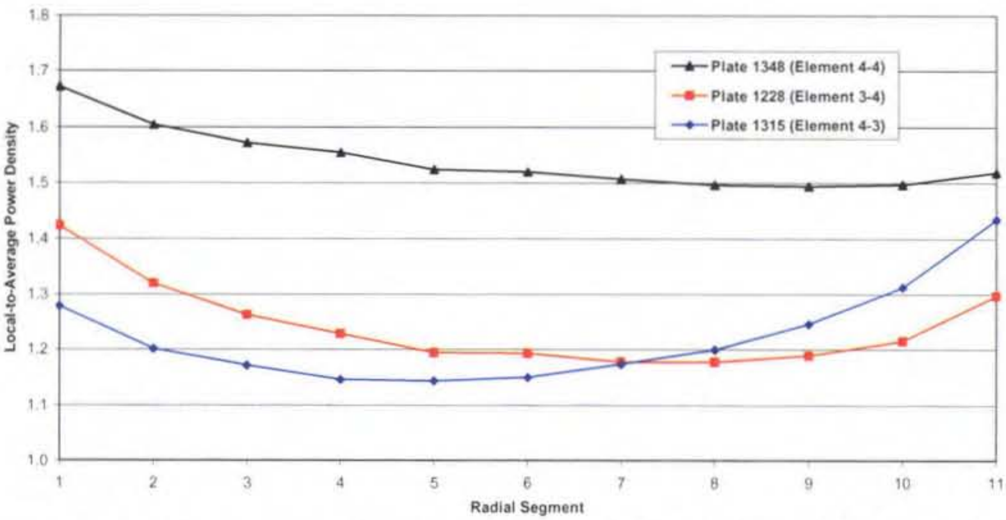


Figure 4-19: Radial Power Profiles in Plates 1345, 1228 and 1315 for Banked Rod Critical Configuration in PUR-1

Table 4-9: Axial and Radial Heating Profile for Plate 1348 of Bundle 4-4 for Banked Critical Configuration.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W at 1 kW)	σ
1	1.88595	5.88645	3.04	0.99%
2	5.88645	9.88695	4	0.86%
3	9.88695	13.88745	5.17	0.76%
4	13.88745	17.88795	5.95	0.70%
5	17.88795	21.88845	6.76	0.66%
6	21.88845	25.88895	7.2	0.64%
7	25.88895	29.88945	7.47	0.63%
8	29.88945	33.88995	7.71	0.62%
9	33.88995	37.89045	7.62	0.63%
10	37.89045	41.89095	6.97	0.65%
11	41.89095	45.89145	6.41	0.68%
12	45.89145	49.89195	5.63	0.74%
13	49.89195	53.89245	3.85	0.86%
14	53.89245	57.89295	1.72	1.25%
15	57.89295	61.89345	1.2	1.48%
Total			80.7	**0.21%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	σ
1	24.0838	24.6253	1.672	0.48%
2	24.6253	25.1668	1.604	0.48%
3	25.1668	25.7082	1.572	0.48%
4	25.7082	26.2497	1.554	0.49%
5	26.2497	26.7912	1.523	0.49%
6	26.7912	27.3327	1.519	0.49%
7	27.3327	27.8742	1.506	0.49%
8	27.8742	28.4157	1.496	0.50%
9	28.4157	28.9571	1.493	0.50%
10	28.9571	29.4986	1.496	0.50%
11	29.4986	30.0401	1.518	0.50%

[1]Positions correspond to MCNP model of PUR-1.

Table 4-10: Axial and Radial Heating Profile for Plate 1228 of Bundle 3-4 for Banked Critical Configuration.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W at 1 kW)	σ
1	1.88595	5.88645	2.36	1.07%
2	5.88645	9.88695	3.11	0.94%
3	9.88695	13.88745	3.95	0.83%
4	13.88745	17.88795	4.65	0.77%
5	17.88795	21.88845	5.23	0.73%
6	21.88845	25.88895	5.57	0.70%
7	25.88895	29.88945	5.84	0.70%
8	29.88945	33.88995	5.98	0.69%
9	33.88995	37.89045	5.76	0.69%
10	37.89045	41.89095	5.4	0.71%
11	41.89095	45.89145	4.98	0.74%
12	45.89145	49.89195	4.36	0.80%
13	49.89195	53.89245	3.42	0.89%
14	53.89245	57.89295	2.55	1.05%
15	57.89295	61.89345	1.95	1.17%
Total			65.11	**0.22%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	σ
1	24.0838	24.6253	1.424	0.52%
2	24.6253	25.1668	1.320	0.52%
3	25.1668	25.7082	1.262	0.53%
4	25.7082	26.2497	1.229	0.54%
5	26.2497	26.7912	1.194	0.54%
6	26.7912	27.3327	1.193	0.55%
7	27.3327	27.8742	1.178	0.54%
8	27.8742	28.4157	1.178	0.55%
9	28.4157	28.9571	1.189	0.54%
10	28.9571	29.4986	1.215	0.54%
11	29.4986	30.0401	1.297	0.54%

[1]Positions correspond to MCNP model of PUR-1.

Table 4-11: Axial and Radial Heating Profile for LEU Plate 1315 of Bundle 4-3 for Banked Critical Configuration.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W at 1 kW)	σ
1	1.88595	5.88645	2.33	1.07%
2	5.88645	9.88695	3.03	0.96%
3	9.88695	13.88745	3.89	0.85%
4	13.88745	17.88795	4.63	0.79%
5	17.88795	21.88845	5.11	0.75%
6	21.88845	25.88895	5.43	0.72%
7	25.88895	29.88945	5.67	0.70%
8	29.88945	33.88995	5.74	0.70%
9	33.88995	37.89045	5.58	0.70%
10	37.89045	41.89095	5.23	0.73%
11	41.89095	45.89145	4.9	0.76%
12	45.89145	49.89195	4.36	0.81%
13	49.89195	53.89245	3.42	0.90%
14	53.89245	57.89295	2.65	1.02%
15	57.89295	61.89345	2.1	1.13%
Total			64.07	**0.23%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	σ
1	16.3894	16.9309	1.279	0.54%
2	16.9309	17.4724	1.201	0.55%
3	17.4724	18.0138	1.172	0.56%
4	18.0138	18.5553	1.147	0.56%
5	18.5553	19.0968	1.144	0.57%
6	19.0968	19.6383	1.151	0.56%
7	19.6383	20.1798	1.174	0.55%
8	20.1798	20.7213	1.199	0.55%
9	20.7213	21.2627	1.246	0.54%
10	21.2627	21.8042	1.313	0.52%
11	21.8042	22.3457	1.435	0.51%

[1]Positions correspond to MCNP model of PUR-1.

4.5.3 Reactor Core Physics Parameters

For the model, the material composition for the plates was U_3Si_2-Al, and 6061 aluminum cladding. An addition of 20 parts per million of a boron-equivalent was added to the 6061 cladding material to account for impurities in the alloy. The MCNP calculated maximum thermal neutron fluxes in the fuel region are 2.01E10 n/(cm2*s) peak in the fuel region, and 1.38E10 n/(cm2*s) average in the fuel region.

Eigenvalue calculations were performed with MCNP5, typically using from 25 to 50 million neutron histories. These calculations yielded a reactor k_{eff} with a 1-σ uncertainty of ±17 pcm (0. 017% $\Delta k/k$) for 25 million histories; the uncertainty was reduced to ±12 pcm (0. 012% $\Delta k/k$) when 50 million histories were employed. The 1-σ uncertainty of the eigenvalue calculations reduces by the square root of the number of histories. The uncertainty of the reactivity feedback coefficients can be calculated as the square root of the sum of the squares of the eigenvalue calculations. Based on the eigenvalue calculations with 25 to 50 million histories, the uncertainties of the water temperature and water void coefficients were found to be on the order of 15% to 30% when calculated over the expected 10 to 20 °C perturbations of water temperature.

It was decided that the calculational uncertainty should be reduced by extending the number of histories in the MCNP eigenvalue calculations. Eigenvalue calculations for the nominal core state and certain other cases which exhibit only small perturbations to the core k_{eff} were performed with 200 to 300 million neutron histories. This reduced the 1-σ uncertainty of the eigenvalue calculations to around 5 pcm (0.005% $\Delta k/k$).

4.5.3.1. Control Rod Worths and Excess Reactivity

Excess reactivity of the LEU core in PUR-1 was determined to be 0.00468 (0.47%) $\Delta k/k$ in a 190 fuel plate core (16 dummies), including a reactivity bias of 0.32% $\Delta k/k$. This value is within the Technical Specification limit of 0.6% for excess reactivity.

Control rod worths were calculated for the core and compared with measured data. The calculated and measured control rod worth values are shown in Table 4-12. The calculated calibration curves were done utilizing MCNP5 for the core, and Figure 4-20, Figure 4-21, and Figure 4-22 show the calculated control rod worth curves for each of the control rods, along with the measured values for each of the rods.

Table 4-12: Comparison of HEU and LEU calculated control rod worths.

	LEU Calculated (Δk/k)	**LEU Measured (Δk/k)**
Shim Safety 1	0.0377±0.0003	0.0393
Shim Safety 2	0.0189±0.0003	0.0222
Regulating Rod	0.0023±0.0003	0.0027

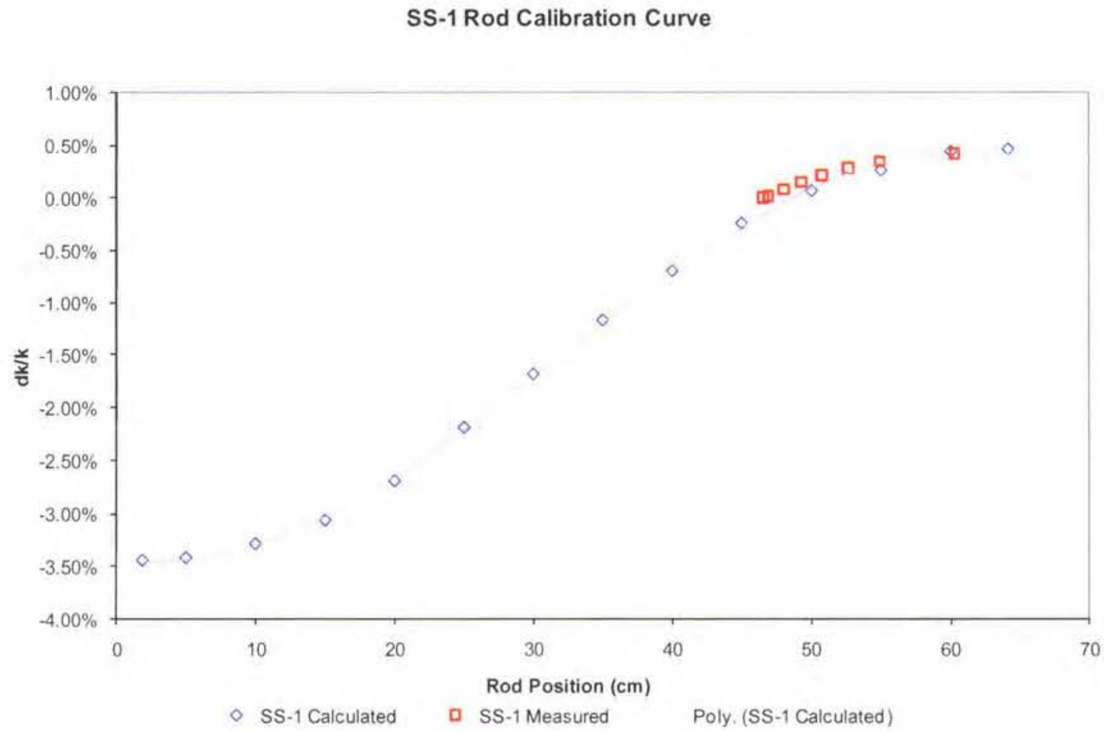


Figure 4-20: Calibration curve for SS-1 rod with calculated and measured values.

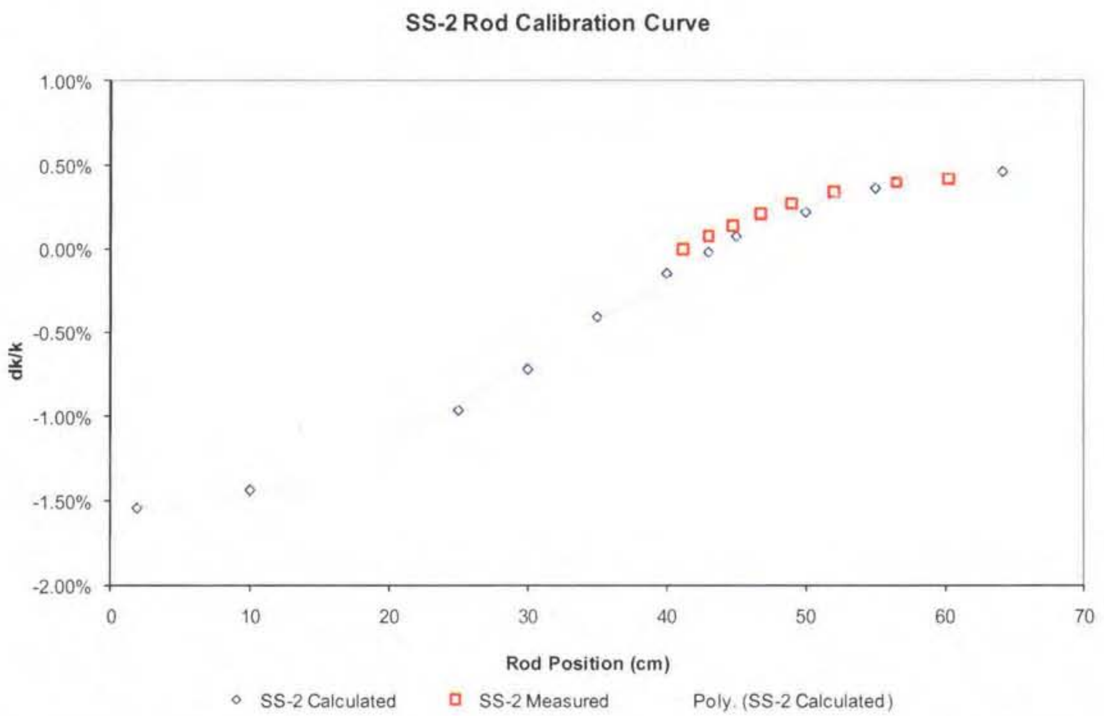


Figure 4-21: Calibration curve for SS-2 rod with calculated and measured values.

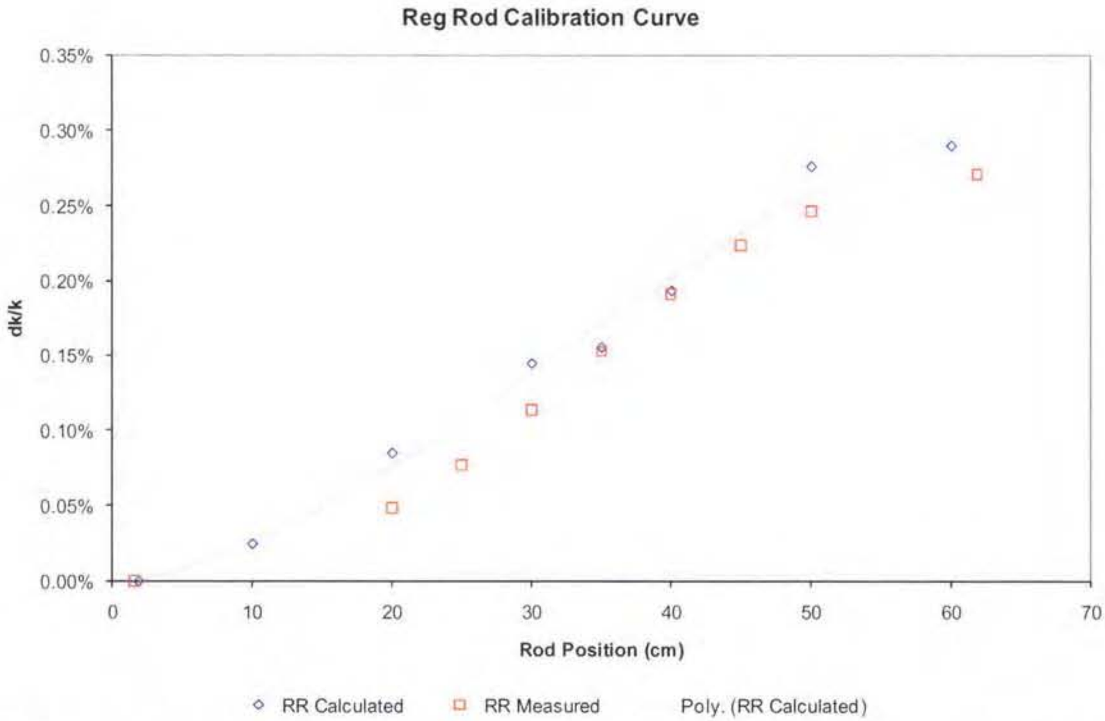


Figure 4-22: Calibration curve for RR with calculated and measured values.

Using the rod worth curves in preceding figures, the maximum reactivity insertion rates were determined by finding the maximum slope, or rate of change, of the curve. The comparison of calculated and measured maximum reactivity insertion rates for PUR-1 are shown in Table 4-13.

Table 4-13: Comparison of calculated and measured maximum reactivity insertion rates.

	Maximum Reactivity Insertion Rates for Control Rods $\left(\frac{\Delta k}{k \cdot s}\right)$	
	Calculated	**Measured**
Shim-safety 1	1.75E-04	2.31E-4
Shim-safety 2	8.75E-05	1.42E-4
Regulating Rod	4.66E-05	5.39E-5

4.5.3.2. Shutdown Margin

The shutdown margin was calculated using MCNP5 and has been measured. These data are shown in

Table 4-14. These values meet the technical specification (TS 3.1.a).

Table 4-14: Comparison of calculated and measured shutdown margins.

	Calculated	Measured
SS-2 Worth (Δk/k)	-1.89%	-2.22%
kexcess (Δk/k)	0.351%	0.42%
Shutdown Margin (Δk/k)	-1.58%	-1.80%

4.5.3.3. Other Core Physics Parameters

Reactivity coefficients and reactor kinetics parameters were calculated for the PUR-1 model core. These values are used to estimate the core reactivity response to changes in properties of the fuel temperature, coolant/moderator temperature, or coolant/moderator density. They are therefore essential for analyses of reactivity-induced transients. These calculations were performed with the MCNP5 code using the same core model as for the core design and power distribution analyses.

The reactor kinetics parameters evaluated for PUR-1 were the effective delayed neutron fraction, β_{eff}, and the prompt neutron lifetime, ℓ. The effective delayed neutron fraction is calculated using two eigenvalue calculations from MCNP5. Normal calculations of k_{eff} include both prompt and delayed neutrons. A additional calculation of k_{eff} is performed with delayed neutrons turned off in MCNP, yielding a k_{eff} that depends only on prompt neutrons, which is denoted by k_{eff}^{prompt}. The effective delayed neutron fraction is then defined as

$$\beta_{eff} = 1 - \frac{k_{eff}^{prompt}}{k_{eff}}.$$

The delayed neutron fraction was calculated using the formulation:

$$\beta_{eff} = 1 - \frac{k_{eff}^{prompt}}{k_{eff}} = \frac{k_{eff} - k_{eff}^{prompt}}{k_{eff}}.$$

The eigenvalue calculations were performed using MCNP5. The value for β_{eff} is calculated as shown below:

k_{eff} = 1.00379 ± 0.00010

k_{eff}^{prompt} = 0.99589 ± 0.00015

β_{eff} = 1 – 1.00379/0.99589 = $7.870\times10^{-3} \pm 1.79\times10^{-4}$ (2.3%)

The bias of the PUR-1 MCNP model is $\Delta\rho_{bias}$= 0.32% Δk/k. This was determined by comparing the results of eigenvalue calculations for several cases with the control rods at measured critical positions. Accounting for the bias in the core model, the β_{eff} is calculated as:

$$\beta_{eff}^{biased} = \frac{k_{eff} - \Delta\rho_{bias} - (k_{eff}^{prompt} - \Delta\rho_{bias})}{k_{eff} - \Delta\rho_{bias}} = \frac{k_{eff} - k_{eff}^{prompt}}{k_{eff} - \Delta\rho_{bias}}.$$

The resulting biased value is obtained then as:

β_{eff}^{biased} = (1.00379 – 0.99589)/(1.00379 – 0.0032) = 7.895x10^{-3} ± 1.79x10^{-4} (2.3%)

The reactivity insertion accident analyses were performed using the *unbiased* β_{eff} values, which are slightly smaller (0.3%) than the effective delayed neutron fraction determined by accounting for the bias in the MCNP model. Consequently, the reactivity insertion accident analyses were performed with a value of β_{eff} that gives more conservative results.

The prompt neutron lifetime is calculated using the "1/v insertion method," where a uniform concentration of a 1/v absorber such as ^{10}B is included at a very dilute concentration everywhere in the core and the reflector. The prompt neutron lifetime, ℓ, is calculated by

$$l = \lim_{N \to 0} \left[\frac{\frac{k_1 - k_0}{k_1}}{N_1 \sigma_a v} \right],$$

where k_1 is the k_{eff} of the system with a uniform concentration, N_1, of a 1/v absorber, and σ_a is the infinitely-dilute absorption cross section of the absorber for neutrons at speed v. For this effort, the ^{10}B absorption cross section is assumed to be σ_a=3837 barns for a neutron speed of v=2200 m/s.

Reactivity coefficients provide an estimate of the reactivity response to changes in state properties, given as:

$$\Delta\rho = \alpha_x \cdot \Delta x$$

where α_x is the reactivity coefficient due to a unit change in property x, and Δx is the value change for property x. The reactivity coefficients are calculated assuming that simultaneous changes in multiple state properties are separable. These are calculated from core eigenvalue calculations with independent perturbations to state properties, as shown here:

$$\alpha_x = \frac{\Delta\rho}{\Delta x} = \frac{k_1 - k_0}{k_1 k_0} \cdot \frac{1}{(x_1 - x_0)}.$$

PUR-1 has operated at a maximum power of 1 kW and is cooled by natural convection. The nominal conditions for the core used in the reactor design model assumed fresh fuel (i.e., no fission products given the low burnup of the PUR-1), isothermal conditions of 20°C, and impurities in the fuel, clad, and graphite based on the best-available data.

The reactivity coefficients are calculated assuming separability of the reactivity feedback effects due to changes in fuel temperature, water temperature, and water density. After establishing a nominal state based on the conditions given above and a critical rod configuration determined in the reactor design analysis, cases with perturbations to the temperatures or water density were evaluated.

The fuel temperature changes are assumed to occur uniformly throughout the reactor (i.e. the same temperature perturbation in all fuel plates), while the water temperatures or densities were perturbed in different zones of the reactor depending on the proximity to the fuel plates. In the time immediately after the initiation of a reactivity induced transient, the water inside the fuel assembly can (shown in light blue in Figure 4-23) would heat up and have a feedback effect on the transient. However, the water between the cans and also in the space between the control rod and guard plates (shown in orange in Figure 4-23) would take some time to be heated as a result of a power increase from the transient due to the slow water circulation time with natural convection cooling. It would take even longer for the water in the reactor tank to be heated. Consequently, cases were evaluated with the water temperature or density perturbed:

- within the fuel assembly (light blue regions in Figure 4-23)
- within the fuel assembly and the water between the assemblies (orange regions in Figure 4-23), and
- within the fuel assembly, the water between the assemblies, and the pool or reflector water (all of the water in the MCNP5 model).

For the evaluation of the reactivity induced transients, only the reactivity coefficients calculated by perturbing the fuel assembly water will be used. A summary of the values determined in the analyses of the HEU and LEU reactor physics parameters and reactivity coefficients is presented in

Table 4-15.

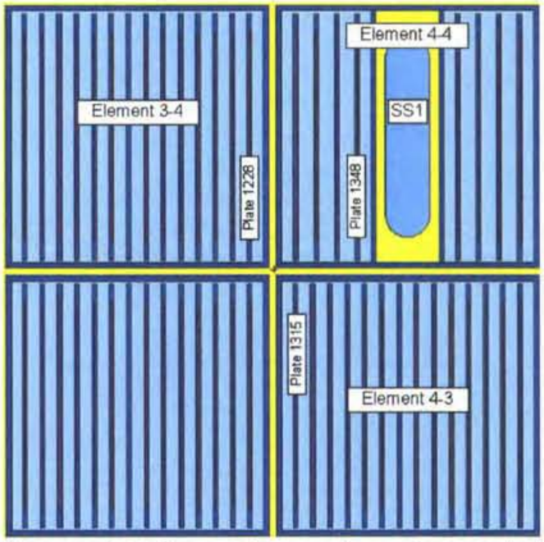


Figure 4-23: Figure showing water regions for perturbation models

Table 4-15: Other core physics parameters.

	LEU (calculated)
$\alpha_{fuel}\left(\frac{\%\Delta k}{k\cdot{}^{\circ}C}\right)$	-8.05E-04
$\alpha_{moderator}\left(\frac{\%\Delta k}{k\cdot{}^{\circ}C}\right)$	-9.05E-03
$\alpha_{void}\left(\frac{\%\Delta k}{k\cdot\%void}\right)$	-1.93E-01
β_{eff}	0.784%
ℓ (μs)	81.3

Table 4-16: Water and Fuel Coefficients for the PUR-1 Core.

PUR-1 LEU Core Design Water Temperature Coefficient (α_{water})				
20 to 30 °C	-9.051E-05	$\Delta\rho/^{\circ}C$	±	9%
PUR-1 LEU Core Design Water Void Coefficient (α_{void})				
0 to 0.60% void	-1.933E-03	$\Delta\rho$/% void	±	7%
PUR-1 LEU Core Design Fuel Temperature Coefficient (α_{fuel})				
20 to 127 °C	-8.053E-06	$\Delta\rho/^{\circ}C$	±	10%
PUR-1 LEU Core Design Effective Delayed Neutron Fraction (βeff)				
		0.00784	±	0.00008

For the LEU core, two different critical rod configurations were determined from the reactor design analysis. A rod configuration with the SS2 rod inserted at 43 cm, and the

SS1 and regulating rods fully withdrawn, was found to result in the largest peak-to-average power density. This critical rod configuration was modeled in the LEU core reactivity coefficients and kinetics parameters calculations.

[redacted]

Figure 4-24: PUR-1 Core Layout with LEU Fuel.

Figure 4-26 shows the effects of the fuel and water temperature, and water density perturbations on the core reactivity. The core has a negative fuel temperature coefficient because of the Doppler effect on the U-238 capture resonances in the fuel. The fuel temperature coefficient is smaller than that from water temperature or density effects, but it is a non-negligible parameter for the LEU accident analyses.

The behavior of the LEU core reactivity due to water temperature and density perturbations was quite similar to that for the HEU core. The LEU core does have a slightly harder neutron spectrum under nominal conditions, so the spectrum hardening due to the water temperature increase should have a smaller feedback effect on the core reactivity. On the other hand, coolant voiding in the LEU core results in greater neutron leakage because of the harder spectrum, so the negative reactivity feedback effect due to the reduced water density is greater for the LEU core.

Table 4-16 gives more detail for the reactivity feedback coefficients for water temperature and density, and fuel temperature perturbations for the LEU core. As can be seen, the water temperature coefficient becomes more negative as the water temperature is increased. The temperature coefficient calculated over the range from 20°C to 30°C for the "fuel assembly" water should be used for the accident analyses. It should be noted that the water temperature coefficient over this range was actually larger than the HEU core. For the water density feedbacks, the coefficient calculated over the range from 40°C to 60°C is the most conservative value for the accident analyses.

Temperature coefficients of reactivity were calculated assuming separability of the reactivity feedback effects due to water temperature, water density (void), and fuel temperature. This is a common practice and makes it easy to understand the inherent shutdown mechanisms that are responsible for affecting reactivity-induced transients. Not only are the feedback coefficient treated as separable, the reactivity coefficients were also calculated under non-isothermal conditions because different regions of the reactor will heat up at different rates during a power increase.

Three distinct regions containing water in the PUR-1 were considered.

1. The water between the fuel plates; this is called the "fuel assembly" water.
2. The water between the fuel element cans and also the water in the control elements between the control rod guard plates; this is called the "inter-assembly" water.
3. The water in the reactor tank; this is called the "reflector" water.

Water temperature coefficients were calculated by adjusting the temperature of the water in each of these regions and calculating the impact on the core reactivity. Reactivity feedback coefficients due to the perturbation of the fuel assembly water temperature, the fuel assembly plus inter-assembly water temperature, and fuel assembly, inter-assembly, and reflector water temperature were calculated. It is important to note that only the feedback effect due to heating of the fuel assembly water (between the fuel plates) was considered in the accident analyses, because only this water would experience an immediate heating due to power increases during the transient. The other regions containing water would take much longer to heat up.

Heating the water increases the thermal motion of the hydrogen atoms in the water. The result is to increase the energy of neutrons which are in "thermal equilibrium" with the hydrogen moderator, thus hardening the neutron spectrum in the reactor. At higher neutron energies, the U-235 fission cross section is reduced. Thus, there is a negative reactivity feedback effect due to heating of the water between the fuel plates.

Increasing the temperature of the reflector (tank) water was found to have a positive reactivity feedback effect over a small temperature range from nominal conditions. This is due to a decrease in the neutron absorption in the reflector as the temperature is increased and the neutron spectrum hardens. For the HEU core, the feedback coefficient due to heating all water in the reactor tank from 20 to 30 °C was calculated to be $2.38 \times 10^{-3} \pm 2.11 \times 10^{-3}$ % $\Delta k/k/^{\circ}C$. However, it would take a long time for any transient to heat the reflector water, so it is judged that there are no significant safety issues related to the positive reactivity feedback coefficient when all the water is heated. It should be noted that for larger increases in the temperature of the reflector water, the reactivity feedback effect is negative.

The void coefficient was also converted to the unit of $\Delta\rho/^{\circ}C$. The reason for expressing the void coefficient in the alternative units was to facilitate a comparison of the water void and temperature reactivity feedback effects, which were treated as separable. This comparison is illustrated in Table 4-15 and Table 4-16. When the reactor coolant within the fuel element (i.e., between the fuel plates, which is the region of interest for accident analyses in the PUR-1) is heated, the reactivity feedback from the water temperature increase is slightly larger.

The unit conversion to $\Delta\rho/^{o}C$ was accomplished by equating the void (water density) perturbation to a corresponding water temperature perturbation at 1.5 atmospheres. This is the water pressure in the PUR-1 core, which is at the bottom of a 15 foot tank of water (the actual tank depth is 17 feet, but the distance to the bottom of the core to the waterline is 15 feet. The density of water at 1.5 atmospheres as a function of temperature is shown in Figure 4-25.

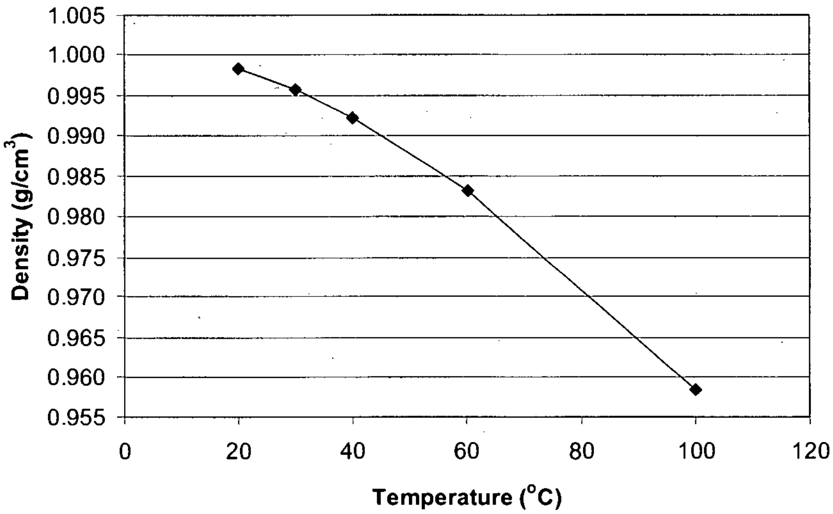


Figure 4-25: Density of sub-cooled water at 1.5 atmospheres.

MCNP allows for two adjustments on neutron scattering reactions based on the temperature of the medium. For neutron energies above 4 eV, the code adjusts the elastic scattering cross sections of nuclei in the medium using a free gas thermal treatment. The code user can specify the temperature of each cell within the model and the cross sections are adjusted if the specified temperature differs from the temperature of the processed nuclear data in the cross section library.

For modeling elastic and inelastic scattering events for neutrons below 4 eV, an $S(\alpha,\beta)$ treatment is employed. These data are available in the MCNP libraries for certain materials; light water and graphite are of interest for the PUR-1 analyses. $S(\alpha,\beta)$ data for graphite at 20 ^{o}C were employed. Furthermore, $S(\alpha,\beta)$ which had previously been evaluated for light water at 20, 30, 60, 100, and 150 ^{o}C were also used.

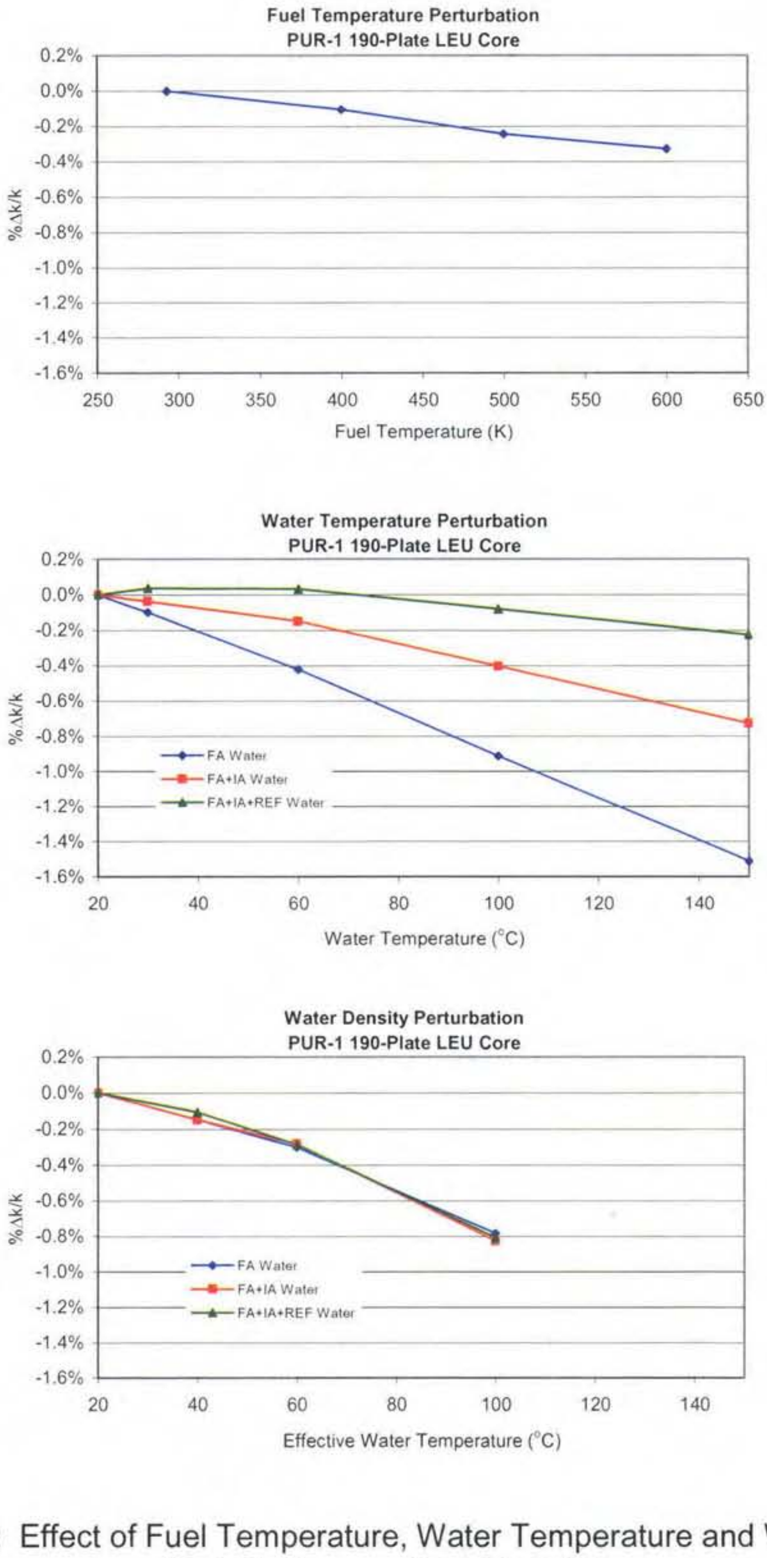


Figure 4-26: Effect of Fuel Temperature, Water Temperature and Water Density Perturbations on Core Reactivity.

4.6 Thermal-Hydraulic Design

In this section, the results of the thermal-hydraulic analyses are discussed in order to demonstrate that the PUR-1 LEU core design provides the cooling capacity necessary to ensure fuel integrity under all anticipated reactor operating conditions. Analyses for behavior under hypothetical accident scenarios are presented in Section 13.

4.6.1 NATCON Code Description

Thermal-hydraulic analyses were performed using the computer code NATCON[1,2], which can be used to analyze the steady-state thermal-hydraulics of plate type fuel in a research reactor cooled by natural convection. The reactor core is immersed in a pool of water that is assumed to be at a constant average temperature.

NATCON computes coolant flowrate, axial temperatures in the coolant and fuel plate surface and centerline, and the approach to onset of nucleate boiling (ONB). Other safety related parameters such as the Onset of Nucleate Boiling Ratio (ONBR) and Departure from Nucleate Boiling Ratio (DNBR) are calculated as well. And an automatic search for the power at ONB can be performed.

Flow is driven by density differences in the coolant that are the result of coolant heating by the fuel. Resulting buoyant forces are counter-balanced by viscous forces that result from the flow. Hot channel factors may also be introduced for determining safety margins. NATCON v2.0 documentation is included as Appendix 1 of this document. It includes information on the calculation of hot channel factors, inputs, and use of the code.

4.6.2 Fuel Element and Fuel Assembly Geometry

In PUR-1, each fuel plate element is loaded into its own assembly container, or can, as described in Section. Cross section views of the two different types of assemblies, standard and control, are shown in Figure 4-27 and Figure 4-28.

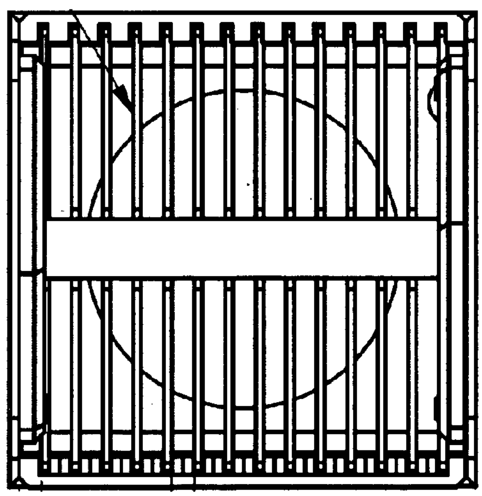


Figure 4-27: Standard LEU fuel assembly.

Figure 4-28: Control LEU fuel assembly.

Two types of channels are encountered in the PUR-1 fuel assemblies. One is the channel between plates, and the other is the channel between the last plate of an element and the assembly can wall. The plate-to-plate and plate-to-assembly wall channel thicknesses are fixed by the spacers on the wall of the assembly. It should be noted that the plate-to-wall channel is heated on only one side, so it can be conservatively assumed that half of the heat from the fuel plate associated with the fuel channel heats the coolant. Table 4-17 summarizes the channel types and thicknesses in PUR-1.

Table 4-17: Channel Types and Thickness in PUR-1 Assemblies

	Plate-to-plate (mils)		Plate-to-wall (mils)	
	Standard	**Control**	**Standard**	**Control**
LEU Asssemblies	144±15	181±15	127±8	127±8
[1] This is the smaller of the two bolt heads on the HEU elements.				

In the thermal-hydraulic analyses, the peak power plates identified in Section 4.5.2 were analyzed using NATCON. The relative power densities in each fuel plate were obtained from detailed MCNP5 criticality calculations, also described in Section 4.5.2. In the NATCON analysis, the relative axial power profiles of the individual plates were utilized in each respective case.

Hot Channel Factors are used by NATCON to account for dimensional variations inherent in the manufacturing process, as well as variations in other parameters that affect thermal-hydraulic performance. The geometry dimensions used in the NATCON models for the HEU and LEU models are shown in

Table 4-18. And the hot channel factors are listed in Table 4-19.

Table 4-18: Model Dimensions for the Thermal-Hydraulic Models

Number of Axial Nodes	14	
Number of Plates	[redacted]	
Thermal Conductivity (W/m*K) Fuel Meat Clad	 80 180	
Pool Temperature (°C)	30	
	Inches	mm
Fuel Meat Height Width Thickness	 [redacted]	 [redacted]
Channel Height Width	 25.110 2.832	 637.79 71.933
Channel Thickness		
Clad Thickness	0.015	0.381
Distance assembly can extends above fuel plate	0.450	11.430

Table 4-19: Hot Channel Factors for the LEU core

Uncertainty	Type of Tolerance	Tolerance Fraction	Hot Channel Factors		
			FBULK Coolant Temp. Rise	FFILM Film Temp Rise	FFLUX Heat Flux
Fuel meat thickness	Random	0.000	1.000	1.000	1.000
U-235 Homogeneity	Random	0.200	1.000	1.200	1.200
U-235 Mass per plate	Random	0.030	1.014	1.030	1.030
Power Density	Random	0.100	1.046	1.100	1.100
Channel Thickness	Random	0.110	1.180	1.111	1.000
Flow Distribution	Random	0.200	1.200	1.000	1.000
Random Uncertainties	**Combined**		**1.322**	**1.251**	**1.226**
Power Measurement F_Q	Systemic	0.500	**1.500**		
Flow friction factor F_W	Systemic	0.048	**1.048**		
Heat Transfer Coeff. F_H	Systemic	0.200	**1.200**		
1. Plate-to-plate channel, 20 mil uncertainty on 181 mil channel thickness.					

The following calculations were done to determine the core power distribution and the ONB power for the new design of the PUR-1 LEU fuel assemblies. The power distribution analyses discussed below show that the LEU plate 1348 is the most limiting. The ONB power calculation given below includes the impact of the power density variation along the width of the plate itself

For the most limiting LEU plate, 1348 (located plate-to-plate in control assembly 4-4 as shown in Figure 4-23), the ONB power is found to be 94.2 kW, using a 20 mil uncertainty on the 181 mil thick channel. The 20 mil uncertainty used is conservative compared to the 15 mil uncertainty given in Table 4-17 for the evaluated core design. This ONB power is shown in Table Q27-. Reducing the channel thickness increases the ONB power because the new LEU design is approaching the optimum channel thickness (≈ 100 mil) which gives the highest ONB power.

Table 4-20: ONB Power With Hot Channel Factor FW to Account for Hydrodynamically Developing Flow, and the Effect of Viscosity Temperature Dependence on FBULK (Most Limiting Fuel Plate)

	LEU Plate 1348[2]
Channel Type	Plate-to-Plate, 20 mil uncertainty, on 181 mil Channel
Hot Channel Factor FW	1.048
Hot Channel Factor FBULK	1.322[8]
Hot Channel Factor FFILM	1.357[8]
ONB Power (kW)	94.2
ΔP Due to Wall Shear, Pa	14.88
Inlet + Outlet Loss ΔP, Pa	2.01
Total Friction ΔP, Pa	16.89
ΔP Due to Buoyancy, Pa	16.89
Flow rate in hot channel, g/s	16.28
Reynolds Number at Channel Outlet	698

1. Other hot channel factors used for LEU plate 1348 are taken from Table 4-26 of the conversion proposal: FQ = 1.500, FH = 1.200, FFLUX = 1.226

2. The ratio of apparent friction parameter Capp for developing laminar flow to the friction parameter C for fully developed laminar flow, Capp/C is 1.0985 for the 197 mil thick channel in the LEU core. This ratio was calculated using Eqs. (341) and (576) of Shah and London [Ref. 2]. FW in the NATCON code input was set to 1.09850.5 because the frictional pressure drop is multiplied by a factor FW2 by the code.

3. For the LEU core (20 mil uncertainty on 197 mil thick channel), the hot channel factor for bulk coolant temperature rise, FBULK, calculated ignoring the temperature-dependence of water viscosity is 1.308 (calculated using the formula in the revised Appendix 1).

4. The value of FFILM that accounts for the power density variation over the plate width.

5. FBULK has changed to 1.322 due to 20 mil uncertainty on 181 mil channel thickness in the LEU core design as shown in Table 4-19. The FFILM in Table 4-19 (1.251) was increased by a factor of 1.085 to account for the power density variation along the width of plate 1348 (shown in Table 4-19). This results in FFILM = 1.357 (=1.0850x1.251) that is used here.

Table 4-19 shows the hot channel factors for the new LEU core design, which were calculated using the equations in the revised Appendix 1 that includes the effect of temperature-dependent water viscosity. A conservative uncertainty of 20 mil (rather than 15 mil shown in Table 4-4) on the 181 mil channel thickness (the most limiting fuel plate 1348 in control assembly 4-4) was used in finding the hot channel factors. To calculate the ONB power, the NATCON code was run using

(i) the hot channel factors shown in Table 4-19,

(ii) a total of [redacted] fuel plates,

(iii) a channel thickness of 181 mil,

(iv) a radial power factor of 1.5414 (=the ratio of 8.07 kW power in plate 1348 to 5.2356 kW at 1 kW per average plate), and

(v) the axial power shape for plate 1348 shown in Table 4-9.

To account for the power density variation along the width of plate 1348, the FFILM in Table 4-19 (1.251) was increased by a factor of 1.085 (= 1.672/1.5412 = the maximum-to-average power density ratio variation over the width of plate 1348, shown in Table 4-9). This results in FFILM =1.085x1.251 = 1.357 that was used in NATCON to calculate the ONB power.

The NATCON code calculates the Darcy-Weisbach friction factor f = C/Re for fully developed laminar flow, using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section. To account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel, an apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2]. The ratio C_{app}/C was found to be 1.0897 at a Reynolds number of 800 at the exit of the 181 mil channel in the new LEU design. Since the NATCON code multiplies the fully developed friction factor by FW^2, the hot channel factor FW equals 1.044 (= $1.0897^{0.5}$). A higher value of 1.048 for FW was used in the NATCON calculation to be conservative.

The ONB power is found to be 94.2 kW which is shown in Table 4-20 for comparison.

The margin to incipient boiling shown in Table 4-28 was calculated at the present operating power of PUR-1 (i.e., 1 kW), and it is the smallest value of the temperature difference (T_{ONB} - T_w) over the coolant channel length in the hottest channel where T_W is cladding surface temperature with all hot channel factors applied, and T_{ONB} is the local onset-of-nucleate-boiling temperature. This basically gives an idea of how far below the onset of nucleate boiling condition the reactor is operating. This definition can be written as an equation as follows:

$$\text{Margin to ONB} = \text{Minimum } T_{incp}(p, q''(z)F_{flux}) - \left[F_{bulk}\left\{T(z)-T_0\right\} + F_{film}\left\{T_{wall}(z)-T(z)\right\}\right]$$

where

$T(z)$ = Bulk coolant temperature at axial position z in the channel heated by the plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h

$T_{wall}(z)$ = Cladding surface temperature at axial position z in the channel heated by a plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h

$q''(z)$			=	Heat flux at position z for the plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$p(z)$			=	Absolute pressure in the channel at axial position z
$T_{incp}(p(z), q''(z)F_{flux})$			=	Onset of nucleate boiling temperature at absolute pressure p(z) and heat flux $q''(z)F_{flux}$
P_{op}			=	Operating power of the reactor (e.g., 1 kW for PUR-1)
N			=	Number of fuel plates in the core (e.g., 190 for PUR-1 LEU core)
T_0			=	Coolant temperature at the channel inlet
F_W			=	Hot channel factor for flow in the channel
F_Q			=	Hot channel factor for reactor power
F_h			=	Hot channel factor for Nusselt number
F_r	=	RPEAK	=	Radial power factor of the plate cooled by the channel
F_{film}	=	FFILM	=	Hot channel factor for temperature drop across the coolant film on cladding surface
F_{flux}	=	FFLUX	=	Hot channel factor for heat flux
F_{bulk}	=	FBULK	=	Hot channel factor for bulk coolant temperature rise in the channel

4.6.3 Thermal Hydraulic Analysis Results

The NATCON/ANL V2.0 code was used to determine the thermal-hydraulics performance of the PUR-1. First, the code was used to compute the power at which the ONB is reached for the plates being examined in each of the HEU and LEU cores. This was done to identify the limiting channel. Then the limiting channel was evaluated under nominal operating conditions for both the HEU and LEU cores. The ONB results provide verification that the Safety Limit (SL) and Limiting Safety System Settings (LSSS, trip points) of the Technical Specifications will indeed assure safe operation of PUR-1 for the LEU core.

4.6.3.1. NATCON Analyses

The reactor pool temperature varies throughout the year from about 22°C to 30°C depending on the ambient temperature and humidity conditions in the reactor room. In all of the following calculations, the higher value 30°C was used.

The power search function of NATCON was used to determine the power level at the Onset of Nucleate Boiling for both the HEU and LEU cores. Table 4-21 provides a summary of the ONB powers for each of the cases analyzed. For the HEU core, the limiting channel/plate was the plate-to-plate (P-T-P) case, with an ONB power of 76.3 kW. For the LEU core, the limiting channel/plate was plate 1348 with the plate-to-plate channel, which had an ONB power of 96.1 kW.

Table 4-21: ONB Powers for HEU and LEU Cores

	LEU Plate 1348		LEU Plate 1215
Channel Type	P-T-P	P-T-W	P-T-P
ONB Power (kW)	96.1	187.8	165.6
1. Plate-to-Plate spacing 2. Plate-to-Wall spacing			

Using NATCON, the thermal-hydraulics parameters of the HEU and LEU cores at the nominal operating conditions were also calculated. All hot channel factors are included in these calculations. These results are shown in Table 4-22.

Table 4-22: Operating Conditions for PUR-1 as Determined by NATCON

	LEU (Plate 1348) P-T-P (1 kW)	**LEU (Plate 1348) P-T-P (10 kW)**
Max. Fuel Temp. (°C)	32.3	39.1
Max. Clad Temp. (°C)	32.3	39.1
Coolant Inlet Temp. (°C)	30.0	30.0
Coolant Outlet Temp. (°C)	32.0	36.0
Margin to incipient boiling (°C)	78.49	70.0
Coolant Velocity (mm/s)	5.41	17.4
Coolant Mass Flow Rate (kg/s)	0.0019	0.0062

4.6.3.2. Safety Limits for the LEU Core

In PUR-1, the first and principal physical barrier protecting against the release of radioactivity is the cladding of the fuel plates. The 6061 aluminum alloy cladding has an incipient melting temperature of 582 °C. However, measurements (NUREG 1313) on irradiated fuel plates have shown that fission products are first released near the blister temperature (~550 °C) of the cladding. To ensure that the blister temperature is never reached, NUREG-1537 concludes that 530 °C is an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation. As a result, PUR-1 has proposed a safety limit in its Technical Specifications requiring that the fuel and cladding temperatures should not exceed 530 °C.

4.6.3.3. Limiting Safety System Settings for the LEU Core

Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. When a limiting safety system setting is specified for a variable on which a safety limit have been placed, the setting must be chosen such that the automatic protective actions will correct the abnormal situation before a safety limit is reached. Table 4-23 shows the maximum power, the LSSS and operating power for PUR-1.

Table 4-23: Key Power Levels for Reactor Operation and LSSS for PUR-1

	1 kW	10 kW Proposed
Maximum Power Level Including 50% Uncertainty	1.8 kW	18.75
Limiting Safety System Settings Power Level	1.2 kW	12.0 kW
Operating Power Level	1.0 kW	10.0 kW

During steady-state operation, peak clad temperatures are maintained far below 530°C , as well as below the temperatures required for ONB (see Table 4-22). NATCON was used to determine the minimum power for ONB for the HEU and LEU cores in the limiting channels, as well as the thermal-hydraulic parameters at these calculated powers. The results of these calculations are shown in Table 4-24.

Table 4-24: Power Levels at the Onset of Nucleate Boiling for HEU and LEU Cores.

	LEU
Power level at ONB (kW)	96.1
Max. Fuel Temp. (°C)	112.6
Max. Clad Temp. (°C)	112.5
Coolant Inlet Temp. (°C)	27.0
Coolant Outlet Temp. (°C)	42.6
Onset of Nucleate Boiling Ratio	1.0
Coolant Velocity (mm/s)	53.3
Coolant Mass Flow Rate (kg/s)	0.019

The licensed operating power level of PUR-1 is 1 kW. The LSSS scram setting of 120% power (1.2 kW for 1 kW, 12.0 kW for uprate) is well below the power level of 96.1 kW in the LEU core, at which ONB would occur in the respective limiting channels. Thus, the present LSSS on power at 12 kW (120% normal operating power) will easily protect the reactor fuel and cladding from reaching the Safety Limit under steady state operations.

Chapter 13 (Accident Analyses) analyzes two hypothetical transients based on values of the Technical Specifications for the LEU core. These transients are: (1) Rapid insertion of the maximum reactivity worth of 0.3% Δk/k of all moveable and non-secured experiments, and (2) Slow insertion of reactivity at the maximum allowed rate of 0.04% Δk/(k*s) due to control blade withdrawal.

For the case of the rapid insertion, of 0.3% Δk/k, the reactor scram was initiated based on the power level trip, assuming failure of the period trip. For the case of the slow insertion of 0.04% Δk/(k*s), scram was initiated on the second power level trip, assuming the first power level trip failed. The reason for this is that the period trip is never reached for the case of this slow reactivity insertion.

Thus the selected LSSS is a conservative setting which ensures that the maximum fuel and cladding temperatures do not reach the safety limit of 530 °C for the range of accident scenarios that were analyzed. In summary, the selected LSSS will protect the reactor fuel and cladding from reaching the safety limit of 530°C under any condition of operation.

However, a NATCON thermal-hydraulic calculation for the LEU plate 1348 was performed assuming a hypothetical pool temperature of 35 °C, and a hypothetical inlet loss coefficient of 10.0 (increased from 0.5), while applying all six hot channel factors of the case H. The ONB power was found to be 79.3 kW, indicating a large margin compared to the PUR-1 operating power of 1 kW.

4.7 References

[1] R. S. Smith and W. L. Woodruff, "A Computer Code, NATCON, for the Analyses of Steady-State Thermal-Hydraulics and Safety Margins in Plate-Type Research Reactors Cooled by Natural Convection," Argonne National Laboratory (ANL), ANL/RERTR/TM-12, Dec 1988.

[2] M. Kalimullah, "NATCON v2.0 Instructions", Argonne National Laboratory (ANL), ANL/RERTR, July 2006.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The water process system includes a 30 GPM water pump, a water filter, a demineralizer, flow meter, a chiller, conductivity cells that measure the pool water before and after passing through the demineralizer and appropriate valves. The details of the reactor coolant system are summarized in

Table 5-1: PUR-1 water process system summary.

WATER PROCESS SYSTEM	
Water Capacity, Reactor Pool	6,400 gal.
Pump Type Capacity At 100°F/110 ft Head	Centrifugal 30 gpm
Ion Exchanger Rated Capacity Flow Rate Design Pressure	Mixed bed (replaceable cartridge type) 15, 000 grains as $CaCO_3$ 0-20 gpm 100 psi
Heat Exchanger Cooling Capacity Design Pressure Tube Shell	Shell and tube (water chiller) 36,000 Btu/hr (10,550 W) 150 psi 225 psi
Filter Maximum Flow Pressure Drop	Cartridge Type 80 gpm 2 psi

5.2 Primary Coolant System

The process system is provided with the reactor to control the pool .water quality and temperature. The purification system is designed to limit corrosion and coolant activation by the use of microfilters and ion exchange resins.

The process system is assembled in one unit and is mounted on a base structure near one wall of the reactor room. The components include a circulating pump, a cartridge-type demineralizer, a Cuno filter, a refrigeration compressor-condenser unit, a heat exchanger, flowmeter, conductivity indicator, thermostat, wiring, piping and valves to complete the system. City water is available through a water supply tank to provide a source of makeup pool water. City water is also provided to the compressor-condenser unit as a secondary cooling medium.

The reactor pool system is arranged in the following order: Water from the pool is drawn out from the scupper drain or suction line via PVC pipe leading to the circulating pump; a second source of water for the pump is a water supply tank supplied with city service water and controlled by a float valve. Ball valves for water shutoff and a vacuum cleaning connection are provided in, the pump supply line. From the pump a pipe with ball valve installed leads first to the filter and then to a demineralizer. An adjustable by-pass or throttling valve is inserted in the system to regulate water flow through the demineralizer. A flow indicator and a conductivity indicator are installed as a check on water purity and flow rate from the demineralizer. The water next flows through a stainless steel heat exchanger. The water from the heat exchanger is then returned to the reactor pool. A magnetrol water-level control is located in the reactor pool; this unit controls a solenoid valve in the line from the water supply tank to ensure that the prescribed pool water level is maintained.

5.3 Secondary Coolant System

The secondary coolant The water next flows through a Ross Model 302 stainless steel heat exchanger of 36,000 Btu/hr capacity which serves as a water chiller; included with this system is a thermostatically controlled Copelametic Model W300H compressor-condenser unit with a water cooled condenser. The water from the heat exchanger is returned to the reactor pool.

Experimentally, no temperature increase has been observed with the pool thermometer following 8 hours of operation at 1 kW. However, based on the mass of water as 1.85 10^4 Kg, calculations indicate that the temperature increases after operating the PUR-I at power lever of 10 kW would be 0.465°C/hour (this takes no credit for heat loss to the surrounding sand and gravel or loss by evaporation). The capacity of the chiller system is 10,550 Watts, which will be able to maintain pool temperature as required.

The chiller is designed with three loops to prevent the spread of contamination in an emergency. The pool water passes through the primary loop while a freon refrigerant is in the secondary loop. The third loop uses campus water to remove the heat and is discharged into the campus sewer system. The chance of contamination passing through the three loop system is small.

5.4 Primary Coolant Cleanup System

From the pump a pipe with ball valve installed leads first to the cartridge-type Cuno filter and then to a Barnstead BD-10 demineralizer. An adjustable by-pass or throttling valve is inserted in the system to regulate water flow through the demineralizer. A flow indicator and a conductivity indicator are installed as a check on water purity and flow rate from the demineralizer.

5.5 Primary Coolant Makeup Water System

The water level in the pool must be maintained at least 13 feet above the reactor core during operation. During periods when there are no planned operations, the water level is still maintained at approximately 13 feet. On average, the addition of about 40 gallons of water per week is required to maintain that water level to make up for evaporation. The primary water makeup system consists of a 20 gallon, gravity driven reservoir that is filled with water from the university water system, that is processed through deionizers consisting of each of an anion, cation and mixed-bed tank.

This tank is filled as required, water is added as needed through a solenoid controlled float switch at the top of the reactor pool. Records are kept for the amount of water used with each fill, and maintained at the reactor facility.

5.6 Nitrogen-16 Control System

The main possible source of nitrogen-16 is from the fast neutron interaction with oxygen in the pool water. The nitrogen must then diffuse to the surface of the pool before it is released to the atmosphere. In normal operation, no strong currents are established in the reactor pool and with the short half-life (7.14 seconds), the nitrogen decays before reaching the surface. No nitrogen-16 has been observed in the reactor room.

5.7 Auxiliary Systems Using Primary Coolant

PUR-1 has no auxiliary systems that use primary coolant.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

PUR-1 is located in a high-bay room that serves the function of confinement in case of emergency. The room is kept at negative air pressure (minimum of -0.05 inches of water) at all times when the reactor is operating, and air intake and outlet are through HEPA filters. The ventilation system can be shut down during emergency situations to restrict release of contamination to the surrounding environment.

The LEU MTR fuel type serves as containment for fission products produced in the fuel during reactor operation. This fuel type is widely used, and has shown excellent fission product retention under operating conditions much harsher than what is experienced at PUR-1, both in burnup and water quality. In the unlikely event that radioactive fission products were to leak from the fuel plates, they would be contained in the reactor pool water, which is monitored for contamination regularly, and also filtered through the water process system.

6.2 Detailed Descriptions

6.2.1 Confinement

The Duncan Annex of the Electrical Engineering Building is of brick, concrete block and reinforced concreted construction which was originally designed as a large high voltage laboratory. It was subsequently subdivided into offices, classrooms and laboratories. [redacted] Figure 3.1 shows the floor plan of the Nuclear Engineering laboratories, including the reactor room.

The outside air supply and exhaust are both passed through HEPA filters. The reactor room is maintained at negative air pressure (minimum -0.05 inches of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room and a room air conditioner circulates and cools the reactor room air.

The only floor drain to the sewers is sealed except for a vent opening. This vent is raised about two feel above the floor and has a filtered inverted opening. Condensate from the air conditioner is released to this drain through an opening 12.0 feet above the floor.

During emergency conditions the exhaust systems is shut off and the sealed room will prevent the rapid spread of contamination. During an emergency the air conditioner and the valve on the drain from the condensate holdup tank are shut off with the same switch that shuts off the exhaust system. The condensate is held until it is tested by Radiological Control before it is released to the sewer. If contamination is found, it is disposed of as radioactive liquid waste.

6.2.2 Containment

The PUR-1 reactor does not require containment.

6.2.3 Emergency Core Cooling System

PUR-1 is a low power pool-type reactor, and in no credible accident scenarios would an Emergency Core Cooling System be necessary.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The PUR-1 instrumentation is summarized in Table 7-1. Is consists of 3 operating channels, and 1 safety channel. The three operating channels are the startup range channel, a log power channel, and a linear power channel. The safety channel has its own detector, as well as monitoring the log power channel for period.

Table 7-1: Summary of PUR-1 Instrumentation.

Instrumentation	Detail
Startup Channel	
Detector	Fission chamber
Range	Source level to 10^{-6} N_f
Indicators	Log countrate meter
Range	Decade Scaler
Log-N Period Channel	
Detector	CIC
Range	104 to 10^{10} n/cm^2 -sec
Indicators	Log-N meter
Range	10^{-4} to 300 N_f
Indicators	Period Meter
Range	-30 to +3 sec
Power Channel	
Detector	BF_3 Ion Chamber
Range	10^4 to 10^{10} n/cm^2-sec
Indicator	Linear Level Meter
Range	0 to 100 N_f
Safety Channels	
Detector (period)	CIC
Range	10^4 to 10^{10} n/cm^2 sec
Detector (level)	BF_3 Ion Chamber
Range	10^{-3} to 150 N_f

Rod Position Indicators Detectors Regulating Range Shim Safety Range Indicators Coarse Range Fine Range	10-turn precision potentiometer 0-2500 ohms 10-turn precision potentiometer 0-2500 ohms Voltmeters 0-70 CM 0-70 CM Digital Voltmeter
Area Monitors (3) Detectors Range Indicators	Scintillation Detectors 0.05 to 50 mr/hr Local & Remote Meters

7.2 Design of Instrumentation and Control Systems

The design of the instrumentation and control system for the Purdue University reactor is based on systems in use at such reactors as the Bulk Shielding Reactor, the Tower Shielding Reactor, and the Pennsylvania State University Reactor.

The instrumentation system consists of three operational channels and one safety channel. The operational channels include a counting rate channel, a log-N period channel, and a linear level channel. The log-N period channel sends a signal to a composite safety amplifier and the sigma bus to provide a fast period scram. Scram systems are interconnected into these channels to effect reactor shutdown in the event of an emergency or abnormal condition. An annunciator and alarm system are included to indicate specific trouble.

7.2.1 Channel 1—Start-up Channel

The startup channel is used to monitor the neutron flux. The channel consists of a fission chamber, a preamplifier, a pulse amplifier, a scaler for accurate counting, a log count rate and period amplifier, a log count rate recorder, and shares a period recorder with Channel #2. The range of this equipment if from 10 to 10^4 counts/second with periods from -30 to +3 seconds. In addition to the outputs shown on the recorders, readout is also provided by a log count rate meter and a period meter shared with Channel #2 on the console and instrument panel. The complete reactor power range may be monitored by this instrument by appropriate repositioning of the detector by means of the fission chamber drive mechanism. The fission chamber may be raised into a cadmium shield by means of a drive mechanism similar to the control rod drive units. The controls and position indication for this drive are located on the console. Two set points, specified in the Technical Specification and based on the reactor period, provide for a reactor setback and trip in the event of a short reactor period.

The first activity in the reactor is indicated by means of the counting rate channel. At start-up, the fission chamber is placed near the reactor core (lower limit). Neutron-produced pulses from the fission chamber are amplified and counted on the scaler, an electronic high-speed counting device with a mechanical register. (Smaller pulses produced by any agency other than neutrons, such as gamma radiation, are rejected by a pulse height discriminator circuit.) These amplified pulses are also fed into the log count rate meter where they are integrated and the average counting rate displayed on a logarithmic indication meter calibrated for four decades (1 to 10,000 counts per second).

The average counting rate is also displayed on a remote indicating meter and recorded on a 10 millivolt Speedomax recorder provided with four-cycle logarithmic paper. When the control rod has been pulled out far enough to produce an effective reproduction constant greater than one (1.0), the reactor power level increases exponentially (at low power levels) and a straight line is drawn on the logarithmic recorder chart. The slope of this line is the pile period, or the time required (in seconds) for the power level to change by a factor of e (approximately 2.718).

The reactor period is displayed on a pile period meter located on the log count rate meter and is recorded on the period recorder until the lower limit of the log-N period channel range is reached.

When the counting rate channel is near the limit of its counting range, the log-N period and linear servo channels become the principal means of controlling the reactor, and the fission chamber is withdrawn to a region of lower neutron flux (to keep the recorder on scale).

7.2.2 Channel 2—Log N and Period Channel

The log N channel indicates the reactor power level over the range from 0.0001 to 300 percent power level. The detector for this channel is a compensated ionization chamber followed by a log N amplifier plus period instrumentation with outputs to the log N recorder and to the period instrumentation with outputs to the log N recorder and to the period recorder shared with Channel #1. Indication, in addition to the recorders, is provided by log N and period meters on both console and instrument rack, with the console period meter shared with Channel #1. This channel is not 'on scale' at startup, but will be indicating before the range of the fission is exceeded.

A reactor trip will be initiated if this channel indicates power levels in excess of 120% of the licensed power. Two set points, specified in the Technical Specifications and based on the period, provide for a reactor setback or trip in the event of a short reactor period. In the event of the loss of high voltage to the compensated ion chamber, a trip will be initiated.

7.2.3 Channel 3—Linear Power

The linear level channel is capable of measuring neutron flux in a reactor operating range from shutdown to > 100 kilowatt. The sensing element is BF_3 ionization chamber coupled to a micro-microammeter. The range of the instrument is adjustable by means of a range switch located on the instrument (instrument panel) from 0–10.0x10^{-12} to 0–10.0x10^{-4} amperes. Detector characteristics, however, limit its maximum output to 10^{-4} amperes. This channel will thus read from startup to full power by adjustment of the range switch. The output is recorded on the linear power recorder on the instrument panel and indicated on meters on the console and on the instrument panel.

This channel has two set points that will initiate a reactor set back at either zero or 100% range. These set points insure that the instrument is kept on range at all times during reactor operation. This is also a 120% range set point that will initiate a reactor trip.

The linear servo channel consists of a BF3 ionization chamber (whose output current is proportional to neutron flux) and its power supply, a micro-microampere amplifier, a Speedomax recording controller with a position-adjusting type control unit, and the regulating rod drive unit. Because this channel is the most precise channel in measuring neutron flux level, it is used as the means to obtain uniform neutron flux density.

The BF_3 ionization chamber provides a DC current which is proportional to the neutron flux at the chamber. This current is amplified by the micro-microampere amplifier which provides a 0-10 millivolt input signal to the recording controller. The controller provides a signal to the servo control unit which, through a drive unit, drives the control rod the necessary amount in the correct direction to maintain the power level at the control point set on the controller. Whenever the power level exceeds the control point setting the control rod is inserted into the reactor the amount necessary to decrease the reactivity and restore the power level to the control point setting. Conversely, whenever the power level drops below the control point setting, the control rod is withdrawn the amount necessary to increase the reactivity and restore the power level to the control point setting.

In practice, the operator sets the red control pointer to the desired power level and selects the range on the micro-microampere amplifier which will provide a reading on the linear recorder equivalent to the power level reading desired on the log-N recorder. The rods are then withdrawn until the black indicating pointer coincides with the red control pointer at which time the power is leveled off manually and the control unit is placed on automatic operation.

7.2.4 Log-N Period Channel

The log-N period channel includes a compensated ionization chamber (whose output current is proportional to neutron flux only) and its power supply; a log-N amplifier and a Speedomax recorder which records log-N; and a composite safety amplifier whose components are a period safety preamplifier, sigma amplifier and magnet amplifier.

The compensated ionization chamber supplies a d-c current (proportional to the neutron flux at the chamber) which is amplified by the log-N amplifier. One output of the log-N amplifier is the logarithm of the power level which is indicated on the power level meter and recorded on the log-N recorder. This output is also differentiated and indicated on the pile period meter. (The pile period is the number of seconds necessary for the power level to increase or decrease by a factor of e.) Positive periods from infinity to three seconds, and negative periods from infinity to thirty seconds are indicated.

7.2.5 Channel 4—Safety Channel

This channel utilized a BF_3 ion chamber and feeds directly into the safety amplifiers. The sensitive range of this instrument is from a few percent to at least 150 percent of power, linearly. Its output is indicated on the instrument chassis (instrument panel). The purpose of this channel is solely to provide a trip at the measure value as specified in the Technical Specifications.

The level safety channel, in conjunction with the safety circuit of the log-N period channel, functions to shut down the reactor immediately by dropping both safety rods whenever the power level increases above 120% N_f and/or an abnormally short period occurs. (Set for a period of 7 seconds.)

In the level safety channel a BF_3 ionization chamber supplies a d-c current proportional to the reactor power level to the composite safety amplifier. Since the current produced by neutron flux at high levels of operation is much greater than that produced by gamma radiation or other sources, this chamber is not compensated. Each magnet amplifier supplies current to an electromagnet which holds a safety rod. The output of each sigma amplifier is connected to the sigma bus which is connected to the input of each magnet amplifier. When there is no signal from the ionization chamber, the sigma amplifiers maintain the potential of the sigma bus at 37 volts. However, if the positive period should become abnormally short (7 seconds) or if the neutron flux (power) level should become dangerously high the sigma bus potential is increased which causes the magnet current in both magnet amplifiers to be quickly reduced to zero, thus dropping both safety rods into the reactor core. This is what is meant by a "fast scram."

The same result is also achieved if the sigma bus potential should increase for any other reasons, since the sigma bus is connected to the input of each magnet amplifier. (Note that a fast scram may be initiated automatically by a short period or a high power level.) Indicator lamps in the annunciator system located on the control console indicate whether a fast scram is due to high level or to period.

7.3 Reactor Control System

Four control channels are incorporated into the reactor control system to raise or lower the two shim-safety rods, the regulating rod and the fission chamber. A fifth control channel is provided to raise and lower the source. The control rods may always be lowered with a jam circuit being the only restriction; however, permissive circuits are included in the raise circuits of the control rods and fission chamber to prevent raising the rods under the following circumstances:

Indicator	**Condition**

Source Missing	The log count rate recorder must indicate the presence of a source of neutrons by indicating a count rate of at least two counts per second. A source missing indicator shows when this condition is not fulfilled.
Log Count Rate Recorder On	A relay in the log count rate recorder prevents raising of the control rods or the fission chamber when the log count rate recorder power is turned off, thus prohibiting the disabling of the source missing circuit. The log count rate recorder may be turned off after the log-N recorder comes on scale. This is accomplished by a relay in the log-N recorder circuit.
Source Drive in Operation	Interlock circuits on the source drive switches prevent raising the control rods or fission chamber when the source is being raised or lowered.

A jam circuit is incorporated in the drive circuits to operate a jam indicator light on the console in the event of a mechanical jam in the drive. This indication alerts the operator to the possibility of cable kinking in the source and fission chamber drive units, or mechanical friction in the rod drives.

7.3.1.1. Shim Safety Rod Drive System

The shim-safety rod to be driven is selected by pushing the desired drive indicator lamp. This connects the drive system for that rod and clears any other drive circuit that may be energized. Electrical interlocks prevent the raising of more than one control rod or the fission chamber simultaneously.

The reactor operator controls the rod with the raise-lower switch on the control console. The raise-lower switch operates mercury relays which control the rod drive motors.

Indicator	**Condition**
Upper Limit	The drive unit is at the upper limit of its travel.
Shim	The drive unit is two-thirds out. This point is where the shim-safety rods are set during critical experiment fuel loading.
Engage	The shim-safety rod is attached to the drive electromagnet.
Lower Limit	The drive unit is at the lower limit of its travel.
Drive	The drive unit is connected to the raise-lower switch.

7.3.1.2. Regulating Rod Drive System

The regulating rod drive system is activated by pushing the reg rod drive indicator lamp. This connects the regulating rod system to the raise-lower switch and clears any other drive circuit that may be energized. Electrical interlocks prevent raising more than one control rod or the fission chamber simultaneously.

The reactor operator controls the regulating rod with the raise-lower switch on the control console. The raise-lower switch operates mercury relays which control the rod drive motors.

The regulating rod may be controlled by the servo amplifier if the reactor operator pushes the servo-permit indicator. The servo system is turned off by pushing the servo indicator.

The following indicator lights are provided for the regulating rod:

1. Upper Limit
2. Servo
3. Servo Permit
4. Lower Limit
5. Drive

7.3.1.3. Fission Chamber Drive System

The fission chamber drive system is activated by pushing the fission chamber drive indicator lamp. This connects the fission chamber drive system to the raise-lower switch and clears any other drive circuit that may be energized. Electrical interlocks prohibit raising the fission chamber while the control rods are being driven.

The reactor operator controls the fission chamber with the raise-lower switch on the control console.

The following indicator lights are provided for the fission chamber:

1. Upper Limit
2. Lower Limit
3. Drive

7.3.1.4. Source Drive System

The source is raised or lowered by pushing either the source raise or source lower indicator lamp. The source drive switches are of the momentary type and must be pushed and held on until it is desired to stop the source drive or one of the limits is reached.

The following indicator lights are provided for the source drive:

1. Upper Limit
2. Source-Raise
3. Source-Lower
4. Lower Limit

7.3.1.5. Gang Lower System

When the reactor is to be shut down, it is permissible to lower all control rods simultaneously by pushing the gang lower indicator light and thus activating the gang lower switch and relay. The gang lower relay directly energizes the lower drive relays for the control rods. The gang lower relay is de-energized by momentarily placing the raise-lower switch into either its raise or lower position. In addition, the gang lower relay may be energized by the setback system described in Section 1.5.5.2.

7.3.1.6. Rod Position Indicating System

Four coarse rod position indicators are provided which continuously monitor the positions of the two shim safety rods, the regulating rod and the fission chamber with a resolution of 2 centimeters. A ratiometer is used as a fine position indicator which may be switched to any one of the control rods or fission chamber to provide a position indication with a resolution of 0.1 millimeter. The rod position indicator measures a voltage across a potentiometer connected to the drive unit. The reference voltage for all drive unit potentiometers is supplied by a 68 volt power supply.

7.4 Reactor Protection System

Two types of scrams are included in the control system to effect shutdown of the reactor in the event of emergency conditions. A fast scram can be accomplished when a short period signal or a high power-level signal removes magnet current. The fast scram operation is described in Section 7.2.5.

A slow scram is initiated by various safety circuits located in the reactor system. The slow scram circuit opens the input power line to the magnet amplifier power supplies located in the composite safety amplifiers.

A setback system is included in the reactor control system to insert the control rods into the reactor without producing a scram. This minimizes the necessity of repeating the startup procedure.

When a scram or setback condition occurs, or when other trouble arises, a buzzer alarm will sound, and a lamp or lamps will be lighted on the control console, indicating the source of the trouble. An annunciator acknowledge button is used to turn off the buzzer. If the trouble results in a scram, a scram reset switch must be actuated before magnet power can be reapplied. In the event of trouble other than a scram, the corresponding switch-lamp is pushed to extinguish the lamp and reset the annunciator system after the trouble has been corrected. If the trouble is not corrected, the indicator light will remain lighted. An annunciator test switch is provided for checking the lamps and the buzzer. An evacuation alarm horn is installed in the reactor instrument racks. This horn is activated by pushing the alarm button on the reactor console.

7.4.1.1. Slow Scram System

Opening of a fault switch actuates a slow scram channel which causes primary power to be removed from the two magnet amplifiers located in the composite safety amplifiers, thus resulting in the shim-safety rods dropping into the core. The following conditions result in a slow scram.

Slow Scram Type	Condition
Scram - Console	Manual pushbutton on console depressed.
High Level - Console Monitor	High radiation level at the console radiation monitor.
High Level - Process System Monitor	High radiation level at the process system radiation monitor.
High Level - Rod Drive Area Monitor	High radiation level at the rod drive area radiation monitor.
CIC Power Supply Trouble	Failure of the high voltage power supply for the compensated ion chamber.
Composite Safety Amplifier Trouble	Failure of one or more of the circuits in either composite safety amplifier.

The following scram conditions are initiated by relay type meters located on the auxiliary scram panel. The scram set points are determined and set by the reactor supervisor prior to operation.

Scrams	**Conditions**
Low-Level Period	Period circuit in the log count rate meter indicates a short period.
High-Level Period	Period circuit in the log-N period amplifier indicates a short period.
High-Level Log-N	The log-N channel indicates a high power level.
High-Level - Linear Level	The linear level channel indicates a high power level.

The following permissive circuits are located in the slow scram circuit and must be in operation before the shim-safety rods can be raised:

1. The log-N selector switch must be in the <u>operate</u> position.
2. The log-count-rate meter switch must be in the <u>use</u> position.
3. The key switch must be on.
4. The scram reset button must be pushed and the magnet power lamp energized.

7.4.1.2. Setback System

Four setback circuits are included in the reactor control system. Actuation of any one of these circuits will result in the shim-safety rods and the regulating rod being driven into the core to their lower limits unless the trouble is cleared and the circuit reset before the lower limits are reached. The setback set points are determined and set by the reactor supervisor prior to operation. The following setback conditions are initiated by relay type meters located on the auxiliary scram panel:

Low Level Period	Period circuit in the log count rate meter indicates a short period.
High-Level Period	Period circuit in the log-N period amplifier indicates a short period.
High-Level - Linear Level	The linear level channel indicates a high power level.
High-Level - Safety Amplifier	The level safety amplifier indicates a high power level.

7.5 Control Console and Display Instruments

The reactor console is designed to provide maximum visibility of the instruments and accessibility to the controls and indicators. All indicators and controls necessary for startup and shutdown operations are located in one group in front of the operator.

Colors for the indicator lights on the console show the operator the status of the reactor at a glance. All trip and warning indicators are red or yellow. Operating procedure, as well as interlock, keep the operator from withdrawing the control rods when a warning indicator is showing.

7.6 Radiation Monitoring Systems

Three scintillation type area monitors are installed in the reactor area to monitor the radiation level at the top of the pool, in the pool water flowing through the process system, and at the reactor console. Each area monitor is connected into the slow scram circuit and is equipped with a local lamp and alarm that is activated if the radiation level exceeds the set point. Three remote area monitor meters are mounted on the reactor instrument racks to provide the operator with an indication of the radiation level at the area monitor sites.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The instrumentation and controls of PUR-1 use the standard building electrical power system at 120 Volts on a dedicated circuit. The water process system uses three-phase power at 240 Volts. Room lighting and HVAC are also powered by building power on their own circuits.

8.2 Emergency Electrical Power System

There are no emergency electrical power systems for any of the systems associated with PUR-1. Loss of electrical power during operation will de-energize the magnet amplifiers on the shim-safety rods, which will drop them into the core and shutdown the reactor similar to a scram, leaving the reactor in a safe condition under any credible operating or accident scenario.

Emergency lighting for the room will be activated upon loss of building power, enabling the operators to ensure the safe shutdown condition of the reactor (i.e. shim-safety rods inserted) and direct personnel in the reactor room as necessary.

9 AUXILLARY SYSTEMS

9.1 Heating, Ventilation and Air Conditioning Systems

The outside air supply and exhaust are both passed through HEPA filters. The reactor room is maintained at negative air pressure (minimum 0.05 inches of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room and a room air conditioner circulates and cools the reactor room air.

9.2 Fuel Storage and Handling

[redacted] These rows are separated by a one quarter inch BORAL plate to reduce the k_{eff} of the fuel stored there. There is also a dry storage facility, located within the facility, but outside of the reactor room.

An MCNP model of the in-pool storage racks was constructed. Two cases for the LEU fuel were examined. One case was run with standard LEU assemblies in all of the 18 positions (which is not possible with the anticipated LEU inventory, but was run as a limiting case), and no credit was taken for the ¼" BORAL plate between the two rows. This first case had a calculated eigenvalue of 0.7660±0.0046. The second case was modeled with the ¼" BORAL plate with a boron density of 23.8E21 $^{atoms}/_{cc}$, and 16 standard assemblies with 14 fuel plates in each. The eigenvalue for this calculation was determined to be 0.3319 ±0.00178. Both of these bracketing cases are below the TS 5.3 limit of 0.8, thus TS 5.3 will be met. The geometry of the model is shown in Figure Q37-1 below.

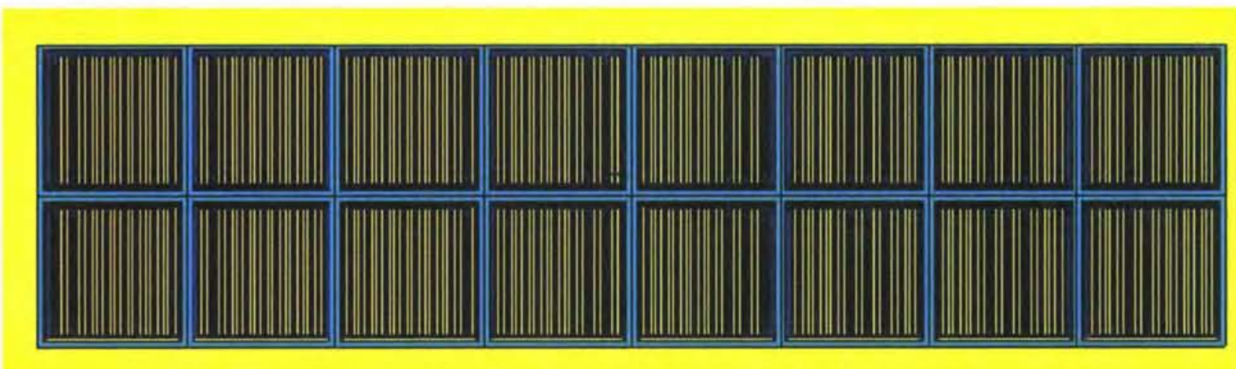


Figure 9-1: MCNP model of in-pool fuel storage rack.

The storage of the extra LEU plates is in the dry storage facility. Parametric eigenvalue calculations have been performed with MCNP5, varying the storage plate spacing in a hypothetical flooded condition to ensure that a critical configuration could not be achieved. These results are shown in Figure 9-2. The maximum k_{eff} obtained in these calculations was 0.41, which is less than the Technical Specification (TS 5.3.1) requirement of less than 0.8. The model used for the dry storage is shown in Figure

9-3.

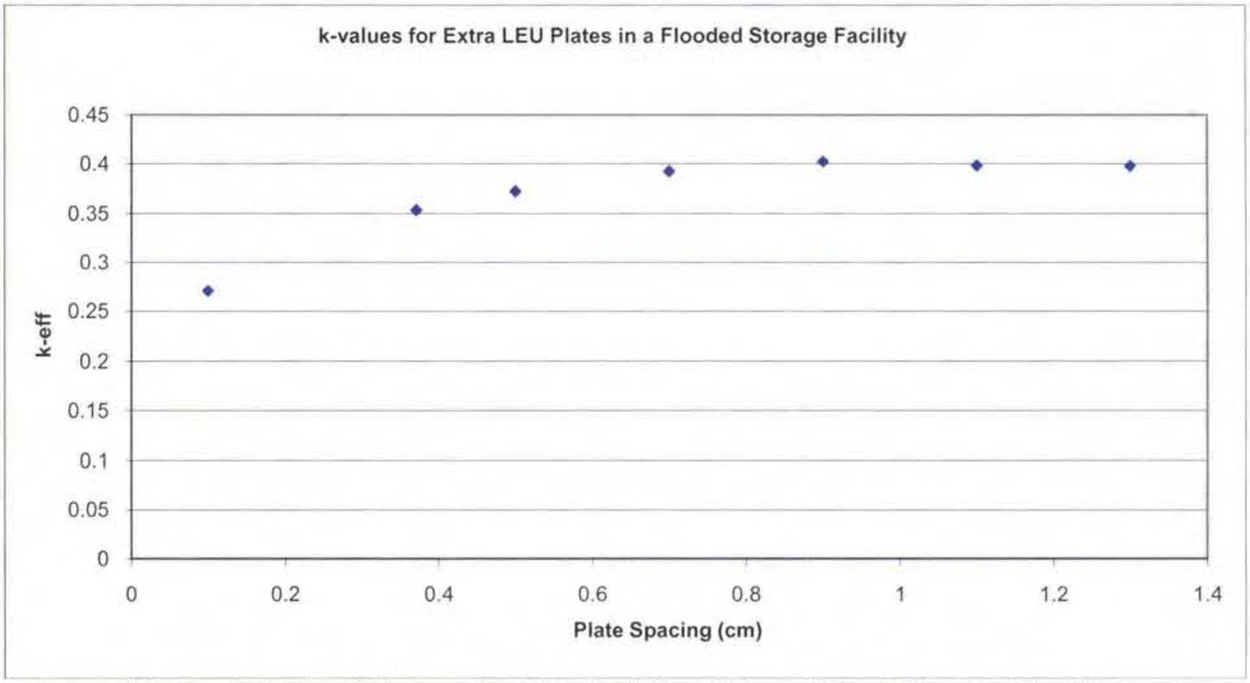


Figure 9-2: k-eff Values for Flooded Condition of Fuel Storage Facility.

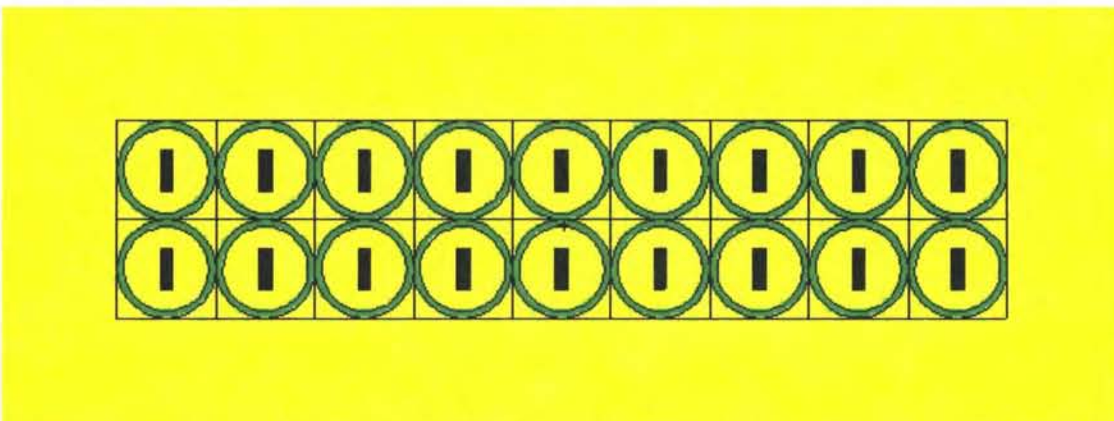


Figure 9-3: MCNP model of dry fuel storage facility.

With all of the LEU fuel plates in the inventory located in the dry storage facility, the calculated eigenvalue of the storage rack is 0.015±0.0003. Should the storage flood with water, the eigenvalue becomes 0.26±0.004. In both of these conditions, the TS limit of $k_{eff}<0.8$ is met. Parametric analyses were performed with spacing of the plates within the tubes, and the above listed cases are the worst case. All movement of fuel is performed under the direction of licensed senior reactor operators.

9.3 Fire Protection Systems

A fire extinguisher, maintained by the Purdue Fire Department, and appropriate for the types of fires that could be encountered in the reactor room, is located within the reactor room.

9.4 Communication Systems

A telephone is located at the console from which the operator can contact the Senior Operator on call, or any other assistance that may be necessary to support operations. Dialing 911 from the console phone will direct a call to Emergency Services.

9.5 Possession and Use of Byproduct, Source and Special Nuclear Material

Purdue University has a Broadscope license, and all byproduct material generated with the reactor falls under this license, as described in Chapter 11.

10 EXPERIMENT FACILITIES AND UTILIZATION

10.1 Summary Description

PUR-1 experimental facilities consist of experiment locations within the graphite reflector on one side of the reactor, and drop tubes located as close as next to the reflector boundary, and as far away as 30 inches. All of the drop tubes are dry air tubes, and the in-reflector facilities are aluminum tubes normally filled with graphite, which can be replaced with experiment capsules.

10.2 Experimental Facilities

The aluminum sample tubes in the irradiation facility are filled with graphite unless they are being used. Replacing the graphite in the six sample tubes with air, as would be expected if samples were inserted, reduces the calculated eigenvalue of the LEU core from 1.00731±0.0002 to 0.996±0.0002 without any reactivity bias, but with the estimated bias of 0.32%, these values would be 1.0041 and 0.993, respectively.

The effect of replacing graphite in the irradiation facility with an aluminum sample holder on the reactor power distribution used in the thermal-hydraulic analyses is well covered by the global hot channel factor of 1.5 for the reactor power measurement uncertainty used in all the reported analyses. The reactor power hot channel factor was used in addition to (1) the radial power factor (used to account for plate-to-plate power variation), (2) the factor used to account for power density variation along the width of the hot plate, and (3) the axial power profile. While replacing the graphite in the irradiation facility with an aluminum sample holder will cause a change in the reactor power distribution, this change will be small compared with the hot channel factors already applied in the thermal-hydraulic analysis. Therefore, no additional thermal-hydraulic analysis is needed.

Other experimental facilities include dry air drop tubes. Two tubes are located next to the core, one 5/8" in diameter, and the other 1.75" in diameter. A 3" PVC drop tube is located in the pool, and a 5" stainless steel drop tube is available that can be located in any grid location. An analysis of the flooding of the 5" drop tube is included in the safety analysis in Chapter 13.

10.3 Experiment Review

Review of new experiments is performed by the reactor staff, and then by the Committee on Reactor Operations, as required by the Technical Specifications.

10.4 References

11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 Radiation Protection Program

Purdue University has a structured radiation safety program. Policies for the program are determined by the University Radiation Safety Committee, which has the mission to ensure the safety of the University and community in the utilization of all radioactive materials and radiation producing devices at the University by faculty, staff or students. This includes all teaching, research, and outreach programs. The program is administered by the Radiation Safety Officer (RSO) and his staff, as part of Radiological and Environmental Management (REM). The staff is equipped with radiation detection instrumentation to determine, control and document occupational radiation exposures at the reactor facility, and all laboratories using radioisotopes at the university under the By-product License 13-02812-04 (Broadscope).

Natural background radiation levels in the West Lafayette area result in an average exposure of about 100 mrem/yr. On the basis of normal reactor use, the maximum potential non-reactor room dose would be less than 1 mrem/yr, so there should be no significant contribution to the background radiation in unrestricted areas.

11.1.1 Radiation Sources

11.1.1.1. <u>Reactor</u>

Radiation from the reactor core if the primary source of radiation directly related to reactor operations. Radiation exposure rates from the reactor core are reduced to acceptable levels by the water in the pool and concrete shielding.

11.1.1.2. <u>External Sources</u>

Sources of radiation associated with reactor use include radioactive isotopes produced for research, activated components of experiments and activated samples.

11.1.2 Radiation Protection Program

Purdue University "Executive Memorandum No. B-14" establishes the University administrative structure for radiation protection, and a "Radiation Safety Manual" is published and maintained by REM, which contains the rules and radiation safety procedures for all laboratories using radioisotopes and/or ionizing radiation, including the reactor. Routine surveys are performed of the reactor room and include analysis of the reactor pool and reactor room air by personnel from REM.

The University has a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques ensures that any credible type of radiation and any significant intensities will be detected promptly and measured correctly.

All reactor-related personnel are required to attend a radiation safety training session before they begin work at the reactor. Written procedures have been prepared that address routine health physics monitoring at the University's research reactor facility, and all reactor personnel are trained in these as well.

11.1.3 ALARA Commitment

The University is committed to the principle of ALARA (As Low as Reasonably Achievable), and REM makes every effort to keep doses to a minimum. All unanticipated or unusual exposures are investigated.

11.1.4 Radiation Monitoring and Surveying

11.1.4.1. Fixed-Position Monitors

The PUR-I has 3 fixed-position radiation area monitors (RAM) with adjustable alarm set points and 1 continuous air monitor (CAM) in the reactor room. The CAM air filters are changed and analyzed semi-monthly.

11.1.4.2. Experimental Monitoring

Wipe tests of exposed surfaces of the reactor room are made monthly. Water samples are taken and counted monthly. All samples and material removed from the reactor are checked for levels of activity and wipe tests made for loose contamination.

11.1.4.3. Personnel Monitoring

TLD badges and TLD finger rings are assigned to all approved reactor personnel. In addition, self reading pocket dosimeters and dose rate instruments are used to administratively keep occupational exposures below regulator limits in 10 CFR 20. Students and visitors are provided self reading pocket dosimeters.

11.1.5 Radiation Exposure Control and Dosimetry

The reactor staff always operates under the principles of ALARA, and personnel exposures are maintained at levels that are well below those required by 10 CFR 20. Personnel exposures for the recent operating history are detailed in Table 11-1 below.

Table 11-1: PUR-1 Personnel Exposures for 2003-2007

	Number of Individuals				
Exposure Range (rem)	**2003**	**2004**	**2005**	**2006**	**2007**
0	0	0	2	3	1
< 0.1	1	1	1	0	2
0.10-0.25	0	0	0	0	0
0.25-0.75	0	0	0	0	0
0.75-1.0	0	0	0	0	0
> 1.0	0	0	0	0	0

11.1.6 Contamination Control

Wipe tests of exposed surfaces of the reactor room are made monthly. Water samples are taken and counted monthly. All samples and material removed from the reactor are checked for levels of activity and wipe tests made for loose contamination.

11.1.7 Environmental Monitoring

11.1.7.1. Airborne Effluent

Argon-41 – Argon-31 is produced by thermal neutron activation of Argon-40 in the air. No detectable traces of Ar-41 from air dissolved in the water or in the isotope irradiation tubes has been observed. No detectable tritium has been observed in the pool water. Even at the design power levels of 10 kW the neutron flux is too low to produce detectable quantities.

The main possible source of nitrogen-16 is from the fast neutron interaction with oxygen in the pool water. The nitrogen must then diffuse to the surface of the pool before it is released to the atmosphere. In normal operation, no strong currents are established in the reactor pool and with the short half-life (7.14 seconds), the nitrogen decays before reaching the surface. No nitrogen-16 has been observed in the reactor room.

A continuous air monitor (CAM), which utilizes a GM tube as a detector, is in operation in the reactor room to indicate long term levels of radiation and to monitor any radioactive particulates released to the air in the room. No airborne radioactivity has been measured at PUR-1. No airborne effluents have been released from the PUR-1 facility.

11.1.7.2. Perimeter monitoring

TLD badges are located at the boundaries of the facility, and are checked for exposure every other month. The doses to these badges are at background levels. Therefore, facility operation has not resulted in public exposures greater than those specified in 10 CFR 20.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

Only very limited contaminated materials are generated by PUR-1. Any contaminated material is disposed of under the Purdue University Broadscope license, as described previously.

11.2.2 Radioactive Waste Controls

Every effort is made to limit the generation of radioactive waste. Very limited amounts of radioactive materials have been produced at PUR-1 in the 46 years of operation.

11.2.3 Release of Radioactive Waste

Disposal of radioactive material is under the Broadscope license, as described previously. No wastes have been released to the environment in an uncontrolled manner.

11.3 Conclusions

12 CONDUCT OF OPERATIONS

12.1 Organization

12.1.1 Structure

The reactor facility is an integral part of the Schools of engineering at Purdue University as shown in Figure 6.1 The reactor supervisor has direct responsibility for the operation of the PUR-I. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, Technical specifications, and other applicable regulations.

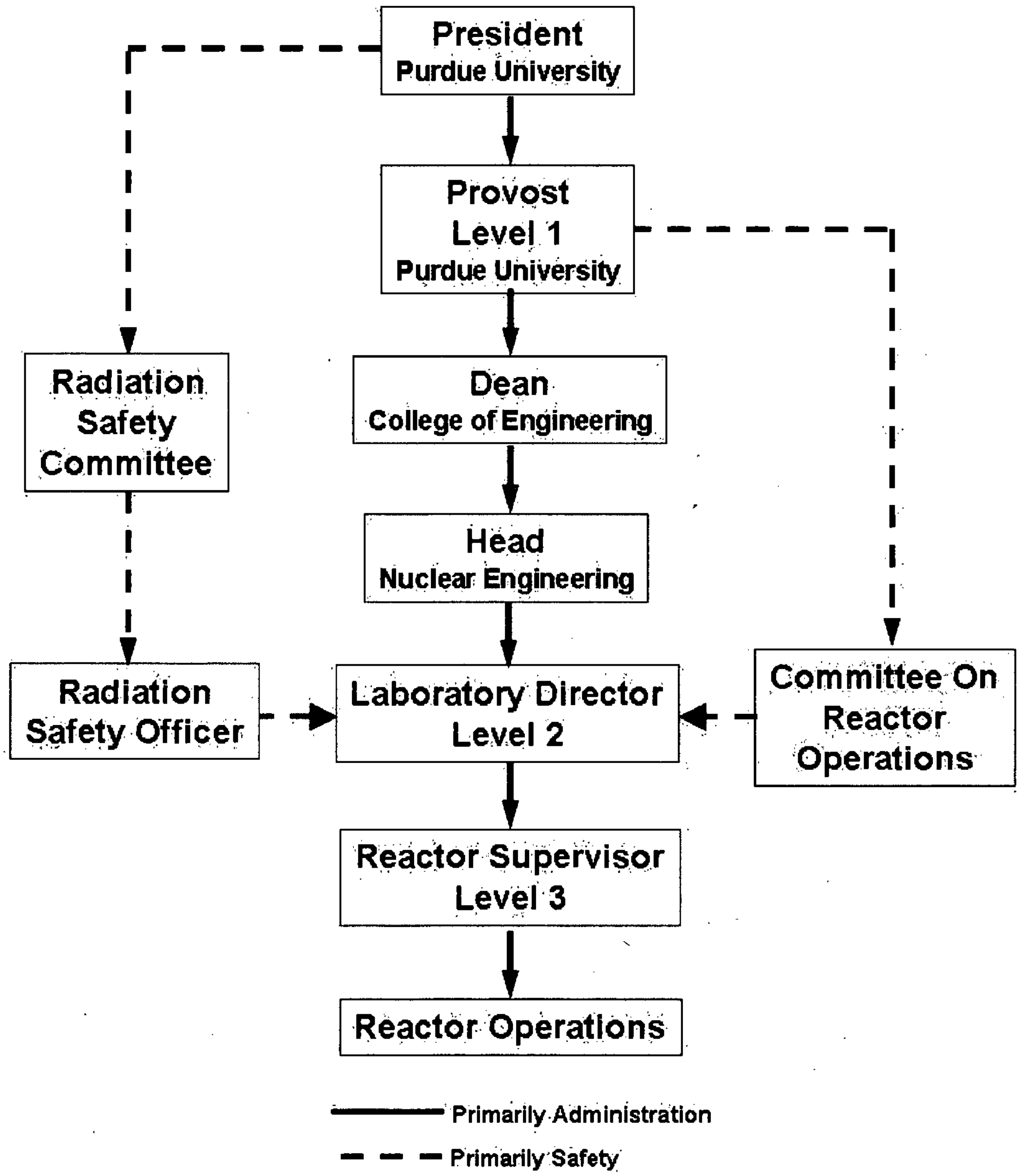


Figure 12-1

12.1.2 Responsibility

The Provost (Level 1) is the individual responsible for the facility's licenses or charter. The Laboratory Director (Level 2) or the designated alternate is responsible for overall reactor facility operation. The Reactor Supervisor (Level 3) shall be responsible for the day-to-day safe operation of the PUR-1. The Reactor Supervisor is be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, including the technical specifications and other applicable regulations.

In all matters pertaining to the operation of the reactor and the administrative aspects of these technical specifications, the Laboratory Director (Level 2) [or the Reactor Supervisor (Level 3) in the absence of the Laboratory Director] shall report to and be directly responsible to the Head of the School of Nuclear Engineering, and the Provost (Level 1). In all matters pertaining to radiation safety they will work with the Radiation Safety Officer.

12.1.3 Staffing, Selection and Training of Personnel

At the time of appointment to the position, the Level 1 Licensee (Insert Level One Title) shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the reactor facility.

At the time of appointment to the position, the Laboratory Director (Level 2) shall have a minimum of five years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job-related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the six years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility-specific training based upon a comparison of the individual's background and capabilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required. The Laboratory Director shall possess a valid Senior Operator License, and meet the certifications requirements of the licensing agency.

Reactor Supervisor (Level 3) - At the time of appointment to the active position, the reactor supervisor shall have a minimum of five years of nuclear experience. He shall have a baccalaureate degree or equivalent experience in an engineering or other scientific field. The degree may fulfill four years of experience on a one-for-one basis. The reactor supervisor shall possess a valid Senior Operator License. During periods when the Reactor Supervisor is absent, these responsibilities may be delegated to a Senior Reactor Operator (Level 4)

Licensed Senior Operator (Level 4) - At the time of appointment to the active position, a senior operator shall have a minimum of a high school diploma or equivalent and should have four years of nuclear experience. A maximum of two years of experience may be fulfilled by related academic or technical training on a one-for-one time basis. He shall hold a valid NRC Senior Reactor Operator's license.

Licensed Operator - At the time of appointment to the active position, an operator shall have a high school diploma or equivalent. He shall hold a valid NRC Reactor Operator's license.

Operator Trainee - An operator trainee shall have all the qualifications to become a licensed operator except for possessing an operator's license.

12.1.4 Radiation Safety

The reactor The reactor facility is an integral part of the Schools of engineering at Purdue University as shown in Figure 6.1 The reactor supervisor has direct responsibility for the operation of the PUR-I. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, Technical specifications, and other applicable regulations.

In all matters pertaining to the administrative aspects of the operation of the reactor, the reactor supervisor reports directly to the Head of the School of Nuclear Engineering or his designated alternate. Financial budgets for the operation of the reactor are handled through the School of Nuclear Engineering.

In all matters pertaining to radiation safety the reactor supervisor is responsible to the Radiological Control committee, usually through the Radiological Control Officer. The Radiological Control Committee was established by the President of the University under an executive memorandum A-50. The duties of the committee, revised under executive memorandum B-14, include responsibility, from the standpoint of safety, for all University programs involving radioactivity or producing radiation. The committee determines the policies and reviews all applications for use of radioactive materials and radiation on the Purdue campus. The Radiological Control Program is administered by the Radiation Control Officer and his staff. The Radiological Control Committee also have several subcommittees reporting directly to it, including the Committee On Reactor Operations (CORO).

Qualifications for the reactor supervisor, reactor staff, and Radiological Control Officers are established in Section 6.1 of the Technical Specifications. The duties of the reactor staff are also specified in Section 6.1 of the Technical Specifications.

12.2 Review and Audit Activities

Review of the facility activities shall be performed by the CORO. Audits of the facility activities shall be performed under the cognizance of the CORO but in no case by the personnel responsible for the item audited. Individual audits may be performed by an individual who needs not be an identified CORO member. These audits shall examine the operating records, procedures, retraining of the facility staff, results of actions taken to correct deficiencies, facility emergency plan, and the facility security plan at intervals designated in the Section 6.2 of the Technical Specifications.

12.3 Procedures

Written procedures for all reactor operations are prepared by the reactor staff and reviewed by the Committee On Reactor Operations (CORO). Those proposed changes to procedures, equipment or systems that change the original intent or use and/or are non-conservative or those that involve an unreviewed safety questions as defined in Section 50.59, 10 CFR are reviewed and approved by CORO before being implemented.

12.4 Required Actions

In the event of a Safety Limit Violation, the following actions shall be taken:

a. The reactor will be shut down immediately and reactor operation will not be resumed without authorization by the Commission.

b. The Safety Limit Violation shall be reported to the Director of the appropriate NRC Office of Inspection and Enforcement (or designee), the Laboratory Director and to the CORO not later than the next work day.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the CORO. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CORO and the Reactor Supervisor within 14 days of the violation, in support of a request to the Commission for authorization to resume operations.

12.5 Reports

12.5.1 Annual Operating Reports

A report covering the operations for the previous year is submitted to the director of the appropriate NRC office by March 31st of each year. It includes:

1. Changes in Plan design and operation
2. Power generation
3. Unscheduled shutdowns
4. Maintenance
5. Changes, tests and experiments

6. Radioactive effluent releases.

12.5.2 Non-Routine Reports

In the event of a reportable occurrence, notification shall be made within 24 hours by telephone and/or telegraph to the Director of Regional Regulatory Operations Office, followed by a written report within ten days. The written report on these Abnormal occurrences, and to the extent possible the preliminary telephone and/or telegraph notification shall:

1. Describe, analyze and evaluate safety implications,
2. Outline the measures taken to assure that the cause of the condition is determined,
3. Indicate that corrective action taken to prevent repetition of the occurrence and/or similar occurrences involving similar components or systems,
4. Evaluate the safety implications of the incident in light of cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

12.5.3 Unusual Events

A written report shall be forwarded within 30 days to the director of the appropriate Regulatory Operations Office in the event of:

1. Discovery of any substantial errors in the transient or accident analysis or in the methods used for such an analysis, as described in the Hazards Summary Report on the bases for the Technical Specifications.
2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications.
3. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

12.6 Records

Records to be maintained for the life of the facility are stored in filing cabinets in the Reactor Supervisor's office, or in storage cabinets in facility spaces. Records to be kept for five years are also kept in the Reactor Supervisor's office. Recent documentation also has electronic versions kept on private network servers. Security associated documentation is not stored on any servers, and is kept in a secure repository.

12.7 Emergency Planning

The PUR-I reactor Emergency Plan (EP) includes the guidelines, policy, and organization required to mitigate the consequences of an emergency. Specific implementation procedures are provided for each type of emergency in the standard operating procedures for the PUR-I reactor. A revised EP has been approved by the NRC, and is maintained as a separate document from this Safety Analysis Report.

12.8 Security Planning

There is a Physical Security Plan (PSP) for the PUR-I reactor facility which describes the physical protection system and the security organization which provides protection against radiological sabotage and detection of theft of special nuclear material from the facility and related laboratories. The physical security plan was submitted as Amendment 4 and Amendment 5 of the reactor license, and is on file with the NRC. It is withheld from public disclosure pursuant to 10 CFR 2.790 (d).

An audit of the PSP is performed biennially, which is overseen by the Committee on Reactor Operations, and the PSP is revised as necessary when changes are recommended in the audit process, or as other external conditions change. A revised PSP has been approved by the NRC, and is maintained as a separate document from this Safety Analysis Report.

12.9 Quality Assurance

The Laboratory Director and the Reactor Supervisor are responsible for maintaining quality assurance for continued safe operation of the PUR-1 reactor. This includes operator performance monitoring through the requalification and training program, continuous evaluation of existing operating procedures, development of new operating procedures, testing and calibration of reactor equipment, repair of existing equipment, and installation of new equipment. The Committee on Reactor Operations (CORO) provides review, audit and oversight functions as defined in the Technical Specifications for these activities. Checklists and other appropriate documentation, including operations logbooks, are kept to maintain a record of these activities.

12.10 Operator Training and Requalification

The PUR – I reactor facility has an NRC-approved operator requalification program that all licensed reactor operators and senor reactor operators must complete as a condition for renewal of their licenses. Persons who are preparing to take the NRC operator's licensing examination participate in essentially the same training program as well as receive extensive 'hands on' reactor operations training at the console. All licensed operators at the PUR-I participate in the program and must satisfactorily complete this program during each license renewal period. Each licensed operator or senior operator includes in his/her license renewal application a statement that he/she has satisfactorily completed the requirements of the requalification program. The requalification program is divided into three major areas which are designed to provide assurance that all operators maintain competence in all aspects of the licensed activities. The three areas are as follows:

1. Lectures followed by examination of various parts of the reactor operations

Technical Specifications, emergency plans, and security plans. Special lectures or makeup studies are used to retrain those operators who demonstrate deficiencies in any part of the examinations.

2. An annual written examination is used to verify the operator's overall knowledge level of reactor operations.
3. An annual evaluation of the operator's performance on the reactor console to actual and/or simulated plant conditions.

At the present time there are only two licensed operators for the facility, both are senior operators and both annually teach courses in reactor operation and the facility. Therefore, in a letter dated March 23, 1982, the facility has received permission to suspend the retaining program for the staff. It will again be reinstated at such a time as the reactor staff is increased.

13 ACCIDENT ANALYSES

In this chapter, details of the analysis of various accident scenarios are presented. The results of some of these analyses validate the safety system settings established in the Technical Specifications for the PUR-I. The potential effects of the accidents on the health and safety of the staff and public are analyzed.

13.1 Accident-Initiating Events and Scenarios

13.1.1 Maximum Hypothetical Accident

In the scenario of this accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I.

13.1.2 Insertion of Excess Reactivity

This accident scenario characterizes the reactor response to an insertion of the maximum allowable excess reactivity for the PUR-1 reactor, 0.6% Δk/k. This transient is examined in four different scenarios, a fast (step) and slow insertion, and each of these with and without scram. In the with scram case, it is assumed that the first scram signal (likely from short period) fails, and the reactor trips on power. The power scram is set for 120% power (12 kW), and it is also assumed that there is a 50% uncertainty in power measurement (actual scram happens at 18 kW), and that the scram signal is delayed by 0.1s before power is cut to the electromagnets that hold the shim-safety rods at their respective heights. It is also assumed that only the least reactive shim-safety rod, SS-2, is able to be inserted into the core (stuck rod scenario).

For the no scram case both rods are stuck at their fully raised position out of the core, which would be similar to the reactor being critical on one shim-safety rod, and that rod being ejected from the core (not a credible accident).

The results for these accidents are summarized in

Table 13-1.

Table 13-1: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	Reactivity Inserted	P_0 (kW)	P_{max} (kW)	Time of Peak Power (s)	$T_{clad,max}$ (°C) at t=0	at Peak Power	Maximum
YES	0.6% Δk/k step	10	40.3	0.187	49.94	49.73	49.94
YES	0.6% Δk/k over 10 s	10	18.4	7.83	49.94	42.30	49.98
NO	0.6% Δk/k step	10	2388	672	49.94	133.08	133.08
NO	0.6% Δk/k over 10 s	10	2388	680	49.94	133.08	133.08

13.1.3 Loss of Coolant

The reactor pool is designed to prevent the possibility of an unintentional drainage. It is constructed of steel and set in a second steel tank with the interstitial region filled with sand. [redacted] Therefore, a sudden loss of coolant is considered to be extremely remote. If the pool drained instantaneously, while the reactor was operating, the loss of water (moderator) would shut the reactor down.

The most severe problem identified in this accident scenario if the removal of decay heat during and after loss of coolant. There is no danger of significant fuel overheating as long as the core stays immersed and heat can be removed by the water. If the core were to become uncovered, heat transfer would occur by natural convection of ambient air. For this case, the amount of heat removed is proportional to the cladding temperature. Decay heat generation after reactor shutdown is shown in Figure 7.1. According to this Figure, the decay power of the PUR-I immediately after the shutdown from full power (1 kW) is about 63 watts. The decay power rapidly decreases as indicated in Figure 7.1, being about 35 watts after 1 minute of cooling. At these power levels, no heating problem exists.

If any accident which is reasonably conceivable, the leakage of water from the reactor pool is expected to be rather slow. In such a case the radiation area monitor mounted directly above the core would detect any additional radiation coming from the core due to the decreasing pool water level. The pool water level is checked during daily routine operations. It is concluded, that a slow leak of pool water would be discovered early and specific actions could be taken to mitigate it consequences.

It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

13.1.4 Loss of Coolant Flow

The Purdue University Reactor is cooled by natural convection, with peak flow rates at the onset of nucleate boiling (ONB, 98.11kW) of 5.41 cm/s, and nominal flow rates at 10 kW of 1.65 cm/s. The only consideration for a loss of coolant flow (LOCF) scenario would be a blockage in a channel. The power density of PUR-1 at 10 kW is such that the individual plate power is very low, and conduction of the heat to adjacent channels would occur, and plate temperatures would remain well below ONB temperatures of 112°C, which is still more than 400°C below the safety limit for the fuel. Therefore this accident is not analyzed.

13.1.5 Mishandling or Malfunction of Fuel

Fuel element maneuvers are always conducted in the reactor pool. They are removed from the core and moved into the storage space, one at a time, using a hand-held fuel handling tool. Annually a fuel element is removed from the pool for inspection. A fuel element weighs about 4.5 kg (9.9 lb) in air and only about 2.8 kg (6.2 lb) in water. Therefore, even if a fuel element should fall from the handling tool during its transfer it is not heavy enough to cause any considerable damage. The most severe consequence likely to occur would be some denting of the end fittings since the fuel element, being an elongated object, would tend to fall in water in a rather upright position.

The PUR-I Standard Operating Procedures define administrative steps which are intended to prevent a fuel handling mishap. They are:

1. All fuel handling is done in accordance with written procedures.
2. Loading operations are done by qualified personnel under direct supervision of a Senior Operator.
3. [redacted] It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

13.1.6 Experiment Malfunction

In this section an analysis is performed to assess the hazard associated with the failure of an experiment in which fissile material has been irradiated in the reactor. In the scenario of this accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I. This accident is analyzed in Section 13.2.1.

The flooding of the 12.7 cm drop tube when placed in position G6 as described in the SER NUREG-1283 resulted in a reactivity insertion of 0.246% $\Delta k/k$, which is below the insertion resulting from the failure of a moveable experiment examined in the conversion proposal. This accident scenario is within the envelope of the examined accident cases, and therefore was not analyzed.

13.1.7 Loss of Normal Electrical Power

The loss of normal electric power at PUR-1 will shutdown the reactor by simulating a scram with a loss of power to the electromagnets that hoist the shim-safety rods. This action will shutdown the reactor from any conceivable operating condition. Since the reactor is cooled by natural convection, there are no shutdown and decay heat issues. There is adequate heat capacity in the reactor pool to address shutdown and decay heat loads. Therefore, this accident scenario is not addressed.

13.1.8 External Events

13.1.8.1. <u>Fire or Explosion</u>

The reactor building is a steel frame structure with concrete block and brick construction. Additionally, [redacted] and the reactor itself below the floor level in the reactor room. The materials surrounding the reactor core are concrete, steel and earth. There is limited combustible material in the reactor room, but it is virtually impossible to exclude all burnable materials. Therefore, some small fire possibility exists. There is portable fire-fighting equipment available, and PUR-1 staff are trained in use of that equipment. Purdue Fire Department is located on campus, and are available for assistance 24 hours a day.

In general, personnel judgement is used in deciding the response to a a fire. If the fire is small and can be easily controlled, little action is necessary to safely shutdown the reactor and control the fire. If the fire is of larger magnitude, the reactor will be secured and outside assistance obtained.

Because of the large volume of water surrounding the core, damage to the fuel is not likely even in a severe fire scenario. Even if the fire involved the control systems, power to those circuits could be cut at the wall of the reactor room, and the control rods would drop and safely shut down the reactor.

Explosions external to the building could affect the reactor and its systems, however explosions external to the reactor pool will have little effect on the reactor due to its location and the construction of the pool. The amount of water covering the core will aid in the reduction of risk to pyrotechnic devices thrown into the pool, either serving to quench the devices, or absorb most of the energy from the detonation.

With these facts, and the location of the reactor in the pool, damage from fire or explosion is not likely.

13.1.8.2. <u>Acts of Sabotage</u>

[redacted] There is a security plan in place, and security systems designed to protect all of the controlled areas in the reactor facility. Purdue Police, a fully functioning police department with arrest authority, is available for assistance, and is staffed 24 hours a day. The PUR-1 Security Plan is reviewed and approved by the U.S. NRC, and provides for in-house procedures to ensure facility protection.

Riots or acts of civil disobedience directed against the reactor facility would be recognizable is sufficient time to ensure safe shutdown of the reactor. Personnel at the reactor could quickly alert local law enforcement, and quickly come under their protection. Then access to the area could be restricted until it is safe to resume normal operations.

13.2 Accident Analysis and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (Failure of a Fueled Experiment)

In this analysis the consequences of a failed experiment generating 1 W were studied. The capsule containing the experiment is assumed to break as it is removed from the reactor. The fission products expected to become airborne are the noble gases and elemental iodine. Other fission products and actinides are not volatile at the temperature 9which is essentially at room temperature) at which the experiment would be performed. The amount of noble gases and radioiodine is assumed to be that specified in Ref. 1, i.e. 100 % of the noble gases and 25 % of the iodine inventory. If the experiment were to break in the reactor pool a credit for the absorption of iodine in water can be taken. However, this is not considered in this analysis.

A conservative assumption that the irradiation time was infinite was made in the analysis. There fore, the fission inventories used in the analysis for some long-lived radionuclides, e.g.Kr-85 or even I-131, are overly conservative. Furthermore, it was assumed that the fission products are instantaneously released and uniformly distributed in the Reactor Room air. The free volume of the Reactor Room is approximately 424 m^3.

The external dose rate (in mrem/hr) due to γ and β-radiation was calculated using the relationships given in Lewis[1]

$$\dot{D}_\gamma = 9.43 \times 10^{11} \times X \times \overline{E}_\gamma$$

where X = radionuclide concentration (Ci/cm^3). $\overline{E}_\gamma$ = average γ-energy per disintegration (MeV) and

$$\dot{D}_\beta = 9.43 \times 10^{11} \times X \times \overline{E}_\beta \qquad (0.1)$$

Where $\overline{E}_\beta$ = the average β-energy per disintegration (MeV). The dose rate, $\dot{D}_\beta$ represents the skin dose. The dose rate to the thyroid (in rem/hr) due to inhalation of the radioiodines is give by

$$D_T = DCF \times B \times X$$

where DCF = dose-conversion factor for the thyroid (rem/Ci), B = breathing rate (cm^3/hr) and X = radioiodine concentration (Ci/cm^3). The standard breathing rate recommended is 1.25×10^6 cm^3/hr.

The calculated saturation activity for each respective radioisotope and its concentration is the Reactor Room after experiment failure is shown in Table 7.5 for an experiment of 1 W. Also shown in this table are the calculated dose rates for the whole-body, skin and thyroid.

Table 13-2: Dose Rates in the Reactor Room from a Failed Fuel Experiment (Power = 1 Watt)

Isotope	As (Ci)	X (Ci/cm3)	E-gamma (MeV)	E-beta (MeV)	DCF (rem/Ci)	DR-gamma (mrem/hr)	DR-beta (mrem/hr)	DR-thyroid (rem/hr)
I-131	2.66E-02	1.57E-11	3.71E-01	1.97E-01	1.00E+06	5.49E+00	2.55E+00	1.96E+01
I-132	3.75E-02	2.21E-11	2.40E+00	4.48E-01	6.60E+03	5.00E+01	8.16E+00	1.82E-01
I-133	5.31E-02	3.13E-11	4.77E-01	4.23E-01	1.80E+05	1.41E+01	1.09E+01	7.04E+00
I-134	5.94E-02	3.50E-11	1.94E+00	4.55E-01	1.10E+03	6.41E+01	1.31E+01	4.82E-02
I-135	4.96E-02	2.77E-11	1.78E+00	3.08E-01	4.40E+04	4.64E+01	7.02E+00	1.52E+00
Kr-83m	5.90E-03	1.39E-11	2.60E-03	1.03E-02		3.41E-02	1.18E-01	
Kr-85m	1.27E-02	3.00E-11	1.51E-01	2.23E-01		4.27E+00	5.50E+00	
Kr-85	2.53E-03	5.97E-12	2.11E-03	2.13E-01		1.19E-02	1.10E+00	
Kr-87	2.00E-02	4.27E-11	1.37E+00	1.05E+00		6.09E+01	4.08E+00	
Kr-88	3.12E-02	7.36E-11	1.74E+00	3.41E-01		1.21E+02	2.07E+01	
Kr-89	3.96E-02	9.34E-11	1.60E+00	1.33E+00		1.41E+02	1.02E+02	
Xe-131m	2.53E-04	5.97E-13	2.00E-02	1.40E-01		1.13E-02	6.88E-02	
Xe-133m	1.35E-03	3.18E-12	3.26E-01	1.55E-01		9.79E-01	4.07E-01	
Xe-133	5.48E-02	1.29E-10	3.00E-02	1.46E-01		3.66E+00	1.55E+01	
Xe-135m	1.77E-02	4.17E-11	4.22E-01	9.74E-02		1.66E+01	3.35E+00	
Xe-135	5.23E-02	1.23E-10	2.46E-01	3.22E-01		2.86E+01	3.27E+01	
Xe-137	5.31E-02	1.25E-10	1.50E-01	1.37E+00		1.77E+01	1.41E+02	
Xe-138	5.70E-02	1.31E-10	1.10E+00	8.00E-01		1.36E+02	8.66E+01	
						7.11E+02	**4.55E+02**	**2.84E+01**

With a γ-dose rate in the Reactor Room as high as 50 mrem/hr any one of the radiation ara monitors would cause an automatic reactor shutdown and audible and visual alarms in the control room. From the past experience, it is know that the reactor building can be evacuated within 1.5 minutes. Therefore, it is assumed in the following analysis that the exposure time to the members of the reactor staff is 1.5 minutes. The resulting radiation doses are: whole-body 22.3 mrem, skin does 13.4 mrem, and the thyroid dose 1.43 mrem.

The radioactive material content, including fission products, of any singly encapsulated experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR 20. This dose limit applies to persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

The radioactive material content, including fission products, of any doubly encapsulated experiment or vented experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 Rem to the whole body or 1.5 Rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 Rem to the whole body or 30 Rem to the thyroid.

This radiation exposure approaches the limits established in the Technical specifications, Sec 3.5.f for a singly encapsulated experiment. This experiment corresponds to the irradiation of 1.1 gm of U-235 in the mid-plane of the isotope irradiation tube located in position F6.

For the radiation calculations outside of the reactor building it was assumed that all fission products released in the reactor building would leak out within 24 hours. Since the reactor room does not have any windows and only a few doors and emergency procedures calls for turning off the air exhaust system, the assumption about the leak rate is considered to be reasonable. Another conservative assumption was made in that no radioactive decay and hence no decrease in the source strength was taken into account while calculation the dose rates outside the reactor building. The radionuclide concentration at a distance of 100 m from the release point was calculated using the atmospheric dispersion faction recommended in Ref.. 1;

$$\frac{X}{Q'} = \frac{1}{\pi \bar{u} \sigma_Y \sigma_Z}$$

Where X is the concentration of radioactive material (Ci/m^3), Q' = source rate (Ci/s), $\bar{u}$ = average windspeed (m/s), σ_Y = lateral plume spread (m) and σ_Z = vertical plume spread (m).

In the above expression for the atmospheric dispersion factor no credit was taken for so-called building wake effects and horizontal plume meandering, both of which help in spreading the radioactive plume. Also, no credit was taken for the fact that air from the basement area is exhausted at a minimum height of 50 feet. Using an average wind speed of 1 m/s and Pasquill type F atmospheric conditions the dispersion factor at 100 m is calculated to be 1.78 x 10^{-4} s/m^3. Actually, the average wind speed is about 3.4 m/s. Therefore, the results of this analysis are conservative at least by a factor of 3).

Table 13-3:Dose Rates at 100 meters.

Isotope	DR-gamma (mrem/hr)	DR-beta (mrem/hr)	DR-thyroid (rem/hr)
I-131	1.95E-03	9.06E-04	6.98E-03
I-132	1.78E-02	2.91E-03	6.49E-05
I-133	5.01E-03	3.88E-03	2.51E-03
I-134	2.28E-02	4.67E-03	1.71E-05
I-135	1.65E-02	2.50E-03	5.41E-04
Kr-83m	1.21E-05	4.20E-05	
Kr-85m	1.52E-03	1.96E-03	
Kr-85	4.23E-06	3.90E-04	
Kr-87	2.17E-02	1.45E-02	

Kr-88	4.30E-06	7.36E-03	
Kr-89	5.02E-02	3.64E-02	
Xe-131m	4.01E-06	2.45E-05	
Xe-133m	3.48E-04	1.45E-04	
Xe-133	1.30E-03	5.54E-03	
Xe-135m	5.91E-03	1.19E-03	
Xe-135	1.02E-02	1.17E-03	
Xe-137	**6.31E-03**	5.03E-02	
Xe-138	4.85E-02	3.08E-02	
	2.10E-01	1.65E-01	1.01E-02

Calculated dose rates at 100 m for an experiment power of 1 W are shown in Table 7.6. If it is assumed that an individual is located at this point for 2 hours following the fission product release from a postulated experiment failure then his/her resulting radiation does to the whole body would be .51 mrem and to the thyroid .02 rem. These doses are only fractions (about 1%) of those which are referred to in 10 CFR 100 in conjunction with the determination of an exclusion area.

It is concluded that experiments using fissile material can be irradiated at the PUR-I within the power limits.

13.2.2 Insertion of Maximum Allowed Excess Reactivity

The analyses of this transient utilizes the reactor physics and reactivity coefficients determined by MCNP5 as described in Chapter 4, and thermal-hydraulic parameters determined by NATCON as described in Chapter 4, and the PARET/ANL[2] code. For this accident, the initial power before the transient was 10 kW.

The original PARET code has been adapted by the Reduced Enrichment for Research and Test Reactors (RERTR) Program to provide transient and thermal-hydraulics analysis for research and test reactors with both plate and pin-type fuel assemblies. The PARET/ANL version of the code has been subjected to extensive comparisons with the SPERT I and SPERT II (light and heavy water) experiments. These comparisons were quite favorable for a wide range of transients up to and including melting of the clad. Revisions of the code include new and more appropriate heat transfer, departure from nucleate boiling (DNB) and flow instability correlations, improved edits, reactor trips, control insertion model, a decay heat power model, and a loss of flow model.

The rapid insertion of the maximum worth of reactivity excess for PUR-1 (0.6% Δk/k) as specified by the Technical Specifications was evaluated. An assumption was made that the period trip (7s) failed, and scram was initiated on the power trip at 12 kW. Since the assumed measurement uncertainty on core power is 50%, a core power trip setting of 18 kW was utilized in the accident calculation. A delay of 0.1 seconds from the sending of scram signal to beginning of control rod motion is assumed. Fuel/coolant channels that are representative of the hottest fuel plates (as identified in Chapter 4) were modeled in the PARET analyses.

Results of this transient are summarized in Table 13-4: Peak power and clad temperature for insertions of 0.6% Δk/k with scram.. The reactor power increases from 1 kW to the trip setting of 1.8 kW in less than 2.5 seconds. There is a negligible increase in the clad temperature as a result of this hypothetical accident. The safety limit is never in danger of being reached.

Table 13-4: Peak power and clad temperature for insertions of 0.6% Δk/k with scram.

SCRAM	Reactivity Inserted	P_0 (kW)	P_{max} (kW)	Time of Peak Power (s)	$T_{clad,max}$ (°C) at t=0	at Peak Power	Maximum
YES	0.6% Δk/k step	10	40.3	0.187	49.94	49.73	49.94
YES	0.6% Δk/k over 10 s	10	18.4	7.83	49.94	42.30	49.98

These results demonstrate the ability of the LSSS to protect the safety limit of fuel temperatures not to exceed 530°C. The maximum temperatures achieved in the fuel are well below temperature of incipient boiling as well. Therefore PUR-1 can maintain the fuel integrity during this accident scenario.

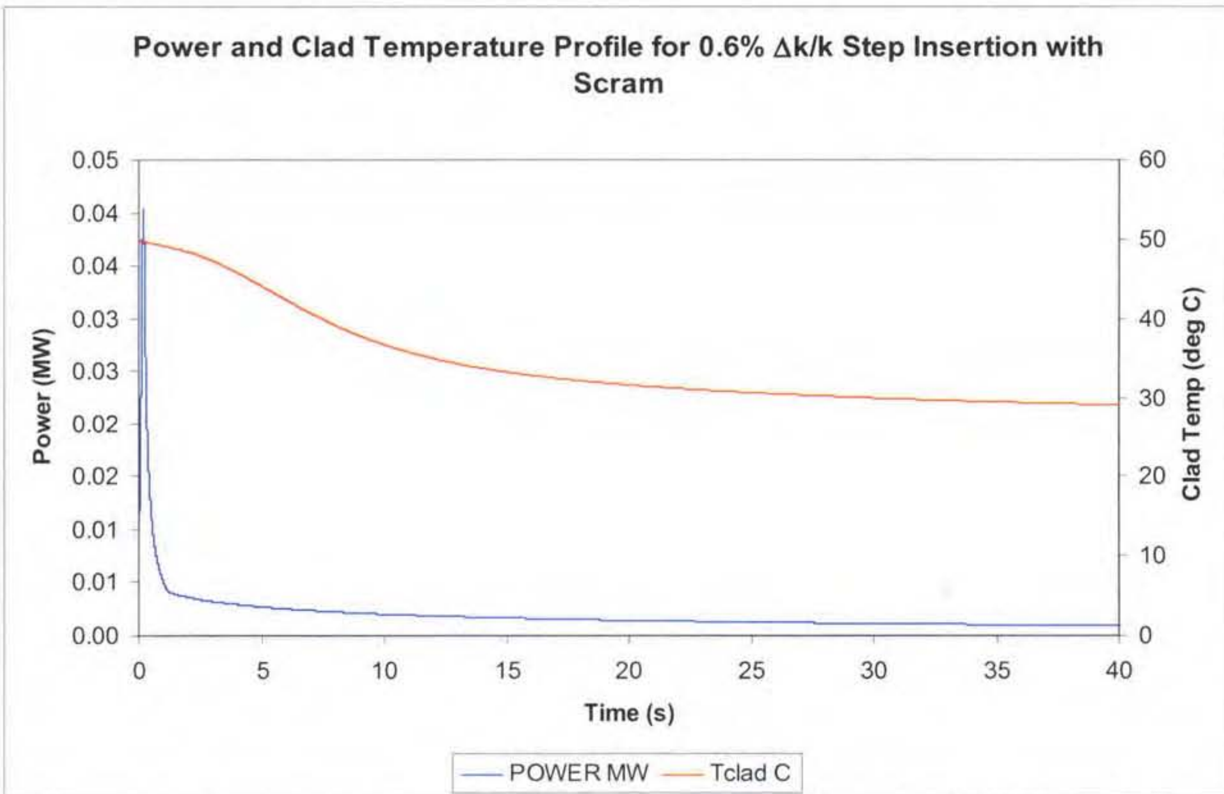


Figure 13-1: Power and Clad Temperatures for 0.6%Δk/k step insertion with scram.

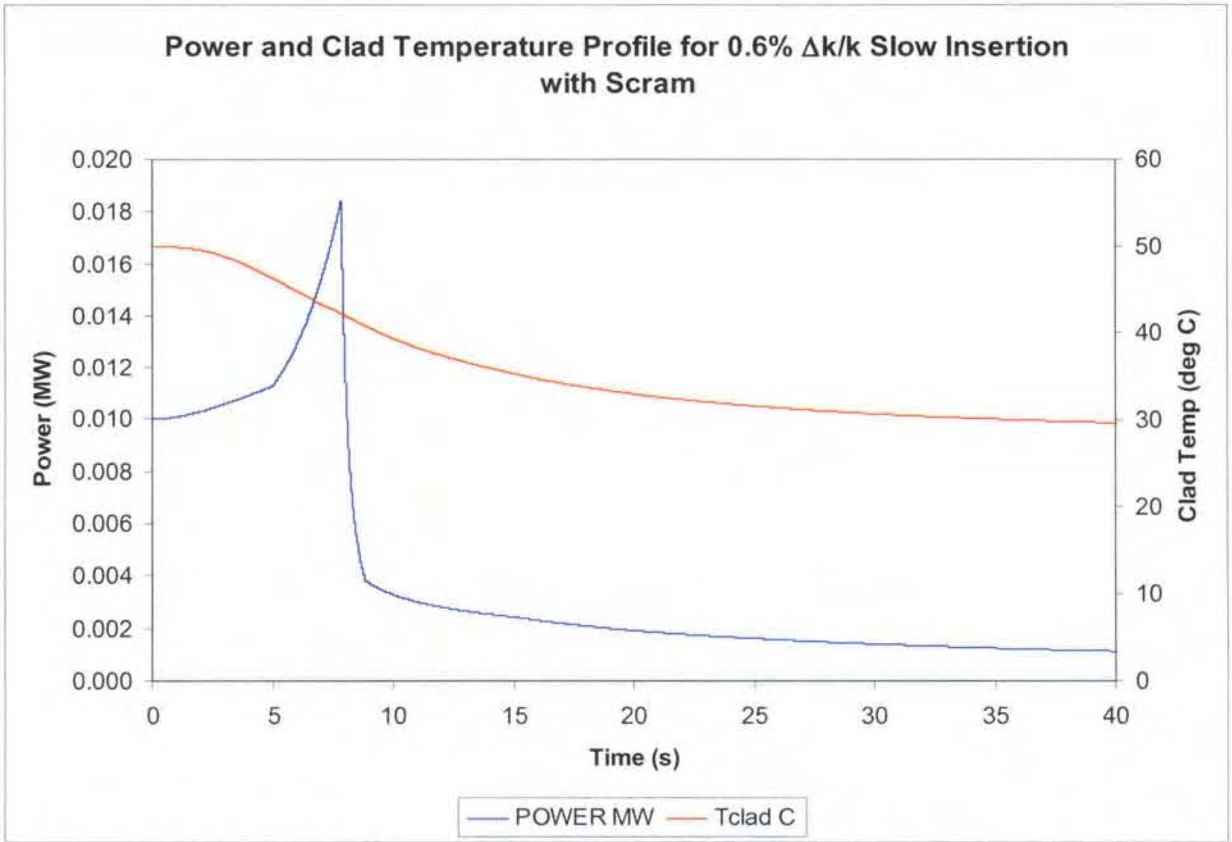


Figure 13-2:Power and Clad Temperatures for 0.6%Δk/k step insertion with scram.

13.2.2.1. Insertion Without Scram

While it is an extremely unlikely event that all reactor protective systems will fail, the analysis of the step and slow insertion of the maximum excess reactivity without scram was analyzed. In both cases it is assumed that both shim-safety rods are completely removed from the core, and are not available for insertion to shut the reactor down. For this accident, the initial power before the transient was 10 kW.

The reactivity coefficients determined in Chapter 4 act in this case to keep the reactor power from increasing without bounds. The power in both cases (step and slow insertion) levels off at about 2.38 MW, with the maximum clad temperature reaching 133.08°C. This temperature is still well below the safety limit of 530°C, so there is no danger of failure of the clad. It should also be noted that emergency boration is available to the operators, and this would act to shut down the reactor in the unlikely case of failure of the reactor protection system. The peak power and clad temperatures are summarized in Table 13-5, and the power and clad temperature history for the accident are shown in Figure 13-3 and Figure 13-4

Table 13-5: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	Reactivity Inserted	P_0 (kW)	P_{max} (kW)	Time of Peak Power (s)	$T_{clad,max}$ (°C)		
					at t=0	at Peak Power	Maximum
NO	0.6% Δk/k step	10	2388	672	49.94	133.08	133.08
NO	0.6% Δk/k over 10 s	10	2388	680	49.94	133.08	133.08

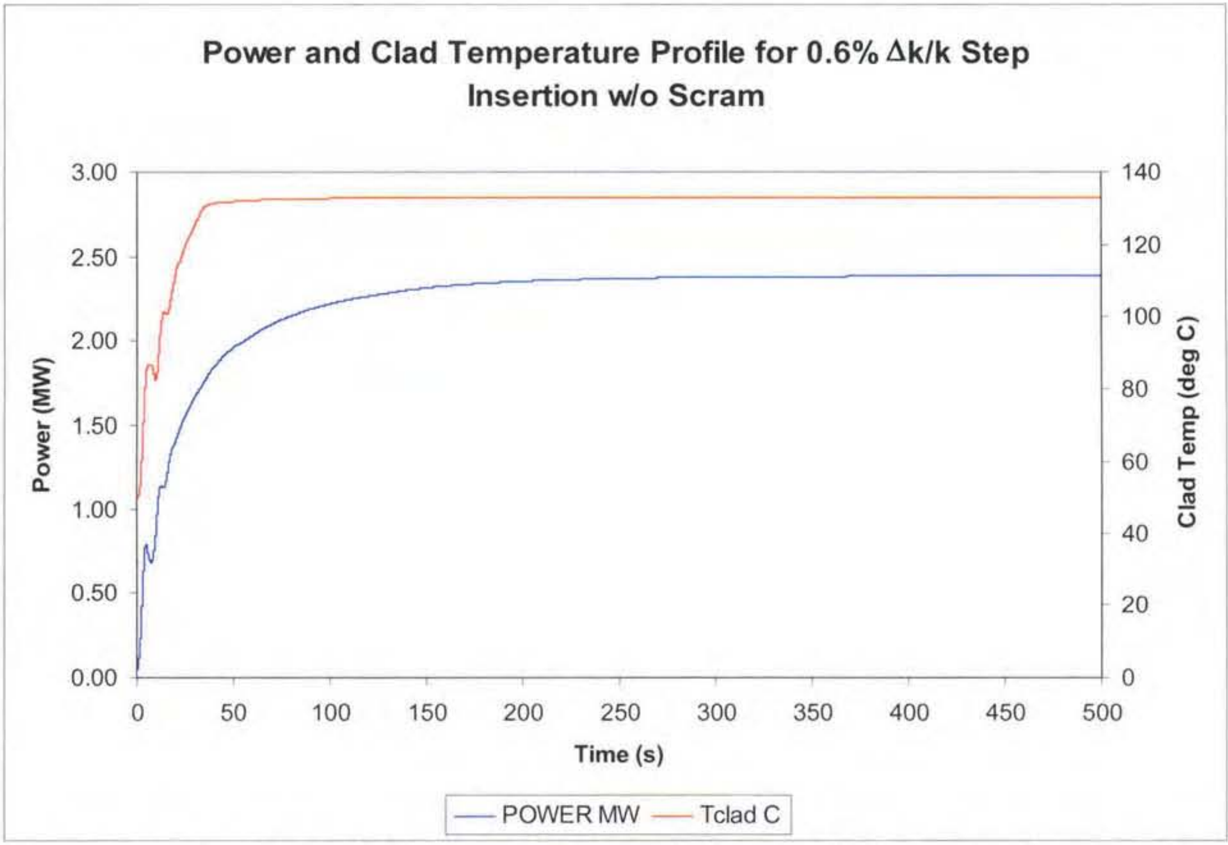


Figure 13-3: Power and Clad Temperatures for 0.6%Δk/k step insertion without scram.

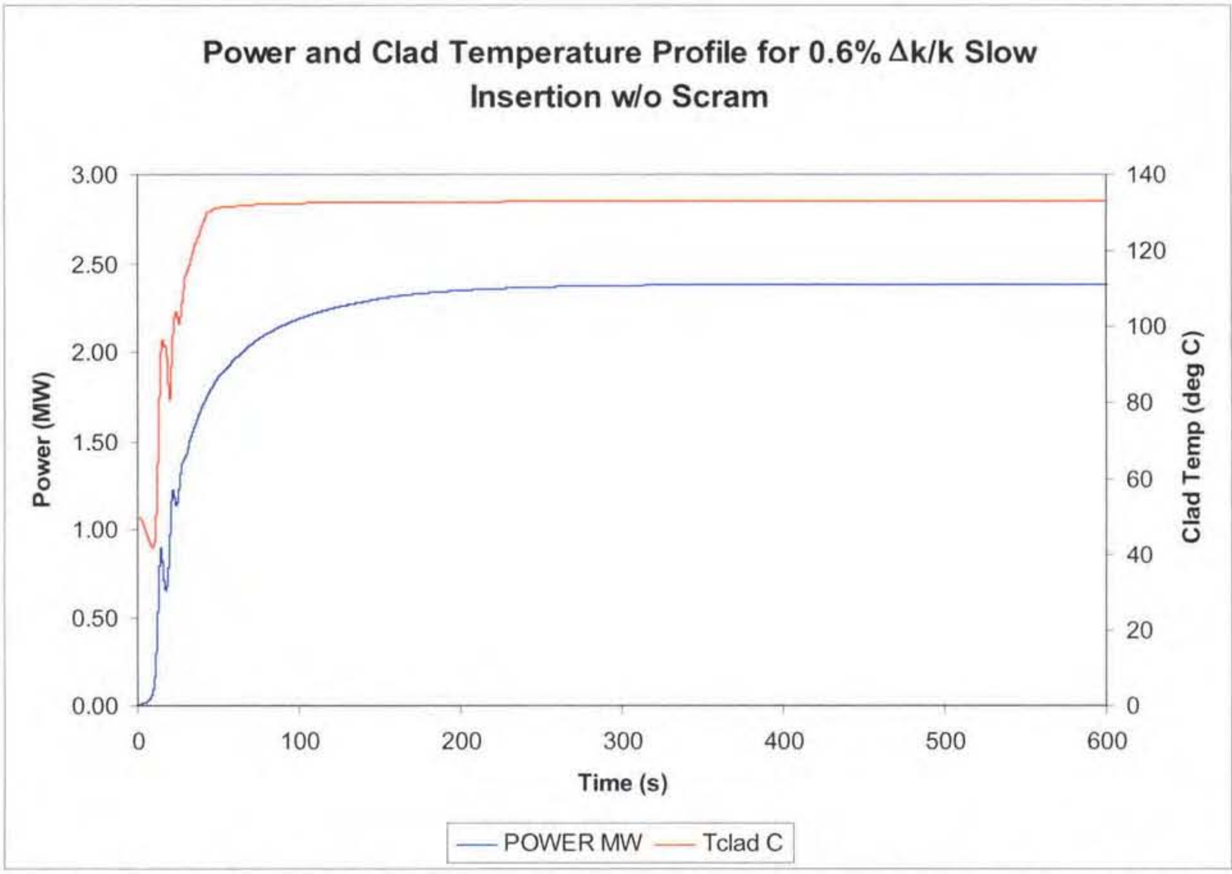


Figure 13-4: Power and Clad Temperatures for 0.6%Δk/k slow insertion without scram.

13.3 Summary and Conclusions

Evaluation of the accident scenarios detailed in this chapter lead to the conclusions that the PUR-1 reactor can operate safely and effectively at 10 kW continuously, with the reactor limiting safety system settings of 110% (11 kW) and 120% (12 kW) adequately protecting the reactor and its systems. Even in the very unlikely event of a complete failure of the reactor protection system and insertion of the maximum allowable reactivity, fuel integrity is maintained and there is little or no risk to the public.

13.4 References

[1] Lewis, E.E., "Nuclear Power Reactor Safety", John Wiley and Sons, Inc, (1977).

W. L. Woodruff and R. S. Smith, "A Users Guide for the ANL Version of the PARET Code, PARET/ANL (2001 Rev.)," Argonne National Laboratory (ANL), ANL/RERTR/TM-16, March 2001.

14 TECHNICAL SPECIFICATIONS

14.1 Summary Description of the Document

The NRC requires each applicant for and holder of a license to operate a non-power reactor to develop technical specifications that state the limits, operating conditions and other requirements for facility operation to protect the environment and health and safety of the facility staff and the public in accordance with 10 CFR 50.36. The technical specifications are typically derived from the facility descriptions and safety considerations contained in the SAR and represent a comprehensive envelope of safe operation.

The technical specifications for PUR-1 were first generated in 1978 to satisfy these requirements. They were made before the publication of the ANSI Standard 15.1, but have been modified over time to meet the evolving format recommendations of that standard.

The technical specifications of the PUR-1 are maintained as a separate document from this safety analysis report.

14.2 Administrative Control of Technical Specifications

The Committee on Reactor Operations (CORO) shall review any proposed changes to the Technical Specifications or licenses.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate a Non-Power Reactor

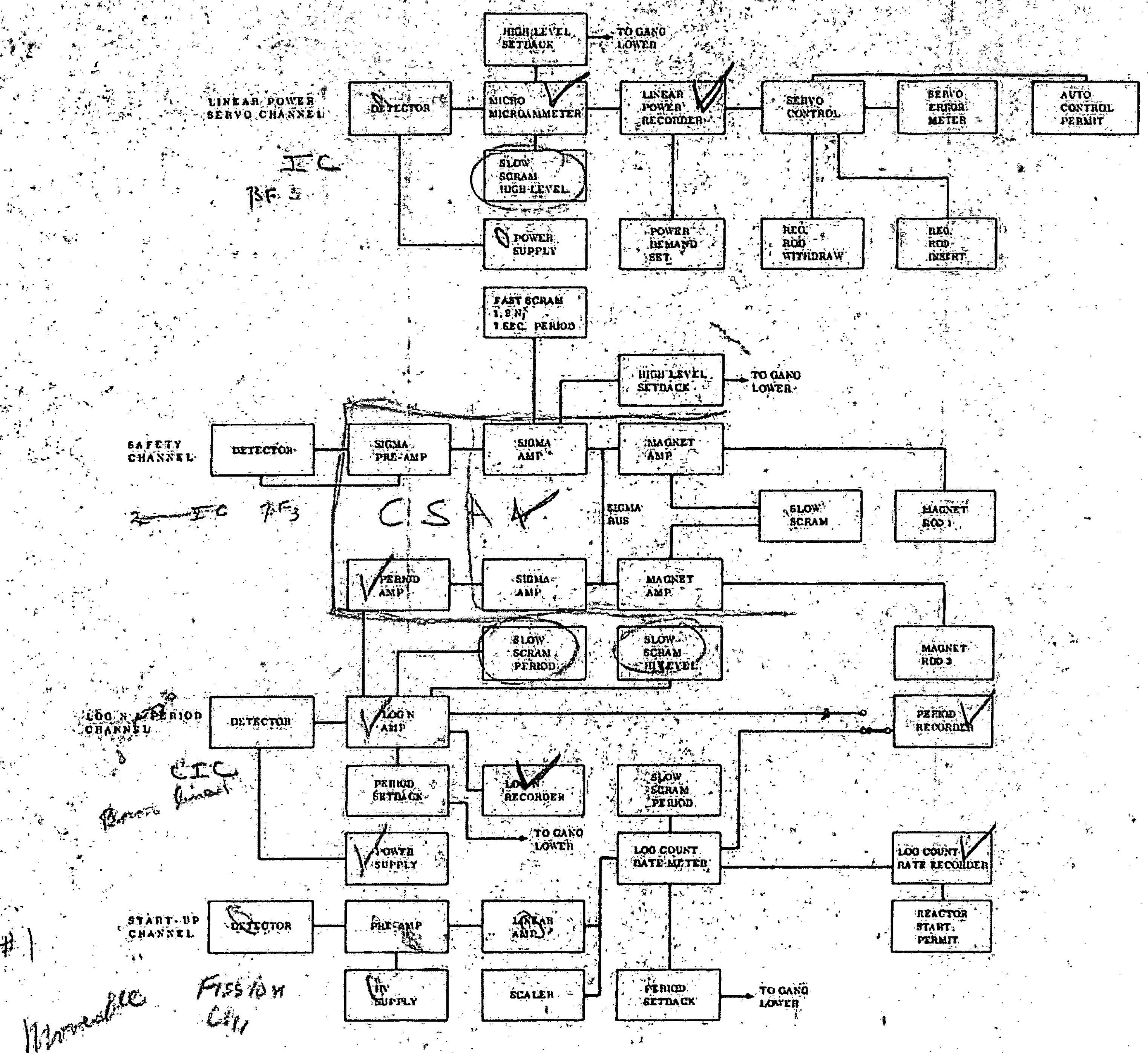
Since the facility is an existing one, the capital costs are low. The operational costs are minimal while the benefits in the education process of nuclear engineering students, radiation health scientists and technicians are great, both to the individual people and to the national interests. No reasonable alternatives exist to the wide versatility of research/training reactors such as the PUR-1 in contributing to the education and scientific knowledge.

Purdue University, as a large research institution, has the capability to continue safe operation of PUR-1 for the foreseeable future.

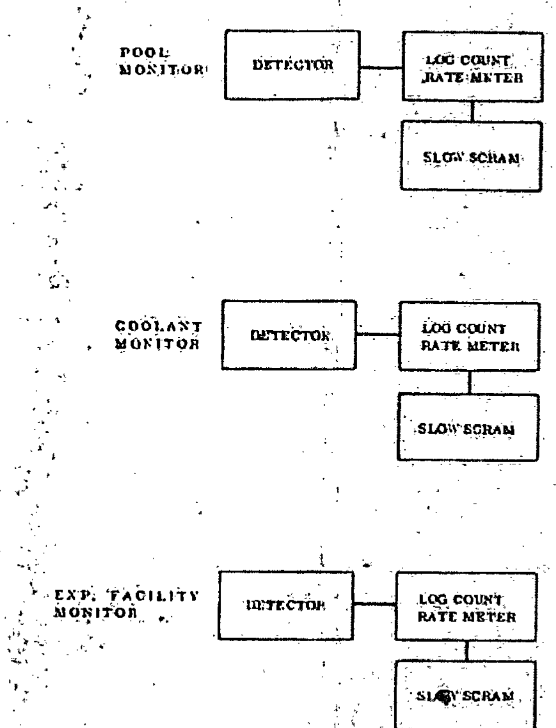
15.2 Financial Ability to Decommission the Facility

Purdue University will provide a letter of intent to show it will meet the financial obligations of decommissioning the facility.

APPENDIX 1: PUR-1 DRAWINGS



NUCLEAR INSTRUMENTATION



AREA MONITORS

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
	LIST OF MATERIAL							

		DATE		LOCKHEED NUCLEAR PRODUCTS LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
		DESIGN		
		CHK.	WEB	
		MATL.		PURDUE UNIVERSITY REACTOR INSTRUMENTATION BLOCK DIAGRAM
		STRESS		
		SUPV.	WEB	
		PROJ.	LBN	TOLERANCES / SCALE / DWG. NO.
		APPD.	WPW	FRACTIONS ± / .XX ± .03 / / PD-11-0003
APPLICATION	QTY. REQ'D	APPD.		ANGLES ± / .XXX ± .010

J. R. EATON

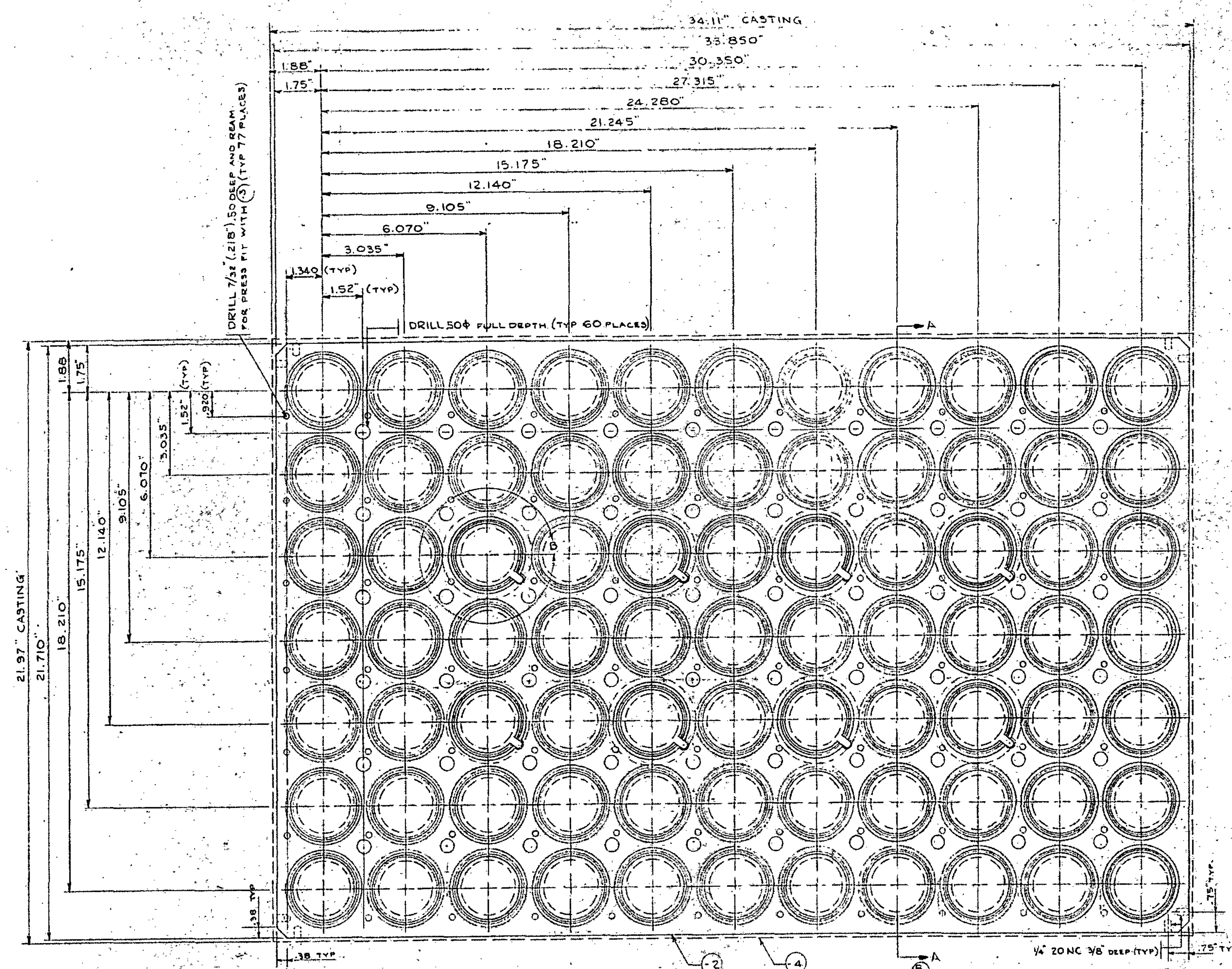
[redacted]

[redacted]

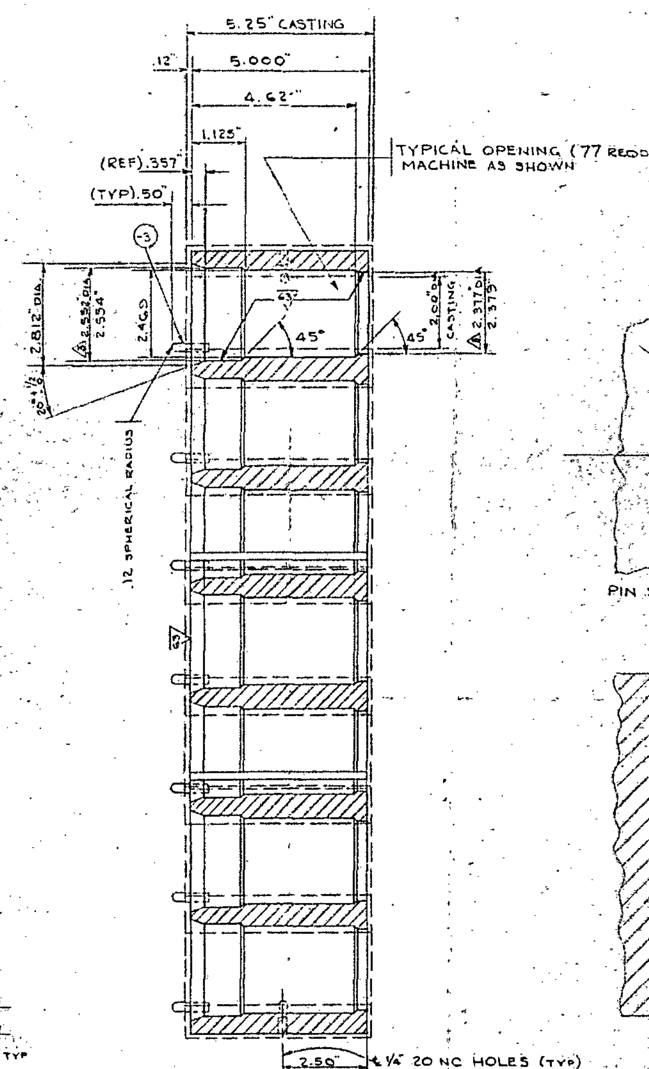
[redacted]

REVISIONS

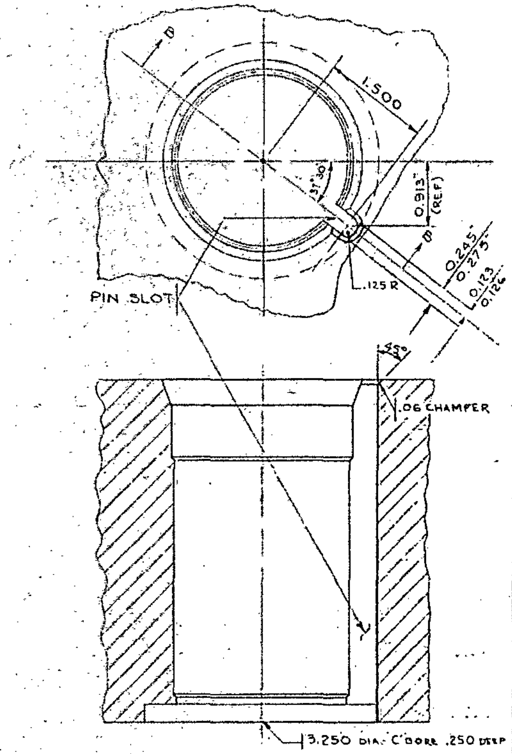
ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	(1) ADDED 5		TCB		10-3-61	
	B	(1) RELOCATED A-A CUTTING PLANE		[illegible]		12-? -61	



PLAN



SECTION A-A



SECTION B-B
SAME AS TYPICAL OPENING
EXCEPT PIN SLOT & COUNTERBORE ADDED
(8 REQD)

DETAIL B
CONTROL ROD OPENING
FULL SIZE

(-1) GRID PLATE ASSEMBLY

3- CASTING TOLERANCE ± .03"
2- CORNER RADII .06"
1- DRAFT ANGLE 2°

CASTING NOTES

(A) 5 6061-T652 (FEDERAL SPEC QQ-A-367e-V) FORGING IS ACCEPTABLE MATERIAL FOR PART -4.
4 - CHROMIC ACID ANODIZE WITH NO DI-CHROMATE SEAL
3 - MUST BE CONCENTRIC WITHIN .002" T.I.R.
2- ANODIZE (-2) PRIOR TO INSTALLATION OF (-3
1- ALL MACHINED SURFACES TO HAVE 125 FINISH UNLESS NOTED

GENERAL NOTES

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
—	-4		CASTING 5	356-T6 ALUMINUM ALLOY	21.97" W 34.11" L 5.00" H	QQ-A-601-3	T6	—
77	-3		DOWEL PIN	304 SS	.25" ϕ x 1" LG		NO	63
1	-2		GRID PLATE	MAKE FROM -4	—	—	—	ANODIZE 4
—	-1		GRID PLATE ASSY	—	—	—	—	—

LIST OF MATERIAL

DATE	7-14-61
DESIGN	[illegible]
CHK.	[illegible signature]
MATL.	
STRESS	
SUPV.	
PROJ.	
APPD.	[illegible signature]
APPD.	
APPLICATION	QTY. REQ'D

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

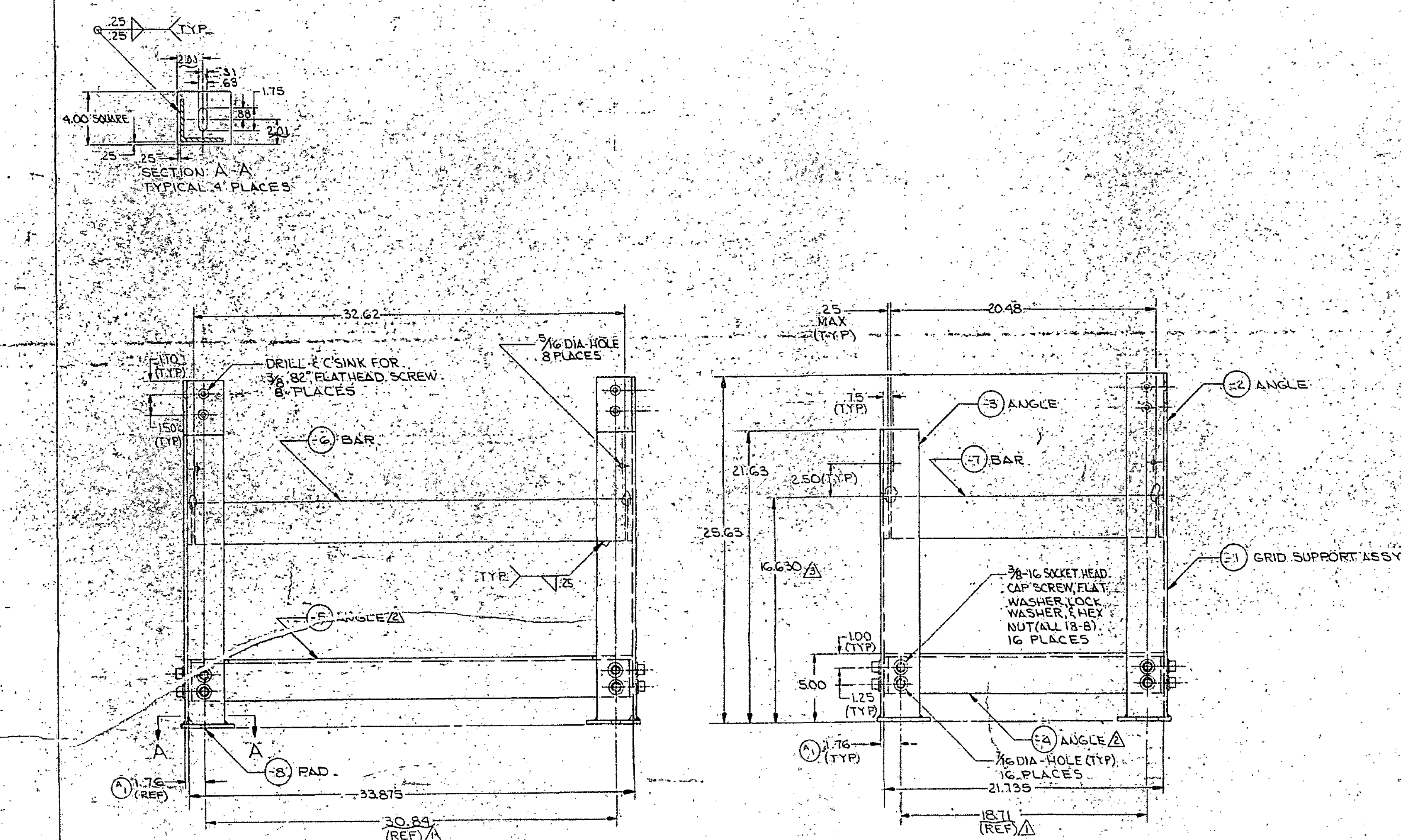
PURDUE UNIVERSITY
REACTOR GRID PLATE

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .03	1/2	117-2031
ANGLES ± 15'	.XXX ± .010		

J. R. [illegible]

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	1 1.76 (WAS) 2.00		HGS		11-15-61	



REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
16	3/8-16		HEX NUT	18-8 STN STL				
16	3/8		LOCKWASHER, MEDIUM					
32	3/8		FLAT WASHER					
16	3/8-16		SOCKET HEAD CAP SCREW	18-8 STN STL	1 LONG			
4	-8		PAD	6061-T6 AL	1/4 PLATE			
2	-7		BAR		3 WIDE x 3/8			
2	-6		BAR		3 WIDE x 3/8			
2	-5		ANGLE		3x3x1/4			
2	-4							
2	-3							
2	-2		ANGLE	6061-T6 AL	3x3x1/4			
	-1		GRID SUPPORT ASSY					

LIST OF MATERIAL

5. ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AWS-STD CODE FOR GAS & ARC WELDING IN BLDG CONSTRUCTION". WELDERS TO HOLD CURRENT CARD.
4. CHROMIC ACID ANODIZE WITH NO DI-CHROMATE SEAL, AFTER WELDING
3 THESE SURFACES TO BE PARALLEL WITHIN ±.010.
2 CUT ENDS OF ANGLES AT 45° FOR FLUSH JOINT.
1 THIS DIMENSION C TO C SPACING OF MOUNTING STUDS.
NOTE:

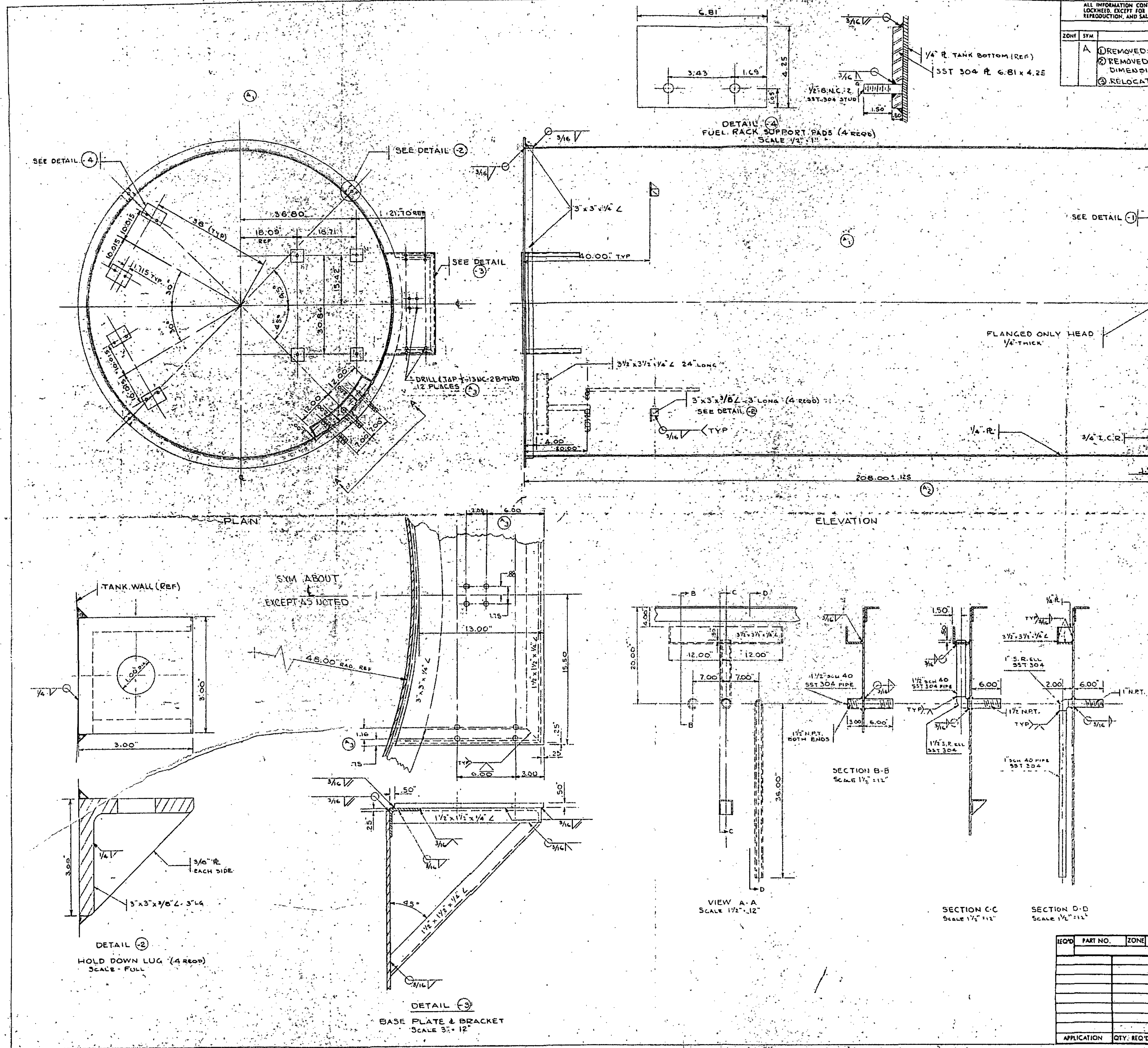
		DATE	7-8-61	
		DESIGN	BRADBURNE	
		CHK.	R H B.	
		MATL.		
		STRESS		
		SUPV.		
		PROJ.		
	1 (-1)	APPD.	[illegible]	
APPLICATION	QTY. REQ'D	APPD.		

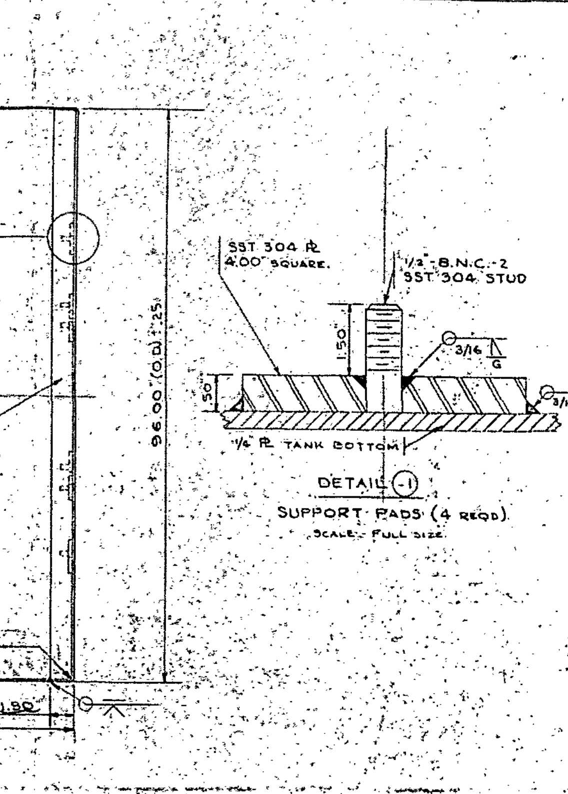
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

PURDUE UNIVERSITY
GRID SUPPORT ASSEMBLY

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ±.03	1/4	117-2002 (A)
ANGLES ±	.XXX ±.010		

J. R. EATCH





REVISIONS.

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	① REMOVED SUPPORT LUGS & DETAIL-5 ② REMOVED NOTE 5 & CORRESPONDING DIMENSION. ③ RELOCATED HOLES IN BASEPLATE		JCB		10-3-61	

NOTES.

1- MATERIAL
TANK HEAD & SHELL - A.S.T.M. A-7 55 T.
STRUCTURAL STEEL - A.S.T.M. A-7 55 T. EXCEPT AS NOTED.

2- WELDING
ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AMERICAN WELDING SOCIETY - STANDARD CODE FOR GAS AND ARC WELDING IN BUILDING CONSTRUCTION". WELDING SHALL BE DONE ONLY BY WELDERS HOLDING A CURRENT CERTIFICATE ISSUED BY A RECOGNIZED, APPROVED TESTING AUTHORITY.
ALL WELDS SHALL BE FULL PENETRATION WELDS UNLESS SHOWN OTHERWISE.

3- INSPECTION
ALL WELDS ON THE INTERNAL SURFACE OF THE TANK SHALL BE GROUND SMOOTH WITHOUT UNDERCUTTING AND SHALL BE INSPECTED USING A LIQUID PENETRANT IN ACCORDANCE WITH THE MANUFACTURERS INSTRUCTIONS. NO VISIBLE CRACKS OR CREVICES SHALL BE PERMITED. ALL DEFECTS SHALL BE REPAIRED BY WELDING IN ACCORDANCE WITH THE WELDING SPECIFICATION. AFTER INSPECTION OF ALL WELDS THE TANK SHALL BE FILLED WITH WATER AND INSPECTED, NO LEAKS ARE PERMITTED.

4- PAINTING
AFTER COMPLETE INSPECTION ALL EXTERNAL SURFACES OF THE TANK SHALL BE GIVEN A SHOP COAT OF "KOPPERS COMPANY, INC." BITUMASTIC MILL UNDERCOAT FOLLOWED BY TWO COATS OF BITUMASTIC No. 50 AS PER MANUFACTURERS INSTRUCTIONS.

(A2)

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH

LIST OF MATERIAL

DATE	7-21-61
DESIGN	R.H.S.
CHK.	[illegible]
MATL.	
STRESS	
SUPV.	
PROJ.	
APPD.	[illegible]
APPD.	

APPLICATION | QTY. REQ'D

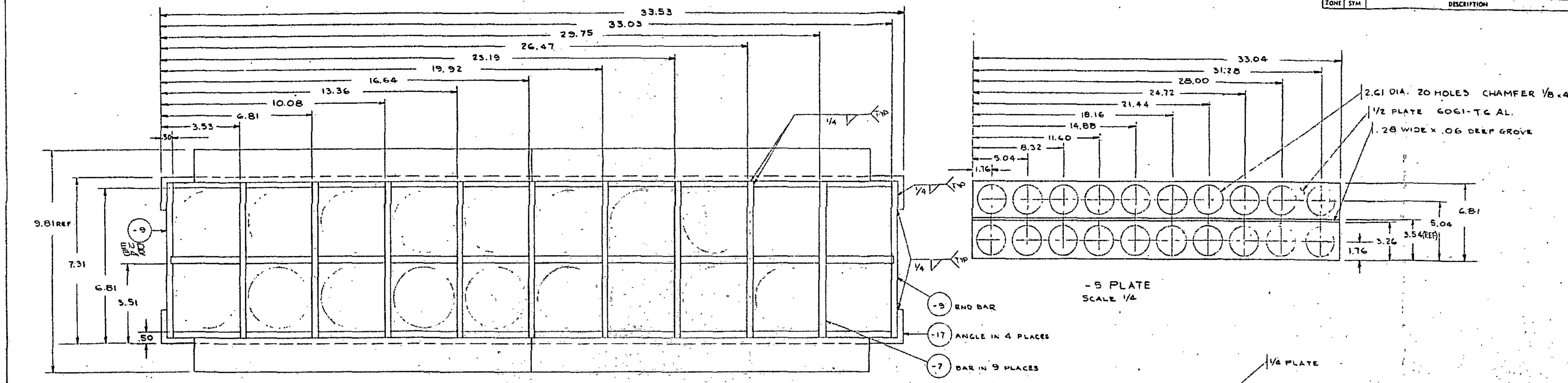
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

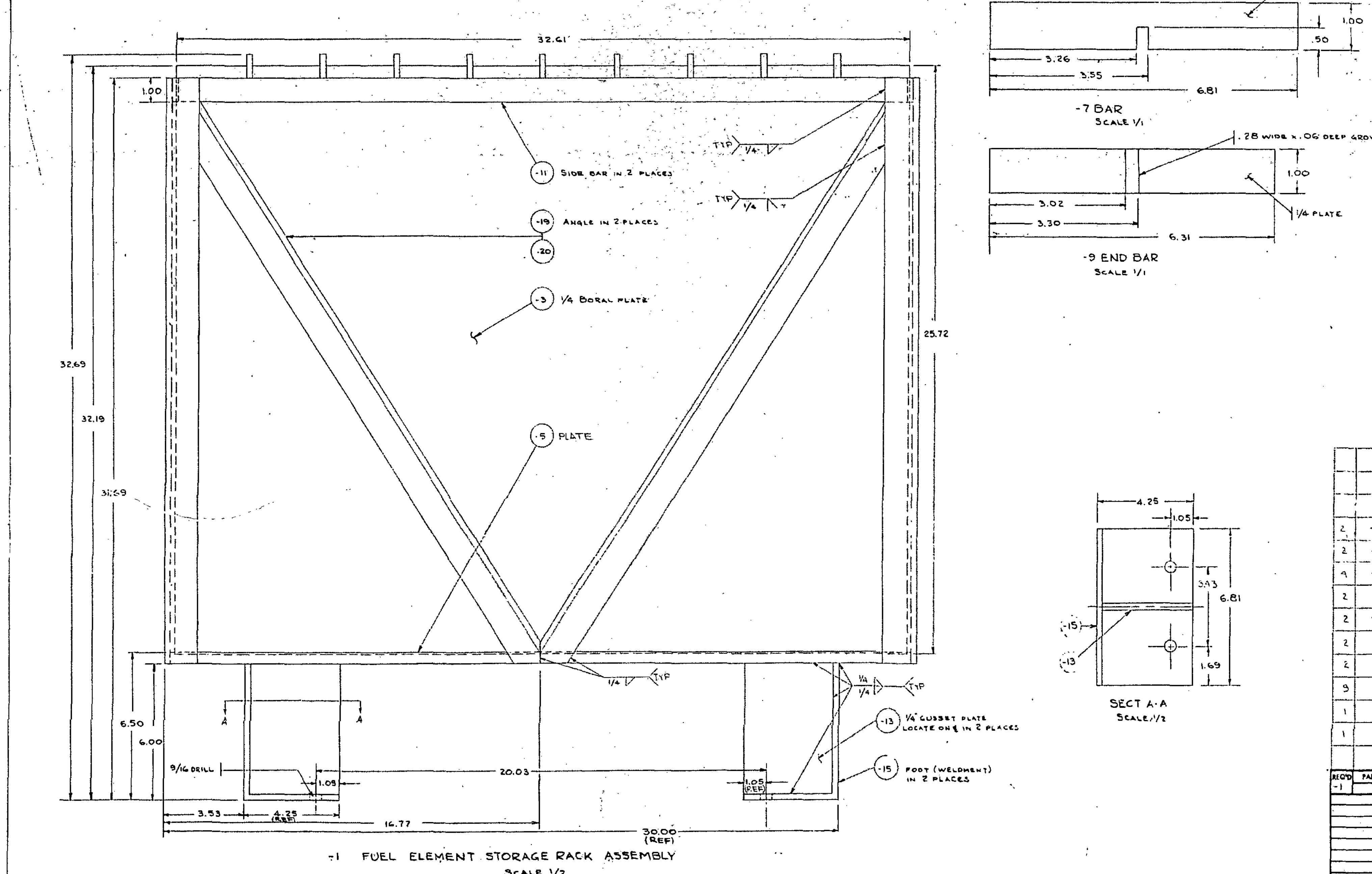
PURDUE UNIVERSITY
REACTOR TANK ASSEMBLY

TOLERANCES.		SCALE	DWG. NO.
FRACTIONS ± 1/32 ANGLES ± 1/2°	.XX ± .03 .XXX ± .010	1" = 12"	117-2003

REVISIONS

ZONE	STM	DESCRIPTION	SERIAL NO	DFTMN	CHK	DATE





NOTES

1- WELDING.
ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AMERICAN WELDING SOCIETY - STANDARD CODE FOR GAS AND ARC WELDING IN BUILDING CONSTRUCTION".
WELDING SHALL BE DONE ONLY BY WELDERS HOLDING A CURRENT CERTIFICATE ISSUED BY A RECOGNIZED, APPROVED TESTING AUTHORITY.
ALL WELDS SHALL BE FULL PENETRATION WE[LDS] UNLESS SHOWN OTHERWISE.

2- BREAK ALL SHARP CORNERS.

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT
2	-20		ANGLE	ALUMINUM	1½" x 1½" x ¼" - 29" LONG CO		
2	-19		ANGLE	ALUMINUM	1½" x 1½" x ¼" - 29" LONG		
4	-17		ANGLE	ALUMINUM	1½" x 1½" x ¼" - 26" LONG		
2	-15		FOOT	ALUMINUM	10" x 7" x ¼"		
2	-13		GUSSET PLATE	ALUMINUM	6" x 4" x ¼"		
2	-11		SIDE BAR	ALUMINUM	1 x 33.28 x ¼"		
2	-9		END BAR	ALUMINUM	1" x 7" x ¼"		
9	-7		BAR	ALUMINUM	1" x 7" x ¼"		
1	-5		PLATE	ALUMINUM	7 x 34.00 x ½	6061-T6	
1	-3		BORAL PLATE	BORAL	25.72 x 32.61 x ¼ THICK		
-1	-1		FUEL ELEMENT STORAGE RACK ASSEMBLY				

LIST OF MATERIAL

DATE	8-1-61
DESIGN	EMB/RHB
CHK.	
MATL.	
STRESS	
SUPV.	
PROJ.	
APPD.	
APPD.	

APPLICATION | 2(-1) QTY. REQ'D

LOCKHEED NUCLEAR PRODUC[TS]
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

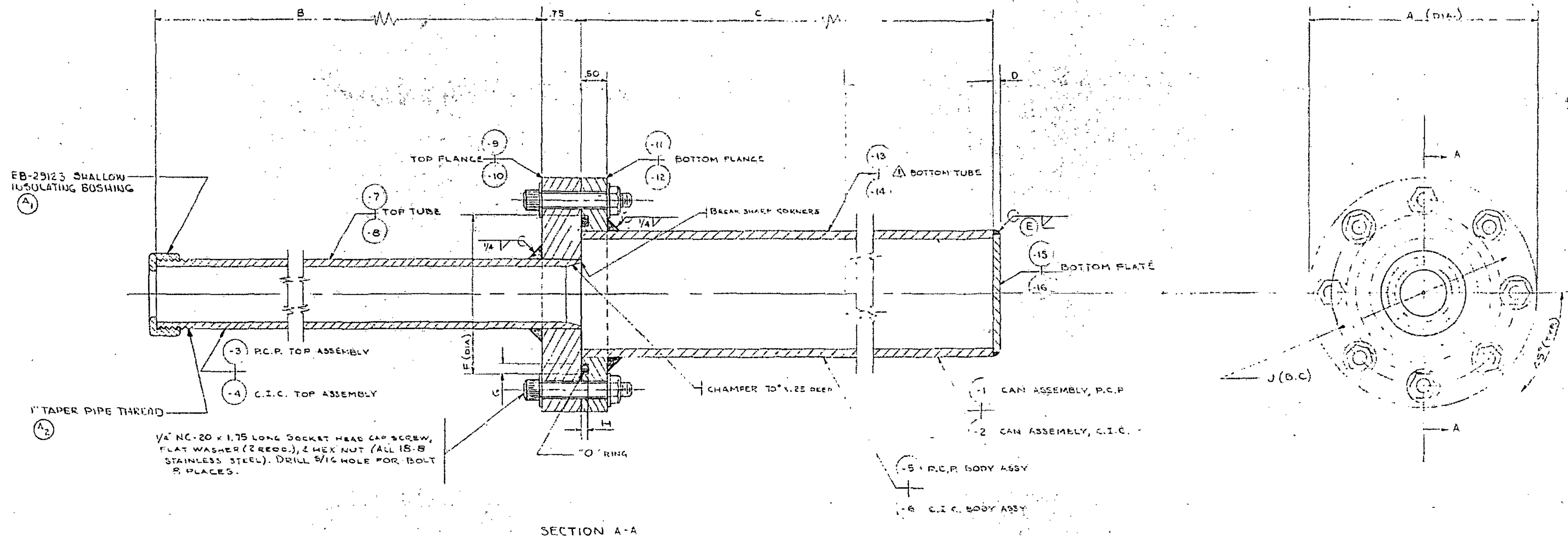
PURDUE UNIVERSITY
FUEL ELEMENT STORAGE RACK

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .03	AS NOTED	117-2004
ANGLES ±	.XXX ± .010		

J. R. EATON

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	(1) ADDED INSULATING BUSHING (2) ADDED TAPER PIPE THREAD		JCB		12-6-61	



PART NUMBER	DIMENSION A	B	C	D	E	F	G	H	J
-7		151±.10							
-9	4.60								3.442
-11	4.60					2.879 +.000/-.029	.053 +.003/-.000	.050 +.008/-.008	3.442
-8		155±.10							
-10	6.32								5.256
-12	6.32					4.693 +.000/-.047	.140 +.003/-.000	.081 +.003/-.000	5.256
-13			32.00		1/8"				
-14			28.50		1/4"				
-15				.12					
-16				.25					

5. BUFF ALL EXTERIOR SURFACES.

4.- STATIC TEST FINAL ASSEMBLY TO 15 P.S.I. NO LEAKS PERMISSABLE.

3.- ALL WELDING TO CONFORM TO THE LATEST ISSUE "A.W.S.- STANDARD CODE FOR GAS AND ARC WELDING IN BUILDING CONSTRUCTION." WELDING TO BE DONE BY WELDER HOLDING A CURRENT CARD ISSUED BY A RECOGNIZED TESTING AUTHORITY.

2.- ALL MACHINE WORK ON FACE OF -11 & -12 BOTTOM FLANGE TO BE DONE AFTER WELDING. FACE OF -9 & -10 TOP FLANGE TO BE TRUE WITHIN ±.002 AFTER WELDING. FINISH OF FACES OF -11, -12, -9, -10 TO BE 32 ALL OTHER MACHINED SURFACES TO BE 125

1. MINIMUM I.D. OF -13 BOTTOM TUBE TO BE 2.06.

NOTE:

-6	-5	-4	-3	-2	REQ'D -1	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
				1	1	EB-29123		SHALLOW INSULATING BUSHING (GRAYBAR ELECTRIC CO. OR EQUAL)					
1						-16		BOTTOM PLATE	6061-T6 AL PLATE	.25 x 4" x 4"			
	1					-15		BOTTOM PLATE	6061-T6 AL SHEET	.125 x 3" x 3"			
1						-14		BOTTOM TUBE	6061-T6 AL STD PIPE	3½" I.P.S. SCH 40 x 30" LG			
	1					-13		BOTTOM TUBE	6061-T6 AL STD PIPE	2" I.P.S. SCH 40 x 33" LG			
1						-12		BOTTOM FLANGE	6061-T6 AL PLATE	.625 x 6.5 x 6.5			
	1					-11		BOTTOM FLANGE	6061-T6 AL PLATE	.625 x 5" x 5"			
		1				-10		TOP FLANGE	6061-T6 AL PLATE	.75 x 6.5 x 6.5			
			1			-9		TOP FLANGE	6061-T6 AL PLATE	.75 x 5" x 5"			
		1				-8		TOP TUBE	6061-T6 AL STD PIPE	1" I.P.S. SCH 40 x 156" LG.			
			1			-7		TOP TUBE	6061-T6 AL STD PIPE	1" I.P.S. SCH 40 x 153" LG.			
				1		2-157		"O" RING (PARKER SEAL CO., OR EQUAL)					
					1	2-39		"O" RING (PARKER SEAL CO., OR EQUAL)					
				8	8			NUT	SST 18-8	1/4" NC-20			
				16	16			WASHER, PLAIN	SST 18-8	1/4" STD			
				8	8			CAP SCREW	SST 18-8	1/4" N.C. 20 x 1.75" LONG			
–				1		-6		C.I.C. BODY ASSY					
	–				1	-5		P.C.P. BODY ASSY					
		–		1		-4		C.I.C. TOP ASSY					
			–		1	-3		P.C.P. TOP ASSY					
				–		-2		CAN ASSY, C.I.C.					5
					–	-1		CAN ASSY, P.C.P.					5

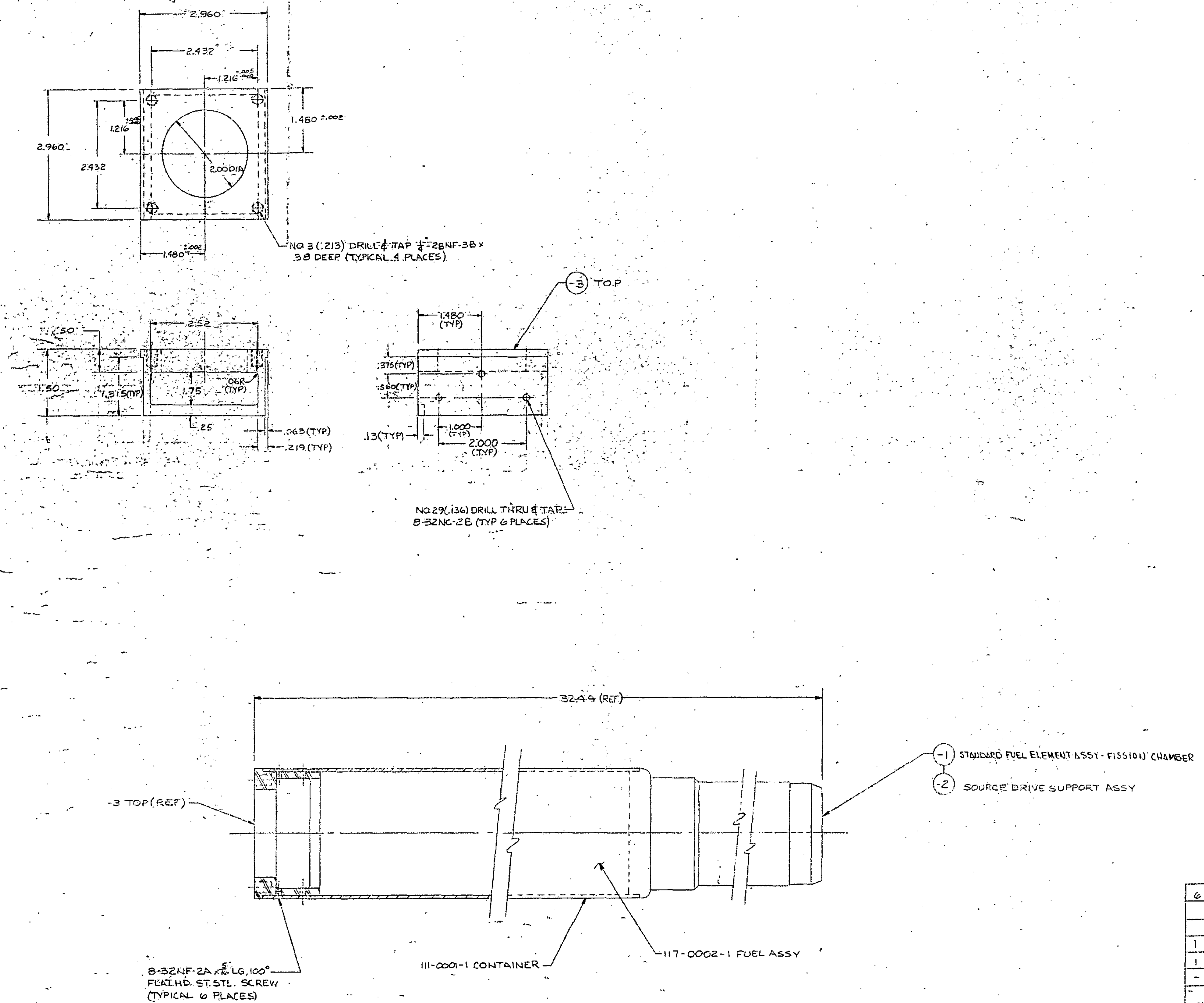
LIST OF MATERIAL

		DATE	9-14-61	**LOCKHEED NUCLEAR PRODUCTS** LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA	
		DESIGN	R.H.B.		
		CHK.			
		MATL.		PURDUE UNIVERSITY ION CHAMBER WATERPROOF CAN ASSY	
		STRESS			
		SUPV.			
	1 (-2)	PROJ.		TOLERANCES: FRACTIONS ± ; .XX ±.03 ; ANGLES ±30' ; .XXX ±.010	SCALE: NONE
	2 (-1)	APPD.			DWG. NO. 117-2005(A)
APPLICATION	QTY. REQ'D	APPD.			

J. R. EATON

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD



⚠1 OMIT SCREWS & -7 HANDLE.

-2	-1	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
6	6	8-32NC-2A		100° FLAT HEAD SCREW	18-8 ST. STL.	5/16" LG.			
	1	117-0002-1		FUEL ASSY					
1	1	111-0001-1		CONTAINER ⚠1					
1	1	-3		TOP	303 ST. STL.	3"x3"x2"			
-		-2		SOURCE DRIVE SUPPORT ASSY					
-	-	-1		STANDARD FUEL ELEMENT ASSY - FISSION CHAMBER					

LIST OF MATERIAL

APPLICATION	QTY. REQ'D		
		DATE	11-10-61
		DESIGN	EMS
		CHK.	J.C. Brothers
		MATL.	
		STRESS	
		SUPV.	
	1 (-2)	PROJ.	W.B. Brown
	1 (-1)	APPD.	[illegible] Hoffman
		APPD.	

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

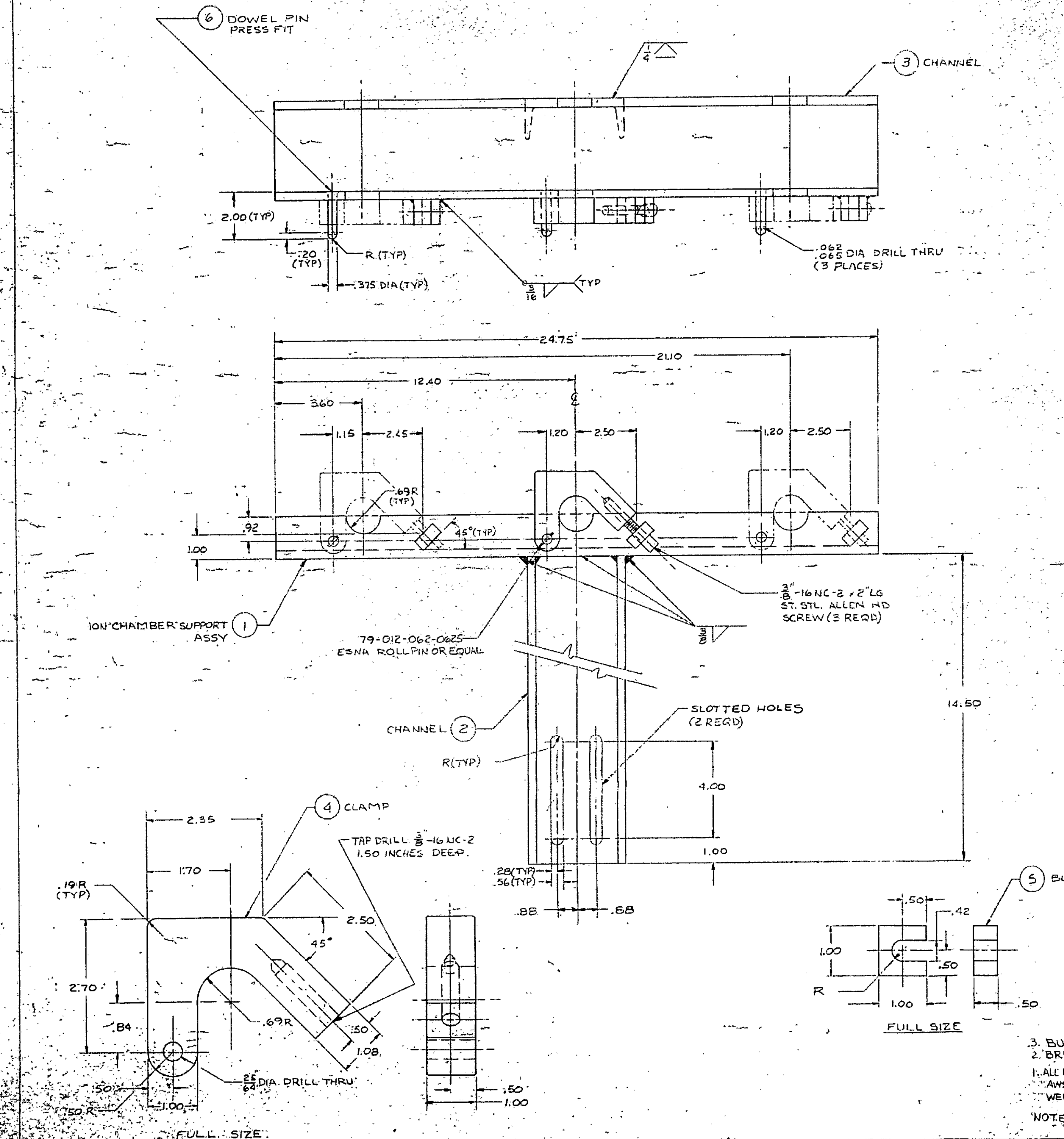
PURDUE UNIVERSITY
SOURCE DRIVE SUPPORT ASSY & STANDARD FUEL ELEMENT ASSY - FISSION CHAMBER

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .01	1/1	117-2006
ANGLES ±	.XXX ± .005		

J. R. EATON

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD



NOTE:
1. ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AWS STD CODE FOR GAS & ARC WELDING IN BLDG CONSTRUCTION." WELDERS TO HOLD CURRENT CARD.
2. BREAK SHARP CORNERS.
3. BUFF FOR APPEARANCE.

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
3			ALLEN HD SCREW	ST STL 18-8	3/8"-16NC-2 x 2" LONG			
3	79-012-062-0625		ESNA ROLL PIN					
3	-6		DOWEL PIN	ST STL 18-8	.375DIA x 2"			
3	-5		BLOCK	6061-T6 AL.	1" x 1" x .50			
3	-4		CLAMP	ST STL 18-8	4.5 x 35 x 1"			
1	-3		4U2.58 CHANNEL	6061-T6 AL	24.75 LG.			
1	-2		4U2.58 CHANNEL	6061-T6 AL	12.5 LG			
-	-1		SUPPORT ASSY					
-1	LIST OF MATERIAL							

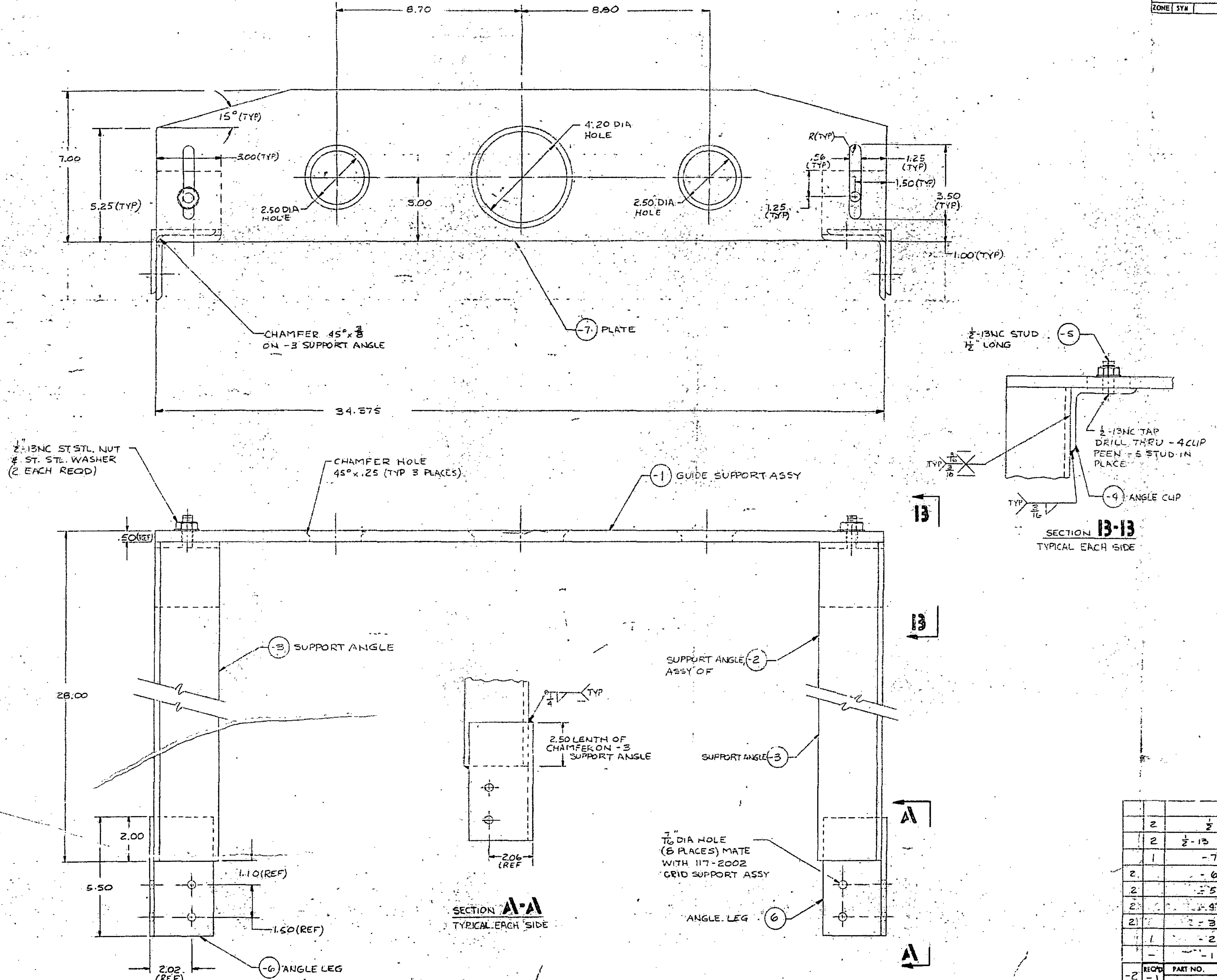
DATE	10-16-61	**LOCKHEED NUCLEAR PRODUCTS**	
DESIGN	SCHAAG	LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA.	
CHK.	[signature]	PURDUE UNIVERSITY ION CHAMBER SUPPORT ASSEMBLY	
MATL.			
STRESS			
SUPV.			
PROJ.		TOLERANCES	SCALE / DWG. NO.
APPD.	[signature]	FRACTIONS ± / .XX ± .03	HALF & NOTED / 117-2007
APPD.		ANGLES ± 1/2° / .XXX ± .010	

APPLICATION	QTY. REQ'D
	1 (-1)

J. R. EATON

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD



NOTE:
1. CHROMIC ACID ANODIZE WITH NO DI-CHROMATE SEAL BEFORE ASSY OF -5 STUD & AFTER WELDING.
2. ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AWS-STD CODE FOR GAS & ARC WELDING IN BLDG CONSTRUCTION". WELDERS TO HOLD CURRENT CARD.

LIST OF MATERIAL

-2	-1 REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
	2	1/2		WASHER	18-8 ST. STL.				
	2	1/2-13		NUT	18-8 ST. STL.				
	1	-7		PLATE	6061-T6 AL	1/2" x 10" x 35"			
2		-6		ANGLE LEG	6061-T6 AL	3" x 3" x 1/4" ∠ 5 1/2" LG			
2		-5		STUD	18-8 ST. STL.	1/2 DIA x 1 1/2 LG			
2		-4		ANGLE CLIP	6061-T6 AL	3" x 3" x 1/4" ∠ 3" LG			
2		-3		SUPPORT ANGLE	6061-T6 AL	3" x 3" x 1/4" ∠ 28" LG			
	1	-2		SUPPORT ANGLE, ASSY OF		—			
	—	-1		GUIDE SUPPORT ASSEMBLY					

		DATE	10-18-61
		DESIGN	Schaag
		CHK.	[illegible signature]
		MATL.	
		STRESS	
		SUPV.	
		PROJ.	
	1 (-1)	APPD.	J. B. Brown
APPLICATION	QTY. REQ'D	APPD.	

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

PURDUE UNIVERSITY
ION CHAMBER GUIDE SUPPORT ASSY

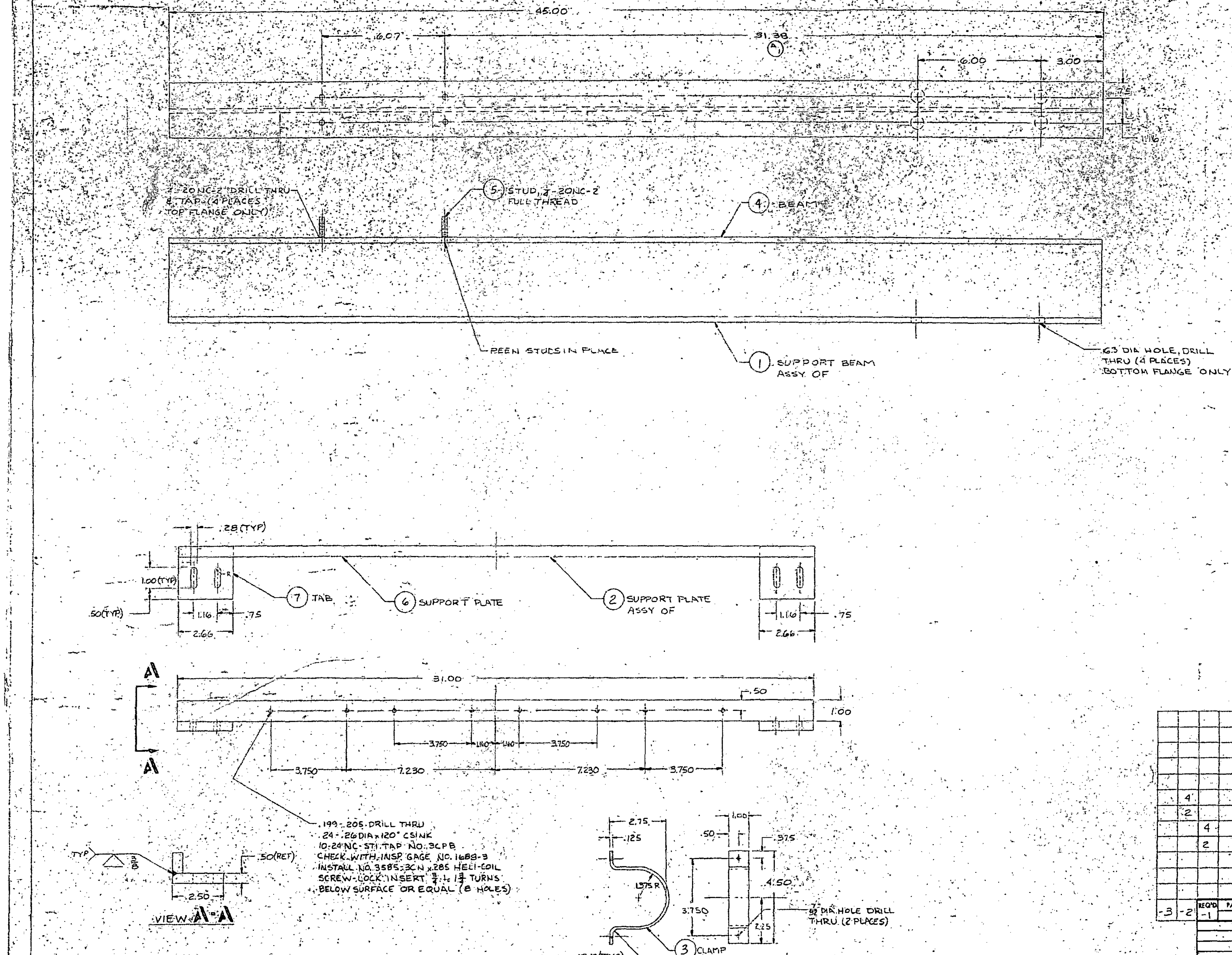
TOLERANCES		SCALE	DWG. NO.
FRACTIONS ± ; ANGLES ± 2°	.XX ± .03 ; .XXX ± .010	HALF	117-2008

J. R. EATON

IF THIS DOCUMENT IS NOT SUPPLIED UNDER THE REQUIREMENTS OF A UNITED STATES GOVERNMENT CONTRACT, THE FOLLOWING LEGEND SHALL APPLY:

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN.	CHK.	DATE	APPD.
	A	(1) 31.38 (WAS) 32.63		/48		11-15-61	



NOTE:

1. ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AWS-STD CODE FOR GAS & ARC WELDING IN BLDG CONSTRUCTION." WELDERS TO HOLD CURRENT CARD.
2. BREAK SHARP CORNERS.

-3	-2	-1	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
	4		-7		TAB	6061-T6 AL	1/2" x 2 1/2" x 2 3/4"			
	2		-6		SUPPORT PLATE	6061-T6 AL	1/2" x 1" x 31"			
		4	-5		STUD	18-8 ST STL	1/4" DIA x 1 1/4"			
		2	-4		"I" BEAM	6061-T6 AL	4I 2.72			
			-3		CLAMP	6061-T6 AL	3/8" x 1" x 6"			
			-2		SUPPORT PLATE ASSY OF					
			-1		SUPPORT BEAM ASSY OF					

REQ'D — LIST OF MATERIAL

APPLICATION	QTY. REQ'D		
		DATE	10-17-61
		DESIGN	Schaaf
		CHK.	[illegible]
		MATL.	
		STRESS	
	5 (-3)	SUPV.	
	2 (-2)	PROJ.	
	2 (-1)	APPD.	J B Brown
		APPD.	

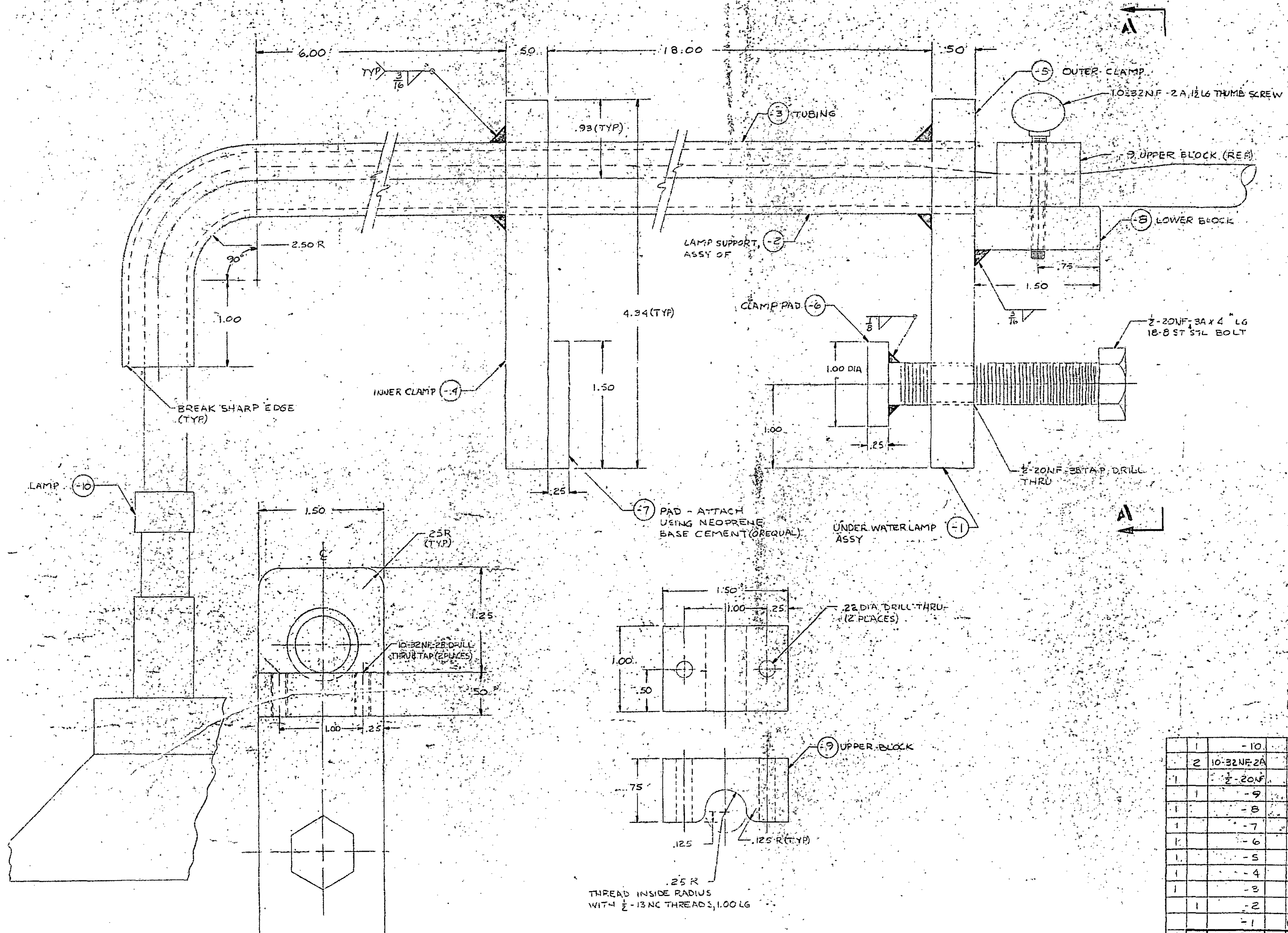
LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

PURDUE UNIVERSITY
DRIVE MECHANISM SUPPORT

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .03	HALF	117-2011 (A)
ANGLES ±	.XXX ± .010		

J. R. EATON

REVISIONS							
ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD.



	REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
	1	-10		LAMP	⚠4				
	2	10-32NF-2A		THUMB SCREW	18-8 ST. STL.	1 1/2 LONG			
1		1/2-20NF		BOLT	18-8 ST. STL.	4 LONG			
	1	-9		UPPER BLOCK	PHENOLIC	3/4" x 1" x 1 1/2"			
1		-8		LOWER BLOCK	18-8 ST. STL.	1/2 x 1 1/2 x 1 1/2			
1		-7		PAD	HARD RUBBER	1/4 x 1 1/2 x 1 1/2			
1		-6		CLAMP PAD	18-8 ST. STL.	1/4" x 1" DIA			
1		-5		OUTER CLAMP	18-8 ST. STL.	1/2" x 1 1/2" x 4 1/2"			
1		-4		INNER CLAMP	18-8 ST. STL.	1/2" x 1 1/2" x 4 1/2"			
1		-3		TUBING	18-8 ST. STL.	1/2" SCH. 10 PIPE			
	1	-2		LAMP SUPPORT, ASSY OF					
		-1		UNDER WATER LAMP ASSY					
-2	-1			LIST OF MATERIAL					

⚠4 UNDER WATER LAMP, 10" CHROME, 1000 WATT BULB, 25 FT. CABLE, CRAFTWELD EQUIPMENT CORP., 2626 JACKSON AVE., LONG ISLAND CITY, N.Y.

3. BREAK ALL SHARP CORNERS.

2. BUFF ALL EXTERIOR STAINLESS STEEL SURFACES WITH SUITABLE COMPOUND

1. ALL WELDING SHALL CONFORM TO THE LATEST ISSUE "AWS-STD CODE FOR GAS & ARC WELDING IN BLDG. CONSTRUCTION". WELDERS TO HOLD CURRENT CARD.

NOTE:

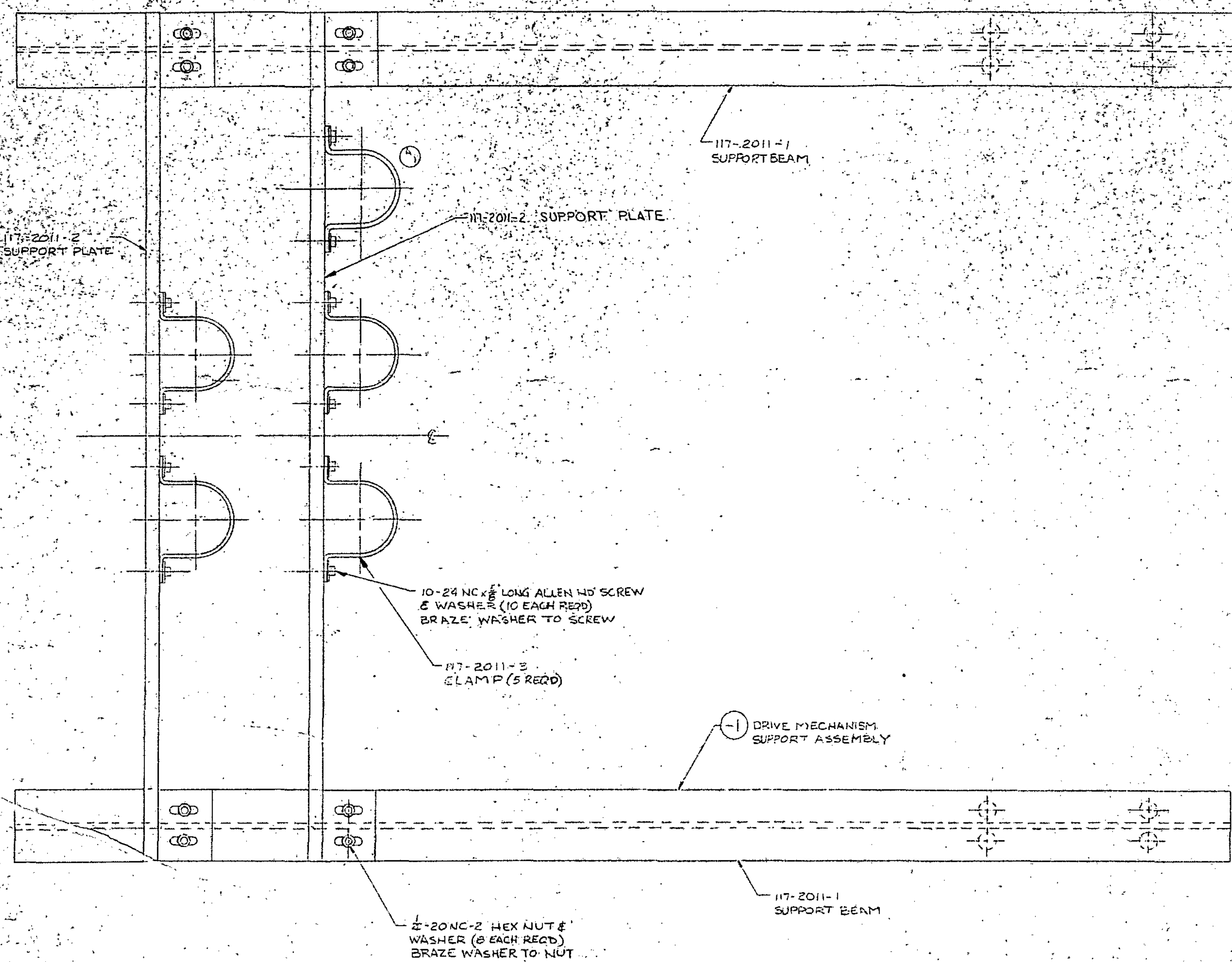
DATE	10-24-61	**LOCKHEED NUCLEAR PRODUCTS**	
DESIGN	SCHAAG	LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA	
CHK.	W. Bradburne		
MATL.		PURDUE UNIVERSITY	
STRESS		UNDER WATER LAMP ASSEMBLY	
SUPV.			
PROJ.		TOLERANCES	SCALE / DWG. NO.
APPD. (1(-1) QTY. REQ'D)		FRACTIONS ± 1/64 .XX ± .03	2/1
APPD. (APPLICATION)		ANGLES ± 1/2° .XXX ± .010	117-2012

J. B. EATON

IF THIS DOCUMENT IS NOT SUPPLIED UNDER THE REQUIREMENTS OF A UNITED STATES GOVERNMENT CONTRACT, THE FOLLOWING LEGEND SHALL APPLY:

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	(1) RELOCATED 117-2011-3 CLAMP ON 117-2011-2 SUPPORT PLATE		JCB		12-5-61	



NOTE: 1. BUFF BEFORE ASSY TO A GLOSSY FINISH USING A SUITABLE COMPOUND.

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
10	NO. 10		WASHER	18-8 ST. STL.				
10	10-24		ALLEN HD. SCREW	18-8 ST. STL.				
8	1/4		WASHER	18-8 ST. STL.				
8	1/4 20NC		NUT, HEX	18-8 ST. STL.				
5	117-2011-3		CLAMP					
2	117-2011-2		SUPPORT PLATE					
2	117-2011-1		SUPPORT BEAM					
	-1		DRIVE MECHANISM SUPPORT ASSY					
-1	LIST OF MATERIAL							

APPLICATION	QTY. REQ'D		
		DATE	10-27-61
		DESIGN	SCHAAG
		CHK.	J.C. Bodline
		MATL.	
		STRESS	
		SUPV.	
		PROJ.	
	1(-1)	APPD.	J.B. Brown
		APPD.	

LOCKHEED NUCLEAR PRODUCTS
LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

PURDUE UNIVERSITY
DRIVE MECHANISM SUPPORT ASSY

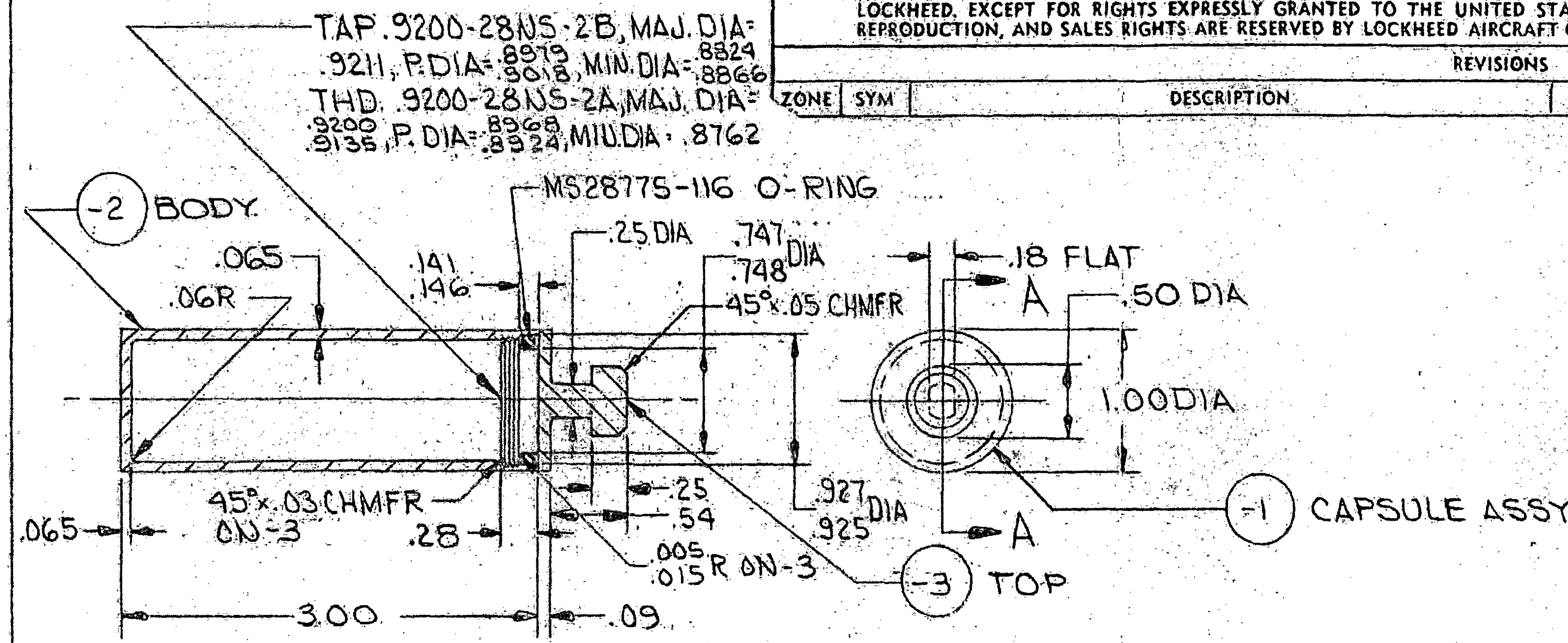
TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .03	1/2	117-2013 (A)
ANGLES ±	.XXX ± .010		

J. R. EATON

IF THIS DOCUMENT IS NOT SUPPLIED UNDER THE REQUIREMENTS OF A UNITED STATES GOVERNMENT CONTRACT, THE FOLLOWING LEGEND SHALL APPLY: ALL INFORMATION CONTAINED IN OR DISCLOSED BY THIS DOCUMENT IS CONSIDERED CONFIDENTIAL AND PROPRIETARY BY LOCKHEED, EXCEPT FOR RIGHTS EXPRESSLY GRANTED TO THE UNITED STATES GOVERNMENT; ALL DESIGN, MANUFACTURE, USE, REPRODUCTION, AND SALES RIGHTS ARE RESERVED BY LOCKHEED AIRCRAFT CORPORATION, GEORGIA DIVISION, MARIETTA, GEORGIA.

REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD



1	MS28775-116		O-RING	PARKER SEAL CO. OR EQUAL				
1	-3		TOP	6061-T6 AL	1.00 DIA x 1			
1	-2		BODY	6061-T6 AL	1.00 DIA x 3			
	-1		CAPSULE ASSY					
REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH

-1

LIST OF MATERIAL

2. ASSY TO BE LEAK FREE UP TO 20 PSI.

1. $\sqrt{32}$ FINISH ALL MACHINE SURFACES.

NOTE:

		DATE	9-7-61
		DESIGN	BRADBURNE
		CHK.	JBB
		MATL.	
		STRESS	
		SUPV.	
		PROJ.	
	36 (-1)	APPD.	[illegible]
APPLICATION	QTY. REQ'D	APPD.	

LOCKHEED NUCLEAR PRODUCTS

LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA

CAPSULE ASSY - ISOTOPE, PURDUE UNIVERSITY

TOLERANCES		SCALE	DWG. NO.
FRACTIONS ±	.XX ± .01	1/1	117-2014
ANGLES ± 30'	.XXX ± .005		

J. R. EATON

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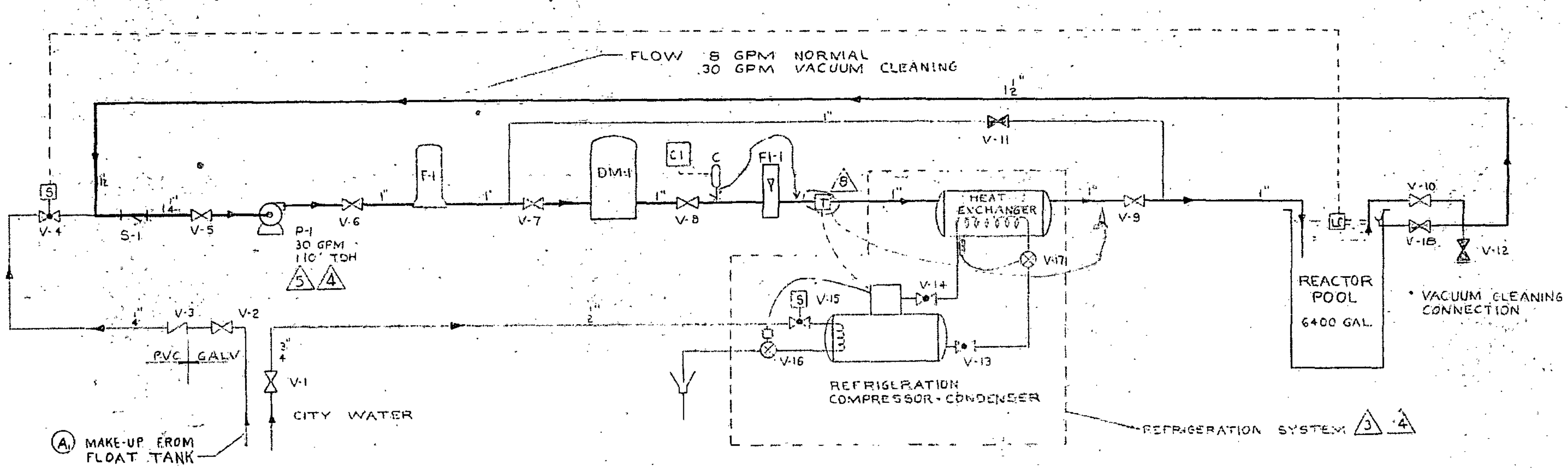
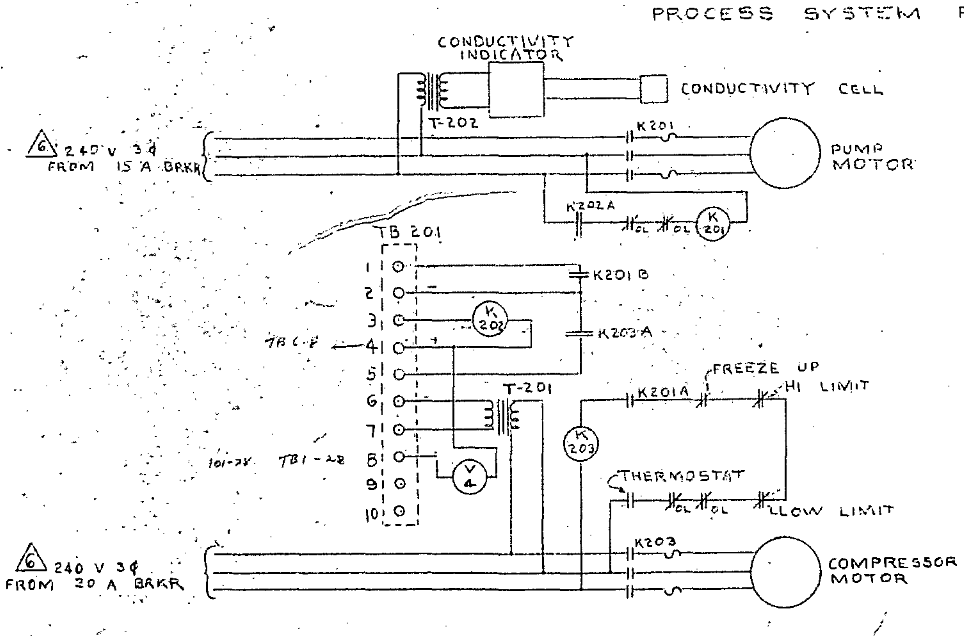
REVISIONS

ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD
	A	A. CHANGED MAKE-UP CONNECTION		EMB		12-5-61	



PROCESS SYSTEM FLOW DIAGRAM



PROCESS SYSTEM - WIRING DIAGRAM

	REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
	1	T		THERMOSTAT, BARBER-COLEMAN TC 202, RANGE 10°-90° 1.5° DIFF.					
	1	TB 201		TERMINAL BOARD, 10 TERMINALS					
	1	T 202		TRANSFORMER 230V/115V, 80 VA, IND. INST. TR-1 OR EQUAL					
	1	T 201		CONTROL TRANSFORMER 230V/24V, 2 AMP STANCOR P6377					
	1	K 203		MAGNETIC STARTER (PART OF REFRIG. SYSTEM)					
	1	K 202		RELAY, POTTER & BRUMFIELD TYPE KRP 11D, 24VDC. DPDT					
	1	K 201		MAGNETIC STARTER, SIZE 0 GE CR 7006, 3 POLE WITH 2 AUX. CONTACTS					
13	1	V-18		BALL VALVE 1½ PVC, SWE TUBE TURNS					
13 10	1	LC		LEVEL CONTROL, MAGNETROL TYPE TF 203					
4 3	1			REFRIGERATION SYSTEM 36,000 BTU/HR					
	1	CI		CONDUCTIVITY INDICATOR, IND INST RE 194, 0-10 MICRO MHO.					
	1	C		CONDUCTIVITY CELL, INDUSTRIAL INSTRUMENTS CEL-C001 WITH 1" ST. STL PIPE FITTING, PYREX CELL					
	1	DM-1		CARTRIDGE TYPE DEMINERALIZER BARNSTEAD WITH MIXED BED CARTRIDGE					
4 5	1	P-1		PUMP 30 GPM, 110' TDH					
11	1	FI-1		FLOW INDICATOR, BROOKS ROTAMETER FV-1110					
12	1	F-1		FILTER, CUNO MODEL 3AX2 WITH NO 2278 C.I CARTRIDGE					
	1	S-1		STRAINER, 1½" TYPE 316 SS, SCREWED END, LESLIE FIG 7500					
	1	V-17		THERMOSTATIC EXPANSION VALVE					
	1	V-16		WATER VALVE					
	1	V-15		SOLENOID VALVE, CONDENSER COOLING WATER					
	1	V-14		COMPRESSOR SUCTION SHUT OFF VALVE					
	1	V-13		LIQUID LINE SHUT OFF VALVE					
13	1	V-12		BALL VALVE, 1½" PVC, S.W.E, TUBE TURNS					
	1	V-11		1"					
13	1	V-10		1½"					
	1	V-9		1"					
	1	V-8		1"					
	1	V-7		1"					
	1	V-6		1"					
	1	V-5		1½"					
	1	V-4		SOLENOID VALVE, POOL MAKE-UP WATER ¼" ASCO #82626 24 VDC COIL					
	1	V-3		CHECK VALVE ¾" PVC, SWE, TUBE TURNS					
	1	V-2		GATE VALVE					
	1	V-1		GATE VALVE					

LIST OF MATERIAL

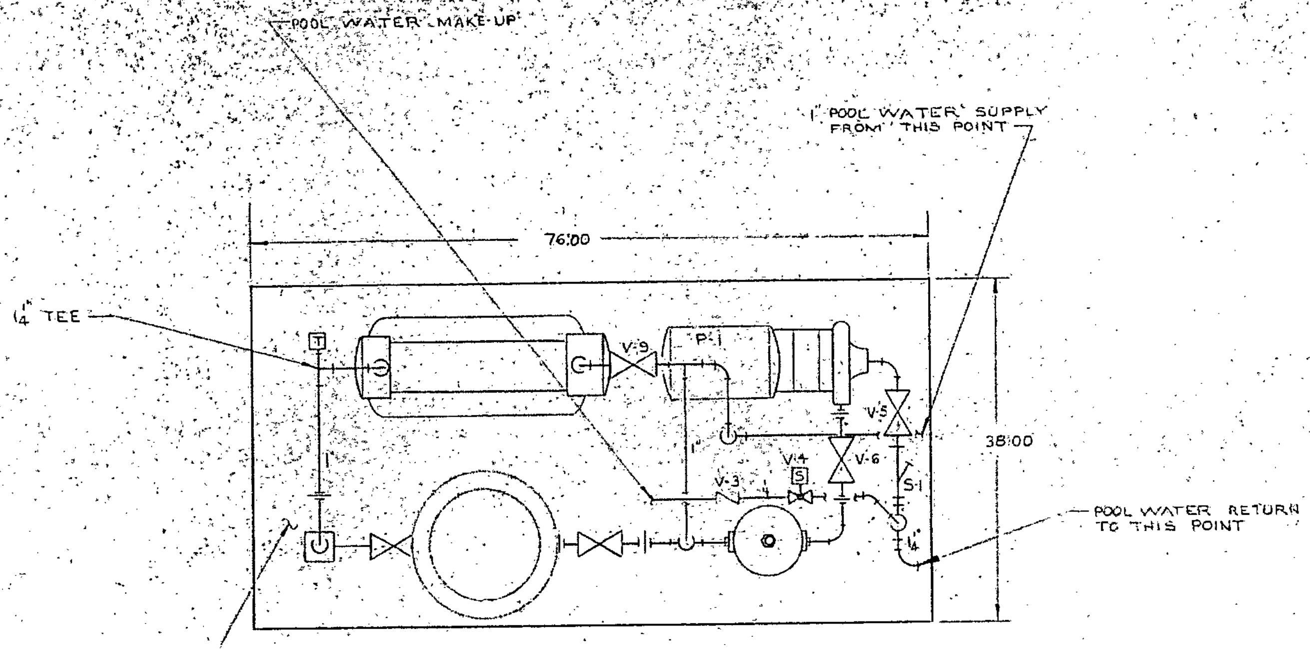
13. VENDOR TO FURNISH AND SHIP AS LOOSE MATERIAL
12. FILTER TO HAVE C.I HEADS WITH STAINLESS STEEL SHELL AND TRIM.
11. SIZE NO 10 WITH 1½ SCREWED ALUMINUM END FITTINGS, ROD GUIDED STAINLESS STEEL FLOAT, COVER PLATES
10. TANK FLOAT SWITCH WITH STANDARD STAINLESS STEEL TRIM, SWITCH TO CLOSE ON DROP IN LEVEL. DISTANCE TO ACTUATING LEVEL 7" ±. MOUNTING CONNECTION 1" NPT
9. COMPONENTS SPECIFIED ARE INDICATIVE OF QUALITY DESIRED. APPROVED EQUALS MAY BE USED. APPROVAL IS TO BE BY L.N.P.
8. USE STAINLESS STEEL THERMOWELL
7. SEE DRAWING 117-2031 FOR EQUIPMENT LAYOUT
6. SEE DRAWING 117-3012
5. PUMP TO BE ALLIS CHALMERS 1¼ x 1¼ H.H., C.I BODY, 316 SS TRIM, MECHANICAL SEAL, 30 GPM 110 FT. TDH. OR EQUAL
4. POWER SUPPLY FOR PUMP AND COMPRESSOR TO BE 240 V-3φ-60~ 3 WIRE
3. REFRIGERATION SYSTEM TO PROVIDE 36000 BTU/HR FLOW RATE 10 GPM, MAX TEMP OF EFFLUENT TO BE 75°F. HEAT EXCHANGER TO BE ROSS MODEL 502 SSCF OR EQUAL CONDENSER TO BE WATER COOLED.
2. PIPING FOR DEMINERALIZED WATER TO BE SCH 40 PVC, HIGH IMPACT, SOLVENT WELDED. THRD'D NIPPLES SCH 80
1. PIPING FOR CITY WATER TO BE GALVANIZED WITH 125# VALVES AND FITTINGS

NOTE

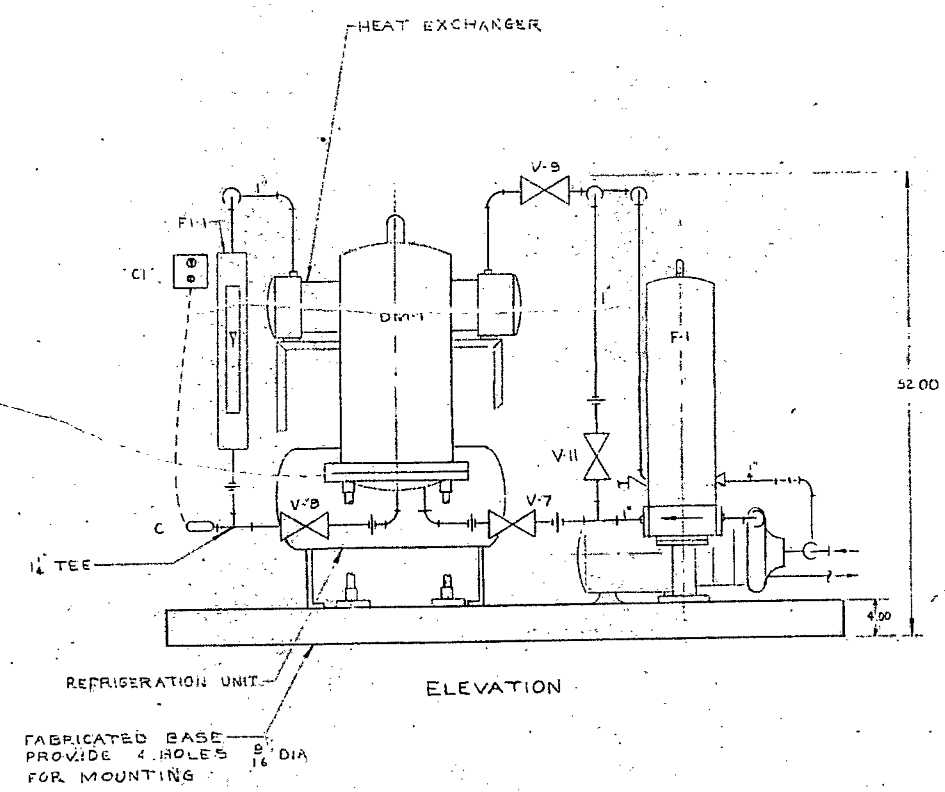
DATE	7-3-61	**LOCKHEED NUCLEAR PRODUCTS** LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
DESIGN	EM BURNETT	ADVANCED TRAINING REACTOR PROCESS SYSTEM FLOW DIAGRAM
CHK.		
MATL.		
STRESS		
SUPV.		
PROJ.		TOLERANCES: FRACTIONS ± .XX ± .03; ANGLES ± .XXX ± .010
APPD.	[signature]	SCALE:
APPLICATION / QTY. REQ'D		DWG. NO. 117-2030

U. R. EATON

REVISIONS							
ZONE	SYM	DESCRIPTION	SERIAL NO.	DFTMN	CHK	DATE	APPD



PROCESS SYSTEM
GENERAL ARRANGEMENT
PLAN



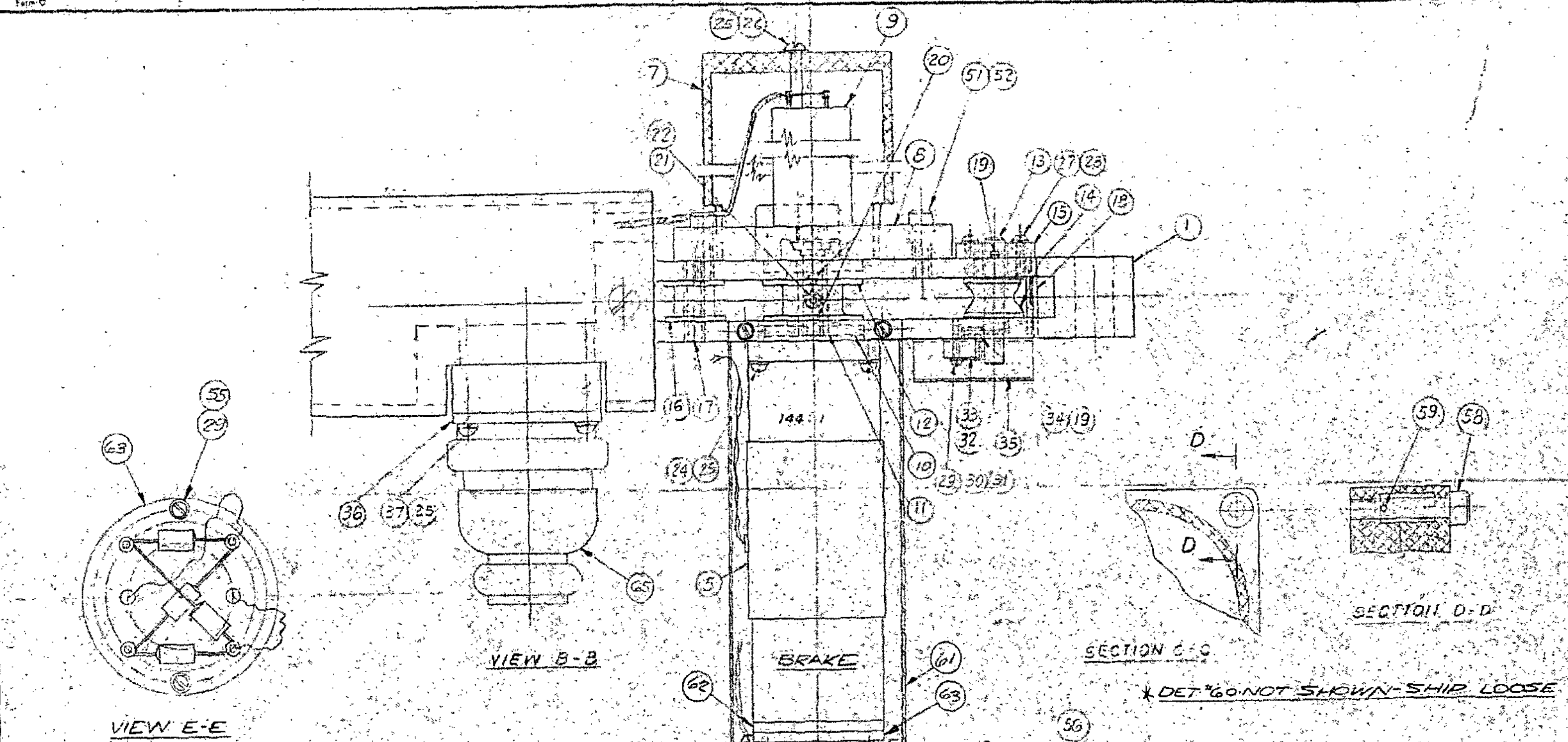
ELEVATION

6 SYSTEM SHALL BE DRAINED AND ALL OPENINGS SEALED WITH "CAPLUGS" BEFORE SHIPMENT.
5 DEMINERALIZER CARTRIDGE & FILTER CARTRIDGES MUST BE REMOVED BEFORE PRESSURE TEST
4 SYSTEM SHALL BE PRESSURE TESTED AT 75 PSIG
3. ALL EQUIPMENT AND PIPING SHALL BE ADEQUATELY SUPPORTED TO PREVENT DAMAGE DURING OPERATION AND SHIPPING
2. DRAWING SHOWS GENERAL ARRANGEMENT OF MAJOR COMPONENTS. LOCATIONS MAY BE CHANGED TO SUIT EQUIPMENT FURNISHED. SYSTEM SHALL BE CONTAINED WITHIN DIMENSIONS SHOWN
1. FOR COMPONENT SPECIFICATIONS SEE DRG 117-2030

REQ'D	PART NO.	ZONE	DESCRIPTION	MATERIAL	SIZE	MATL. SPEC.	HEAT TREAT	FINISH
LIST OF MATERIAL								

APPLICATION	QTY. REQ'D			
		DATE	7-3-61	LOCKHEED NUCLEAR PRODUCTS LOCKHEED AIRCRAFT CORP. MARIETTA, GEORGIA
		DESIGN	EM BURNETT	ADVANCED TRAINING REACTOR PROCESS SYSTEM EQUIPMENT LAYOUT
		CHK.		
		MATL.		
		STRESS		
		SUPV.		
		PROJ.		TOLERANCES / SCALE / DWG. NO.
		APPD.	[signature]	FRACTIONS ± / .XX ±.06 / 1/8 / 117-2031
		APPD.		ANGLES ± / .XXX ±.010

J. R. EATON

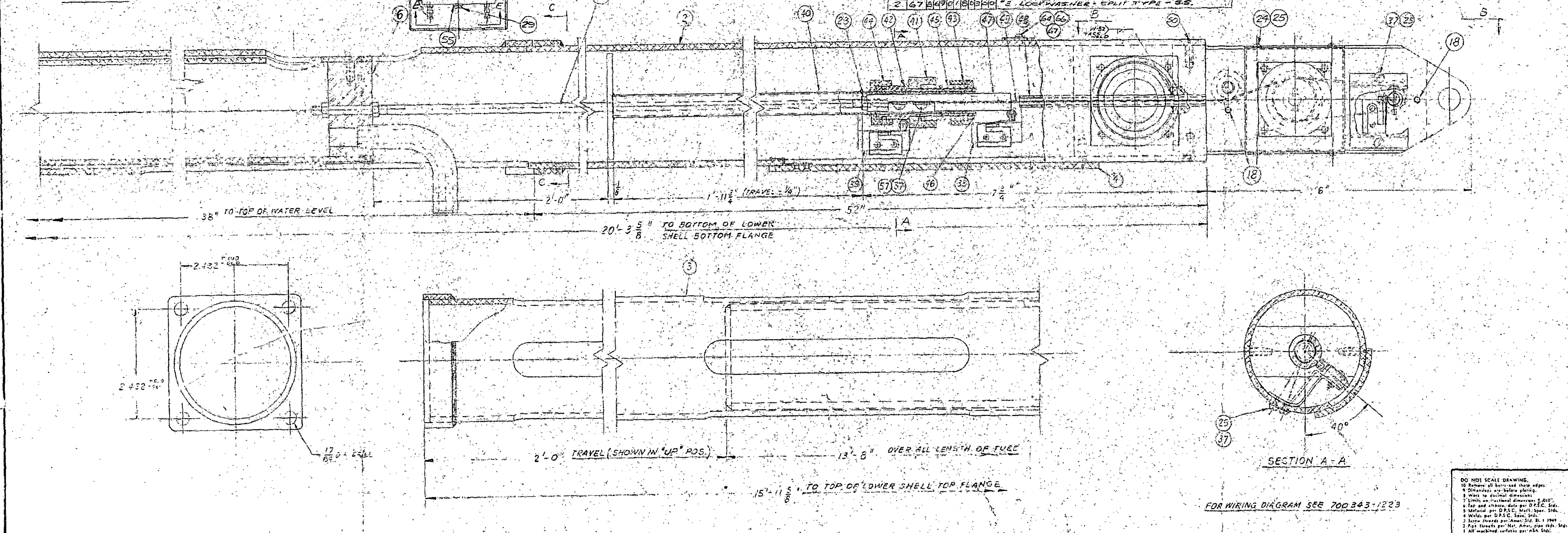


MATERIAL LIST

DWG. No.

No. REQD.	DET. No.	PART NUMBER (1 2 3 4 5 6 7 8 9 0)	DESCRIPTION
1	34	7002531114	SPRING - COMPRESSION
1	35	7002591118	COVER - JAM SWITCH
1	36	8123743003	RECEPTACLE - CANNON #FK-24-32 S
12	37	8656910606	#6-32 x 1/4 RD. HD. MACH. SCREW STNL. STL.
1	38	7002571029	LIMIT SWITCH ASS'Y. - UPPER
1	39	7002581028	LIMIT SWITCH ASS'Y. - LOWER
1	40	7002561020	TUBE ASS'Y. - SWITCH ACTUATOR
1	41	7002421118	BEARING BLOCK
1	42	7002351114	SLEEVE
1	43	7002391113	RETAINER - SPRING
2	44	8650140600	#6-32 x 3/16 SOC. HD. SET SCR. FLT. POINT STNL. STL.
1	45	7002311111	SPRING - COMPR.
1	46	9079200025	RETAINING RING - TRUARC #5100-62-W
1	47	7002551112	ACTUATOR - LIMIT SWITCH
1	48	7002541113	TUBE - CABLE GUIDE
1	49		1/32 - 3x7 WIRE ROPE (MACWHYTE) S.S. 56" LG
2	50	8656671012	#10-24 x 3/8 FLT. HD. MACH. SCREW STNL. STL.
2	51		#10-24 - 5/8 SOC. HD. CAP SCREW - STNL. STL.
2	52	8690401000	#10 - LOCK WASHER - INT. - SHKPRF. STAINL. STL.
6'	53		ALPHA WIRE CORP. SHIELDED CABLE TYPE 1333 [illegible]
10'	54	1A-2030-0	#22 GA. [illegible]
1	55	8690600200	#2 LOCKWASHER SHKPRF. INT. STN. STL.
1	56	7003451114	CONNECTING ROD
2	57	8690310600	#6 LOCKWASHER - SHAKEPROOF - EXT. S.S.
4	58	7001240113	CAP SCREW - FLANGE S.S.
4	59	8664651517	ROLL PIN - 79-012-062-0343 S.S.
1	60	8406300004	CAPACITOR - 2 MFD. - 330 V.A.C. SANGAMO ELECTRIC CO. TYPE P033-5
1	61	7003791022	MOTOR COVER ASS'Y
1	62	7003801110	RECTIFIER INSULATING STRIP
1	63	7003821027	RECTIFIER ASS'Y
1	64	1028000111	NAMEPLATE
1	65	8123723029	PLUG - CANNON #FK-24-21C
2	66	8656910309	#3-48 x 3/16 RD. HD. MACH. SCR. - S.S.
2	67	8670160300	#3 LOCKWASHER - SPLIT TYPE - S.S.

No. REQD.	DET. No.	PART NUMBER (1 2 3 4 5 6 7 8 9 0)	DESCRIPTION	MATL	LAB.
1	1	7003391138	BRACKET - DRIVE MOUNTING		
1	2	7002151034	SHELL ASS'Y - UPPER		
1	3	[illegible]	SHELL ASS'Y - LOWER		
1	4	7003231029	COVER ASS'Y.		
1	5	7002281129	MOTOR ASS'Y. - TYPE FC - GLOBE INDUST.		
1	6	7002321037	TUBE ASS'Y.		
1	7	7003331126	COVER ASS'Y. - HELIPOT		
1	8	7003551117	BRACKET - MOUNTING - HELIPOT		
1	9	7003501115	HELIPOT		
2	10	7001611118	BEARING		
1	11	7003401113	SHAFT - TUBULAR		
1	12	7003301111	DRUM		
1	13	7002151115	SHAFT - PIVOTING		
1	14	7002471113	ROLLER - JAM SWITCH ACTUATOR		
1	15	7002431117	RETAINER - PIN		
1	16	7002491111	ROLLER - GUIDE		
1	17	8636452532	ROLL PIN #79-048-250-1000 S.S.		
2	18	8636451932	ROLL PIN #79-026-125-1000 S.S.		
2	19	8636451518	ROLL PIN #79-012-062-0562 S.S.		
2	20	8636451712	ROLL PIN #79-032-094-0375 S.S.		
1	21	8660151005	#10-24 x 1/4 SOC. HD. SET SCREW NYLOK NUT MSCB1033-2P		
1	22	8660127408	#4-40 x 1/4 SOC. HD. SET SCREW - HALF DOG PT.		
3	23	8630201000	#10-24 HEX NUT - STAINL. STL.		
8	24	8656910612	#6-32 x 1/2 RD. HD. MACH. SCREW - STNL. STL.		
20	25	8690390600	#6 LOCKWASHER - SHKPRF. - INT. S.S.		
2	26		#6-32 x 3/4 RD. HD. MACH. SCREW - S.S.		
2	27	[illegible]	#4-40 x 3/8 RD. HD. MACH. SCREW STNL. STL.		
2	28	8690390400	#4 LOCKWASHER - SHKPRF. - INT. S.S.		
4	29	8656910216	#2-56 x 1/4 RD. HD. MACH. SCREW STNL. STL.		
2	30	8630200200	#2-56 HEX NUT " "		
2	31	8690250200	#2 LOCKWASHER - SPLIT " "		
1	32	8473420009	MICRO SWITCH #1SM1		
1	33	8473050005	ACTUATOR - MICRO SWITCH #JS-2		



FOR WIRING DIAGRAM SEE 700343-1223

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ±.010".
6 Tap and c'bore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. B1.1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

No.	DATE	REVISION	BY	APPVD	MF
A	1/5/62	52" WAS [illegible] TOP OF SHELL TO FLANGE JOINT WAS [illegible] FROM FLANGE [illegible] WATER LEVEL WAS [illegible] 15'-11 5/8" TO TOP OF LOWER SHELL TOP FLANGE WAS 12'-3 5/8" - 13'-8" LENGTH OF TUBE ASS'Y (DET #6) WAS 13'-6 5/8"	CWA	R.E.	
B		ADDED DET #55, 61, 62, 63, & 65; AT DET #29 QUAN. WAS 4; DET #25 QUAN. WAS 16; DET #23 QUAN. WAS 2; DET #65 WAS SHOWN	FHS		

No.	DATE	REVISION	BY	APPVD	MF
B	2/23/62	IN DOTTED: ADDED VIEW E-E	FHS	RE	
C	2/27/62	REVISED HELIPOT COVER - DET #7 WAS 700333-1021. DET #26 WAS 2 1/2 LONG. DETAIL #21 WAS 1/2 LONG	CWA	RE	
D	3/14/62	HARDWARE #ON DET. 32 WAS 847342-0006	OF	REW	
E	3/22/62	ADDED DET'S. #64, 66, & 67	RC	[illegible]	

RELEASED TO SHOP

MATERIAL SPECIFICATION | FINISHING SPEC. | HEAT TREAT

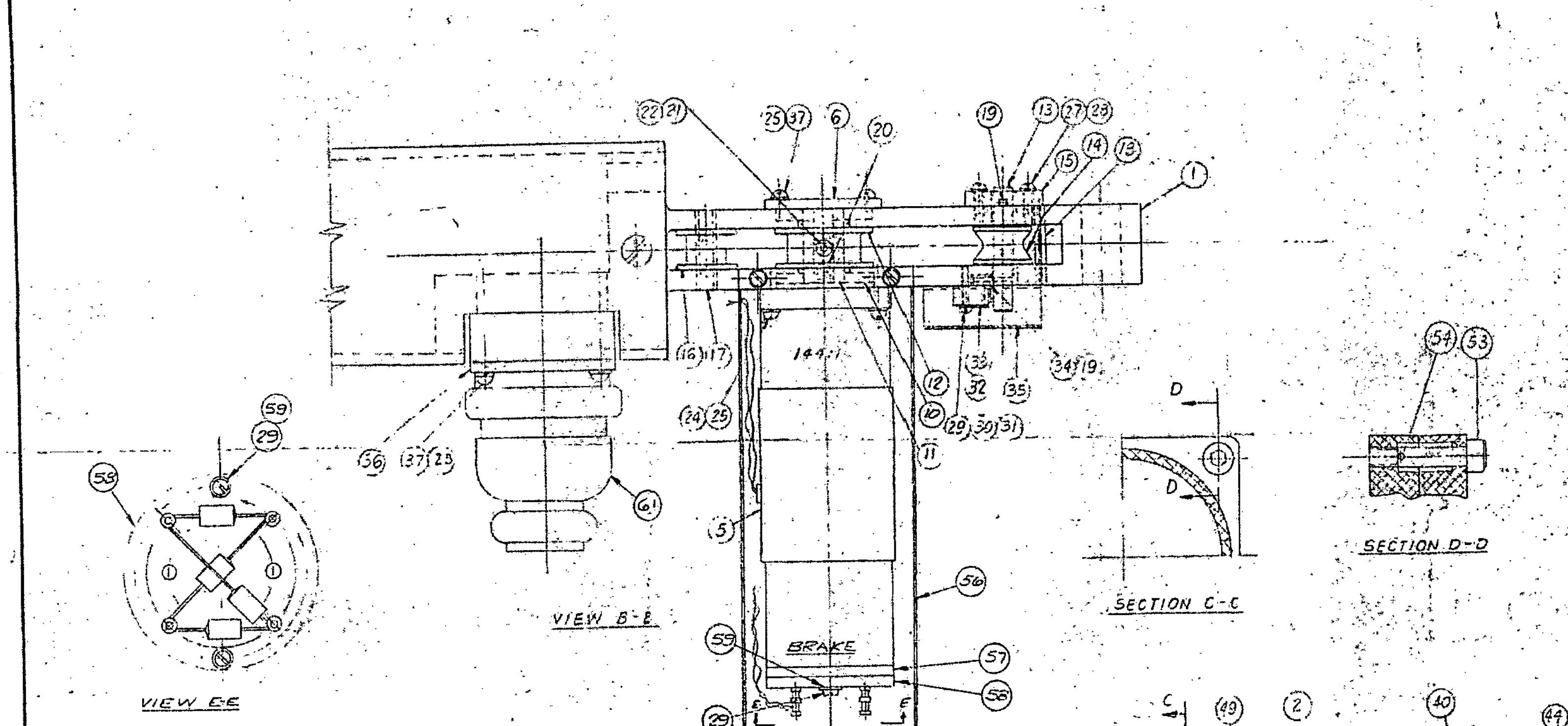
DRAWN BY [illegible] | APPVD BY [illegible]
CHECKED CWA | SCALE FULL

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

DRIVE ASS'Y.
FISSION CHAMBER

DATE STARTED | DATE FINISHED 11-29-61

DRAWING AND/OR PART NUMBER: 7002361041



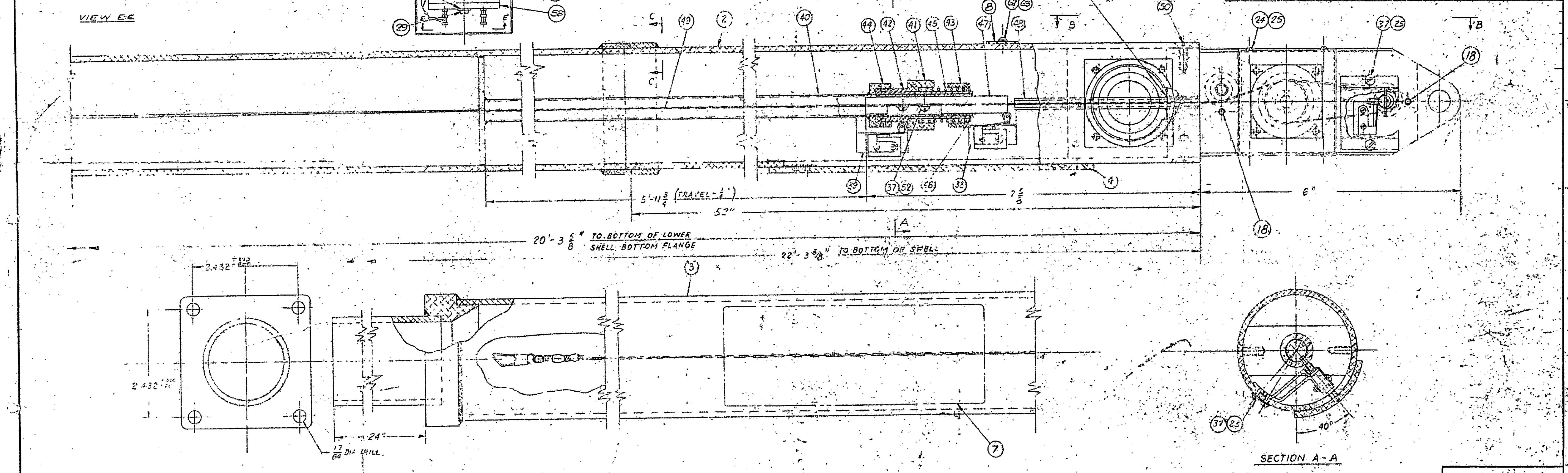
No. REQD.	DET. No.	PART NUMBER	DESCRIPTION
1	34	7002531114	SPRING - COMPRESSION
1	35	7002591118	COVER - JAM SWITCH
1	36	8423740009	RECEPTACLE - CANNON # FK-24-32 S
16	37	8656510609	#6-32 x 1/4 RD. HD. MACH. SCREW STNL. STL.
1	38	7002571029	LIMIT SWITCH ASS'Y. - UPPER
1	39	7002581028	LIMIT SWITCH ASS'Y. - LOWER
1	40	7003271025	TUBE ASS'Y. - SWITCH ACTUATOR
1	41	7002421118	BEARING BLOCK
1	42	7002351119	SLEEVE
1	43	7002391113	RETAINER - SPRING
2	44	8660140606	#6-32 x 3/16 SOC. HD SET SCR. FLT. POINT STNL. STL.
1	45	7002311111	SPRING - COMPR.
1	46	9079200025	RETAINING RING - TRUARC # 5100-62-W
1	47	7002551112	ACTUATOR - LIMIT SWITCH
1	48	7002541113	TUBE - CABLE GUIDE
1	49	7003381022	1/32 - 3x7 WIRE ROPE ASS'Y.
2	50	8656671012	#10-24 x 3/8 FLT. HD. MACH. SCREW STNL. STL.
10	51	UA-2030-D	22 GA. TEFLON INSUL. WIRE (COLORS PER WIRING DIAGRAM) FOR MOTOR WIRING
2	52	8690310600	#6 LOCKWASHER - SHAKEPR. EXT. - S.S.
4	53	7001240113	CAP SCREW - FLANGE S.S.
4	54	8636451511	ROLL PIN 79-012-062-0343 - S.S.
1	55*	8106300004	CAPACITOR - 5 MFD - 330 V A.C. - SANGAMO ELECTR. CO. TYPE 8033-5
1	56	7003791022	MOTOR COVER ASS'Y
1	57	7003801110	RECTIFIER INSULATING STRIP
1	58	7003821027	RECTIFIER ASS'Y
2	59	8690300200	#2 LOCKWASHER - SHKPRF. INT. STNL. STL.
	60		
1	61	8423720029	PLUG - CANNON # FK-24-21C
2	62	8656910306	#3-48 x 3/16 RD. HD. MACH. SCR. STN. STL.
2	63	8690180300	#3 LOCKWASHER - SPLIT TYPE STN. STL.

* DET. 55 NOT SHOWN - SHIP LOOSE

MATERIAL LIST

DWG. No.

No. REQD.	DET. No.	PART NUMBER	DESCRIPTION	MATL.	LAB.
1	1	7002411134	BRACKET - DRIVE MOUNTING		
1	2	7003181034	SHELL ASS'Y - UPPER		
1	3	7003251035	SHELL ASS'Y - LOWER		
1	4	7003231029	COVER ASS'Y.		
1	5	7002281124	MOTOR ASS'Y. - TYPE FC - GLOBE INDUST.		
1	6	7003221111	PLATE - BEARING RETAINER		
1	7	7003941024	COUNTER WEIGHT (PER ORDER)		
1	8	1026020111	NAMEPLATE		
2	10	7001611118	BEARING		
1	11	7002841116	SHAFT - TUBULAR		
1	12	7002351117	DRUM		
1	13	7002461115	SHAFT - PIVOTING		
1	14	7002471113	ROLLER - JAM SWITCH ACTUATOR		
1	15	7002431117	RETAINER - PIN		
1	16	7002491111	ROLLER - GUIDE		
1	17	8636452532	ROLL PIN # 79-048-250-1000 STNL. STL.		
1	18	8636451932	ROLL PIN # 79-028-125-1000		
2	19	8636451518	ROLL PIN # 79-012-062-0562		
1	20	8656451713	ROLL PIN # 79-022-094-0375		
1	21	8650161009	#10-24 x 1/4 NYLOCK SET SCREW # MSS CR1024-2P		
1	22	8660120408	#4-40 x 1/4 SOC. H.D. SET SCREW - HALF DOG PT. S.S.		
8	24	8656910616	#6-32 x 1/2 RD. HD. MACH. SCREW - STNL. STL.		
22	25	8690300600	#6 LOCKWASHER - SHKPRF. - INT. STNL. STL.		
2	27	8656691812	#4-40 x 3/8 RD. HD. MACH. SCREW STNL. STL.		
2	28	8690190400	#4 LOCKWASHER - SHKPRF. - INT. - CAD. PLT.		
4	29	8656910216	#2-56 x 1/2 RD. HD. MACH. SCREW STNL. STL.		
2	30	8630200200	#2-56 HEX NUT		
2	31	8690250200	#2 LOCKWASHER SPLIT		
1	32	8473920009	MICRO SWITCH # 1SM1		
1	33	8473050005	ACTUATOR - MICRO SWITCH # JS-2		



FOR WIRING DIAGRAM SEE DWG. 700342-1224

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ±.010".
6 Tap drill, etc. per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. B1.1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

No.	DATE	REVISION	BY	APPVD	MF
A	1/8/62	52" DIM. FROM TOP OF SHELL FLANGE JOINT WAS 1'4" ... (2FT LONG EXTENSION) BELOW BOTTOM FLANGE ... 2 3/4 O.D.	CWA	R.B.	
B	2/22/62	ADDED DET # 56, 57, 58, 59, & 61; AT DET # ... QUAN. WAS 4; AT DET # 29 QUAN WAS 2; DET # 61 WAS SHOWN IN DOTTED; AT DET # 25 ADDED QUAN; ADDED VIEW E-E	FHS	R.F.	
C	3/16/62	HARDWARE No. ON DET. 32 WAS 847342-0006	DF	REH	
D	3/22/62	DET # 2 WAS 700331-1037 - ADDED DETS # 28 & 29	RC	REH	

700237-1040

MATERIAL

REFERENCE DRAWINGS

RELEASED TO SHOP

MATERIAL SPECIFICATION | FINISHING SPEC. | HEAT TREAT

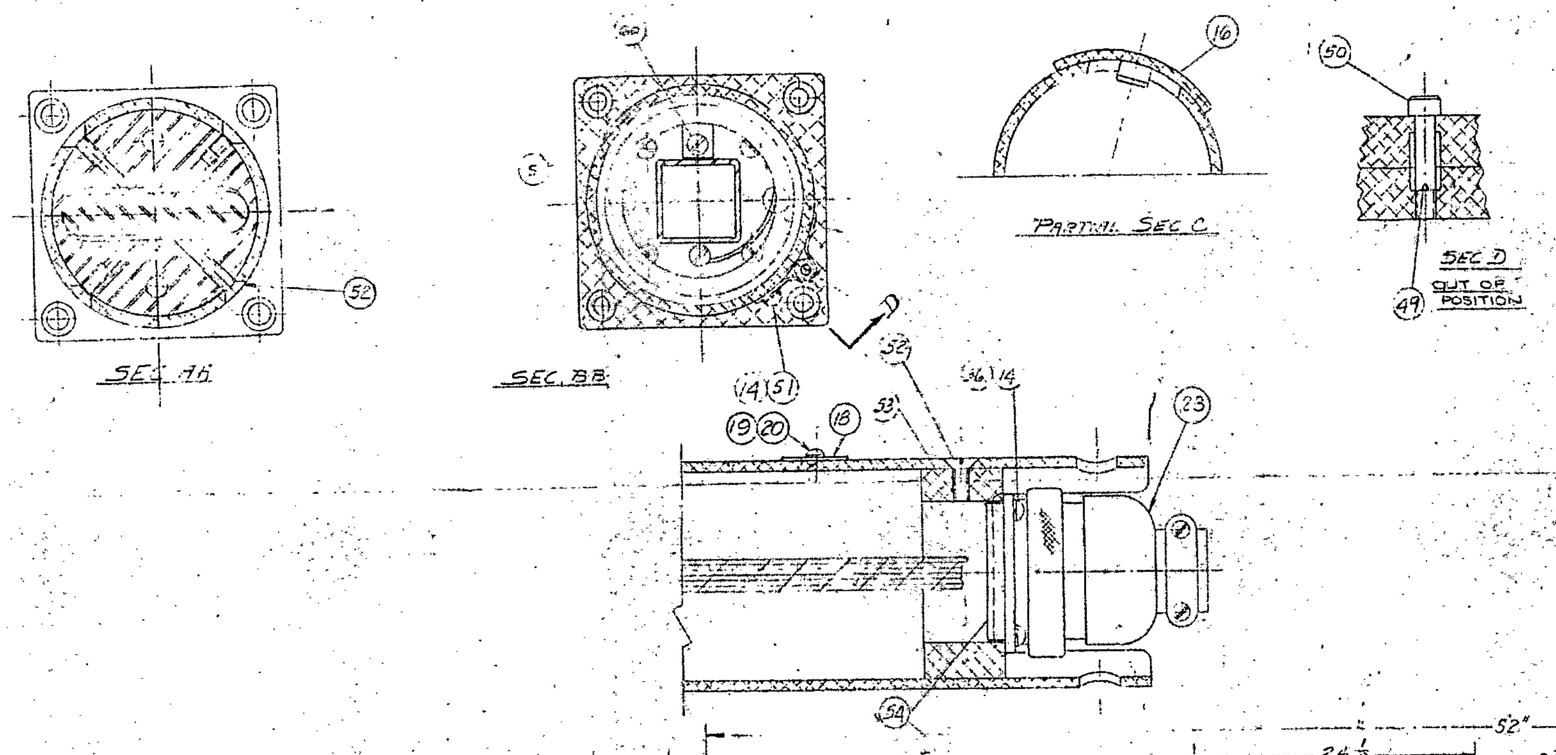
DRAWN BY | APPVD BY
CHECKED CWA | SCALE FULL

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

DRIVE ASS'Y. SOURCE

DRAWING AND/OR PART NUMBER: 7002371040

DATE STARTED | DATE FINISHED 11-30-61



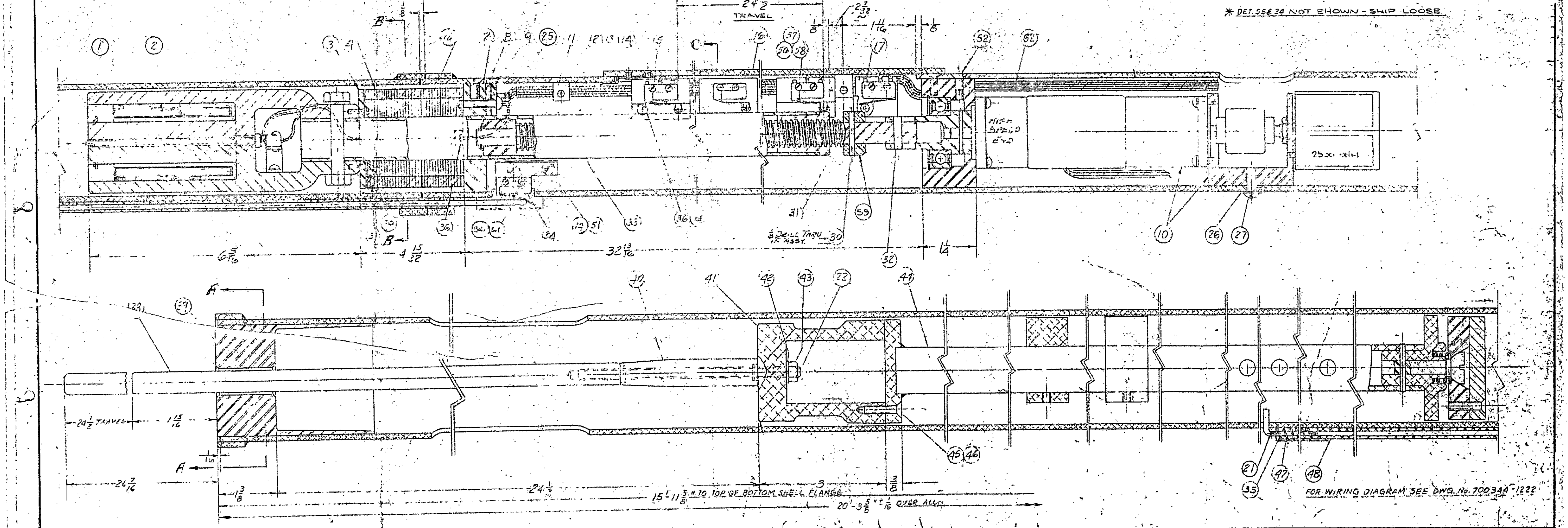
MATERIAL LIST — DWG NO

DET NO.	NO REQ	PART NUMBER	DESCRIPTION	MTL	LAB
30	1	863645-1932	ROLL PIN-79-028-125-1000	SS	
31	1	SK-8381-P	SHAFT ASSY-SCREW.		
32	1	863645-2224	ROLL PIN-79-040-187-0750	SS	
33	1	SK-8380-P	TUBE ASSY.-GUIDE		
34	1	EX-9507-P	SWITCH ASSY - SCRAM ROD		
35	1	EX-9508-D	SPRING - SCRAM ROD		
36	16	865691-0608	SCREW-RD HD. MACH-6-32 x 1/2	SS	
37	2	SK-10016-D	RETAINER - WIRE - CABLE		
38	1	700296-1121	ELEMENT-SHIM SAFETY		
39	1	700295-1123	SLEEVE - PISTON		
40	1	700297-1120	COLLAR - LOCATING		
41	1	700298-1129	PISTON		
42	2	869018-2500	LOCKWASHER-1/4" (SPLIT)	SS	
43	2	863020-2500	NUT-HEX-1/4-20	SS	
44	1	700299-1037	ROD ASSY - SPACER		
45	3	869031-1900	LOCKWASHER-SHKPF-#10-TYPE 15-1510-00	SS	
46	3	865657-1024	SCREW-FLAT HD. MACH-10-24X3/4	SS	
47	1	EX-9493-D	CLAMP-GUARD-SCRAM ROD		
48	1	EX-9496-D	GUARD-SCRAM-ROD		
49	4	863645-1511	ROLL PIN-79-012-062-0343	SS	
50	4	700124-0113	CAP SCREW-FLANGE.	SS	
51	6	865691-0606	SCREW-RD HD MACH-6-32X3/16	SS	
52	14	865687-1012	SCREW-FLAT HD. MACH.-10-24X3/8	S.S.	
53	1	SK-9027-D	RING - CONN. RET.		
54	1	842374-0009	RECEPTACLE-(CANNON) FK-24-32S		
*55	1	840630-0064	CAPACITOR-5.00MFD.-330V.AC (SEE SPEC)		
56	8	865691-0216	SCREW-RD.HD.MACH - 2-56 X1/2	SS	
57	8	863020-0200	NUT-HEX-2-56	SS	
58	8	869025-0200	LOCKWASHER #2 (SPLIT)	SS	
59	1	700140-0113	ACTUATOR-SWITCH-TOP		
60	2	700225-1112	RETAINER-WIRE-SCREW	SS	
61	1	700127-0110	ENCLOSURE-TOP-ACTUATOR ROD		
62	50'	UA-2030-D	22 GA. TEFLON INSUL WIRE (COLORS PER WIRING DIAGRAM)		

MATERIAL LIST — DWG. No.

DET. No.	No. REQD.	PART NUMBER	DESCRIPTION	MATL	LAB.
1	1	700315-1037	SHELL ASSY-BOTTOM		
2	1	7-13685	MAGNETIC CORE ASSY.		
3	1	700110-0119	RETAINER-WIRE-ACTUATOR		
4	1	700222-1121	CABLE-CONDUCTOR-FLEXIBLE		
5	4	865687-0612	SCREW-FLAT HD. MACH.-6-32X3/8		
6	1	700291-1035	SHELL ASSY. - TOP		
7	1	700117-0112	RING-GUIDE		
8	1	700106-0115	INSULATOR-TERMINAL		
9	1	700105-1023	RETAINER & TERMINAL ASSY.		
10	1	700293-1035	MOTOR & HELIPOT ASS'Y.		
11	3	SK-9026-D	WIRE CLIP		
12	9	863020-0600	NUT-HEX-6-32	SS	
13	3	865687-0612	SCREW-FLAT HD.MACH-6-32X3/8	SS	
14	17	869031-0600	LOCKWASHER-SHKPF-#6 EXT.	SS.	
15	3	700175-1028	SWITCH ASSY.		
16	1	700167-1028	COVER ASSY - SWITCH		
17	1	700111-1025	SWITCH ASSY - JAM		
18	1	102802-0111	NAME PLATE		
19	2	865691-0206	SCREW-RD. HD. MACH. #3-48x3/16	SS.	
20	2	869018-0300	#3 LOCKWASHER - SPLIT	SS.	
21	1	EX-3415-D	ROD - SCRAM		
22	2	700301-1116	ROD - CONNECTING		
23	1	842372-0029	CONNECTOR-RECEPT. CANNON # FK-24-21C	7/8	
*24	1		GE THYRECTOR DIODE TYPE # 6RS21SA6D6-PEAK VOLTS 310		
25	2	865691-0620	#6-32x5/8 RD.HD.MACH. SCREW	S.S.	
26	1	865691-1018	#10-24x9/16 RD. HD. MACH SCREW	SS	
27	1	869018-1000	#10 LOCK WASHER - SPLIT TYPE	SS	

* DET. 55 & 24 NOT SHOWN - SHIP LOOSE



FOR WIRING DIAGRAM SEE DWG. No. 700344-1222

No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF

UNLESS OTHERWISE SPECIFIED
DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ±.010".
6 Tap and c'bore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. B.1.1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.

700225-1044

REFERENCE DRAWINGS — MATERIAL — MATERIAL SPECIFICATION — FINISHING SPEC. — HEAT TREAT

DRAWN BY — APPVD BY — CHECKED — SCALE

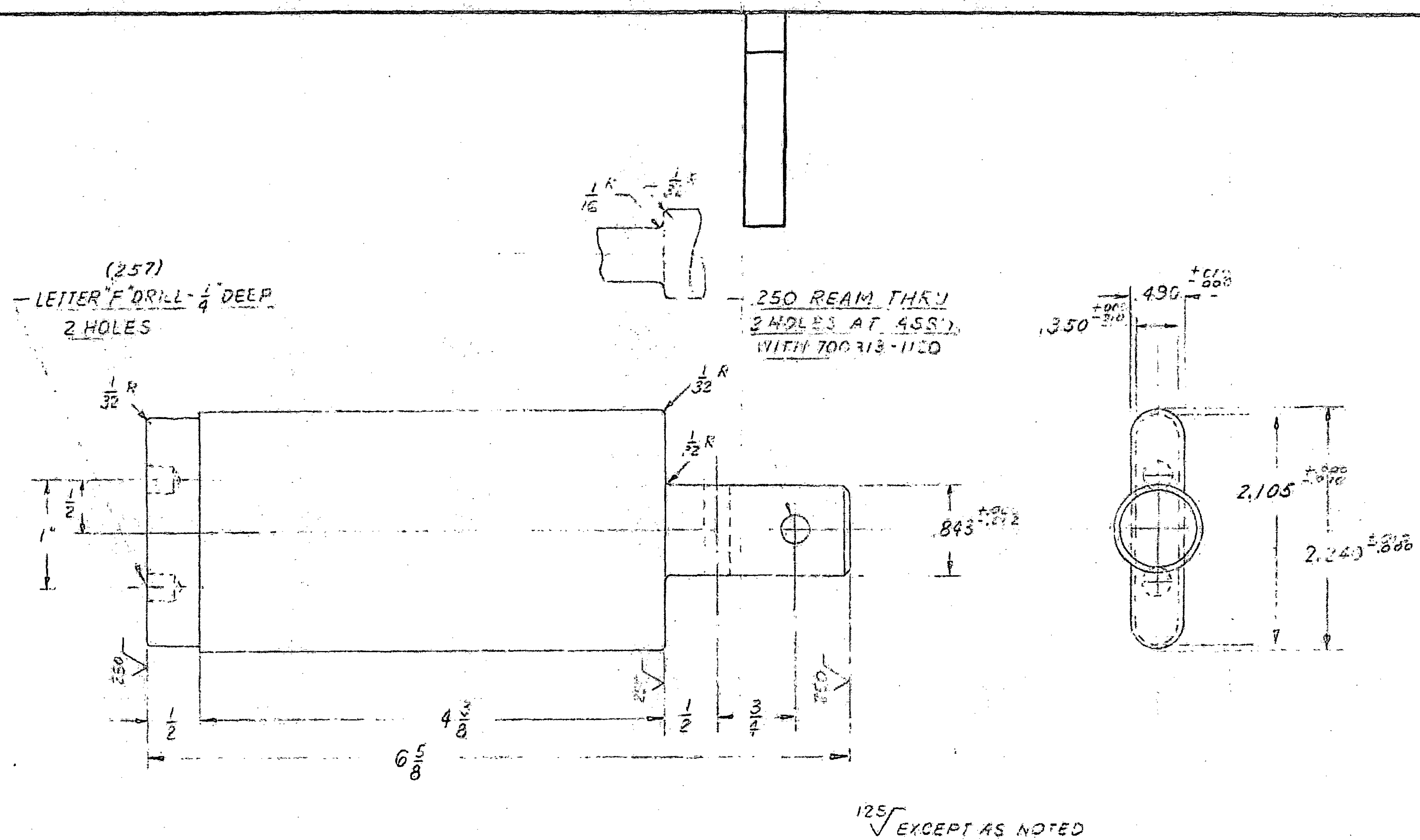
DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

TITLE: DRIVE ASS'Y. SHIM SAFETY ROD.

DATE 11-17-61

DRAWING AND/OR PART NUMBER: 700290-1044

Form P



DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ± .010".
6 Tap and ctrbore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. Bl. 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.

UNLESS OTHERWISE SPECIFIED

No.	DATE	REVISION	BY	APPVD	MF

MATERIAL	MATERIAL SPECIFICATION	FINISHING SPEC.	HEAT TREAT
ALUMINUM BAR STOCK	ALLOY - 2[illegible] - T4	CLEAR ANODIZE CHROMIC ACID METHOD	

REFERENCE DRAWINGS: [illegible]

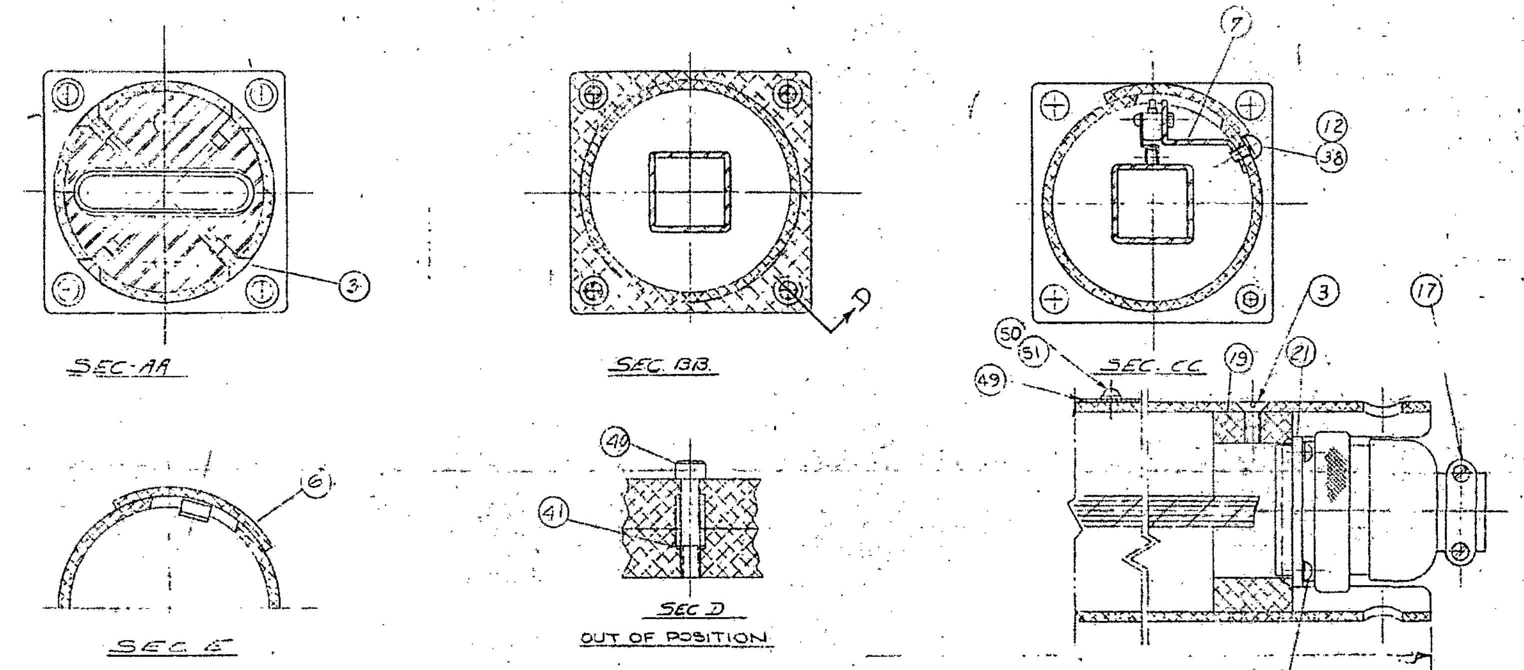
DRAWN BY: [illegible] | APPVD. BY: [illegible] | CHECKED: CWA | SCALE: FULL

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

PLUG - UPPER
REGULATING ROD

DATE STARTED:
DATE FINISHED 11-15 61

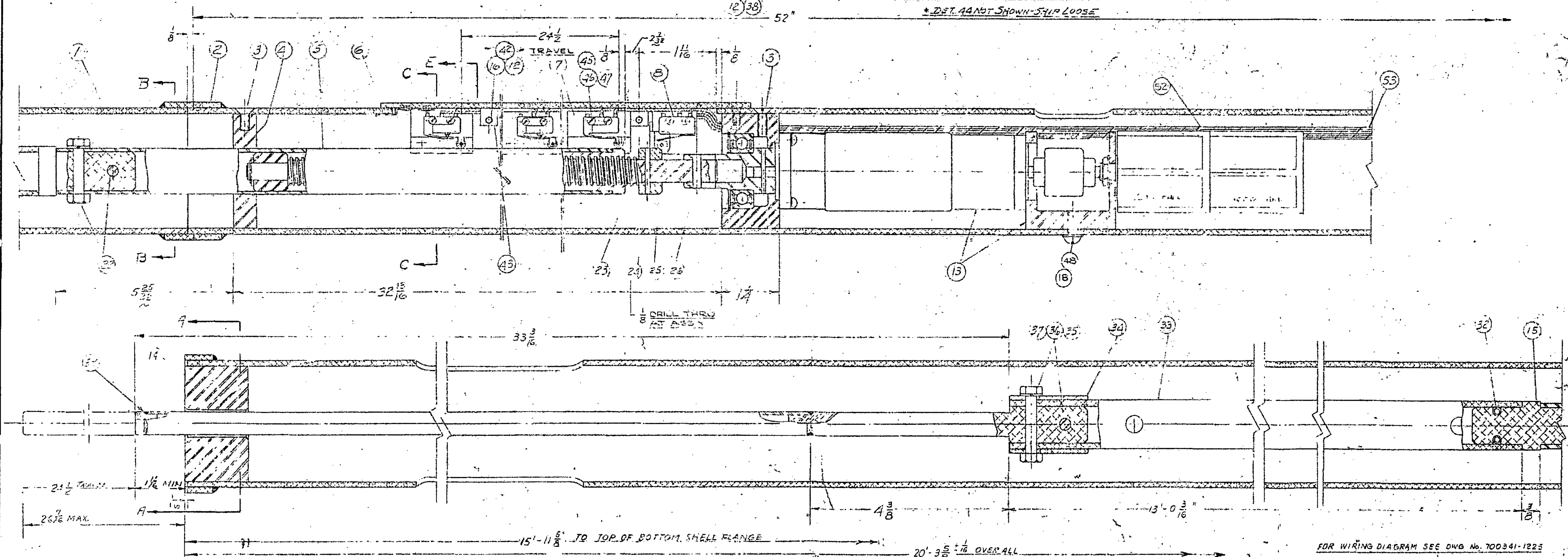
DRAWING AND/OR PART NUMBER: 7[illegible] - [illegible]10[illegible]



MATERIAL LIST				DWG NO	
DET NO	NO REQ	PART NUMBER	DESCRIPTION	MAT	LAB
32	2	863645-2230	ROLL PIN-79-040-187-0937	SS	
33	1	700313-1120	ROD-REG. MOUNTING PIPE		
34	1	700189-0115	COUPLING-REG. ROD		
35	4	863020-2500	NUT-1/4-20	SS	
36	2	865006-2536	CAP SCREW-HEX. HD 1/4-20 x 1 1/2	SS	
37	4	869018-2500	LOCKWASHER-1/4 (SPLIT)	SS	
38	12	865691-0608	SCREW-RD. HD. MACH.-6-32 x 1/4	SS	
39	1	700302-1032	ROD ASSY-REGULATING COMP		
40	4	700124-0113	CAP SCREW-FLANGE	SS	
41	4	863645-1511	ROLL PIN-79-012-062-0343	SS	
42	3	865687-0612	SCREW-FLAT HD. MACH-6-32 x 3/8	SS	
43	3	SK-9026-D	WIRE CLIP	SS	
44	1		CAPACITOR-6.00MFD.-330V-AC.-(SEE SPEC)	SS	
45	8	865691-0216	SCREW-RD. HD. MACH. 2-56 x 1/4	SS	
46	8	863020-0200	NUT-HEX.-2-56	SS	
47	8	869025-0200	LOCKWASHER #2 (SPLIT)	SS	
48	1	869018-1000	LOCKWASHER #10 (SPLIT)	SS	
49	1	102802-0111	NAMEPLATE		
50	2	865691-0306	SCREW-RD HD MACH. 3-48-3/16 LG	SS	
51	2	869018-0300	LOCKWASHER #3 (SPLIT)	SS	
52	6'		ALPHA WIRE CORP. SHIELDED CABLE TYPE 1878 DESIGNATION-M.W.C. 18(19)30 SIZE 18, .120" O.D.		
53	50'	UA-2030-D	22 GA. TEFLON INSUL. WIRE (COLORS PER WIRING DIAGRAM)		

MATERIAL LIST				DWG. No.	
DET. No.	No. REQD.	PART NUMBER	DESCRIPTION	MATL	LAB.
1	1	700312-1930	SHELL ASSY-BOTTOM		
2	1	700306-1036	SHELL ASSY-TOP		
3	13	865687-1012	SCREW-FLAT HD. MACH.-10-24 x 3/8	SS	
4	1	700117-0112	RING-GUIDE		
5	1	SK-8419-P	TUBE ASSY.-GUIDE		
6	1	700167-1028	COVER ASSY.-SWITCH		
7	3	700175-1028	SWITCH ASSY.-		
8	1	700111-1025	SWITCH ASSY-JAM		
9					
10					
11					
12	15	869031-0600	LOCKWASHER-SHAKEPR. #6-EXT.	SS	
13	1	700310-1024	MOTOR & HELIPOT ASS'Y.		
14	1	700311-1114	PLATE-GUIDE-ROD		
15	1	700130-0112	PLUG-CONN. REG. ROD		
16	3	863020-0600	NUT-HEX-6-32	SS	
17		842372-0029	CONNECTOR-CANNON # FK-24-21 C 7/S		
18	1	865691-1018	SCREW-RD. HD. MACH.-10-24 x 5/16	SS	
19	1	SK-9027-D	RING-CONN. RET.		
21	1	842374-0003	RECEPTACLE (CANNON) FK-24-32S		
22	2	865006-2524	CAP SCREW-HEX. HD.-1/4-20 x 1 1/4	SS	
23	1	SK-8361-P	SHAFT ASSY-SCREW		
24	1	863645-1932	ROLL PIN-79-028-125-1000	SS	
25	1	700140-0113	ACTUATOR-SWITCH-TOP		
26	1	863645-2224	ROLL PIN-79-040-187-0750	SS	

* DET. 44 NOT SHOWN-SHIP LOOSE



FOR WIRING DIAGRAM SEE DWG No. 700341-1225

No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF	No.	DATE	REVISION	BY	APPVD	MF

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ± .010".
6 Tap and counterbore data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. BL 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

700224-1045

MATERIAL | REFERENCE DRAWINGS | MATERIAL SPECIFICATION | FINISHING SPEC. | HEAT TREAT

CHECKED: 1 W.A. | SCALE 12-1-0

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

TITLE: DRIVE ASS'Y. REGULATING ROD

DATE 11-15-61

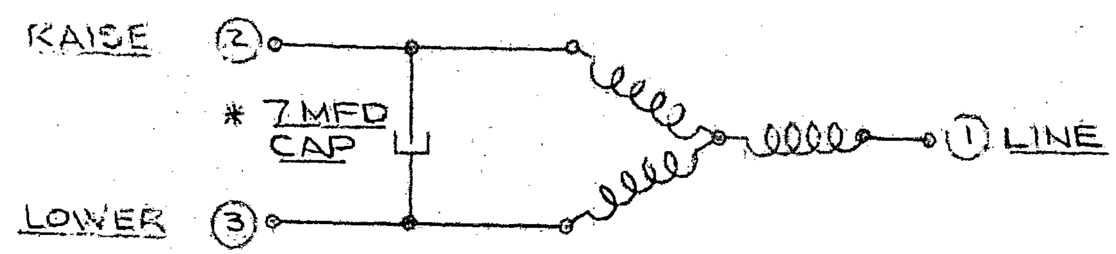
DRAWING AND/OR PART NUMBER: 7003071045

Form P

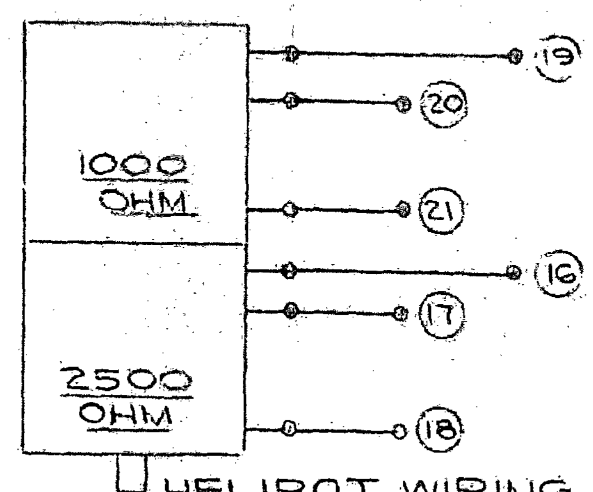
7003411225

PIN NUMBERS	DESIGNATION
1	MOTOR - YELLOW - (LINE)
2	MOTOR - WHITE - (RAISE)
3	MOTOR - BLUE - (LOWER)
4	JAM SWITCH - (COMMON) - WHITE
5	JAM SWITCH - N.O. - BROWN
6	JAM SWITCH - N.C. - RED
7	UPPER SWITCH - (COMMON) - VIOLET
8	UPPER SWITCH - N.O. - GRAY
9	UPPER SWITCH - N.C. - BLUE
10	2/3 UP SWITCH - (COMMON) - GREEN
11	2/3 UP SWITCH - N.O. - ORANGE
12	2/3 UP SWITCH - N.C. - RED
13	LOWER SWITCH - (COMMON) - GREEN
14	LOWER SWITCH - N.O. - GREEN
15	LOWER SWITCH - N.C. - RED
16	2500 OHM HELIPOT (SLIDER) (SHIELDED WIRE)
17	2500 OHM HELIPOT (UP) (SHIELDED WIRE)
18	2500 OHM HELIPOT (DOWN) (SHIELDED WIRE)
19	1000 OHM HELIPOT (SLIDER) (SHIELDED WIRE)
20	1000 OHM HELIPOT (UP) (SHIELDED WIRE)
21	1000 OHM HELIPOT (DOWN) (SHIELDED WIRE)
22	
23	GROUND TO MECHANISM.
24	

* 7 MFD. CAPACITOR SUPPLIED BY D.P.S. MOUNTED REMOTE TO DRIVE IN CUSTOMERS PANEL.



MOTOR CIRCUIT



HELIPOT WIRING

NOTES:

1. SLIDER MOVES TOWARD (UP) - TERMINAL № 20 ON 1000 OHM HELIPOT & TERMINAL № 17 ON 2500 OHM HELIPOT WHEN ROD MOVES "UP".
2. SLIDER MOVES TOWARD (DOWN) - TERMINAL № 21 ON 1000 OHM HELIPOT & TERMINAL № 18 ON 2500 OHM HELIPOT WHEN ROD MOVES "DOWN".

LOCKHEED AIRCRAFT - D.P.S. ORDER № 21039
PURDUE UNIVERSITY
CONNECTOR (DRIVE) - # FK-24-32S

No.	DATE	REVISION	BY	APPVD	MF
A	1-8-62	ADD "GROUND TO MECH." TO PIN # 23	EWA	R.E	

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ± .010".
6 Tap and ctrbore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. Bl. 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

MATERIAL	MATERIAL SPECIFICATION	FINISHING SPEC.	HEAT TREAT

REFERENCE DRAWINGS: 700307 - 104.5

DRAWN BY: CWM | APPVD. BY: R.E | CHECKED: R.E | SCALE: NONE

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

WIRING DIAGRAM FOR REGULATING ROD

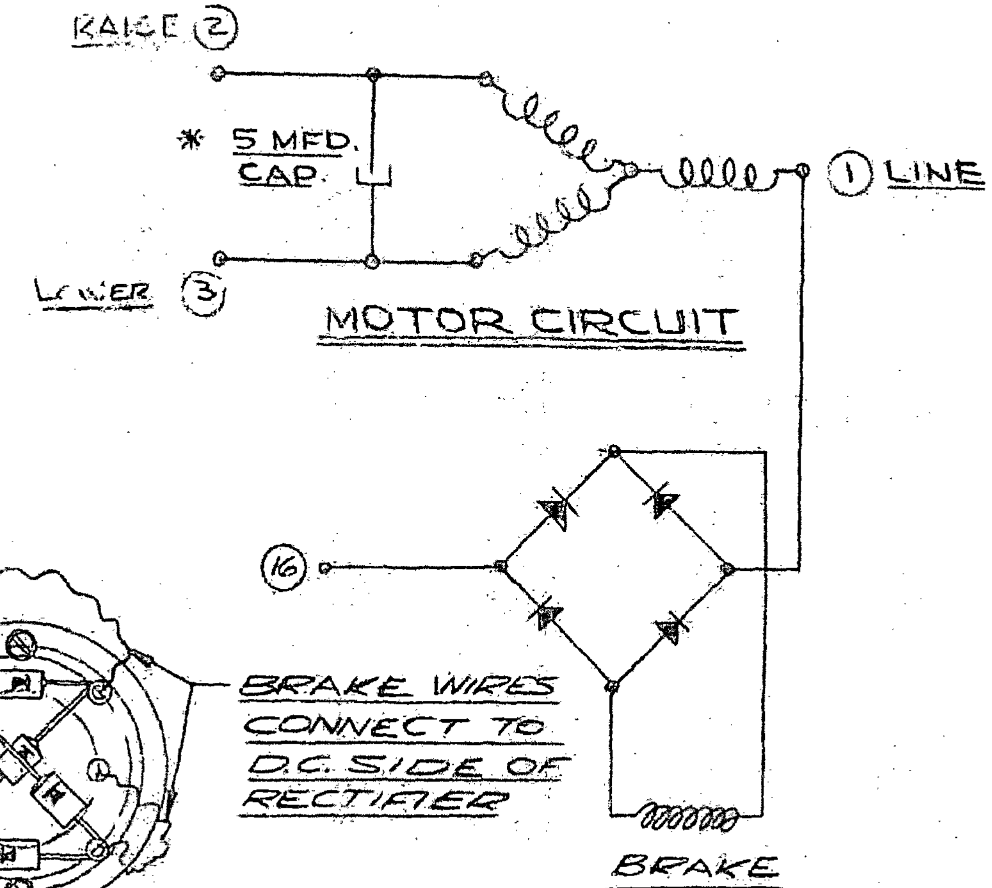
DATE STARTED: | DATE FINISHED: 11-17-61

1	2	3	4	5	6	7	8	9	0
7	0	0	3	4	1	1	2	2	5

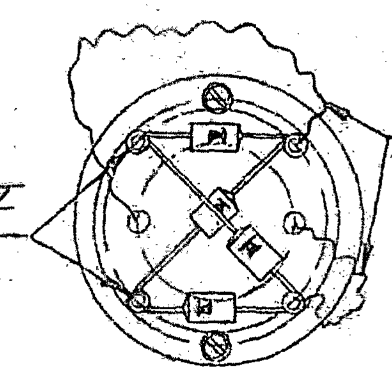
DRAWING AND/OR PART NUMBER

Form P

PIN NUMBER	DESIGNATION
1.	MOTOR - BRAKE-YELLOW-(LINE)
2.	MOTOR - BLUE - (RAISE)
3.	MOTOR - WHITE - (LOWER)
4.	SAFETY SWITCH-(COMMON)-GREEN
5.	SAFETY SWITCH-N.O. - ORANGE
6.	SAFETY SWITCH-N.C. - BLACK
7.	UPPER SWITCH-(COMMON)-VIOLET
8.	UPPER SWITCH - N.O. - GRAY
9.	UPPER SWITCH - N.C. - BLUE
10.	LOWER SWITCH-(COMMON)- BROWN
11.	LOWER SWITCH - N.O. - RED
12.	LOWER SWITCH - N.C. - ORANGE
13	
14	
15	
16	BRAKE-RED
23	GROUND TO MECHANISM
24	



A.C. SIDE OF RECTIFIER-CONNECT TO PINS #1 & 16



BRAKE WIRES CONNECT TO D.C. SIDE OF RECTIFIER

REAR VIEW OF MOTOR

* 5 MFD CAPACITOR SUPPLIED BY D.P.S. MOUNTED REMOTE IN CUSTOMER'S PANEL.

LOCKHEED AIRCRAFT - D.P.S. ORDER Nº 21039
(PURDUE UNIVERSITY)
CONNECTOR-(DRIVE) - # FK-24-32S

No.	DATE	REVISION	BY	APPVD	MF
A	1-8-62	ADDED "GROUND TO MECHANISM" TO PIN #23	CWM	R.E.	
B	2-23-62	ADDED "BRAKE" TO PIN #1 & TO MOTOR CIRCUIT; ADDED PIN #16 & REAR VIEW OF MOTOR	FHS	P.U.	
C	3-?-62	PIN NO. 2 WAS WHITE; NO. 3 WAS BLUE	RC	[illegible]	

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ± .010".
6 Tap and ctrbore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. Bl. 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

MATERIAL	MATERIAL SPECIFICATION	FINISHING SPEC.	HEAT TREAT
700237-1040			

REFERENCE DRAWINGS

DRAWN BY: CWM — APPVD. BY: R.E. — CHECKED: CWA — SCALE: NONE

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

WIRING DIAGRAM FOR SOURCE DRIVE

DATE STARTED — DATE FINISHED 11-17-61

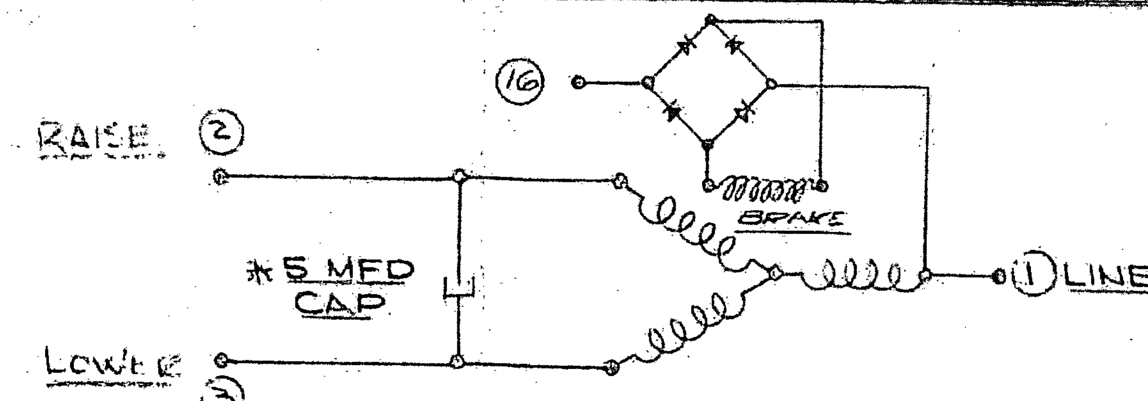
1	2	3	4	5	6	7	8	9	0
7	0	0	3	4	2	1	2	2	4

DRAWING AND/OR PART NUMBER

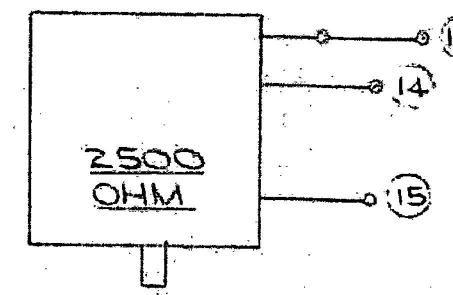
Form P

PIN NUMBERS	DESIGNATION
1.	MOTOR - BRAKE-YELLOW-(LINE)
2.	MOTOR - WHITE - (RAISE)
3	MOTOR - BLUE - (LOWER)
4.	SAFETY SWITCH - (COMMON) - GREEN
5	SAFETY SWITCH - N.O. - ORANGE
6	SAFETY SWITCH - N.C. BLACK
7.	UPPER SWITCH - (COMMON) - VIOLET
8.	UPPER SWITCH - N.O. - GRAY
9	UPPER SWITCH - N.C. - BLUE
10	LOWER SWITCH (COMMON) - BROWN
11	LOWER SWITCH - N.O. - RED
12	LOWER SWITCH - N.C. - ORANGE
13	2500 ohm HELIPOT (SLIDER) - (SHIELDED WIRE)
14	2500 ohm HELIPOT (UP) - (SHIELDED WIRE)
15	2500 ohm HELIPOT (DOWN) - (SHIELDED WIRE)
16	BRAKE - RED
23	GROUND TO MECHANISM
24.	

* 5 MFD. CAPACITOR SUPPLIED BY D.P.S. MOUNTED REMOTE IN CUSTOMERS PANEL.



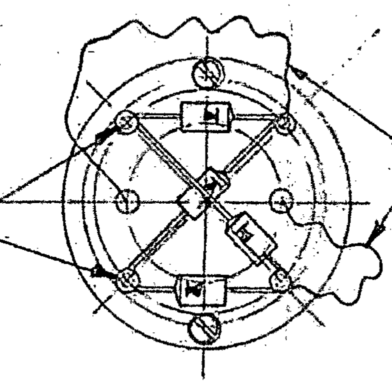
MOTOR CIRCUIT



HELIPOT WIRING

NOTE
1. SLIDER MOVES TOWARD (UP) TERMINAL Nº 14 WHEN ROD MOVES UP
2. SLIDER MOVES TOWARD (DOWN) TERMINAL Nº 15 WHEN ROD MOVES DOWN.

A.C. SIDE OF RECTIFIER-CONNECT TO PINS #1 & 16



BRAKE WIRES CONNECT TO D.C. SIDE OF RECTIFIER

REAR VIEW OF MOTOR

LOCKHEED AIRCRAFT - D.P.S ORDER Nº 21039
PURDUE UNIVERSITY
CONNECTOR (DRIVE) #FK-24-32S

No.	DATE	REVISION	BY	APPVD	MF
A	1-5-62	ADDED "GROUND TO MECHANISM" TO PIN #23	CWA	RE	
B	2-23-62	ADDED BRAKE TO PIN #1 TO MOTOR CIRCUIT; ADDED PIN #16 & REAR VIEW OF MOTOR	FHS	PJ	

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ±.010".
6 Tap and ctrbore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. Bl. 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

MATERIAL		MATERIAL SPECIFICATION	FINISHING SPEC.	HEAT TREAT
700236	1041	DRAWN BY CWM; APP'D. BY RE; CHECKED CWA; SCALE NONE	DIAMOND POWER SPECIALTY CORP. LANCASTER, OHIO	
REFERENCE DRAWINGS		DATE STARTED	DATE FINISHED	

WIRING DIAGRAM FOR FISSION CHAMBER DRIVE

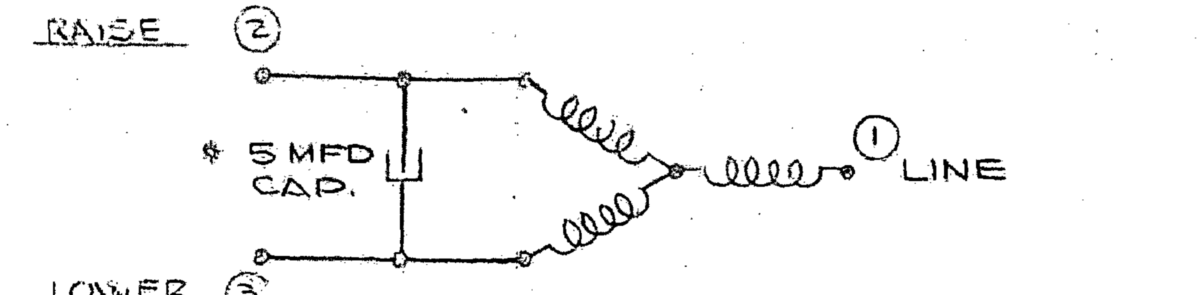
1	2	3	4	5	6	7	8	9	0
7	0	0	3	4	3	1	2	2	3

DRAWING AND/OR PART NUMBER

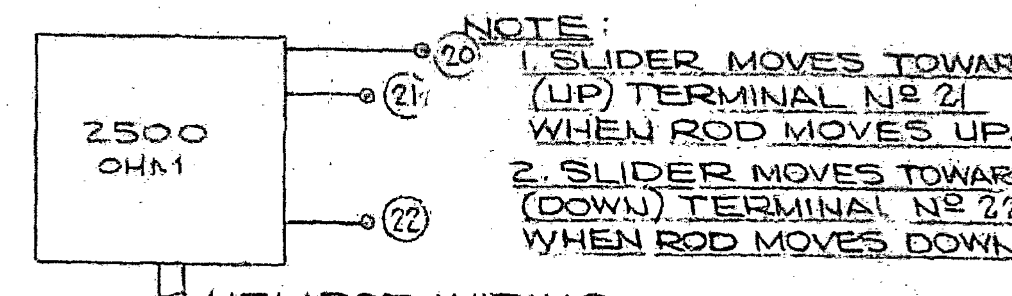
Form P

PIN NUMBERS	DESIGNATION
1.	MOTOR - YELLOW - (LINE)
2.	MOTOR - WHITE - (RAISE)
3.	MOTOR - BLUE - (LOWER)
4.	JAM SWITCH (COMMON) WHITE
5.	JAM SWITCH - N.O. - BROWN
6.	JAM SWITCH - N.C. - RED
7	UPPER SWITCH - (COMMON) - VIOLET
8	UPPER SWITCH - N.O. - GRAY
9	UPPER SWITCH - N.C - BLUE
10.	1/2 UP SWITCH - (COMMON) - GREEN
11.	1/2 UP SWITCH - N.O. - ORANGE
12.	1/2 UP SWITCH - N.C. - BLACK
13.	LOWER SWITCH - (COMMON) - YELLOW
14.	LOWER SWITCH - N.O. - GREEN
15.	LOWER SWITCH - N.C. - RED
16.	ENGAGE SWITCH - (COMMON) RED
17	ENGAGE SWITCH - N.O. - YELLOW
18.	BOTTOM SWITCH - (COMMON) - WHITE
19.	BOTTOM SWITCH - N.C. - BLUE
20.	2500 OHM HELIPOT (SLIDER) - (SHIELDED WIRE)
21.	2500 OHM HELIPOT (UP) - (SHIELDED WIRE)
22.	2500 OHM HELIPOT (DOWN) - (SHIELDED WIRE)
23.	GROUND TO MECHANISM.
24.	COAXIAL - MAGNET - 2 WIRES - GREEN & ORANGE

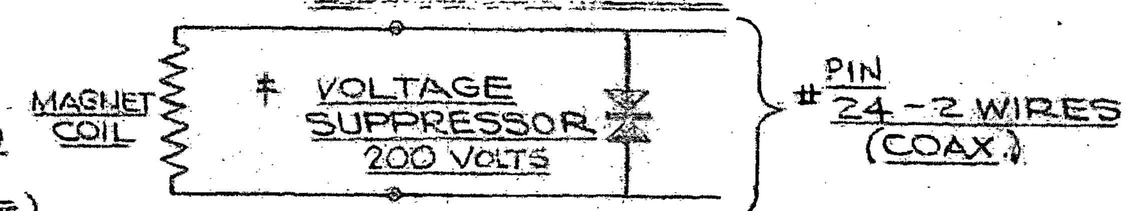
700344-1222



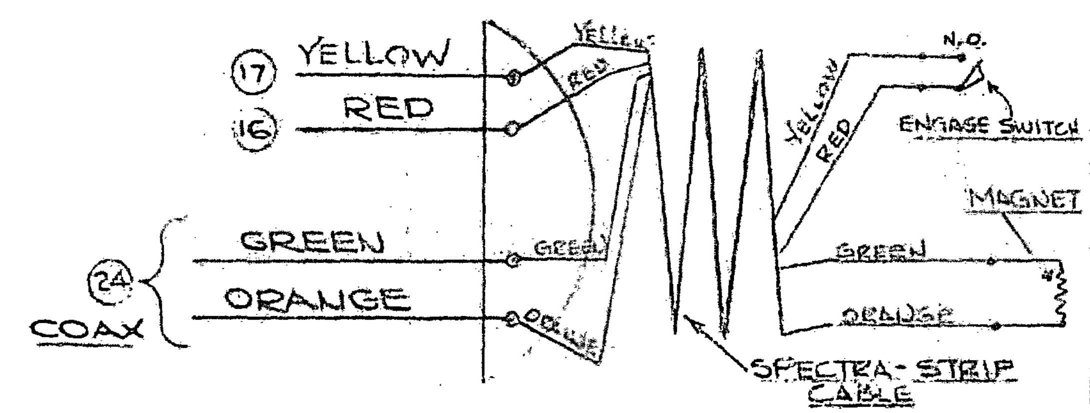
MOTOR CIRCUIT



HELIPOT WIRING



MAGNET COIL



LOWER TERMINAL BOARD

* 5 MFD. CAPACITOR SUPPLIED BY D.P.S MOUNTED REMOTE TO DRIVE IN CUSTOMERS PANEL.

‡ 200 VOLT SURGE SUPPRESSOR SUPPLIED BY D.P.S MOUNTED REMOTE TO DRIVE IN CUSTOMERS PANEL

LOCKHEED AIRCRAFT - D.P.S. ORDER # 21039
(PURDUE UNIVERSITY)
CONNECTOR (DRIVE) - # FK-24-32S

No.	DATE	REVISION	BY	APPVD	MF
A	1-B-62	ADDED "GROUND TO MECHANISM" TO PIN #23		R.E.	

DO NOT SCALE DRAWING.
10 Remove all burrs and sharp edges.
9 Dimensions are before plating.
8 Work to decimal dimensions.
7 Limits on fractional dimensions ± .010".
6 Tap and ctrbore. data per D.P.S.C. Stds.
5 Material per D.P.S.C. Mat'l. Spec. Stds.
4 Welds per D.P.S.C. Spec. Stds.
3 Screw threads per Amer. Std. Bl. 1 1949
2 Pipe threads per Nat. Amer. pipe thds. Stds.
1 All machined surfaces per ASA Stds.
UNLESS OTHERWISE SPECIFIED

MATERIAL: 700290-1044
REFERENCE DRAWINGS

MATERIAL SPECIFICATION | FINISHING SPEC. | HEAT TREAT

DRAWN BY: CWM | APPVD. BY: R.E. | CHECKED: [illegible] | SCALE: NONE

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

WIRING DIAGRAM FOR SHIM SAFETY ROD

1	2	3	4	5	6	7	8	9	0
7	0	0	3	4	4	1	2	2	2

DATE STARTED | DATE FINISHED 11-17-61 | DRAWING AND/OR PART NUMBER

APPENDIX 2: NATCON INFORMATION

A 2.1 Hot Channel Factors in the NATCON Code Version 1.0

The NATCON code version 1.0 [Ref. ANL/RERTR/TM-12] uses three hot channel factors (FQ, FW, FH). Using the source code and documentation, the factor FH used in NATCON is found to be the same as the factor FNUSLT used by E. E. Feldman. Table 1 shows the tolerances and uncertainties included in each of the six hot channel factors used by E. E. Feldman. The correspondence between the NATCON hot channel factors and E. E. Feldman's six hot channel factors is as follows.

Feldman's Hot Channel Factor		**NCATCON Input Variable**
System-wide Factors:		
FFLOW	a factor to account for the uncertainty in total reactor flow	FW (approximately)
FPOWER	a factor to account for the uncertainty in total reactor power	FQ
FNUSLT	a factor to account for the uncertainty in Nu number correlation	FH
Local Factors:		
FBULK	a hot channel factor for local bulk coolant temperature rise	FBULK (new input)
FFILM	a hot channel factor for local temperature rise across the coolant film	FFILM (new input)
FFLUX	a hot channel factor for local heat flux from cladding surface	FFLUX (new input)

Hot Channel Factors in the NATCON Code Version 2.0

Sections 2.1 and 2.2 develop, for laminar natural convection, two thermal-hydraulic relationships that are used in section 2.3 to obtain formulas for the hot channel factors from user-supplied manufacturing tolerances and measurement uncertainties. The results of section 2.3 are summarized here for convenience. The first three are local/random hot channel factors, and the last three are system-wide. An example of the use of these hot channel factors is given in section 4, with NATCON running instructions in section 3, and the new input description in section 5.

$$\text{FBULK} = 1 + \sqrt{\left\{(1+u_1)^{\frac{1}{2+\alpha}}(1+u_2)^{\frac{1}{2+\alpha}}\left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1\right\}^2 + u_6^2}$$

FBULK is higher (conservative) if the temperature dependence of water viscosity is ignored.

$$\text{FFILM} = 1+\sqrt{u_1^2+u_2^2+u_3^2+u_4^2+u_5^2}$$

$$\text{FFLUX} = 1+\sqrt{u_1^2+u_2^2+u_3^2+u_4^2}$$

FQ = 1 + u_7

FW = 1 + u_8

FH = 1 + u_9

where

- u_1 = Fractional uncertainty in neutronics calculation of power in a plate
- u_2 = Fractional uncertainty in U-235 mass per plate = $\Delta m / M$
- u_3 = Fractional uncertainty in local (at an axial position) fuel meat thickness
- u_4 = Fractional uncertainty in U-235 local (at an axial position) homogeneity
- u_5 = Fractional uncertainty in coolant channel thickness = $(t_{nc} - t_{hc}) / t_{nc}$
- u_6 = Fractional uncertainty in flow distribution among channels
- u_7 = Fractional uncertainty in reactor power measurement
- u_8 = Fractional uncertainty in flow due to uncertainty in friction factor
- u_9 = Fractional uncertainty in convective heat transfer coefficient, or in the Nu number correlation
- M = Nominal mass of U-235 per plate, gram
- Δm = Tolerance allowed in U-235 mass per plate, gram

The code obtains, for an input nominal reactor power CPWR, a thermal-hydraulic solution using the three systematic hot channel factors FW, FQ and FH. If the user-input reactor power is zero, then the code itself chooses the nominal power from a series of power levels (10 kW, 100 kW, 200 kW, and so on increasing in steps of 100 kW). This thermal-hydraulic calculation is done for a hot plate power of CPWR*FQ*(Radial power peaking factor RPEAK)/(Total number of fuel plates in standard and control assemblies). Also, the frictional resistance to flow is multiplied by FW^2, and the convective heat transfer coefficient found for laminar flow in a rectangular channel is divided by FH. The random hot channel factors FBULK, FFILM and FFLUX are not used in this solution.

Having obtained the above solution, the random hot channel factors FBULK, FFILM and FFLUX are applied to the temperatures obtained, using the following equations. The temperatures calculated with all six hot channel factors are printed after the above solution. The onset of nucleate boiling ratio, ONBR, is computed using the temperatures with all six hot channel factors applied (using the equation below). If the user-input nominal power is zero, then the last nominal power for which the code prints a solution is that at which the ONBR is 1.0.

$T_{i,6hcf} = T_0 + (T_i - T_0)*FBULK$

$T_{wall,i,6hcf} = T_{i,6hcf} + (T_{wall,i} - T_i)*FFILM$

$T_{max,i,6hcf} = T_{wall,i,6hcf} + (T_{max,i} - T_{wall,i})*FFLUX$

$$ONBR = \frac{\left(T_{incp,i} - T_0\right)}{\left(T_{wall,i,6hcf} - T_0\right)}$$

where

T_0 = Bulk water temperature at the coolant channel inlet, i.e., the pool temperature, °C

T_i = Bulk water temperature in node i of the channel with only systematic hot channel factors applied, °C

$T_{wall,i}$ = Cladding surface temperature in node i with only systematic hot channel factors applied, °C

$T_{max,i}$ = Fuel meat centerline temperature in node i with only systematic hot channel factors applied, °C

$T_{i,6hcf}$ = Bulk water temperature in node i of the channel with all six hot channel factors, °C

$T_{wall,i,6hcf}$ = Cladding surface temperature in node i with all six hot channel factors, °C

$T_{max,i,6hcf}$ = Fuel meat centerline temperature in node i with all six hot channel factors, C

$T_{incp,i}$ = Incipient boiling temperature in node i with only systematic hot channel factors applied, C

Flow Rate in a Coolant Channel versus Power of a Fuel Plate

NATCON is a laminar natural circulation code. The flow rate is calculated in the code by balancing the buoyancy pressure force to the laminar friction pressure drop. Following this concept, an analytical relationship is developed here (with some approximation) for the coolant flow rate in a single coolant channel in terms of the power generated in a fuel plate and the channel geometrical dimensions. The analytical relationship is needed for obtaining hot channel factors.

The hot channel factor FW used in the code to account for the uncertainty in coolant flow rate is actually applied to the laminar friction factor in the code, that is, the laminar friction factor is multiplied by FW^2. It is not applied directly to the flow rate. The relationship developed here explains how this technique works.

ρ_1 , T_1 at channel outlet

|

|

|

| L = Channel height containing hot coolant (hotter than pool), m

| P = Power in a single fuel plate or the two half plates, W

| W=Upward flow rate in a single channel, kg/s

|

|

ρ_0 , T_0 at channel inlet

Schematic of what the code analyses, that is, a single rectangular coolant channel heated by a half of a fuel plate on each side (right and left sides).

The above schematic shows what the code analyses, that is, a single rectangular coolant channel heated by a half of a fuel plate on each side (right and left sides). See Fig. 1 for details. The buoyancy pressure force is caused by the decrease in water density due to heating in the channel. The temperature dependence of water density can be written as

$$\rho(T) = \rho_0 - \rho_0 \beta (T - T_0) \qquad (1)$$

where

T_1 = Bulk water temperature at channel outlet, C

ΔT = $T_1 - T_0$ = Temperature rise in channel from inlet to outlet, C

ρ_0 = Water density at channel inlet, i.e., the water density in the pool, kg/m^3

β = Volumetric expansion coefficient of water, per C

$\bar{\rho}$ = Average coolant density in the channel, kg/m^3

L = Channel height that contains hotter coolant (hotter than pool), m. It is the sum of heat generating length of fuel plate, non-heat generating fuel plate length at top, and the assembly duct length above the top of fuel plate

g = Acceleration due to gravity, 9.8 m/s^2

The buoyancy pressure force is given by

$$\text{Buoyancy } \Delta p = (\rho_0 - \bar{\rho})\, g\, L \tag{2}$$

The average coolant density $\bar{\rho}$ is given by

$$\bar{\rho} = 0.5\,(\rho_0 + \rho_1) = \rho_0 - 0.5\,\rho_0\,\beta\,(T_1 - T_0) = \rho_0 - 0.5\,\rho_0\,\beta\,\Delta T \tag{3}$$

$$\text{Buoyancy } \Delta p = 0.5\,\rho_0\,\beta\,\Delta T\, g\, L \tag{4}$$

The coolant temperature rise ΔT can be written in terms of the input power P generated in a fuel plate, as shown by Eq. (5) below, and then the buoyancy Δp of Eq. (4) can be written in terms of the input power P, as shown by Eq. (6).

$$\Delta T = P / (W\, C_p) \tag{5}$$

$$\text{Buoyancy } \Delta p = \frac{\rho_0 \beta g L P}{2 W C_P} \tag{6}$$

Ignoring the minor losses at channel inlet and outlet, the laminar frictional pressure drop in the channel is written below as Eq. (9) after using the laminar friction factor given by Eq. (7), and after replacing the coolant velocity by mass flow rate using Eq. (8). The parameter C in Eq. (7) is a constant for a given channel cross section, but it depends upon the channel cross section aspect ratio width/thickness, and varies from 57 for aspect ratio 1.0 (square channel) to 96 for an infinite aspect ratio (infinitely wide channel).

$$f = C / R_e \tag{7}$$

$$W = \bar{\rho}\, A\, V \tag{8}$$

$$\text{Frictional } \Delta p = \frac{\bar{\rho} f L_C V^2}{2D} = \frac{C \bar{\mu} L_C W}{2 \bar{\rho} A D^2} \tag{9}$$

where

f = Moody friction factor for laminar flow in the channel

R_e = Reynolds number in the channel = $\bar{\rho}\, VD/\bar{\mu}$

A = Flow area of the channel cross section, m^2

D = Equivalent hydraulic diameter of the channel cross section, m

L_C = Total coolant channel length causing frictional pressure drop, m.

V = Coolant velocity averaged over the channel cross section, m/s

W = Coolant mass flow rate in the channel, kg/s

$\bar{\mu}$ = Average coolant dynamic viscosity in the channel, $N\text{-}s/m^2$

$\mu(T)$ = Temperature-dependent dynamic viscosity of water, $N\text{-}s/m^2$

μ_0 = $\mu(T_0)$ = Coolant dynamic viscosity at the channel inlet temperature T_0

For the PUR-1 reactor, the temperature dependence of the dynamic viscosity of water over the temperature range 27 °C ≤ T ≤ 50 °C can be approximated as follows.

$$\mu(T) = \mu(T_0)\,(1+T-T_0)^{-\alpha} \qquad (10)$$

where $\alpha = 0.12$, $T_0 = 27$ °C, $\mu(T_0) = 0.875\times10^{-3}$ $N\text{-}s/m^2$

The average coolant dynamic viscosity $\bar{\mu}$ used in Eq. (9) can be set equal to the viscosity at the average coolant temperature ($T_0 + 0.5\Delta T$) in the channel. Putting this temperature in Eq. (10), the average viscosity $\bar{\mu}$ is found to be

$$\bar{\mu} = \mu(T_0)\,(1+0.5\Delta T)^{-\alpha}. \qquad (11)$$

Equation (11) indicates that the average viscosity $\bar{\mu}$ can be set equal to $\mu(T_0)$ if ΔT is just a few °C (this is the case for the PUR-1 reactor at the operating power of 1 kW). If ΔT is greater than a few °C, i.e., $1 \ll 0.5\Delta T$ (this is the case for the PUR-1 reactor at an ONB power of about 100 kW), then Eq. (11) simplifies to the following.

$$\bar{\mu} = \mu(T_0)\,(0.5\Delta T)^{-\alpha} \qquad \text{if } \Delta T \gg 2\ °C \qquad (12a)$$

$$\bar{\mu} = \mu(T_0) \qquad \text{if } \Delta T \ll 2\ °C \qquad (12b)$$

Substituting Eq. (12a) into Eq. (9), the frictional Δp becomes

$$\text{Frictional } \Delta p = \frac{C\mu_0 (\Delta T)^{-\alpha} L_C W}{2^{1-\alpha}\bar{\rho} AD^2} = \frac{C\mu_0 L_C W}{2^{1-\alpha}\bar{\rho} AD^2}\left(\frac{WC_P}{P}\right)^{\alpha} \tag{13}$$

Equating the frictional Δp of Eq. (13) to the buoyancy Δp of Eq. (6) to find the steady-state coolant flow rate W in the channel, one obtains Eq. (14) below. Equation (14) can be rewritten as Eq. (15).

$$\frac{\rho_0 \beta g L P}{2WC_P} = \frac{C\mu_0 L_C W}{2^{1-\alpha}\bar{\rho} AD^2}\left(\frac{WC_P}{P}\right)^{\alpha} \tag{14}$$

$$W^{2+\alpha} = \frac{\rho_0 \bar{\rho} AD^2 \beta g L P^{1+\alpha}}{2^{\alpha} C\mu_0 L_C C_P^{1+\alpha}} \tag{15}$$

Equation (15) relates the fuel plate power to the channel flow rate in natural circulation. It is used to find the dependence of the flow rate on the parameter C in the laminar friction factor (at constant power). All parameters in this equation are constant ($\bar{\rho}$ is also practically constant) except the parameter C in the laminar friction factor. Based on Eq.(15), the relationship between the flow rate W and the parameter C is given by Eq. (16) below.

$$W \propto \left(\frac{1}{C}\right)^{\frac{1}{2+\alpha}} \tag{16}$$

Equation (16) shows that the friction factor parameter C is multiplied by a factor $(FW)^2$, the coolant flow rate W will be reduced by the factor $(FW)^{\frac{2}{2+\alpha}}$. This has been verified by actually running the NATCON code for the PUR-1 reactor. Since α is small (α = 0.12 for the PUR-1 reactor), 2/(2+α) is nearly 1.0, and the flow rate W is reduced approximately by the factor FW.

Bulk Coolant Temperature Rise versus Power of a Fuel Plate

Equation (5) expresses, for laminar natural circulation, the bulk coolant temperature rise in terms of fuel plate power, coolant flow rate and specific heat. Putting the value of flow rate obtained in Eq. (15) into Eq. (5), the bulk coolant temperature rise is given by Eq. (17) below, purely in terms of power and the geometrical dimensions of the channel. The right hand side of Eq. (17) is rearranged into two factors in Eq. (18), such that the second factor is sensitive to power and channel geometrical dimensions that usually have manufacturing tolerances and measurement uncertainties, and the first factor is insensitive to power and channel geometrical dimensions.

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C P}{C_P \rho_0 \bar{\rho} A D^2 \beta g L} \right)^{\frac{1}{2+\alpha}} \tag{17}$$

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C}{C_P \rho_0 \bar{\rho} \beta g L} \right)^{\frac{1}{2+\alpha}} \left(\frac{P}{AD^2} \right)^{\frac{1}{2+\alpha}} \tag{18}$$

The nominal flow area and hydraulic diameter of a rectangular coolant channel are given by

$$A = t_{nc} w_{nc} \tag{19}$$

$$P_w = 2 (t_{nc} + w_{nc}) \tag{20}$$

$$D = 4 A / P_w = 2 t_{nc} w_{nc} / (t_{nc} + w_{nc}) \tag{21}$$

where

t_c	= Channel thickness (spacing between fuel plates), m
t_{nc}	= Nominal channel thickness (spacing between fuel plates), m
t_{hc}	= Minimum channel thickness in hot channel (spacing between fuel plates), m
w_c	= Channel width, assumed not to change from its nominal value, m
P_w	= Wetted perimeter of the nominal channel, m
P_{nc}	= Power generated in a fuel plate, without applying manufacturing tolerances, W
P_{hc}	= Power generated in a fuel plate, after applying manufacturing tolerances, W

Because the channel thickness t_c is much smaller than the channel width w_c in most experimental reactors, Eq. (21) reduces to

$$D \approx 2 t_c \tag{22}$$

Using the channel area and hydraulic diameter given by Eqs. (19) and (22) into Eq. (18), the bulk coolant temperature rise can be written in terms of power, channel thickness, and channel width. This is the desired relationship for use in finding hot channel factors.

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C}{C_P \rho_0 \bar{\rho} \beta g L} \right)^{\frac{1}{2+\alpha}} \left(\frac{P}{4 w_c t_c^{\,3}} \right)^{\frac{1}{2+\alpha}} \tag{23}$$

Formulas for Hot Channel Factors

For use in the NATCON version 2.0, six hot channel factors (three global/systemic and three local/random) are obtained from 9 manufacturing tolerances and measurement uncertainties u_1, u_2,.... u_9 that are defined below. These are fractional uncertainties rather than percent. Of these nine uncertainties, those affecting a particular hot channel factor are indicated in Table 1. The systemic hot channel factors are given by Eqs. (24) through (26), and the random hot channel factors are given by Eqs. (27) through (29). A utility Fortran computer program NATCON_HCF and a Microsoft spreadsheet NATCON.HotChanFactors.xls have also been developed to compute the hot channel factors using these formulas.

$$FQ = 1 + u_7 \tag{24}$$

$$FW = 1 + u_8 \tag{25}$$

$$FH = 1 + u_9 \tag{26}$$

The ratio of the power generated in hot plate to its nominal power, caused by the uncertainties in neutronics-computed power and in U-235 mass per plate, can be written as

$$\frac{P_{hc}}{P_{nc}} = (1+u_1)(1+u_2) \tag{27}$$

The ratio of bulk coolant temperature rise in hot channel to the temperature rise in the nominal channel, caused by the uncertainties in neutronics-computed power, U-235 mass per plate, and channel thickness, is obtained from Eq. (23). Only the quantity in the second parentheses is important here because the quantity in the first parentheses is insensitive to these uncertainties.

$$\frac{\Delta T_{hc}}{\Delta T_{nc}} = \left(\frac{P_{hc}}{P_{nc}}\right)^{\frac{1}{2+\alpha}} \left(\frac{t_{nc}}{t_{hc}}\right)^{\frac{3}{2+\alpha}} = (1+u_1)^{\frac{1}{2+\alpha}}(1+u_2)^{\frac{1}{2+\alpha}}\left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} \tag{28}$$

The uncertainty in flow distribution is assumed to reduce the channel flow to (1- u_6) times the flow without this uncertainty, and therefore the bulk coolant temperature rise is increased by the factor (1+ u_6). This uncertainty in bulk coolant temperature rise is statistically combined with that given by Eq. (28) to obtain the following formula for the hot channel factor FBULK for input to the NATCON version 2.0.

$$FBULK = 1 + \sqrt{\left\{(1+u_1)^{\frac{1}{2+\alpha}}(1+u_2)^{\frac{1}{2+\alpha}}\left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1\right\}^2 + u_6^{\,2}} \tag{29}$$

The temperature drop across coolant film on the cladding surface at an axial location is given by Eq. (30). Here the heat flux q" (W/m^2) on the cladding surface is replaced by t_f q'''/2 in terms of the volumetric power density q''' (W/m^3) in the fuel meat.

$$\Delta T_{film} = \frac{q''}{h} = \frac{t_f\, q'''}{2h} \tag{30}$$

The convective heat transfer coefficient h (W/m^2-C) is given by Eq. (31). Here the laminar Nusselt number N_u is independent of flow rate, and varies only slowly with the aspect ratio (width/thickness) of coolant channel. The main variation of the heat transfer coefficient with channel thickness is due to the denominator of Eq. (31). The numerator of Eq. (31) is considered to be constant.

$$h = \frac{N_u K_{cool}}{D} = \frac{N_u K_{cool}}{2\ t_c} \tag{31}$$

Using Eq. (31) for the heat transfer coefficient, the temperature drop across coolant film can be written as Eq. (32).

$$\Delta T_{film} = \frac{q'''\ t_f\ t_c}{N_u\ K_{cool}} \tag{32}$$

Equation (32) states that ΔT_{film} is directly proportional to the fuel meat thickness (having uncertainty u_3), the channel thickness (having uncertainty u_5), and the power density in meat. The uncertainty in power density is caused by three uncertainties, that is, u_1, u_2 and u_4. Statistically combining these five uncertainties gives the following formula for the hot channel factor FFILM for input to the NATCON version 2.0.

$$\text{FFILM} = 1 + \sqrt{u_1^2 + u_2^2 + u_3^2 + u_4^2 + u_5^2} \tag{33}$$

The uncertainty in the heat flux at the cladding surface is included in the hot channel factor FFILM given by Eq. (33). A hot channel factor FFLUX for the heat flux alone can be found from Eq. (34) for heat flux in terms of the power density q''' in the fuel meat and the thickness of the meat. The fractional uncertainty in heat flux is the sum of fractional uncertainties in power density and meat thickness, as given by Eq. (35).

$$q'' = \frac{q'''\, t_f}{2} \tag{34}$$

$$\frac{\delta q''}{q''} = \frac{\delta q'''}{q'''} + \frac{\delta t_{fuel}}{t_{fuel}} \tag{35}$$

In Eq. (35), the uncertainty in power density is caused by three uncertainties, that is, u_1, u_2 and u_4. The uncertainty in the meat thickness is given by u_3. Statistically combining these four uncertainties gives the following formula for the hot channel factor FFLUX for input to the NATCON version 2.0.

$$FFLUX = 1+\sqrt{u_1^2+u_2^2+u_3^2+u_4^2} \quad (36)$$

The uncertainty in the temperature drop ΔT_{metal} from fuel meat centerline to cladding surface is not important in the case of the PUR-1 reactor because ΔT_{metal} is very small compared to ΔT_{film}. For example, ΔT_{metal} is 0.05 °C and ΔT_{film} is 34.5 °C at 100 kW without any hot channel factors.

Table 1. Uncertainties Included in the Six Hot Channel Factors Used in NATCON Version 2.0 (X implies that an uncertainty affects a hot channel factor)

	Uncertainty Fraction	FQ	FW	FH	FBULK	FFILM	FFLUX
Local or random uncertainties							
1	Neutronics calculation of power in a plate, u_1				X	X	X
2	U-235 mass per plate, u_2				X	X	X
3	Local fuel meat thickness, u_3					X	X
4	U-235 axial homogeneity, u_4					X	X
5	Coolant channel thickness, u_5				X	X	
6	Flow distribution among channels, u_6				X	X	
System-wide uncertainties							
7	Reactor power measurement uncertainty, u_7	X					
8	Flow uncertainty due to uncertainty in friction factor, u_8		X				
9	Heat transfer coefficient uncertainty due to uncertainty in Nu number correlation, u_9			X			

The following information was presented as answers to Requests for Additional Information (RAIs) during the conversion process. This information is not presented in the SAR chapters on thermal hydraulics.

Question 28

28. Appendix 1. From the information in Appendix 1 it is not clear how insignificant are the channel inlet and outlet losses when compared to the wall shear. Please clarify.

Response:

The information in Appendix 1 was used only to obtain hot channel factors for input to a more detailed thermal-hydraulic calculation using the NATCON code [Ref. 8 of the conversion proposal]. Therefore, Appendix 1 is a simplified modeling of what is calculated in detail in NATCON, and it is used only for the purpose of obtaining closed-form equations from which hot channel factors could be found. Appendix 1 does not include the minor losses. The minor losses calculated by NATCON are reported below, and found to be about 16% of the total frictional pressure drop in the HEU core, and 14% of the total frictional pressure drop in the LEU core (see Table Q27-1).

The pressure drop due to inlet and outlet losses were calculated (by the NATCON code) using loss coefficients of 0.5 and 1.0 respectively. The pressure drop due to wall shear along the channel length is found by summing the pressure drop for each axial mesh which is calculated using temperature-dependent coolant viscosity and density for the axial mesh (14 mesh intervals were used over the channel length in all calculations). The pressure drops are calculated by NATCON assuming fully developed laminar flow in a rectangular cross-section channel, and then multiplied by a factor FW^2 (FW squared) where FW is an input which may be used to account for the increased pressure drop due to hydrodynamically developing laminar flow. In the calculations presented in the conversion proposal, FW was always set to 1.0, and thus the increased pressure drop due to developing laminar flow was not included. It is included in the calculations presented here (Table Q27-1). The method used is described below.

For the most limiting fuel plate in Table 4-27 of the conversion proposal for each core (HEU and LEU), a comparison of the pressure drops due to inlet plus outlet loss and wall shear, with and without the effect of developing laminar flow, are tabulated in Table Q27-1.

NATCON calculates the Darcy-Weisbach friction factor $f = C/Re$ for laminar flow, using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section (values of parameter C are given in the response to Question number 29). An apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2 listed at the end of all responses] to account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel. The ratio C_{app}/C was found to be 1.1105 for the 207 mil HEU channel, 1.0985 for the 197 mil LEU channel. Since the NATCON code multiplies the fully developed friction factor by FW^2 as mentioned above, the input FW equals 1.054 and 1.048 for the HEU and LEU channels respectively. NATCON calculations were done using these values of FW, and the pressure drops due to inlet plus outlet loss and wall shear are compared in Table Q27-1 (column B for the HEU channel, and column F for the LEU channel).

Table Q27-1 shows that the pressure drops due to wall shear and minor losses are 84% and 16%, respectively, of the total pressure drop in the HEU channel at its ONB power; and the pressure drops due to wall shear and inlet plus outlet loss are 86% and 14%, respectively, of the total pressure drop in the LEU channel at its ONB power.

Question 29

29. Appendix 1. From the information in Appendix 1 it is not clear what is the functional dependency of the laminar friction parameter C to the channel cross-section dimensions. Provide a reference for the evaluation of C.

<u>*Response:*</u>

The following values (rows 1 and 2 of Table Q29-1) of the parameter C for fully developed laminar flow in a channel of *rectangular cross section* versus the width-to-thickness aspect ratio (w_c/t_c) of the channel are used in the NATCON code that was used in the thermal-hydraulics calculations. The table starts from the square cross section (aspect ratio = 1.0) and goes to the infinite value of the aspect ratio (parallel plates). In order to find the parameter C for the aspect ratio of the PUR-1 reactor, the NATCON code simply interpolates between the tabulated values. The original author of the code obtained these values from an old Reference [E. R. G. Eckert and T. F. Irvine, Heat Transfer Laboratory, University of Minnesota (1957)] but these values are also given in a textbook by Frank Incropera [Ref. 3]. These values are obtained from the closed-form analytical solution for the fully developed laminar velocity distribution in a rectangular channel summarized by R. K. Shah and A. L. London [Ref. 2]. Equation (341) in [Ref. 2] is a fitted equation to easily find the parameter C. It should be noted that the aspect ratio used in [Ref. 2] is channel thickness-to-width ratio (the reciprocal of that used in NATCON and shown below in Table Q29-1), and the friction factor in [Ref. 2] should be multiplied by 4 to get the Darcy-Weisbach friction factor that is used in NATCON and tabulated below.

Table Q29-1. Friction Parameter C Used in the NATCON Code

w_c/t_c	1.0	2.0	3.0	4.0	5.0	6.3	8.0	11.0	15.0	18.0	100.0
C in NATCON	58.0	63.0	69.0	72.5	77.0	80.0	83.0	85.0	88.0	89.0	96.0
C in Ref. 3	57.0	62.0	69.0	73.0			82.0				96.0
C in Ref. 2	56.9	62.2	68.4	72.9	76.3	79.5	82.4	85.6	88.1	89.3	94.7

Question 30

30. Appendix 1. From the information in Appendix 1 in both the calculation of the channel flow and the calculation of the bulk coolant temperature rise the ratio of the coolant kinematic viscosity to density (μ/ρ) was assumed to be insensitive to temperature. Please demonstrate the validity of this assumption.

Response:

The information in Appendix 1 was used only to obtain hot channel factors for input to a more detailed thermal-hydraulic calculation using the NATCON code [Ref. 8 of the conversion proposal]. NATCON does account for the temperature dependence of coolant viscosity and density in the calculation of the channel flow and the calculation of the bulk coolant temperature rise. Therefore, Appendix 1 is a simplified modeling of what is calculated in detail in NATCON, for the purpose of obtaining closed-form equations from which hot channel factors could be found.

As suggested in the question, water viscosity is temperature-dependent, i.e., it decreases with rising temperature. Appendix 1 was revised to account for the effect of temperature dependence of viscosity on hot channel factors, and the revised Appendix 1 is enclosed herewith. The temperature dependence of the dynamic viscosity of water over the temperature range 27 °C ≤ T ≤ 50 °C (adequate for the PUR-1 reactor) can be written as follows.

$$\mu(T) = \mu(T_0)\,(1+T-T_0)^{-\alpha} \qquad \text{(A1)}$$

where $\alpha = 0.12$

T_0 = 27 °C = Pool temperature of PUR-1

$\mu(T_0) = 0.875\text{x}10^{-3}$ N-s/m ^{2}

$\mu(T)$ = Temperature-dependent dynamic viscosity of water, N-s/m^2

As shown in the revised Appendix 1, the revised relationship between the flow rate W in a channel and the friction parameter C is given by Eq. (A2). The revised formula for hot channel factor FBULK for bulk coolant temperature rise is given by Eq. (A3).

$$W \propto \left(\frac{1}{C}\right)^{\frac{1}{2+\alpha}} \qquad \text{(A2)}$$

$$\text{FBULK} = 1 + \sqrt{\left\{(1+u_1)^{\frac{1}{2+\alpha}}(1+u_2)^{\frac{1}{2+\alpha}}\left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1\right\}^2 + u_6^{\,2}} \qquad \text{(A3)}$$

The exponent on the right hand side of Eq. (A2) changed from 0.5 (in the conversion proposal ignoring temperature dependence of μ) to the revised value 1/2.12 = 0.4717. There exponents in Eq. (A3) for FBULK also changed, e.g., from 3/2 to 3/2.12 = 1.415. As a result of this revision, the hot channel factor FBULK decreased from 1.312 (in the conversion proposal) to 1.301 for the most limiting fuel plate 262 in the HEU core. Similarly, FBULK decreased from 1.321 (in the conversion proposal) to 1.308 for the most limiting fuel plate 1348 in the LEU core. The effect of ignoring the temperature dependence of viscosity is conservative.

NATCON calculations were done with these revised values of FBULK along with a value of FW > 1.0 to account for the increased friction due to developing laminar flow (in response to Question number 33). The results are shown in Table Q27-1 (column C for the HEU core, and column G for the LEU core).

As a consequence of the two effects (i.e., increased friction due to developing laminar flow and the temperature dependence of viscosity) on hot channel factors FW and FBULK, the ONB power of the HEU core changes from 76.3 kW (reported in the conversion proposal) to 75.9 kW, and the ONB power of the LEU core changes from 96.1 kW (reported in the conversion proposal) to 95.8 kW. The effect is small for the PUR-1 reactor.

Question 32

32. Appendix 1. Equation (30) has two terms and the conversion proposal states that the expression within the parenthesis on the right hand side of the equation varies slowly compared to the heat flux t_{fuel} q'''/2. Demonstrate the validity of the statement with reference to the PUR-1 fuel plate.

Response:

Equation (30) of Appendix 1 is for finding a hot channel factor for the temperature drop from the meat mid-plane to cladding surface (ΔT_{metal}). This temperature drop is very small compared to the temperature drop from the cladding surface to bulk coolant (ΔT_{film}). For example, in the PUR-1 HEU fuel plate 262 without hot channel factors, ΔT_{metal} is 0.07 °C and ΔT_{film} is 46.98 °C (at meat mid-height) at a high power of 100 kW. Similarly, in the PUR-1 LEU fuel plate 1348 without hot channel factors, ΔT_{metal} is 0.05 °C and ΔT_{film} is 34.5 °C at a power of 100 kW. Therefore, the hot channel factor for ΔT_{metal} is not important for PUR-1. The important hot channel factor is the factor FFILM for ΔT_{film}. In the case of PUR-1, ΔT_{film} is the bigger component (bigger than the bulk coolant temperature rise) in the total temperature rise from the inlet temperature to the cladding surface temperature at the axial level experiencing the onset of nucleate boiling. The hot channel factor FFILM found by Eq. (29) of Appendix 1 in the conversion proposal remains unchanged. It depends on the uncertainties in $q'''t_{fuel}$ and channel thickness (as shown in Eq. 28), but not on the uncertainty in $[t_{fuel}/(4K_{fuel}) + t_{clad}/K_{clad}]$.

In short, PUR-1 is not limited by the fuel peak temperature, but by the onset of nucleate boiling, and the uncertainty in $[t_{fuel}/(4K_{fuel}) + t_{clad}/K_{clad}]$ is not important for PUR-1. We believe that the hot channel factor FFILM has been determined accurately.

Question 33

33. Section 4.7.2. According to Appendix 1 the systematic uncertainty in flow rate is accounted for by applying the hot channel factor Fw to the laminar friction factor C. Explain the reason for the value of the flow friction factor Fw being unity in Tables 4-25 and 4-26.

<u>Response:</u>

As suggested in the question, a value of FW (hot channel factor for flow) greater than 1.0 should be used to account for the increased frictional pressure drop due to the hydrodynamically developing laminar flow in the entrance region of the coolant channel, otherwise the code (NATCON) accounts only for the fully developed frictional pressure drop. This has been done now and the results are presented in Table Q27-1. Since each coolant channel creates its own buoyancy to drive its own coolant flow, there is no uncertainty due to *redistribution* of a total reactor flow rate. The loss coefficients of 0.5 and 1.0 at channel inlet and outlet are used in the calculations. To account for the reduction in flow rate due to the hydrodynamically developing laminar flow in the channel, the values of FW were calculated for the most limiting channels in the HEU and LEU cores as follows.

NATCON calculates the Darcy-Weisbach friction factor f = C/Re using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section (values of parameter C are given in the answer to Question number 29). These values of parameter C are for the fully developed laminar flow in a rectangular cross-section channel. An apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2] to account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel. The ratio C_{app}/C was found to be 1.1105 for the 207 mil HEU channel, and 1.0985 for the 197 mil LEU channel. Since the NATCON code multiplies the fully developed frictional factor by FW^2, the input FW equals 1.054 and 1.048 for the HEU and LEU channels respectively. The flow reduction factor is input factor FW or more accurately $FW^{2/(2+\alpha)} = FW^{0.9434}$ (noting that α = 0.12 for the PUR-1 reactor as mentioned in the revised Appendix 1 enclosed herewith).

The results of using these values of FW in NATCON calculations (excluding the effect of temperature dependence of μ on hot channel factors) are shown in Table Q27-1. The ONB power of the HEU core changes to 75.8 kW from 76.3 kW reported in the conversion proposal. The ONB power of the LEU core changes to 95.7 kW from 96.1 kW reported in the conversion proposal.

The channel flow indeed gets reduced by the factor $FW^{0.9434}$ as expected. For the HEU plate 262, the flow reduces from 0.02083 kg/s to 0.01989 kg/s (see Table Q27-1) when the input hot channel factor FW is changed from 1.0 to 1.054. The expected reduced flow should be $0.02083/(1.054)^{0.9434}$ = 0.01982 kg/s which is close to the NATCON-calculated value of 0.01989 kg/s. For the LEU plate 1348, the flow reduces from 0.01912 kg/s to 0.01834 kg/s (see Table Q27-1) when the input FW is changed from 1.0

to 1.048. The expected reduced flow should be $0.01912/(1.048)^{0.9434}$ = 0.01829 kg/s which is close to the NATCON-calculated value of 0.01834 kg/s.

Question 36

36. Table 4-28. Define the parameter "margin to incipient boiling."

Response:

The margin to incipient boiling shown in Table 4-28 was calculated at the nominal operating power of PUR-1 (i.e., 1 kW), and it is the smallest value of the temperature difference (T_{ONB} - T_w) over the coolant channel length in the hottest channel where T_W is cladding surface temperature with all hot channel factors applied, and T_{ONB} is the local onset-of-nucleate-boiling temperature. This basically gives an idea of how far below the onset of nucleate boiling condition the reactor is operating. This definition can be written as an equation as follows:

Margin to ONB = Minimum $T_{incp}(p,q^{"}(z)F_{flux}) - \left[F_{bulk}\left\{T(z)-T_0\right\} + F_{film}\left\{T_{wall}(z)-T(z)\right\}\right]$

where

$T(z)$	= Bulk coolant temperature at axial position z in the channel heated by the plate power of $P_{op}F_r\ F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$T_{wall}(z)$	= Cladding surface temperature at axial position z in the channel heated by a plate power of $P_{op}F_r\ F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$q"(z)$	= Heat flux at position z for the plate power of $P_{op}F_r\ F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$p(z)$	= Absolute pressure in the channel at axial position z
$T_{incp}(p(z), q"(z)F_{flux})$	= Onset of nucleate boiling temperature at absolute pressure p(z) and heat flux $q"(z)F_{flux}$
P_{op}	= Operating power of the reactor (e.g., 1 kW for PUR-1)
N	= Number of fuel plates in the core (e.g., 190 for PUR-1 LEU core)
T_0	= Coolant temperature at the channel inlet
F_r	= RPEAK = Radial power factor of the plate cooled by the channel
F_W	= Hot channel factor for flow in the channel
F_Q	= Hot channel factor for reactor power
F_h	= Hot channel factor for Nusselt number

F_{film} = FFILM = Hot channel factor for temperature drop across the coolant film on cladding surface

F_{flux} = FFLUX = Hot channel factor for heat flux

F_{bulk} = FBULK = Hot channel factor for bulk coolant temperature rise in the channel

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Specification

Specification for Purdue University Standard and Control Fuel Elements - Assembled for the Purdue University Reactor

Idaho National Laboratory

The INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance.

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Idaho National Laboratory

SPECIFICATION FOR PURDUE UNIVERSITY STANDARD AND CONTROL FUEL ELEMENTS – ASSEMBLED FOR THE PURDUE UNIVERSITY REACTOR	Identifier: SPC-382 Revision: 1 Effective Date: 01/24/07 Page: i of ii

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1. SUMMARY

1.1 General

This *specification* (see def.) defines the materials, components, testing, inspection, certain processes, *quality control* (see def.) requirements and acceptance criteria for the fabrication of standard and control *fuel elements* (see def.) and fuel element containers for the Purdue University Reactor at Purdue University at West Lafayette, Indiana.

2. APPLICABLE CODES, PROCEDURES, AND REFERENCES

2.1 Standards, Specifications, Drawings and Attachments

The applicable portions of the following documents as defined herein, form a part of this specification. Where there is a conflict between the documents cited and the latest revision thereof, the *supplier* (see def.) shall notify the *purchaser* (see def.) of the conflict and use the latest revision in effect at the signing of the contract, unless otherwise directed by the purchaser.

2.1.1 Specifications and Standards

National Codes and Standards

ASTM E 1742-00	Standard Practice for Radiograph Examination
ASTM E 1417-99	Standard Practice for Liquid Penetrant Examination
MIL-C-45662	Calibration System Requirements
RDT F6-2T	Welding of Reactor Core Components, Sections 1,2,3 and 6

American Society for Testing and Materials (ASTM)

ASTM B 209-00	Standard Specification for Aluminum and Aluminum-Alloy Sheet and Plate
ASTM B 210-04	Aluminum and Aluminum-Alloy Drawn Seamless Tubes
ASTM B 211-00	Standard Specification for Aluminum and Aluminum-Alloy Bar, Rod and Wire
ASTM B 214-99	Standard Test Method for Sieve Analysis of Granular Metal Powders
ASTM B 221-00	Standard Specification for Aluminum

	and Aluminum-Alloy Extruded Bars, Rods, Wires, Profiles and Tubes
ASTM B 241-02	Aluminum and Aluminum-Alloy Seamless Pipe and Seamless Extruded Tube
ASTM E 8-00	Methods of Tension Testing of Metallic Materials
ASTM E 29-93a (1999)	Recommended Practice for Indicating Which Places of Figures are to be Considered Significant in Specified Limiting Values
ASTM E 2016-99	Standard Specification for Industrial Woven Wire Cloth

American Welding Society (AWS)

AWS A5.10-1995	Aluminum and Aluminum Alloy Welding Rod and Bare Electrodes

American National Standards Institute (ANSI)

ANSI B46.1-1994	Surface Texture
ANSI Y14.5-1994	Dimensioning and Tolerancing for Engineering Drawings

American Society of Mechanical Engineers (ASME)

ASME Section V – 2001, without addendum	Boiler and Pressure Vessel Code Section V
ASME Section IX - 2001	Boiler and Pressure Vessel Code Section IX
ASME NQA-1-1997	Quality Assurance Requirements for Nuclear Facility Applications

Idaho National Laboratory (INL)

TRTR-11	Specification for Low Enriched Uranium Metal in Test Reactor Fuel
TRTR-14	Specification for Reactor Grade Low Enriched Uranium Silicide Fuel Powder
IN-F-4-TRA	Specification for Aluminum Powder for Matrix Material in Test Reactor Fuel
STD 7022A	Cleanliness Acceptance Levels for Nuclear or Non-Nuclear Service Components

American Society for Nondestructive Test (ASNT)

SNT-TC-1A (1996 or later)	American Society For Nondestructive Testing (ASNT) Recommended Practice

2.1.2 Drawings (INL)

635454	Purdue University Test Research and Training Reactor Graphite Reflector Assembly and Source Drive Assembly
635455	Purdue University Test Research and Training Reactor Standard Fuel, Partial, & Dummy Element Assemblies
635456	Purdue University Test Research and Training Reactor Control Fuel Element Assembly and Dummy Control Fuel Element Assembly
635457	Purdue University Test Research and Training Reactor Fission Chamber Fuel Element Assembly
635458	Purdue University Test Research and Training Reactor Standard Fuel Container Assembly
635459	Purdue University Test Research and Training Reactor Control Fuel Container Assembly
635460	Purdue University Test Research and Training Reactor Irradiation Facility Assembly
635461	Purdue University Test Research and Training Reactor Capsule Holder and Capsule Insert Assemblies and Details
635462	Purdue University Test Research and Training Reactor Graphite Container Assembly, and Source Drive Container Assembly
635463	Purdue University Test Research and Training Reactor Fuel Plate Assembly and Dummy Fuel Plate Detail
635464	Purdue University Test Research and Training Reactor Container Tube Assembly and Details
635465	Purdue University Test Research and

	Training Reactor Graphite Block Detail And Graphite Capsule Holder Detail
635466	Purdue University Test Research and Training Reactor Miscellaneous Details
635467	Purdue University Test Research and Training Reactor Source Drive Nozzle Assembly and Source Drive Top
635468	Purdue University Test Research and Training Reactor Nozzle Preliminary Machined and Fission Chamber Top

3. TECHNICAL REQUIREMENTS

3.1 Production Qualification

The supplier is required to qualify the processes or portions of the process or be exempt from same by written approval of the purchaser. In *qualification* (see def.), only materials that comply with this specification shall be used. Qualification processes, equipment, and operator qualification/training programs shall be identical to those used during *production* (see def.). To qualify, the supplier must demonstrate, to the satisfaction of the purchaser, that the process is capable of producing a product, which satisfies all the requirements of the specifications. Assembly of production fuel elements shall not be initiated until: (1) all required data, to assure compliance with the qualification requirements, has been submitted to the purchaser; (2) data and records required by Section 6.3 have been submitted; and (3) written approval of qualification has been received by the supplier from the purchaser.

3.1.1 Fuel Plate Qualification:

Fuel plate (see def.) qualification shall be satisfied by supplier production of a minimum of two consecutively produced plate *lots* (see def.), in lot quantities of 24 *plates* (see def.). The plates shall be made using low enriched uranium in the form of *Silicide* (see def.) powder, which have a yield of no less than 65% acceptable fuel plates meeting all applicable requirements of this specification. The supplier may combine the results of two consecutive lots into a production run in determining the 65% yield requirement provided that there have been no changes in the *manufacturing* (see def.) *procedure* (see def.) between lots which would require *requalification* (see def.) in accordance with Section 3.1.3.

In the event that fuel plate qualification has been performed by the supplier meeting all the requirements of this specification during the past twelve months, and qualified operators are performing the fabrication, fuel plate qualification requirements listed above will be waived.

Fuel plates made in *development* (see def.) (prior to and including qualification runs that fail to meet the 65% yield requirements) will not be used in fabricating production fuel elements without prior approval of the purchaser.

3.1.2 Fuel Element Qualification:

The supplier shall fabricate 1 *dummy standard fuel element assembly* (see def.), which shall meet the requirements of this specification.

3.1.3 Requalification:

The supplier shall notify the purchaser of any proposed process change. A changed process may not be used in production until the supplier has met all the requirements of Section 3.1.3, submits the results and data of the requalification effort, and receives written approval from the purchaser.

Requalification for any fuel plate attribute to the requirements of the specification will be required when the processes, materials, fuel loadings, equipment or equipment operators (welding and rolling) which have been previously qualified are changed, unless the supplier can demonstrate to the satisfaction of the purchaser by engineering explanation or proof test that such changes will have no detrimental effect on the product.

Requalification for compacting, *pack* (see def.) assembly, and rolling mill operators can be less than qualification basis, since the procedure has already been established. Candidate operators who are not qualified for compacting operations, pack assembly operations, and hot/cold rolling mill operations must demonstrate their abilities in performing the individual operations they are assigned.

An operator must qualify by processing two lots of fuel plates with minimum lot size of 24, for the operation he is assigned to qualify, before performing any production operation independently. Each lot of fuel plates shall be processed through final inspection, with a minimum yield rate of 90% acceptable fuel plates required for the operator to be termed qualified.

NOTE: *Failure of an operator to qualify, because of fuel plate deviations, must be based on deviations related to the operation being qualified. The purchaser on a case-by-case basis will determine the quantities and sizes of requalification fuel plates selected to be destructively examined.*

3.1.4 Operator Qualification:

Operator qualification will be accomplished via an approved supplier internal qualification program for the following operations:

A. Arc melting

B. Compacting

C. Pack assembly

D. Hot rolling

E. Cold rolling

F. Final machining.

3.1.4.1 In addition to the operations specified above, the supplier shall also show evidence of the training and competency of those individuals who perform any of the following fuel element fabrication and inspection activities:

A. Powder sieving, weighing, and testing

B. Compact weighing, visual and dimensional inspection

C. Fuel plate/element and component cleaning

D. Fuel plate annealing operations

E. Dimensional inspection of plates, elements, and subcomponents

F. Metallographic sample preparation and inspection

G. Visual inspection of plates, elements, and subcomponents

H. Void volume inspection

I. Fluoroscope inspection of fuel plates

J. Radiography and inspection of fuel plate radiographs

K. Ultrasonic testing and interpretation.

The individuals performing these operations shall have specific requirements imposed on them that will demonstrate their knowledge and ability to perform their respective assignments. Documented evidence of the training of these individuals shall be maintained and shall be made available to the purchaser upon request.

3.2 Materials

The material requirements for the components comprising the fuel element are as specified on Drawings per Section 2.1.2 and requirements of this section.

3.2.1 Fuel Bearing Plates

3.2.1.1 Fuel Cores: The *fuel cores* (see def.) of the fuel plates shall be uranium silicide powder dispersed in aluminum alloy powder which meet the requirements of IN-F-4-TRA and TRTR-14, per Section 2.1.1 of this specification.

3.2.1.2 Frames and Covers: Aluminum for the frames and cover plates shall conform to ASTM B209, Alloy 6061-O. The aluminum plate stock used for frame and cover plates shall be certified by the supplier to contain less than 30 PPM boron, 80 PPM cadmium, and 80 PPM lithium.

The subcontractor shall furnish certified physical properties and chemical analyses of ingots or plates of the 6061 materials to INL.

3.2.2 Aluminum Weld Filler Metal:

All aluminum weld filler metal shall be ER4043 as required by Specification AWS A5.10-1995.

3.2.3 Dummy (Non-Fueled) Plate:

Dummy (non-fueled) plates (see def.) shall be fabricated from aluminum Type 6061-O, that meets the requirements of Section 3.2.1.2.

3.2.4 Material Requirements

All material used or contained in the product shall comply with all the requirements of this specification and Drawings per Section 2.1.2 unless exempted by written document by the purchaser.

3.3 Mechanical Requirements

3.3.1 Fuel Plate Requirements

3.3.1.1 Fabrication: The supplier shall furnish the details of his fuel plate rolling schedule and component cleaning process to the purchaser for approval prior to use in production per 6.3.1.

Compacting details shall include silicide – aluminum compacting pressure and compacting press dwell time.

After hot rolling, each fuel plate shall be blister annealed per Section 4.6.1 and then cold rolled to final thickness at room temperature. After cold rolling operation, the fuel plates shall be subjected to program annealing. The rolling schedule shall contain, at a minimum, the following:

A. Nominal plate reduction

B. Minimum number of hot roll passes

C. Nominal inter-pass reduction and target thickness

D. Hot rolling furnace temperature

E. Preheat time for all hot roll passes

F. Final hot roll plate thickness

G. Type and frequency of roll lubricant utilized

H. Nominal cold roll reduction.

I. Final cold roll thickness.

Fuel plate *cladding* (see def.) thickness required by Section 3.3.1.4 and fuel core homogeneity requirements of Section 4.4 are independent requirements that must be met.

3.3.1.2 Core Configuration: No fuel particles are allowed within the fuel free zones located at the ends of the plates as shown on Drawing 635463.

The nominally unfueled area of each fuel plate as defined by Drawing 635463 may contain random fuel particles defined as flaking and limited in size, location, and spacing per this Section, as determined by Section 4.5.

The presence of fuel particles detected between the maximum fuel core outline and fuel plate edges and ends is allowed provided they do not violate the following restrictions:

— One or more fuel particles, which fit in a rectangle whose area is not more than $4x10^{-4}$ in^2 is acceptable

AND

— The fuel particle(s) are no closer than 0.080 in. to any other particle edge to edge

AND

— No particle is closer to the plate edge or end than the major dimension of the particle.

Stray fuel particles (see def.) that violate the above requirements may be removed from fuel plate edges by filing, provided the following:

— The filed out area is no deeper into the edge of the plate than 0.050 in., no longer than 0.250 in.

AND

— Each filed area is at least 1.0 in. apart

Filing of fuel plate ends, for the removal of stray particles, is not allowed, unless previously approved by the purchaser.

3.3.1.3 Internal Defects and Bond Integrity: Metallurgical bond, as determined by Section 4.6 is required at interface areas of the finished fuel plates, specifically fuel core-to-clad and clad-to-frame. The presence of grain growth across the fuel matrix-cladding interface and across the aluminum frame-cladding interface of at least 50% is required. Fuel core defects in excess of 0.06 in. in any dimension as determined by Section 4.7 are not allowed.

3.3.1.4 Cladding Thickness: During production, all plates will be subjected to UT min-clad inspection. The standard will be calibrated at the nominal 0.008-inch scan depth. The gage will then be adjusted to a 0.010 inch scanning depth and the fuel plates will be scanned at 0.010 inch. Fuel plate UT traces, which display min-clad indications at the 0.010-inch depth, shall be visually compared with the 0.008-inch Standard trace. Fuel plates for which the UT reports show a comparable density of indications, or worse, than the indications displayed on the standard UT report are unacceptable. Fuel plates, which fail the 0.010-inch UT scan, shall be rescanned at 0.008 inch. Only fuel plates which are acceptable when rescanned at 0.008 inch shall be submitted to the Purchaser and User for evaluation.

3.3.2 Non-fueled (dummy) plates:

The supplier shall use a cold rolling method to obtain plate thickness. Non-fueled (dummy) plates shall be subjected to program anneal.

3.3.3 Fuel Element Requirements

3.3.3.1 Welding: All welding shall be performed using procedures and welding personnel qualified in accordance with ASME Section IX or the criteria defined in Appendix B. Quality acceptance of production welds shall be in accordance with Appendix B, Section 5.

3.4 Physical Properties

Fuel plates shall have a core of U_3Si_2 and aluminum and completed fuel plates and fuel elements shall have fuel loadings per Sections 3.4.1.2, and 3.4.1.5.

3.4.1 Fuel Plate Requirements

3.4.1.1 Fuel Core: The fuel core shall consist of 19.75 ±0.2 weight % enriched uranium silicide powder dispersed in aluminum powder. The uranium silicide powder shall be -100, +325 U.S. standard mesh particles. However, a blend may contain up to 35 weight percent of -325 U.S. standard mesh particles. Any powder particles greater than 100 mesh particles shall be reground such that they will go thru the 100 mesh sieve. The fuel core shall be fabricated according to standard powder-metallurgical and roll-bonding techniques. The supplier shall provide to the purchaser, a written procedure for pack assembly and the initial rolling step which describes the method used to prevent excessive oxidation that causes non-bond of fuel core to the cladding.

3.4.1.2 Fuel Loading: By using the approved supplier's method of assigning U-235 content, per a detailed description as to the weighing procedure by which the supplier proposes to assign fuel plate U-235 content. Each fuel plate shall contain [redacted]. The weight of each core shall be measured and recorded to within 0.01 gram U-235 based upon weight of the final compact and chemical and isotopic analysis of the constituents.

3.4.1.3 Fuel Homogeneity: Fuel homogeneity requirements are located in section 4.4.

3.4.1.4 Void Volume: In the qualification process, all fuel plates shall be inspected for void volume using the method described in Section 4.2. The percent voids in the fuel cores of all fuel plates shall be determined by the inspection procedure developed by the supplier. The percent voids in the fuel cores shall be at least 3.0% and not more than 11.0%.

3.4.1.5 Fuel Element Requirements

3.4.1.6 Fuel Loading: Assigned fuel loading for each fuel element shall b[redacted]. Each Control Fuel Element shall contain [redacted]. Control limits for the method used to measure this weight are established at the 95% confidence level for a significant population of measurements of a particular standard. The U-235 enrichment shall be 19.75 ±0.2 weight % of total uranium per specification TRTR-11.

3.5 Surface Condition

Fuel plates and completed fuel elements must comply with the surface condition requirements of Section 3.5.1, 3.5.2, and 3.5.3 and drawings of Section 2.1.2, per ANSI B46.1. Sanding, or any other finishing procedure that will smear the aluminum surface, will not be allowed on fuel plates unless approved by the purchaser.

3.5.1 Surface Defects

3.5.1.1 Compliance with surface finish and defect requirements shall be established by 100% visual inspection of all fuel plates and elements. The surface of the finished fuel plates shall be smooth and free of gouges, scratches, pits, or removal of metal in excess of 0.005 inch in depth. Dents in the fuel plate shall not exceed 0.012 inch in depth or 0.25 inch in diameter. If there is evidence of dogboning in the plates, surface defects in the *dogbone* (see def.) area shall not exceed 0.003 inch in depth. No degradation of the fuel plates beyond these limits shall be permitted.

3.5.1.2 Fuel Plates shall be free of stringiness, scabs, or cracks. Surface finish shall be as required by Drawing 635463. Compliance with requirements of this section shall be accomplished by visual inspection of all fuel plates and fuel elements.

3.5.1.3 Defects on fuel plate edges or ends are permissible provided they are evaluated and acceptable to the requirements of Paragraph 3.3.1.2.

3.5.1.4 Compliance with surface finish and defect requirements shall be established by 100% visual inspection of all fuel element containers. Fuel element containers shall be free of surface defects such as pits, dents, or scratches in excess of 0.010 inch in depth and 0.12 inch in diameter or equivalent area.

3.5.2 Cleanliness:

The suppliers fabrication, assembly, and storage areas used for the production of Purdue University fuel elements and/or components shall conform to the requirements of *"controlled work area"* (see def.) as defined in Paragraph 1.3.6 of INL Standard 7022A. Cleanliness shall be in compliance with INL Standard 7022A, Paragraphs 1.1, 1.2.3, 3.1, 3.2-b, d, i, 3.3 - d, e, 4.1.3, 4.2, and 4.3. Freon shall not be used to clean fuel elements or components.

As determined by Section 4.10 of this specification, there shall be no foreign materials on the finished fuel plates or surfaces of the finished fuel elements. All oil, metal chips, turnings, dusts, abrasives and spatter, scale, and other particles shall be removed from the fuel surfaces by procedures which assure that the minimum cladding thickness has not been violated. All components shall be cleaned by a method approved by the purchaser.

3.5.3 Contamination:

The surfaces of each fuel plate shall be counted or smeared and counted for alpha-beta-gamma contamination. The alpha count shall be less than five dpm per 100 cm^2, and the beta-gamma count shall be less than 200 dpm per 100 cm^2.

Each fuel element shall be smeared and counted for radioactive contamination. The alpha count shall be less than five dpm per 100 cm^2, and the beta-gamma count shall be less than 200 dpm per 100 cm^2.

3.6 Marking

NOTE: *All fuel plates, fuel assemblies, and fuel element containers will be marked per this section.*

3.6.1 Fuel Plate Identification:

Each finished fuel plate shall be identified, as shown on Drawing 635463, by a combination of numbers and/or letters that will maintain positive identification relative to the complete traceability to the supplier fabrication history, including the basic material lots, heat or metal, manufacturing cycle, and quality control phases. The identification number shall be stamped, etched or vibro-peened at the location specified by Drawing 635463. The depth of the identification characters shall not exceed 0.010 in.

3.6.2 Fuel Assembly Identification:

Each fuel assembly shall have an identifying number such as 07-XX (07 signifying year of fabrication). The number shall be placed on the container assembly as shown on Drawings 635455, 635456 and 635457.

The identification shall be stamped or entered by a method approved by the purchaser, with two inch block characters not in excess of 0.010 inches in depth. Standard assemblies should be labeled: E2, F2, G2, H2, F3, H3, E4, F4, G4, H4, F5, H5. Control assemblies should be labeled: E3, G3, and E5. The fission chamber assembly should be labeled as G5. The source assembly shall be labeled as C3. The spare Standard Assemblies should be labeled: SP-1, SP-2, SP-3. The spare Control Assembly should be labeled as SP-4.

3.6.3 Dummy Element Identification:

The dummy standard fuel element assembly shall have the identifying number DUM-1. The number shall be placed on the container assembly as shown on Drawing 635455. The identification shall be stamped or entered by a method approved by the purchaser, with two inch block characters not in excess of 0.010 inches in depth.

3.7 Storage

All fuel plates, fuel assemblies, and fuel element containers that have received final cleaning per Section 3.5.2 shall be protected in clean polyethylene containers or other containers approved by the purchaser while (1) awaiting final assembly, (2) being transferred into or being maintained in storage, or (3) being prepared for packaging or shipment. Any material exposed to contamination shall be reinspected to the requirements of Section 3.5.

3.8 Fuel Element Surface Treatment

If boehmite treatment is required during fuel element fabrication, the following shall apply. After fuel elements are assembled and inspected they shall be subjected to an environment that will cause an evenly distributed boehmite layer of 0.00006 to 0.0003 in. thickness (averaged over the surface using eddy current instrumentation) to form on all surfaces of the entire assembly.. The treatment process shall be performed under controlled conditions, which shall require the supplier to maintain a record of the thermal history of the autoclave. The records shall include heat charts of recorded time and temperature. Documented evidence of the controls placed on the autoclave shall be maintained by the supplier.

3.8.1 After the boehmite process has been qualified, one fuel element from every 2nd autoclave run shall be inspected following a procedure approved by the Purchaser.

3.8.2 Each fuel element shall have a corresponding aluminum plate coupon, made from fuel plate end crops, placed near the fuel element during the boehmite formation process. The aluminum plate coupon shall be subjected to the same environment as the fuel elements and each coupon measured for boehmite thickness.

3.8.3 Fuel elements and aluminum plate coupons subjected to the boehmite formation process must be carefully handled to preclude scratches, dents, and gouges that would cause removal of boehmite.

3.9 Graphite Reflectors and Graphite Radiation Baskets

Graphite reflector assemblies (see def.) and *irradiation facility assemblies* (see def.) shall be fabricated as per requirements contained in this section and in drawings 635454, 635460, 635461, and 635465.

3.9.1 Material:

All materials used shall comply with all the requirements of this specification and applicable drawings.

3.9.2 Assembly:

The assembly of the graphite reflector assemblies and irradiation facility assemblies shall be as shown on the applicable drawings.

3.9.3 Welding:

All welding shall be performed using procedures and welding personnel qualified in accordance with ASME Section IX or the criteria defined in Appendix B. Quality acceptance of production welds shall be in accordance with Appendix B, Section 5.

3.9.4 Identification:

The graphite reflector assemblies shall have identifying numbers such as GR-X placed on the side of the assembly as shown drawing 635454. The graphite reflector shall be labeled as follows: D1, D2, D3, D4, D5, E1, F1, G1, H1, I1, I2, I3, I4, and I5. The irradiation facility assemblies shall have identifying numbers such as IF-X placed on the side of the assembly as shown on drawing 635460. The irradiation facility assemblies shall be labeled as follows: D6, E6, F6, G6, H6, and I6. The identification shall be stamped or entered by a method approved by the purchaser, with two inch block not in excess of 0.010 inches in depth.

3.9.5 Dimensional Inspection:

Verification of all external dimensions of the graphite reflector assemblies and irradiation facility assemblies shall be by 100% inspection, in accordance with drawings 635454 and 635460. All dimensions of this specification shall apply at a temperature of 75°F ± 5°.

3.9.6 Surface Finish and Defects:

The graphite reflector assemblies and irradiation facility assemblies shall be free of surface defects such as pits, dents, scratches in excess of 0.010 inch deep and 0.12 inch diameter or equivalent area.

3.9.7 Storage:

All graphite reflector assemblies and irradiation facility assemblies shall have received final cleaning and shall be protected in clean polyethylene containers or other containers approved by the purchaser while (a) being transferred into storage, (b) being maintained in storage, or (c) being prepared for shipment or packaging.

4. QUALITY ASSURANCE

The supplier shall document, implement, and maintain a quality program in compliance with ASME NQA-1-1997.

The supplier shall permit the purchaser to conduct pre-award and continuing evaluation of the Quality Program.

Personnel performing NDE examinations, specifically radiographic, ultrasonic, liquid penetrant, and visual shall be certified to American Society for Nondestructive Testing (ASNT) Number SNT-TC-1A and certification documentation shall be made available to the purchaser.

Unless otherwise specified, the supplier shall be responsible for the performance of all tests and inspections required prior to submission to the purchaser of any fuel element for acceptance. Provided, however, that the performance of such tests and inspections is in addition to, and does not limit, the right of the purchaser to conduct such other tests and inspections as the purchaser deems necessary to assure that all fuel elements are in conformance with all requirements of this specification. Except as otherwise specified, the supplier may use for inspection purposes either his own or any commercial laboratory acceptable to the purchaser. Records of all tests and examinations shall be kept complete and available to the purchaser. All test and measuring equipment shall be calibrated per the requirements of Standard MIL-C-45662.

The following applies to specified limits for requirements on core density per Section 3.4.1.1 and 4.2 and all dimensional requirements of this specification. For purposes of determining conformance with this specification an observed value or a calculated value shall be rounded off to the nearest unit in the last right hand place of figures used in expressing the limiting value in accordance with the rounding-off method of the Recommended Practices for Designating Significant Places in Specified Limiting Values (ASTM E29).

The supplier shall prepare for his use and the purchaser's approval an integrated manufacturing and inspection test plan. The plan shall include all manufacturing operations, equipment and tooling used, inspection requirements and gaging used, and mandatory hold points established by the purchaser.

Any materials or fuel element components that are fabricated using equipment, personnel, or processes that are not in accordance with approvals as previously granted by the purchaser are subject to *rejection* (see def.). A report of any such incident must be submitted in accordance with Section 6.3.7.

Fuel element inspection for shipment or rejection will be made by the on-site purchaser's representative at the supplier's plant. Final fuel element acceptance will be made by the purchaser at the User's facility.

4.1 Materials

Compliance with the material requirements of Section 3.2 shall be established by supplier certification. A "Certification of Chemical Analysis" or a certified mill test report shall be supplied to the purchaser for each lot of material used in the fabrication of fuel elements. This certificate shall give the results of the chemical analysis for the material. All fuel element materials shall be traceable.

4.2 Core Density

The density of the fuel cores required in Section 3.4.1.3 shall be determined by the Archimedes principle. During qualification of the fuel plate core void density required by Section 3.4.1.3 shall be determined on all qualification fuel plates submitted. After the particular plate type has been qualified, 100% inspection for void density is not required for production lots of fuel plates. For production lots, three randomly selected fuel plates from each lot shall be inspected for void volume density. Should any one of these plates be discrepant, the entire lot must then be inspected for void volume density. If void density discrepancies appear regularly in the process, the purchaser may request 100% inspection.

The actual core volume shall be calculated by the following formula where: weight units are in grams and volumes in cubic centimeters.

$$V_c = V_p - \frac{W_p - W_c}{\rho_{AL}}$$

where:

V_c = immersion volume of fuel plate core

V_p = volume of fuel plate

ρ_{AL} = density of aluminum used for fuel plate cladding

= 2.715 gms/cc

W_p = weight of plate

W_c = deburred weight of fuel plate core compact

The theoretical core volume shall be calculated by the following formulas:

$$V_{ct} = \left(\frac{WU_3Si_2}{\rho U_3Si_2}\right) + \left(\frac{W_{Al}}{\rho_{Al}}\right)$$

where:

V_{ct}	=	calculated theoretical core volume
WU_3Si_2	=	weight of U_3Si_2 powder in core
W_{Al}	=	weight of aluminum matrix powder in core
ρU_3Si_2	=	density of U_3Si_2 powder as measured
ρ_{Al}	=	density of aluminum powder used for core matrix = 2.710 gms/cc

The void percent in the core shall be calculated using the following formula:

$$V\% = \frac{V_c - V_{ct}}{V_c} \times (100\%)$$

where:

V%	=	percent voids in the fuel plate core

4.3 Fuel Loading

Verification of the fuel loading as specified in Section 3.4.1.2 shall be in conformance to the supplier's procedure required in Section 6.3.1.

In order to determine compliance with the fuel density requirements of Section 4.4, the U-235 loading of the fuel plate, as determined in accordance with the procedures of Section 6.3.1, will be divided by the core volume (Vc) as calculated by the method described in the second paragraph of Section 4.2.

4.4 Fuel Homogeneity

Fuel core homogeneity requirements shall be complied with by a one-piece radiograph of all fuel plates from each fuel plate lot and evaluation of the radiograph by calibrated densitometer measurements. Purchaser approved density standards may be used by the supplier. Fuel plates and density standards shall be exposed simultaneously. Fuel plate density variations shall be determined by comparison of fuel plate areas to corresponding areas of the standard.

All fuel plates shall be inspected for homogeneity. Homogeneity of the fuel plate core shall be determined by radiograph film density measurements with a densitometer having a 0.080 inch aperture.

When determining fuel core density from plate radiographs, the brighter the image on the radiograph, the more dense is the uranium and the lower the number indicated on the densitometer. The darker the image on the radiograph, the less dense is the uranium and the larger the number indicated on the densitometer. A +30% fuel core density and a +20% fuel core density is indicated by the densitometer readings in the suspect area being 30% or 20% lower than the average densitometer readings for all core locations. A –30% or a –20% fuel core density is indicated by the densitometer readings in the suspect area being 30% or 20% higher than the average densitometer readings for all fuel core locations.

Any one-half inch diameter or greater spot in the plate fuel core area, other than the dogbone area shall not be less in fuel density than –20% of the average fuel density for all fuel core locations. To determine the low density of a one-half inch diameter area, the film is maneuvered under the densitometer in the low-density area until the highest number possible is obtained on the densitometer. This number is recorded. Then four readings are taken one-fourth inch from this spot and symmetrical around it. The average of these five readings is compared to the average densitometer readings for all fuel core locations.

If density standards are used, the average densitometer readings of all fuel core locations will be replaced by the nominal density standard and comparisons will be determined between the suspect spot on the radiograph and the –30% and –20% standards. For the +30% and +20% homogeneity overload inspection, compare the nominal density standard to the suspect area. In this case densitometer units from nominal of the fuel plate represent the following percentages: -0.15 = +30%; -0.10 = +20%. Fuel plates exceeding these limits are discrepant.

For rectangular shaped, suspected discrepant areas that are evaluated to the one-half inch criteria, orient the four symmetrical readings such that worst case readings will be taken.

Between the minimum and maximum permissible fuel core length boundary, fuel underload condition shall not be evaluated.

Any indication of un-alloyed uranium as determined by radiography shall be cause for rejection.

Any 0.080 inch diameter spot in the fuel plate dogbone area (area within one inch of each fuel core end) shall not be greater in fuel density than +30% of the average fuel density for all core locations. Any one-half inch diameter area in the dogbone area shall not be less in fuel density than –30% of the average fuel density for all fuel core locations. The actual dogbone shall not be more than one-half inch in the longitudinal direction.

Other than the dogbone areas near ends of fuel core, any one-half inch diameter area shall not be greater in fuel density than +20% of the average fuel density for all fuel core locations. To determine the high density of a one-half inch diameter area, the film is maneuvered under the densitometer in the high-density area until the lowest number possible is obtained on the densitometer. This number is recorded. Then four readings are taken one-fourth inch from this spot and symmetrically around it. The average of these five readings is compared to the average densitometer readings for all fuel core locations.

Unless otherwise specified, purchaser approval of all radiographs is required prior to assembly of fuel plates into elements.

4.5 Core Configuration

Each finish-cut flat fuel plate shall be radiographed in accordance with Appendix A and evaluated for compliance with Section 3.3.1.2.

Visual radiograph inspections will be performed without magnification on a light table having a light intensity of 450 to 600 ft-candles at the table surface and the area darkened to give a light range of 5 to 15 ft-candles 18 in. above the light table with radiograph film in place on the table.

4.6 Bond Integrity

4.6.1 <u>Blister Anneal</u>:

After the fuel plate has been hot rolled, it shall be heated to 900°F±13°F, held at that temperature for a period of 2 hours, -15 minutes, +30 minutes, removed from furnace, and allowed to air cool.

Any blisters, in the fuel core region larger than a 0.060 in. diameter or any blister in the frame region of the fuel plate larger than 0.120 in. diameter shall result in rejection of the associated fuel plate. A maximum of two blisters less than 0.060 in. diameter is allowed in the fuel core area, provided they are more than 0.250 in. apart. A maximum of two blisters in any of the four sides of the *picture frame* (see def.)(a maximum of eight) region smaller than 0.120 in. can be tolerated providing that no blister is any closer to the plate edge or end or to another blister than the major dimension of the blister and no blister is closer to the plate edge or end than 0.050 inch. When there is question as to size or location of the blisters, the acceptance or rejection of the plate shall be determined in the ultrasonic inspection of Section 4.6.2.

4.6.2 Ultrasonic Scanning:

The finished fuel plate area shall be ultrasonically inspected in compliance with ASME Boiler and Pressure Vessel Code, Section V, Article 5, Paragraphs T-110, T-510, T-520, T-521, T-522-a, b, c, e, g, i, j, k, l, o, T-523, T-523-1, and T-534. Any indication of discontinuity in the fuel core region equivalent to that indicated by a 0.060 in. diameter standard or any indication of a discontinuity in the frame region of the fuel plate equivalent to that indicated by a 0.120 in. diameter standard shall result in rejection of the associated fuel plate. Acceptance criteria for number of blisters revealed by ultrasonic scanning are per Section 4.6.1. Any discontinuities, inside the fuel plate, other than blisters and for which acceptance criterion is not already stated, shall be described by the supplier and evaluated by the purchaser.

4.6.3 Metallographic Examination.

During qualification, one fuel plate per lot selected for qualification per Section 3.1.1 will be sectioned per Figure 1, polished and etched, and examined at 50× or above for bond and clad-core-clad dimensions per the requirements of Sections 3.3.1.3 and 3.3.1.4, and Drawing 635463, respectively.

If the fuel plate fails the metallographic examination for grain growth, voids, laminations, core cracking or separation, or foreign particles or materials, then randomly selected another plate in the lot for metallographic examination. If this plate fails the examination, reject the lot.

Fuel plates selected for destruction tests may be rejected fuel plates, providing the attribute to be tested for is not affected by the cause for rejection. Reject fuel plates so used must have purchaser approval before destruct tests are performed.

4.7 Internal Defects

Any internal defect in excess of the requirement of Section 3.3.1.3 in the fuel core, including voids, laminations, U_3Si_2 segregation, clumping, core cracking or separation, or foreign particles or materials, which is identified by any measurement technique, including radiography per Section 4.4, ultrasonic scanning per Section 4.6.2, or metallography per Section 4.6.3, shall be cause for rejection of the fuel plate.

4.8 Surface Finish and Defects

Compliance with requirements of Section 3.5 shall be established by visual inspection of all fuel plates and fuel elements. Out-of-specification defects shall be measured for size and depth and reported to the purchaser.

4.9 Clad-Core-Clad Dimensions

Fuel Plate Qualification requirements of section 3.1.1 shall be established by ultrasonic techniques using the purchaser-supplied, min-clad inspection gage. All fuel plates will be subjected to ultrasonic min-clad inspection with the fuel core region scanned for each plate. Ultrasonic min-clad inspection shall be accomplished by calibration of the min-clad gage, using the Advanced Test Reactor (ATR) Standard (8E0777) scanned at the normal mode of 0.008 inches. The min-clad gage will then be adjusted and the fuel plates will be scanned at a depth of 0.010 inches. Ultrasonic Test (UT) traces showing fuel at the 0.010 inch depth will be compared to the 0.008 inch standard to determine plate acceptability. If the density of indications from fuel plate exceeds the ATR standard density of indications, the plate is rejectable.

NOTE: *The ATR standard is a small piece of an ATR fuel plate that has fuel particles near the surface. It is used on the UT min-clad machine to indicate min-clad indications and compare the density of these indications to any indications noted from a fuel plate being inspected by UT.*

During the fuel plate qualification process, compliance with the requirements of Section 3.3.1.4 shall be established by destructive analysis of one fuel plate per lot in accordance with Figure 1.

After fuel plate qualification, all production plates shall be min-clad ultrasonic inspected at a depth of 0.010 inches. Those plates discrepant at 0.010 inches shall be rescanned at 0.008 inches. Plates which are acceptable when re-scanned at 0.008 inches shall be submitted on Information/Change Request (Form 540.33) to the purchaser.

4.10 Cleanliness

Fuel plate, fuel assembly, and fuel element container cleanliness requirements of Section 3.5.2 shall be established by visual inspection without magnification of all fuel plates, fuel assemblies, and fuel element containers.

4.11 Contamination

The surfaces of each fuel plate and fuel assembly shall be counted or smeared and counted for alpha-beta-gamma contamination and meet the requirements of Section 3.5.3.

4.12 Dimensional

It shall be the supplier's responsibility to assure that fabrication is performed in accordance with all dimensions delineated in the Drawings referenced in Section 2.1.2. Noncomplying design dimensions on fuel plates, fuel assemblies, and fuel element containers (actual measurements) shall be submitted to the purchaser for review and approval. Any discrepant component shall not be used in a fuel element assembly unless approved.

The supplier is to certify to compliance with the design dimensional requirements delineated in the Drawings referenced in Section 2.1.2.

All dimensions of finished fuel plates, fuel assemblies and fuel element containers apply at 75°F±5°F.

4.12.1 <u>Final Dimensional Inspection</u>.

Dimensions required by this specification and drawings of Section 2.1.2 shall be inspected using a purchaser approved sample plan and recorded on an inspection sheet with "in specification" dimensions recorded by check mark, "OK," or actual measurements and "out of specification" dimensions recorded as actual measurements.

4.13 Reactor Components and Spare Fuel Element Parts

Reactor components and spare fuel element parts not assembled into fuel element assemblies are required to be certified. The certification shall consist of material certification, fabrication verification, and supplier certificate of compliance to the specification and drawing requirements. The certification documents shall be submitted to the purchaser and user.

5. PACKAGING AND SHIPPING

Packaging and shipping of the fuel elements shall be performed using a Purchaser approved procedure in compliance with this section.

- The purchaser shall provide shipping containers to protect the fuel elements from damage during shipment and which conform to the applicable requirements of the Departments of Energy and Transportation, and other regulatory agencies having jurisdiction of the shipment of radioactive materials. Re-useable shipping containers will be returned to the Supplier by the User at the Purchaser's expense.
- The Supplier is responsible for loading the fuel elements into shipping containers in a sealed polyethylene sleeve in a cleaned dry condition and free of extraneous materials.
- The Supplier shall take necessary precautions during packaging to prevent damage to the fuel elements during shipment. Each container shall be provided with a tamper-proof seal. Loading and shipping documents for the container shall be prepared in accordance with the applicable regulatory requirements.
- The Supplier shall make arrangements for shipment to the User.

6. NOTES

6.1 Definitions

For the purpose of this specification, the following terms are identified:

Batch. The amount of silicide powder mixture which is handled as a unit or traceable to a common step.

Blended. To mix or mingle constituents of a batch.

Certification. The action of determining, verifying and attesting in writing (signed by a qualified party) to the qualifications of personnel and material.

Cladding. The aluminum covers bonded to the fuel core and the picture frame.

Control Fuel Element Assembly. An assembly consisting of the control fuel element container with eight fuel plates.

Controlled Work Area. A work area to which access of personnel, tools, and materials is limited and physically controlled. Temporary enclosures may be used where adjacent activities produce contamination which is detrimental to the job.

Development. A determination of processes, equipment, and parameters required to produce a product in compliance with this specification.

Dogbone Area. Thickening of the fuel core usually in the last 1/2 in. of the core, which may result in clad thinning in those areas.

Dummy Fuel Element Assembly. An assembly consisting of a fuel element container with unfueled simulated dummy fuel plates.

Dummy Fuel Plate. A non-fueled plate made entirely from the aluminum material specified in this document.

Edge Clad. The distance between the edge of the fuel core and the edge of the finished fuel plate, before any stray particles are removed, in the width direction as determined by radiography of a flat fuel plate.

Failure. A condition where the fabrication process appears to be out of control or a breakdown or damage to equipment creates excessive costs and/or schedule delays.

Fuel Compact. A quantity of uranium silicide powder and aluminum powder, cold compacted by pressing into a solid block for assembly into packs for hot roll and cold roll into fuel plates. The compacts are encased in frames and cover plates to form the pack.

Fuel Assembly. An assembly of fuel plates and hardware components. This includes both the standard and control fuel elements.

Fuel Core. The uranium-bearing region of each Fuel Plate.

Fuel Plate. The Fuel Core complete with aluminum frame and cladding.

Graphite Reflector Assemblies. A component consisting of a graphite container assembly with a graphite block inside.

In-Process Controls. Inspections and tests made during production to ensure that the manufacturing processes, equipment, and personnel are producing a product meeting specified requirements.

Irradiation Facility Assemblies. A component consisting of a round tube attached inside a graphite container assembly with graphite blocks filling the annulus between the tube and container. Inserted within the tube is the isotope capsule assemblies.

Lot. A group of pieces handled as a unit or material traceable to a common processing step.

Manufacture(ing). All fabrication, assembly, test, inspection and quality control processes. Fabrication is a synonym for Manufacture.

Pack. The fuel compact, picture frame, and cover plates, assembled together for hot rolling.

Picture Frame. The window shaped aluminum frame, which holds the fuel compact.

Plates. See Fuel Plates.

Procedure. The detailed description of the series of processes during manufacture and inspection, which follow a regular definite order (not to be construed as an outline).

Production. That phase of the program, following Qualification, during which the product is in Manufacture.

Purchaser. Idaho National Laboratory (INL).

Qualification. A demonstration that the Manufacturing process, equipment and personnel can produce a Product in compliance with this Specification.

Quality Control. The sampling plans, inspections, tests and records required and used during Production to assure that the Product is in compliance with this Specification.

Rejection. Materials, parts, components, or assembly products, which will not be accepted as fulfilling the contract requirements because of noncompliance with this Specification.

Requalification. A demonstration that a single or group of manufacturing processes, equipment and personnel can produce a product in compliance with this specification after the original qualification has been completed and becomes invalid.

Silicide. Uranium metal alloyed with silicon and fabricated per the requirements of Specification TRTR-14. The word "fuel" is a synonym for Silicide.

Specification. All parts and appendixes to this document, its references, drawings, and standards, as may be modified from time to time by contractual document.

Standard Fuel Element Assembly. An assembly consisting of the fuel element container with fourteen (14) fuel plates.

Stray Fuel Particles. Isolated fuel particles lying outside the maximum fuel core outline defined on Drawing 635463.

Supplier. The primary vendor selected by INL to manufacture the product.

User. Purdue University, at West Lafayette, Indiana.

6.2 Purchaser Tests

None

6.3 Submittals

The following data and records shall be supplied to the purchaser in the quantities stated. The purchaser's approval, prior to implementation, is required on those marked with an asterisk. All records and data shall be maintained by the supplier for the duration of the Purdue University fuel element contract.

The granting of approval by the purchaser of design, working drawings, specifications, requests, and other technical data submitted by the supplier under the provisions of the subcontract or specification shall not affect or relieve the supplier from such responsibility as the supplier has with respect to adequacy or correctness of the design, working drawings specifications, reports, and other technical data.

6.3.1 Preproduction:

Documents requiring approval must be submitted prior to production use. The number of copies shall be as specified by the Vendor Data Schedule. These documents include:

- *A detailed description as to the weighing procedure by which the supplier proposes to assign Plate U-235 content as required in Section 4.3.
- Included in the description must be sampling, analytical, and quality control procedures; a statement as to the established accuracy and precision of the assigned fuel plate and fuel element U-235 content; developmental and production data in support of the accuracy and precision estimate; and data which at the 95% confidence level, shows that the method used to assign U-235 values has a bias which is less than 0.2% relative
- *A detailed description as to the manner the supplier will use to verify the fuel Plate U-235 value as required by Section 4.3

- *All fabrication, assembly, cleaning, surface treating, handling, and decontamination procedures (not to be construed as an outline)

- *All production test, inspection, and quality control procedures, including all nondestructive and destructive tests and all standards and sampling section drawings. All data from these tests, including but not limited to: radiographs, metallographic samples, ultrasonic testing traces, and qualification yield rates

- *All packaging, storage and shipping procedures

6.3.2 Pre-repair:

*All repair programs and procedures prior to use.

6.3.3 Manufacturing Schedule:

*A schedule using a purchaser approved technique.

6.3.3.1 Reports.

1. Biweekly qualifications phase summary status report. The first such report shall be initiated 1 month after date of contract award.

2. Three (3) copies of a monthly report detailing program progress against a previously submitted schedule shall be supplied by the supplier to the purchaser. Report type, format and submittal schedule shall be as agreed upon between the purchaser and supplier.

6.3.4 Delivery Submittals:

Three copies (except as noted) of the following data and records shall be sent prior to or accompany the shipments. The supplier shall maintain copies of these records for at least 10 years and until the supplier has received written approval from the purchaser for disposition or disposal:

- Certification of product compliance to the requirements of this specification to include any test data pertaining thereto

- Supplier's core compact data sheets, with individual fuel plate uranium composition data including:

- Serial number with *batch* (see def.) identification

Uranium content
Fuel plate core weight
U-235 enrichment
Total quantity U-235 content
Core void density data

- Individual fuel element composition data, including:

  Uranium content
  U-235 content
  Serial number of each plate in the element

- Radiation count from fuel plate and fuel element exterior as required by Section 3.5.3 and 4.11. The counting period, counter, background, efficiency, and type of counter used shall be reported

- List of all applicable waivers and deviations and related fuel plates or fuel elements

- If performed, documented evidence of the performance and test results of the boehmite formation from the fuel element surface treatment per Section 3.8.

6.3.5 <u>Fuel Plate Radiographs</u>:

Fuel plate radiographs of all accepted fuel plates required by Sections 4.4 and 4.5 and Appendix A shall be sent to the user.

6.3.6 <u>Core Compact Data Sheets</u>:

Supplier's fuel core compact data sheets shall be supplied to the INL Quality Assurance Representative as they are generated.

6.3.7 <u>Report of Production by Unapproved Process</u>:

Whenever the supplier's previously submitted and approved process control limits are exceeded, or any material or fuel element components are fabricated using equipment, personnel, or processes which are not purchaser approved, the time, nature, description, corrective action to be taken, and proposed further corrective action shall be reported immediately by the supplier, with a written report to the purchaser to follow within 10 working days.

1. T Samples. Transverse to be taken equally space along Fuel Core length.
2. L Samples. Longitudinal to be taken at centerline and to include the Dogbone Area.

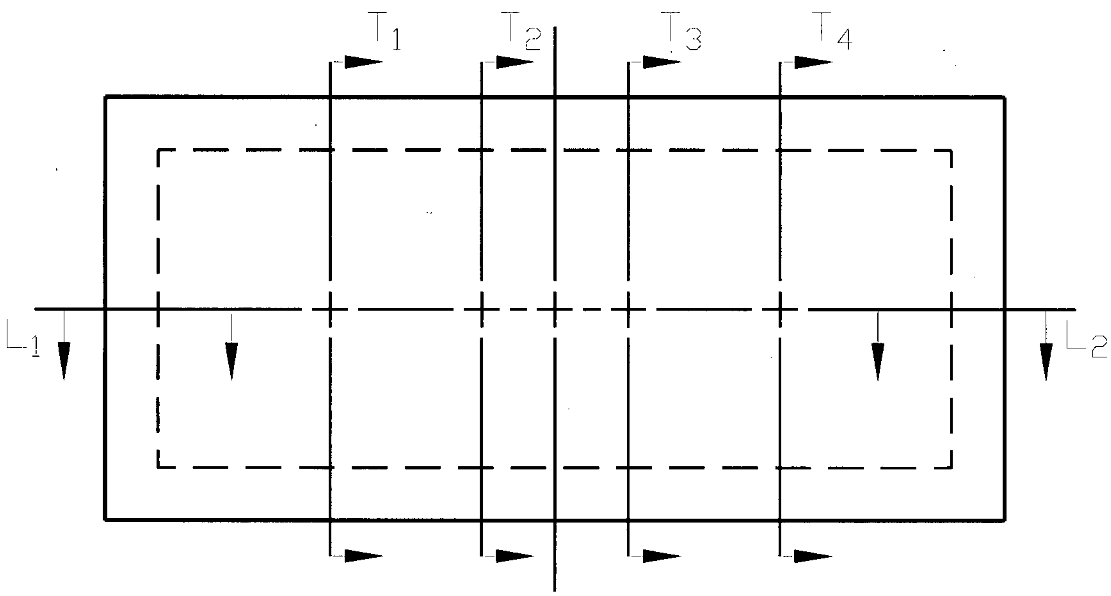


Figure 1. Purdue University Fuel Plate Sampling Procedures For Destructive Tests.

APPENDIX A

Requirements for Radiography of Purdue University Fuel Plates

1. Scope

This specification provides requirements for radiography of Purdue University reactor fuel plates, acceptable film quality and film identification.

2. Requirements

A procedure must be written to specify the details for achieving acceptable fuel plate radiographs. The procedure must include the requirements given in this specification.

2.1 Equipment Setup

The voltage shall be 100 k.v.p. with a focal spot size of 5 mm maximum. The distance between the focal point and the plate shall be at least twice the length of the plate. The focal point shall be centered laterally and longitudinally over the plate or group of plates.

2.2 Film

2.2.1 The image outline shall be clear and sharp; the film shall be free of runs, streaks, scratches, blurs, and cassette defect that will affect the area covered by the fuel plates.

2.2.2 The film density of all points of the radiograph that correspond to the fuel plate border locations outside the plate core shall provide densitometer readings between 1.5 and 2.7. Film density as read over the nominal density standards shall provide densitometer readings between 0.9 and 1.5.

2.2.3 The film shall be extreme sensitivity, extra fine grain, high contrast, double emulsion, industrial x-ray type, (Kodak type M or equal) which is acceptable to the purchaser. Development of the film shall be in accordance with the manufacturer's recommendation.

2.2.4 Film Identification

2.2.5 A system of identification of the film shall be provided by the supplier, which shall show as a minimum:

A. Plate lot number

B. Plate type and serial number

C. Orientation of density standard

D. Density standard identification

E. Date of radiography.

APPENDIX B

Welding Requirements and Qualification for Purdue University Fuel Elements

1. Scope

The requirements for welding and for the evaluation of welds applicable to the Purdue University Fuel Element Container and components are established by this Appendix.

1.1 Application. This document defines requirements for the following:

1.1.1 Welding procedure qualification.

1.1.2 Performance qualification of welders, welding equipment, and special fixturing.

1.1.3 Information to be included in welding procedure specifications.

1.1.4 Application of qualified procedures to production welding.

1.1.5 Destructive testing and nondestructive examination for qualification and for production welding.

1.2 Special Limitations for Applicability. The requirements contained in this appendix are to some degree based on RDT F6-2T. Those requirements applicable to Manual, GTAW, single pass, welding of Plug Joint welds, Corner Joint welds, and Partial Penetration Butt Joint welds have been included in this appendix. The introduction of a new weld design or weld process requiring a change in these limited parameters would require an appropriate review of RDT F6-2T for requirements applicable to the new parameters.

1.3 Definitions.

Arc Strike. Any localized melting, heat affected zones, or change in the contour of the surface of the finished weld or adjacent base metal resulting from an arc or heat generated by the passage of electrical energy between the weld or base metal and a current source; such as welding electrodes, electron beams, ground clamps, high frequency arc, etc.

Automatic Welding. Welding with equipment which performs the entire welding operation without constant observation and adjustment of controls by an operator. The equipment may or may not perform the loading and unloading of the work.

Face of Weld. The exposed surface of a weld on the side from which welding was done.

Face Reinforcement. Reinforcement of weld at the side of the joint from which welding was done.

Heat. A single homogeneous melt of metal or alloy.

Joint Penetration. The minimum depth a groove or flange weld extends from its face into a joint, exclusive of reinforcement.

Machine Welding. Welding with equipment which performs the welding operations under the constant observation and control of an operator. The equipment may or may not perform the loading and unloading of the work.

Position of Welding. The terms related to positions of welding for joint types and welding processes and the position limits are defined in Section IX, ASME Boiler and Pressure Vessel Code.

Repair. The process of restoring a nonconforming item characteristic to an acceptable condition, although it does not conform to a specified requirement.

Rework. The process by which a nonconforming item is made to conform to specified requirements.

Root of a Joint. That portion of a joint to be welded where the members approach closest to each other. In cross section the root of the joint may be a point, a line or an area.

Root of a Weld. The points, as shown in cross section, at which the back of the weld intersects the base metal surfaces.

Root Penetration. The depth a groove weld extends into the root of a joint measured on the centerline of the root cross section.

Root Reinforcement. Reinforcement of weld at the side opposite that from which welding was done.

Root Surface. The exposed surface of a weld on the side opposite that from which welding was done.

Size of a Groove Weld. The joint penetration (depth of chamfering plus root penetration when specified).

Undercut. A groove melted into the base metal adjacent to the toe or root of a weld and left unfilled by weld metal.

Underfill. A depression on the face of the weld or root surface extending below the surface of the adjacent base metal.

Welder and Welding Operator Performance Qualification. The tests to demonstrate a welder's or welding operator's ability to produce welds meeting prescribed standards.

Welder. One who is capable of performing a manual or semiautomatic welding operation (sometimes erroneously used to denote a welding machine).

Welding Operator. One who operates machine or automatic welding equipment.

Welding Procedure Qualification. The test to demonstrate that welds made by a specified procedure can meet prescribed standards.

Welding Procedure Specification. A written welding procedure which specifies the detailed methods and practices to be used in the production of a weldment and how they shall be carried out. A specification includes all elements of a procedure necessary to produce a satisfactory weldment. Examples of some of the elements included in a specification are: material used, preparation of base materials, preheat and postheat cleaning, assembly method and sequence, fixturing, heat treatments, joint welding procedures, preweld and postweld nondestructive examinations, repair, rework, etc.

Welding Procedure. The detailed methods and practices including all joint welding procedures.

2. Reference Document

The following documents are a part of this appendix to the extent specified herein. The issue of a document in effect on the date of the invitation to bid, including any amendments also in effect on that date, shall apply unless otherwise specified. Where this appendix appears to conflict with the requirements of a reference document, such conflict shall be brought to the attention of the purchaser for resolution.

2.1 American Society for Testing and Materials (ASTM) Standards

2.1.1 ASTM E2, Preparation of Micrographs of Metals and Alloys

2.1.2 ASTM E3, Preparation of Metallographic Specimen

2.2 American Society of Mechanical Engineers (ASME) Codes

2.2.1 ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications

2.3 American Welding Society (AWS) Standards

2.3.1 AWS A2.2, Nondestructive Testing Symbols

2.3.2 AWS A3.0, Terms and Definitions

3. Weld Qualification Requirements

3.1 General Requirements

3.1.1 All welding procedures, welders, or welding machine operators shall be qualified in accordance with the provisions identified in this Appendix.

3.1.2 Weld Procedure and Performance Qualification Testing previously qualified to these requirements under other contracts may be used. Existing records to support previously qualified procedures and personnel are subject to review by the purchaser.

3.1.3 Base materials and filler material shall comply with the requirements of the drawings.

3.1.4 Welding processes which satisfy the specified requirements and produce the quality required by this Appendix are permissible. Welding procedures which utilize fluxes and coatings shall not be used.

3.1.5 Fixtures: The capability of fixtures for aligning parts shall be demonstrated before welding of production parts is initiated. If chill bars or blocks are used, the type of material and their location with respect to the joint shall be included in the procedure specification.

3.1.6 Position of Qualification Welds. All procedure and performance qualification test welds shall be made in the same positions as for production welds.

3.1.7 Special Conditions for Qualification Welds: All procedure and performance qualification test welds shall be made under conditions which simulate the actual production welding conditions. These conditions shall include space limitations, joint accessibility, degree of comfort due to heat, position and other handicaps or environmental factors which the welder or welding operator will endure during actual production welding.

3.1.8 Heat Treatment. Weld preheat and postheat treatments shall not be used without prior approval by the purchaser.

3.1.9 Interpass Temperature. For multi-pass weld, the weld interpass temperature shall not be less than 60° F or greater than 350° F without prior approval by the purchaser.

3.1.10 Records. Records of welding, associated processing, and inspection shall be maintained for all welds. Complete records may consist of inspection forms, routings, or reference to Operating Procedures or other documents. These records shall include at least the following:

1. Base Material (Type, material specification, heat or lot number).
2. Filler Material (Type, material specification, heat or lot number).
3. Cleaning procedures.
4. Joint identification and weld maps when applicable.
5. Welding machine type and identification.
6. Welding procedure specification.
7. Welder or welding operator qualification.
8. Procedure and performance qualification.
9. Current-voltage data for machine or automatic welding.
10. Date welds are made.
11. Inert gas mixture, when applicable.
12. Nondestructive examination procedure.
13. Nondestructive examination personnel identification.
14. Examinations and tests (nondestructive and destructive) and the results.
15. Photomacrographs and photomicrographs.
16. Metallographic specimens.
17. If applicable, rework and repair of welds.
18. Disposition of welds.

3.2 Welding Procedure Specification

3.2.1 The welding procedure specification shall meet the requirements of this Appendix, and shall be submitted to the purchaser for information.

3.2.2 The welding procedure specification shall include all essential elements and details, as required by this section, to cover each joint to be welded by the supplier. Each joint shall be identified in the welding procedure specification. The specification shall include a joint design sketch for each joint welding procedure even if the specification references drawing numbers.

3.2.3 The following basic information and essential variables shall be included in sufficient detail to assure that compliance with the requirements of the specification can be verified:

1. Basic Information

    a. Joint Design: (the joint geometry, fit-up, and other required dimensions of the welded joint) tolerances and material thickness.

    b. Method of arc initiation

    c. Electrode size (for gas tungsten arc welding)

    d. Gas type and flow rate (shielding and backing gas)

    e. Welding current range for manual welding

    f. Whether tack welds or fixtures are used for assembly of the joint for welding

    g. Method and frequency of cleaning

    h. Number of weld layers and passes

    i. Whether stringer beads or weave beads are used

2. Essential Variables

    a. General, All Welding Processes.

        i. A change from a base material type or grade (materials of the same nominal chemical analysis and mechanical property range, even though a different product form) to any other base material type or grade. When joints are made between two different types or grades of base material, a procedure qualification shall be made for the applicable combinations of materials, even though procedure qualification tests have been made for each of the two base materials welded to itself.

ii. A change of filler metal type or classification to any other type of classification

iii. A change in welding position.

iv. A change in vertical welding direction, i.e., from upward to downward or vice versa.

v. The addition or omission of integral backing (e.g., "butt-lap" type joint).

vi. The addition or omission of nonfusing metal retainers.

vii. The addition or omission of filler metal to the joint.

viii. Any change in the method by which filler is added, such as preplaced shims, preplaced wire, preplaced consumable inserts, wire feed, or prior weld metal surfacing ("buttering") of one or both joint faces.

ix. The addition or omission or any type of preplaced consumable inserts or joint surfacing.

x. A change in the shape or size of preplaced consumable inserts or joint surfacing.

xi. A change from multiple pass welds to single pass welds.

xii. The omission of inert gas backing during welding, except that requalification is not required where a qualified welding procedure is changed to omit the inert gas backing and then is used only for a single welded butt joint with a backing strip, or a fillet weld. For multiple pass welding, the omission of inert gas backing during welding until three layers or 3/16 of weld metal thickness has been deposited, whichever is greatest.

xiii. A change from one welding process to any other process or combination of welding processes.

b. Manual Welding, All Welding Processes.

i. An increase in the standard size of filler metal from that stated and qualified in the procedure specification.

ii. A change in joint geometry which violates the tolerances given for the joint geometry elements listed below:

Bevel Angle: State in procedure specification. Tolerance: Minus 5%.

Groove Angle: State in procedure specification. Tolerance: Minus 5%

Alignment Tolerance: Assign value in procedure specification. Qualify procedure for single welded joints using maximum permissible misalignment in a portion of the joint.

c. Gas Tungsten Arc Process.

i. A change of electrode material type.

ii. A change in arc starting methods.

iii. A change from a single shielding gas to any other shielding gas or to a mixture of shielding gases or a change in specified composition of gas mixture.

iv. A decrease in shielded gas flow rate of more than ten percent.

3.3 Welding Procedure Qualification

3.3.1 The welding procedure shall be qualified to the requirements of this section.

3.3.2 All welding used in qualifying a welding procedure shall be performed in accordance with a welding procedure specification.

3.3.3 Before any welding is performed on production components, the supplier shall qualify each proposed welding procedure by:

1. Recording all essential elements of the welding procedure in a welding procedure specification (see Section 3.2)

2. Verifying the welding procedure specification by welding test specimens representing each joint to be welded in production and performing nondestructive examination and destructive tests in accordance with the requirements of this Appendix.

3. Submitting to the purchaser, for information, the welding procedure specification and a certified copy of the detailed results obtained from the tests performed on the test welds. The metallographic sections required by this Appendix shall also be submitted to the purchaser.

3.3.4 Essential Variables. The welding procedure shall be set us as a new welding procedure specification and shall be completely re-qualified when any of the changes listed in Section 3.2.3.2 are made in the procedure.

3.3.5 Chart Recordings. Current-voltage-time charts shall be used for each procedure qualification weld for automatic or machine welding. Calibrated current and voltage indicating meters may be substituted for trace chart type equipment for manual welding. The current and voltage ranges shall be recorded for manual welding.

3.4 Welder Performance Qualification

3.4.1 Performance qualification weld tests shall meet the requirements of this section, except that any welder used to qualify the welding procedure shall also be considered qualified and additional performance weld tests are not required.

3.4.2 General.

1. The performance qualification tests are intended to determine the ability of welders to make sound welds.

2. The performance test may be terminated at any stage of the testing procedure whenever it becomes apparent to the supervisor conducting the tests that the welder does not have skill required to produce satisfactory results. In this event, the welder may be re-tested at the discretion of the supplier in accordance with 3.4.3.

3. Each supplier shall maintain a record of the procedures, including the essential variables, under which welders are examined and the results of the examinations.

3.4.3 Qualification of Welders.

1. Each welder shall pass the tests prescribed for procedure qualification except that tensile tests are not required. The essential variables and the test results obtained by each welder shall be recorded in a Performance Qualification Test Report. Any welder who performs acceptable welding procedure qualification tests shall be considered qualified.

2. Renewal of Qualification. Requalification of a welder is required when:

a. 90 or more days have elapsed since he last produced acceptable welds using the specific welding process, or

b. He has not performed acceptable welds using the production welding procedure.

c. Any time there is a specific reason to question a welder's ability to make welds meeting the requirements of this Appendix, requalification shall be required. Only one test weld shall be required for renewal of qualification. If this test weld fails to meet all of the original requirements, then a complete performance requalification shall be required.

3.4.4 Chart Recordings. Current-voltage-time charts shall be used for each procedure and performance qualification weld for automatic or machine welding. Calibrated current and voltage indicating meters may be substituted for trace chart type equipment for manual welding. The current and voltage ranges shall be recorded for manual welding.

3.5 Welding Machine Qualification

3.5.1 Performance qualification weld tests shall meet the requirements of this section, except that any welding machine used to qualify the welding procedure shall also be considered qualified and additional performance weld tests are not required.

1. The performance qualification tests are intended to determine the ability of welding machines to make sound welds.

2. Any time there is a specific reason to question a welding machine's ability to make welds meeting the requirements of this Appendix, requalification shall be required. Only one test weld shall be required for renewal of qualification. If this test weld fails to meet all of the original requirements, then a complete performance requalification shall be required. Welding machines used for the manual welding of any successful procedure or welder qualification tests shall be considered qualified for manual welding of all core components covered in this Appendix.

3.5.2 Welding machines used for the manual welding of any successful welder performance qualification tests shall be considered qualified for manual welding of all components covered in this session.

3.6 <u>Examination & Tests</u>

3.6.1 <u>Type of Test Required</u>. The following tests shall be used for the qualification of welding procedures and / or welders as applicable:

1. Nondestructive examination by a liquid penetrant method.

2. Nondestructive examination by Visual to test for soundness and surface characteristics of the weld.

3. Destructive examination by sectioning for metallographic examination of weld joints and adjacent areas to test for fusion, weld geometry, weld reinforcement, and soundness of the weld.

4. When the purchaser has reason to believe that the quality of any weldment is doubtful, he may require additional inspection.

5. <u>Nondestructive Examination and Tests</u>

    a. <u>Visual</u>. The test weld shall be examined visually prior to welding and after welding in accordance with Section 5.1

    b. <u>Liquid Penetrant</u>. The test weld shall be examined after the final layer in accordance with Section 3.6.2.2 using a color contrast method.

    c. Unless otherwise specified, inspection of procedure and performance qualification welds shall be performed in the final surface condition.

6. <u>Destructive Examination</u>. Each test weld shall be sectioned transversely to metallographically examine a minimum of:

    a. Three section faces for welds on cylindrical components less than 1 ¼ inch in diameter or for welds that are one to four inches long on non-cylindrical components.

    b. Four sections faces for welds in cylindrical components that are greater than 1 ¼ inch in diameter or for welds that are greater than four inches long on non-cylindrical components.

c. One section face for plug welds, arc spot welds, and welds that are less than one inch long on non-cylindrical components.

d. The cross section shall be polished and etched to provide clear definition of the structure in the fusion zone and the heat-affected zones.

e. For welds in (a) and (b) of this paragraph, one cross section shall be made through a weld start and a weld stop area and the remaining sections shall be made at random. For weld described in (c) of this paragraph, the cross section shall be made at the approximate centerline of the weld. Examination of the welds shall be in accordance with Section 3.6.2.3.

3.6.2 Acceptance Criteria for Qualification Test Welds

1. Visual Examination. Visual examination shall be in accordance with Section 5.1.

2. Liquid Penetrant Examination. Unless otherwise specified, final weld surfaces shall be examined using a color contrast method.

   a. For welded joints in materials less than 1/8 inch thick the following relevant indications are unacceptable.

      i. Any cracks.

      ii. Linear indications.

      iii. Indications with dimensions exceeding 1/64 inch.

      iv. Rounded indication separated by ¼ inch or less edge-to-edge.

      v. Five or more rounded indications in any six square inches of weld surface with the major dimension of this area not to exceed six inches with the area being taken in the most unfavorable location relative to the indication being evaluated.

   b. For all welds in materials 1/8 inch thick or greater, the following relevant indications are unacceptable. (Only those indications with major dimensions greater than 1/64 inch are considered relevant for item (iii).)

i. Any cracks.

ii. Any linear indications.

iii. Rounded indications with dimensions exceeding 10 percent of the nominal weld thickness or 1/8 inch, whichever is smaller. Rounded indications separated by 1/16 or less edge-to-edge shall be evaluated as a single indication.

iv. Four or more rounded indications in a line separated by 1/16 inch or less edge-to-edge.

v. Six or more indications in any six square inches of weld surface with the major dimension of this area not to exceed six inches with the area taken in the most unfavorable location relative to the indications being evaluated.

vi. Aligned indications in which the average of the center-to-center distance between any one indication and the two adjacent indications in a straight line is less than 3/16 inch.

3. <u>Metallographic Examination</u> Metallographic examinations shall be performed on qualification test welds at not less than 50X on test welds as required in this Section in accordance with ASTM E.2. Any cross section which is shown by metallographic examination to contain any of the following relevant defects shall be cause for rejection of the test welds.

   a. Any cracks.

   b. Incomplete fusion, or insufficient joint or root penetration.

   c. Any tungsten inclusions, slag inclusions, or porosity having a maximum dimension greater than 20 percent of the weld thickness or 1/32 inch, whichever is smaller.

   d. More than four tungsten inclusions or pores which have a maximum dimension less than in (c) above.

e. Any deviation from specified weld geometry or weld reinforcement.

3.6.3 Test Welds.

1. Procedure and / or welder performance qualification shall be made on test welds which duplicate the production weld joint type and which simulate the conditions to be used in production with respect to orientation, the essential variables listed in Section 3.2.3.2, and the dimensions of the parts to be joined to the extent that they affect heat requirements, relative motions, and distortions. All welding used in qualifying a welding procedure and / or welder performance shall be performed in accordance with the procedure specification.

2. For manual welding, two consecutive test welds shall be made when the weld joint is less than six inches in length. Only one test weld shall be required when the weld joint is 6 inches or greater in length.

3. All test welds shall be tested using the required tests listed in Section 3.6.1. To qualify the procedure specification used in making the test welds, each weld shall pass the required tests.

4. Repair of procedure or performance qualification test weld(s) is prohibited.

4. Production Welding

All production welding shall be accomplished using approved welding procedure specifications and qualified welders and/or welding operators.

5. Quality Acceptance of Production Welds

5.1 All completed production welds shall be visually examined in accordance with the following requirements:

5.1.1 General Visual Inspection Requirements. All visual examinations shall comply with the following:

1. Visual examination shall be made under direct daylight-type fluorescent lighting of at least 100 foot-candles at the work examination area.

2. Visual examination shall be performed with the aid of a 5x (minimum) magnifying glass.

3. The inspection required by this standard shall not be performed by the welder who made the welds. However, if the welder is qualified in accordance with this standard, he may visually inspect his own welds prior to the inspections required by this standard.

4. Personnel performing visual inspection shall have 20-20 vision, natural or corrected, stereo acuity, and shall not be color-blind.

5.1.2 Visual Acceptance Criteria (except for porosity). Visual examination of weld joint preparations and welds shall be performed in accordance with the following requirements to verify conformance to the written welding procedure, the design requirements, and the requirements of this standard:

1. Prior to welding, the weld joint edges and adjacent surfaces shall be examined for:

   a. Proper edge preparation, dimensions, and finish.

   b. Alignment and fitup of the pieces being welded.

   c. Verification of correct material by check of records.

   d. Verification of the cleanliness requirements.

2. After welding, the joint shall be examined in the final surface condition for:

   a. Contour, reinforcement and surface finish of welds.

   b. Degree of underfill, undercut, and overlap.

   c. Arc strikes, weld spatter and impression marking.

   d. Burn-through and fuse-through

3. Weld joints and surfaces which are shown by visual examination to have any of the following defects or areas of nonconformance are unacceptable:

   a. Any nonconformance revealed by 5.1.2.1.

b. Any zone of incomplete fusion.

c. Insufficient joint or root penetration.

d. Any undercutting, underfill, or burn through.

e. Any concavity on the face side of groove welds.

f. Any arc strikes, weld spatter, and impression marking.

g. Any visible inclusions, porosity, cracks, and unfilled craters.

4. Machined welds shall meet the drawing requirements.

5. All welds shall be free from surface markings resulting from mishandling, punching, scratching, etc., which exceed the specified surface requirements.

6. All welds shall be free of dross, or slag.

7. All welds shall be free of oxidation due to improper shielding and overheating which produce black or gray spalling or loose particles. Iridescent temper films and the dark metallic vapor deposits which may occur adjacent to the welds are acceptable. These films and deposits shall be removed by approved cleaning procedures when accessible.

5.1.3 Visible unacceptable porosity is as follows:

1. Four or more pores with a major dimension of 0.048 inches or more randomly positioned.

2. A single pore with a major dimension of 0.064 inches or more.

3. Six or more pores with a major dimension of 0.016 inches or greater in one weld.

4. Four or more porosity with a major dimension of 0.016 inches or greater, in line separated by less than 0.063 inches from edge to edge.

5.2 Repair of a defective weld by welding shall be limited to two attempts. Unacceptable defects shall be removed and re-examination made using liquid penetrant color contrast method to assure complete removal of the defect. If the removal of the defects results in reducing the thickness of the weld metal below the thickness of the base metal, the area shall be rewelded using a welding repair procedure which has been approved by the Purchaser. Whenever a defect is removed and subsequent repair by welding is not required, the excavated area shall be blended into the surrounding surface to remove any sharp notches, crevices or corners. Completed repairs shall be visually re-examined per Section 5.1. Records shall be maintained on all repairs and shall include the following:

5.2.1 Location of joint.

5.2.2 Location of defect.

5.2.3 Description of defect, including type and size.

5.2.4 Reference to approved repair procedure.

5.2.5 Inspections before and after repair and the results thereof.

5.2.6 Identification of repair welders or welding operators.

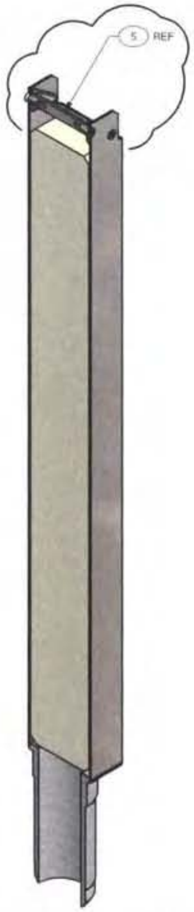
NOTES:

1. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
2. MARK ASSEMBLY IDENTIFICATION PER SPECIFICATION SPC-382.
3. REMOVE ALL BURRS AND SHARP EDGES.

2	1	SHEET	REV STATUS OF SHEETS
2	2	REV	

REVISIONS

REV	DESCRIPTION	EFFECTIVE DATE
1	REVISED DESCRIPTION OF ITEM 3; ADDED SHEET 2; ADDED ITEMS 2, 6, & 8; SEE DAR-509420	1/24/07
2	ADDED ITEM 9, CHANGED MATERIAL ITEM 7 CHANGED ITEM 5, SEE DAR-514751	4/4/07



-1 SECTIONED ISOMETRIC REFERENCE ONLY
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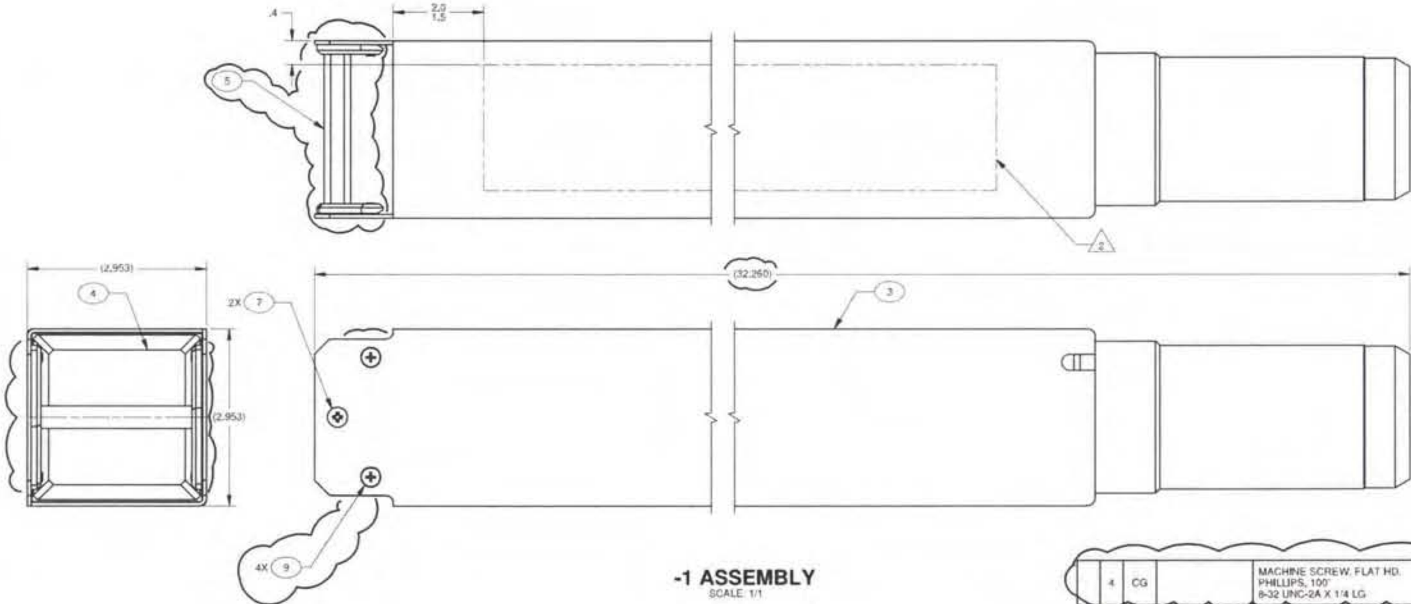


-1 ASSEMBLY
SCALE: 1/1

-2	-1	SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL/SPECIFICATION OR VENDOR NAME	ITEM NO.
	4	CG		MACHINE SCREW, FLAT HD. PHILLIPS, 100° 8-32 UNC-2A X 1/4 LG	316 SST OR 304 SST	9
1			635467-2	SOURCE DRIVE TOP		8
6	2	CG		MACHINE SCREW, FLAT HD. PHILLIPS, 100° 8-32 UNC-2A X 1/2 LG	316 SST OR 304 SST	7
1			635462-3	SOURCE DRIVE CONTAINER ASSEMBLY		6
	1		635464-2	BAIL ASSEMBLY		5
	1		635465-1	GRAPHITE BLOCK		4
	1		635462-1	GRAPHITE CONTAINER ASSEMBLY		3
		SC	-2	SOURCE DRIVE SUPPORT ASSEMBLY		2
		SC	-1	GRAPHITE REFLECTOR ASSEMBLY		1

QTY REQD — PARTS LIST

DASH NO.	NEXT ASSY
-2	FINAL
-1	FINAL

APPLICATION

DIMENSIONING AND SYMBOLOGY ARE AMERICAN NATIONAL STANDARD UNLESS OTHERWISE SPECIFIED SURFACE ROUGHNESS DIMENSIONS AND TOLERANCES ARE IN INCHES
DO NOT SCALE DRAWING

SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE: 6/1/06

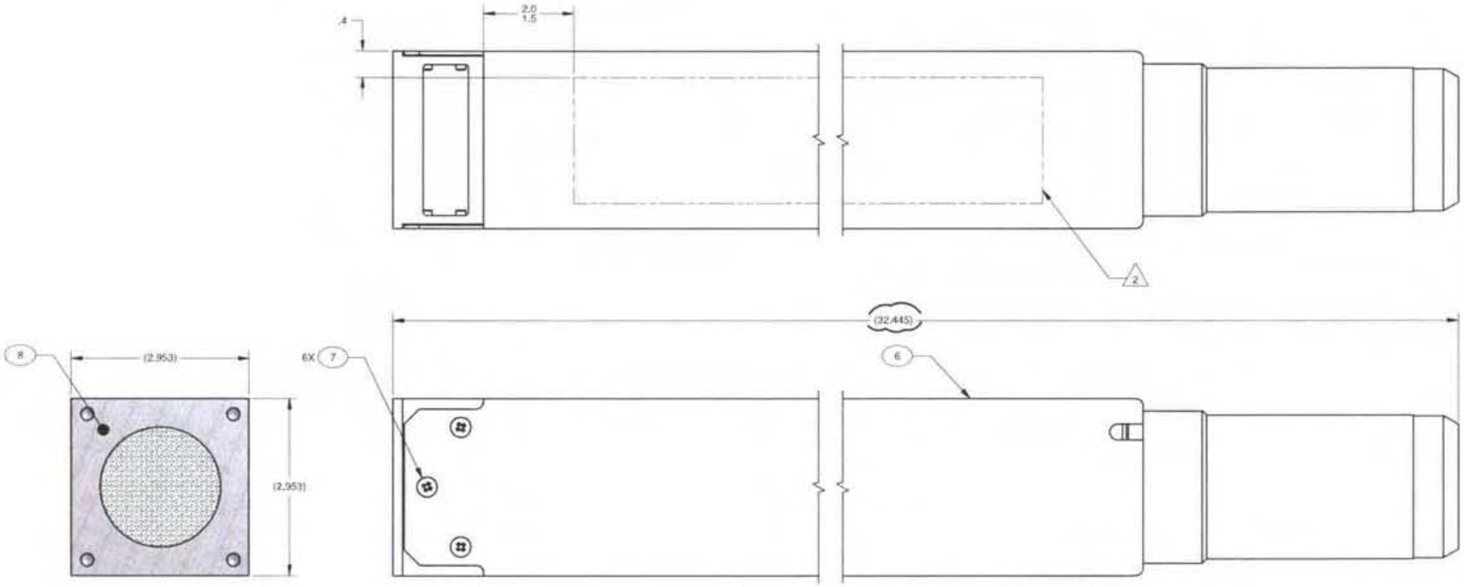
INL Idaho National Laboratory

PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
GRAPHITE REFLECTOR ASSEMBLY AND
SOURCE DRIVE ASSEMBLY

SIZE D | CAGE CODE 01MF3 | INDEX CODE NUMBER: AREA 540, TYPE 9999, CL 43, ORIG 051 | DWG-635454 | REV 2

SCALE: NOTED | SHEET 1 OF 2

SD8032



-2 ASSEMBLY
SCALE: 1/1

-2 SECTIONED ISOMETRIC
SCALE: NONE

SIZE	CAGE CODE	AREA	TYPE	CL	DSEQ	DWG-	REV
D	01MF3	540	9999	43	061	635454	2
SCALE: NOTED						SHEET 2	

SD8032

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[redacted]

[redacted]

[redacted]

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D

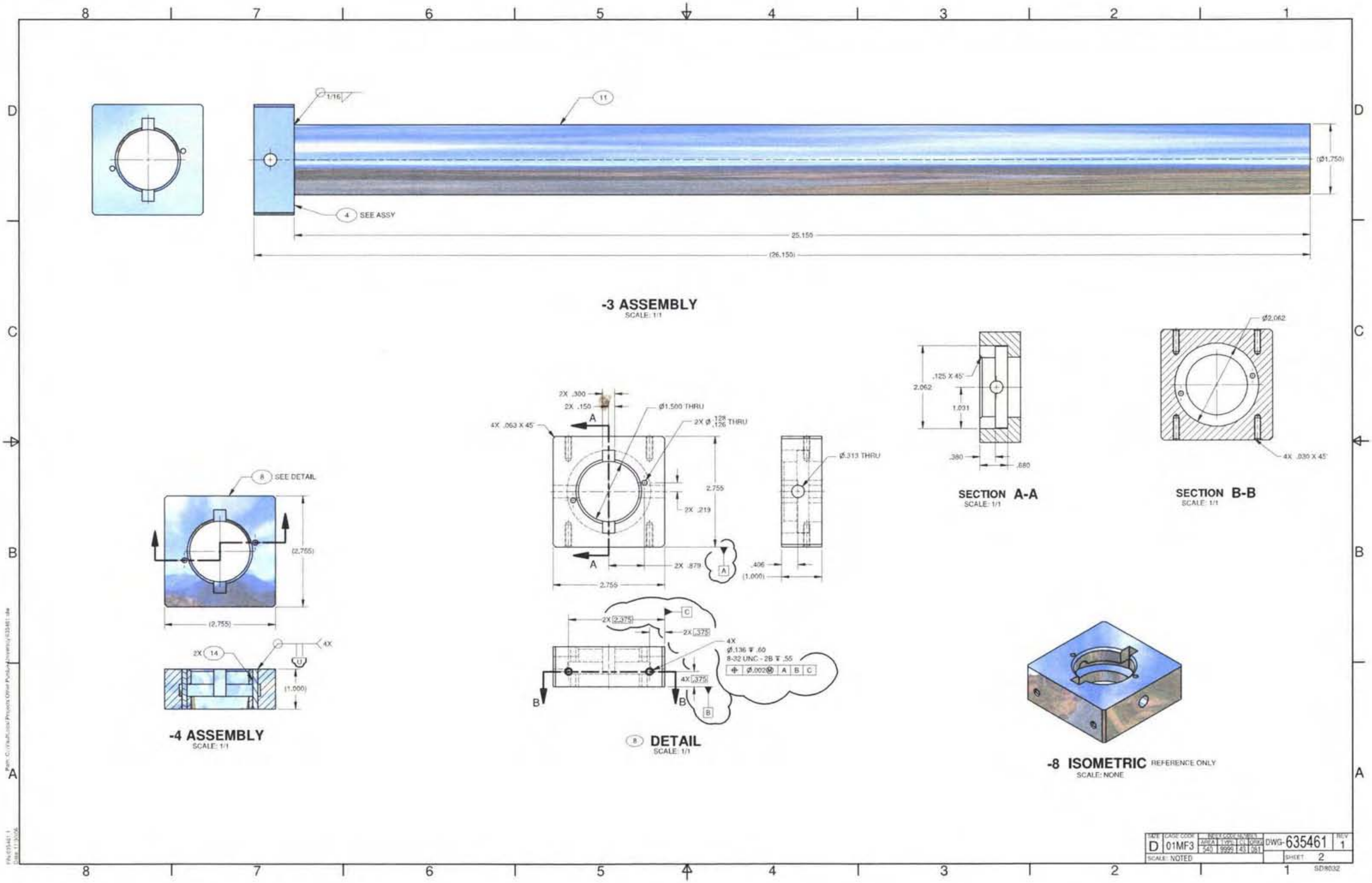
C

B

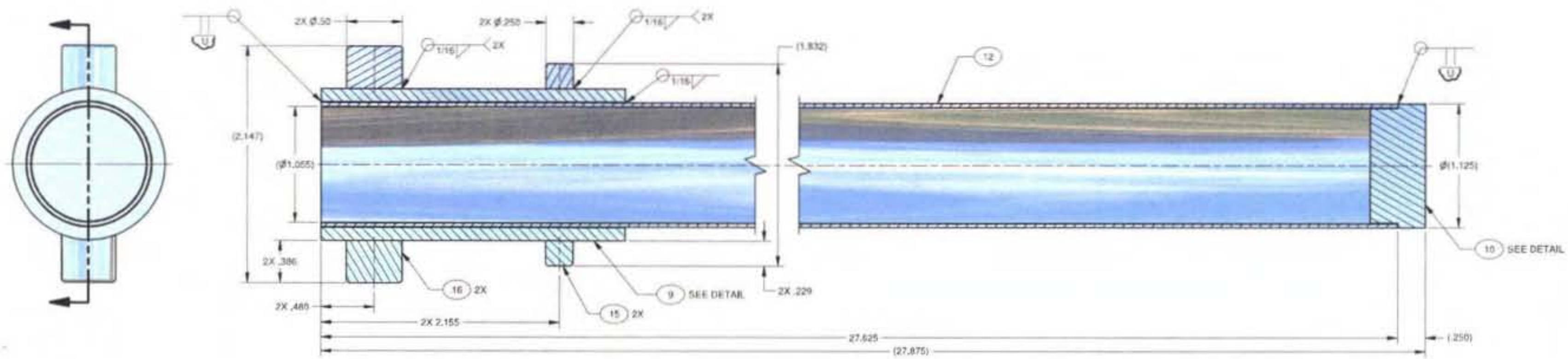
A

[redacted]

[redacted]

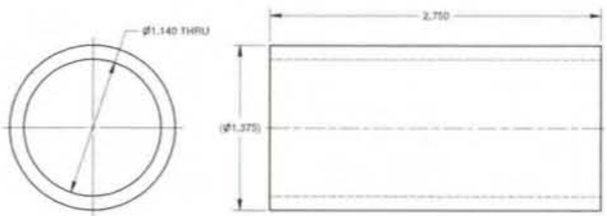


11
1/16
4 SEE ASSY
(Ø1.750)
25.150
(26.150)
-3 ASSEMBLY
SCALE: 1/1
8 SEE DETAIL
(2.755)
(2.755)
2X 14
4X
(1.000)
-4 ASSEMBLY
SCALE: 1/1
2X .300
2X .150
Ø1.500 THRU
4X .063 X 45°
2X Ø .128 .126 THRU
Ø.313 THRU
2.755
2X .219
2X .879
A
.406
(1.000)
2.755
2X 2.375
2X .375
C
4X
Ø.136 ▼ .60
8-32 UNC - 2B ▼ .55
⊕ Ø.002Ⓜ A B C
4X .375
B
8 DETAIL
SCALE: 1/1
.125 X 45°
2.062
1.031
.380
.680
SECTION A-A
SCALE: 1/1
Ø2.062
4X .030 X 45°
SECTION B-B
SCALE: 1/1
-8 ISOMETRIC REFERENCE ONLY
SCALE: NONE
D 01MF3
540 9999 43 061
DWG-635461
REV 1
SCALE: NOTED
SHEET 2
SD8032

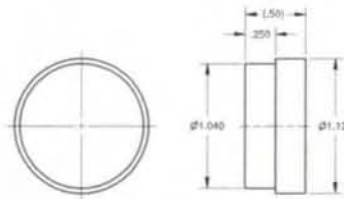


-2 ASSEMBLY
SCALE: 2/1

-2 ISOMETRIC REFERENCE ONLY
SCALE: NONE



9 **DETAIL**
SCALE: 2/1



10 **DETAIL**
SCALE: 2/1

SIZE	CAGE CODE	DWG	REV
D	01MF3	DWG-635461	1
SCALE: NOTED		SHEET 3	

SD8032

[redacted]

D

C

B

A

Path: C:\VaultLocal Projects\Other Purdue University\635463.idw

File:635463_4

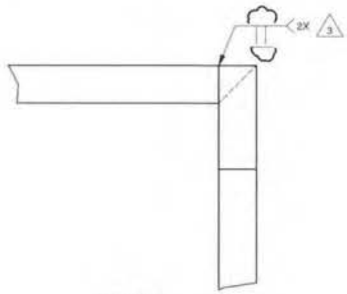
[redacted]

NOTES:

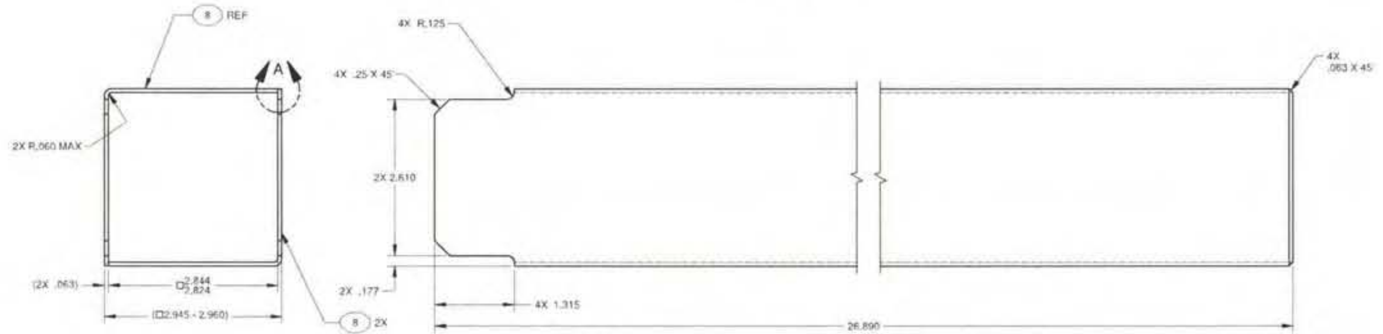
1. REMOVE ALL BURRS AND SHARP EDGES.
2. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
3. WELDING MAY BE PREFORMED AT FABRICATOR'S OPTION AS PART OF NEXT ASSEMBLY PROCESS (DWG 635462).
4. WELD PER SPECIFICATION SPC-382 APPENDIX B, USING ITEM 10 (WELD FILLER METAL).

3	2	1	SHEET	REV STATUS OF SHEETS
	2	2	REV	

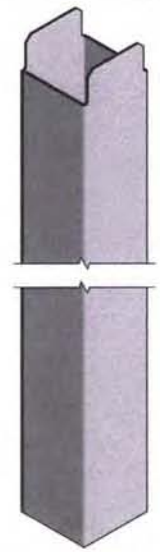
REVISIONS		
REV	DESCRIPTION	EFFECTIVE DATE
1	REVISED TITLE; CANCELED SHEETS 2 & 3; REMOVED ITEMS 4, 5, & 6; SEE DAR-509420	1/24/07
2	REINSTATED SHEET 2, REMOVED ITEM 3, ADDED -2 ASSY, SEE DAR-514751	4/4/07



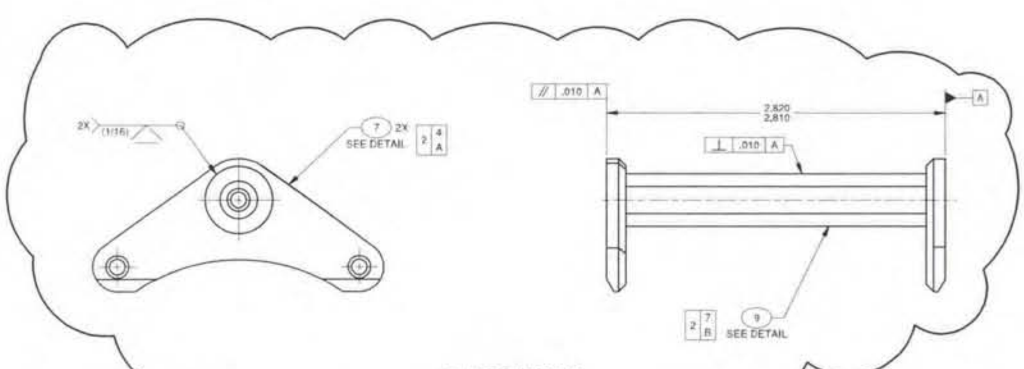
VIEW A
SCALE: 10/1



-1 ASSEMBLY
SCALE: 1/1



-1 ISOMETRIC REFERENCE ONLY
SCALE: NONE



-2 ASSEMBLY
SCALE: 2/1

QTY REQD -2	QTY REQD -1	SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION OR VENDOR NAME	ITEM NO.
	AR	SC		WELD FILLER METAL	ER 4043 AWS A5.10	10
AR		SC			BAR, Ø1/2 ASTM B211 AL 6061-T6511 OR AL 6061-T651	9
	AR	SC			SHEET AL 6061-T4 ASTM B209	8
AR					SHEET, .160 THK AL 6061-T6 ASTM B209	7
			-6	REMOVED		6
			-5	REMOVED		5
			-4	REMOVED		4
			-3	REMOVED		3
		SC	-2	BAIL ASSEMBLY		2
		SC	-1	CONTAINER TUBE ASSEMBLY		1

PARTS LIST

DASH NO.	NEXT ASSY
-2	635455
-2	635454
-1	635462
APPLICATION	

DIMENSIONING AND SYMBOLOGY ARE AMERICAN NATIONAL STANDARD UNLESS OTHERWISE SPECIFIED SURFACE ROUGHNESS "3" DIMENSIONS AND TOLERANCES ARE IN INCHES. DO NOT SCALE DRAWING

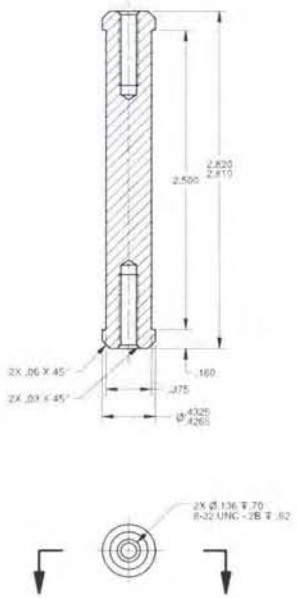
SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE: 6/1/06

INL Idaho National Laboratory

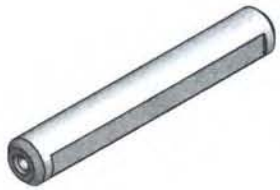
PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
CONTAINER TUBE
ASSEMBLY AND DETAILS

SIZE	CAGE CODE	INDEX CODE NUMBER AREA / TYPE / CL / ORIG	DWG-	REV
D	01MF3	540 / 9999 / 43 / 061	635464	2

SCALE: NOTED
SHEET 1 OF 2
SD8032

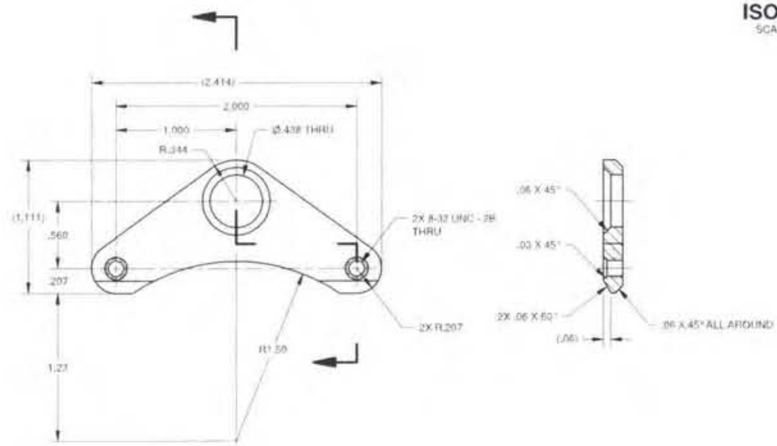


9 **DETAIL**



ISOMETRIC
SCALE: NONE

ISOMETRIC
SCALE: NONE



7 **DETAIL**

SIZE	CAGE CODE	AREA	TYPE	DIV	DEPT	DWG- 635464	REV 2
D	01MF3	540	9999	43	081		
SCALE: 2/1						SHEET 2	

SD8032

D

C

B

Path: C:\Vault\Local Projects\Other Purdue University\635465.idw

A

File:635465-1
Date:

[redacted]

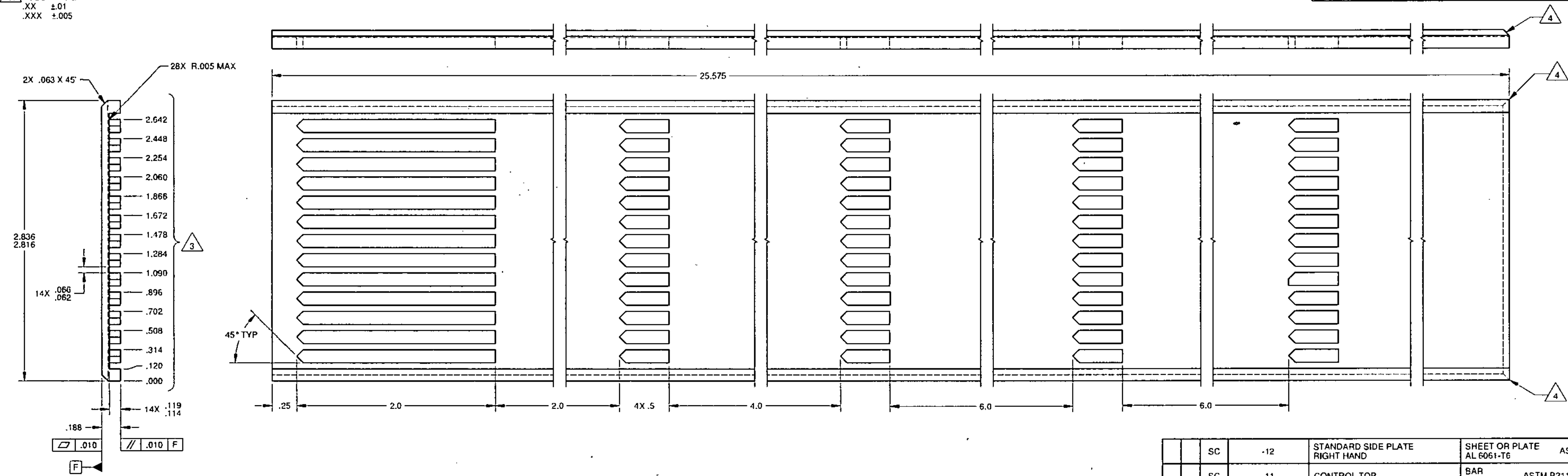
4	3	2	1	SHEET	REV STATUS OF SHEETS
3	3	3	3	REV	

REVISIONS

REV	DESCRIPTION	EFFECTIVE DATE:
1	REMOVED ITEMS 1 THRU 5; ADDED SHEETS 2 THRU 5; ADDED ITEMS 6 THRU 11; SEE DAR-509420	1/24/07
2	SEE DAR-514536	2/5/07
3	REMOVED ITEM 10, REMOVED BOTTOM, REMOVED NOTE 5, ADDED NOTE 6, CHAMFER FROM ITEM 11. SHEETS 1, 2, & 4 TOTALLY REVISED. SEE DAR-514751	4/4/07

NOTES:

1. REMOVE ALL BURRS AND SHARP EDGES.
2. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
3. ALL GROOVE LOCATIONING DIMENSIONS ARE ± .002 AND TOLERANCE SHALL NOT BE ACCUMULATIVE UNLESS OTHERWISE SPECIFIED.
4. WELD PREP AS NECESSARY.
5. REMOVED
6. TOLERANCE:
   .XX ±.01
   .XXX ±.005



6 DETAIL LEFT HAND (SHOWN)

12 DETAIL RIGHT HAND (OPPOSITE)

QTY REQD	SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION OR VENDOR NAME	ITEM NO.
	SC	-12	STANDARD SIDE PLATE RIGHT HAND	SHEET OR PLATE AL 6061-T6 ASTM B209	12
	SC	-11	CONTROL TOP	BAR AL 6061-T651 ASTM B211 OR B221	11
		-10	REMOVED		10
	SC	-9	SPACER PLATE	SHEET AL 6061-T6 ASTM B209	9
	SC	-8	EDGE PLATE	SHEET AL 6061-T6 ASTM B209	8
	SC	-7	CONTROL SIDE PLATE LEFT HAND	SHEET OR PLATE AL 6061-T6 ASTM B209	7
	SC	-6	STANDARD SIDE PLATE LEFT HAND	SHEET OR PLATE AL 6061-T6 ASTM B209	6
		-5	REMOVED		5
		-4	REMOVED		4
		-3	REMOVED		3
		-2	REMOVED		2
		-1	REMOVED		1

PARTS LIST

QTY REQD	SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION OR VENDOR NAME	ITEM NO.
	SC	-13	CONTROL SIDE PLATE RIGHT HAND	SHEET OR PLATE AL6061-T6 ASTM B209	13

PARTS LIST CONTINUED

DASH NO.	NEXT ASSY	DASH NO.	NEXT ASSY
		-12	635458
		-11	635456
		-9	635459
		-8	635459
		-8	635458
		-7	635459
-13	635459	-6	635458
APPLICATION		APPLICATION	

DIMENSIONING AND SYMBOLOGY ARE AMERICAN NATIONAL STANDARD UNLESS OTHERWISE SPECIFIED
SURFACE ROUGHNESS 125/
DIMENSIONS AND TOLERANCES ARE IN INCHES
TOLERANCES .X = ± .1
DECIMALS .XX = ± .03
.XXX = ± .010
FRACTIONS ± 1/8
ANGULAR ± 2
DO NOT SCALE DRAWING

SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE: 6/1/06

INL Idaho National Laboratory

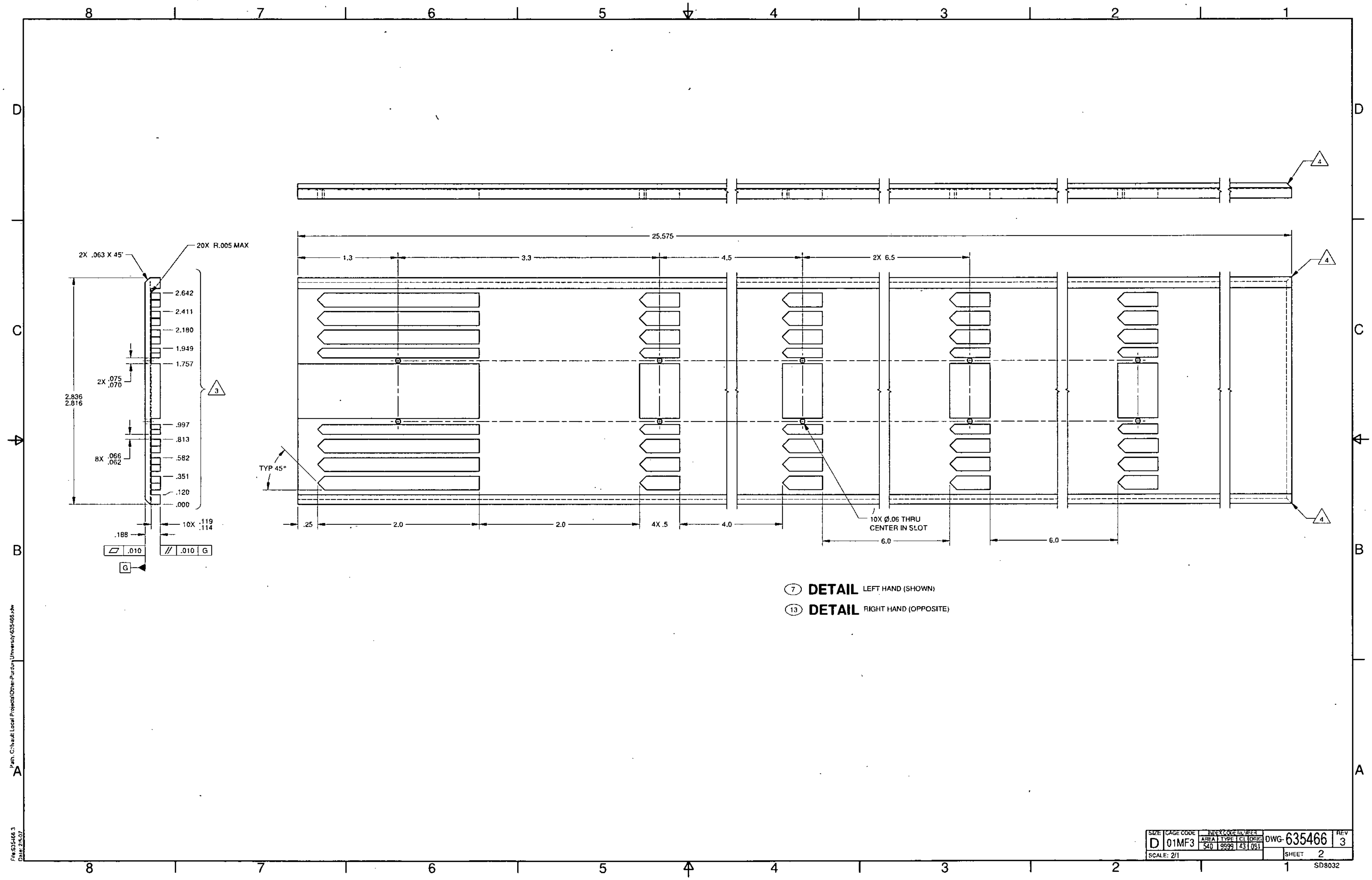
PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
MISCELLANEOUS DETAILS

SIZE	CAGE CODE	INDEX CODE NUMBER AREA	TYPE	CL	ORIG	DWG-	REV
D	01MF3	540	9999	43	081	635466	3

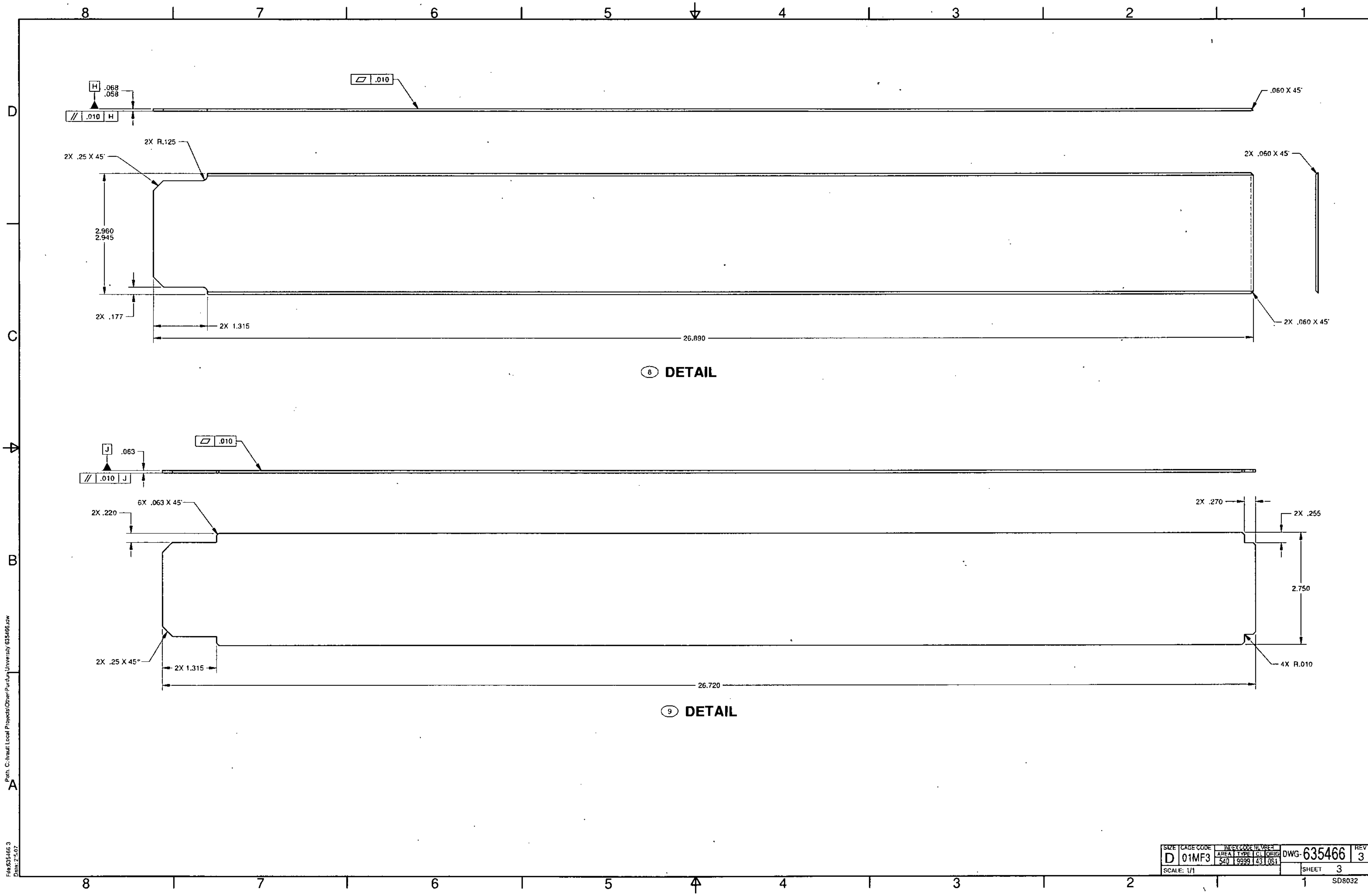
SCALE: 2/1 SHEET 1 OF 4

SD8032

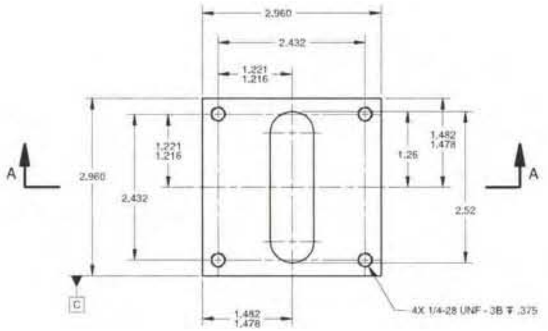
Path: C:\vault Local Projects\Other\Purdue University 635466.idw
File: 635466 3
Date: 2-5-07

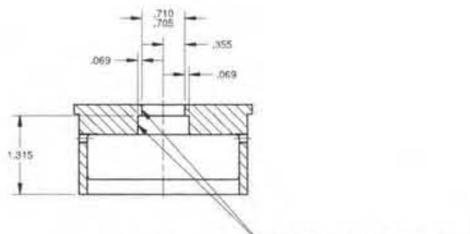


2X .063 X 45°
20X R.005 MAX
2.642
2.411
2.180
1.949
1.757
2X .075 .070
2.836 2.816
.997
.813
8X .066 .062
.582
.351
.120
.000
.188
10X .119 .114
.010
// .010 G
G
25.575
1.3
3.3
4.5
2X 6.5
TYP 45°
.25
2.0
2.0
4X .5
4.0
10X Ø.06 THRU
CENTER IN SLOT
6.0
6.0
7 DETAIL LEFT HAND (SHOWN)
13 DETAIL RIGHT HAND (OPPOSITE)
SIZE D
CAGE CODE 01MF3
AREA 540 TYPE 9999 CL 43 ORIG 051
DWG- 635466
REV 3
SCALE: 2/1
SHEET 2
SD8032

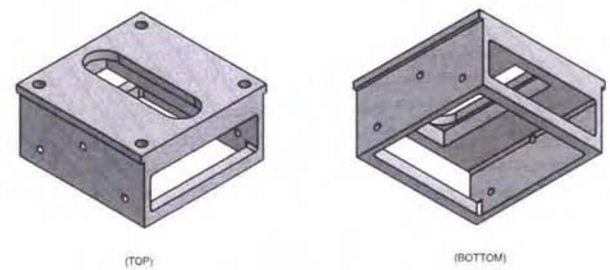


H
.068
.058
.010
// .010 H
.060 X 45°
2X R.125
2X .25 X 45°
2X .060 X 45°
2.960
2.945
2X .177
2X 1.315
26.890
2X .060 X 45°
8 DETAIL
J
.063
.010
// .010 J
6X .063 X 45°
2X .220
2X .270
2X .255
2.750
2X .25 X 45°
2X 1.315
4X R.010
26.720
9 DETAIL
SIZE D
CAGE CODE 01MF3
INDEX CODE NUMBER AREA 540 TYPE 9999 CL 43 ORIG 061
DWG- 635466
REV 3
SCALE: 1/1
SHEET 3
SD8032
Path: C:\Ivault Local Projects\Other\Purdue University 635466.idw
File:635466 3
Date: 2-5-07

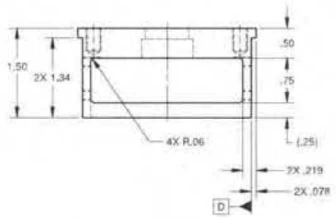


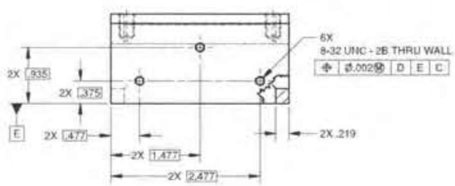


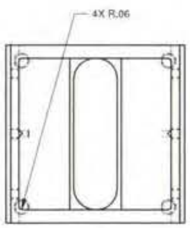
SECTION **A-A**



-11 ISOMETRIC REFERENCE ONLY
SCALE: NONE







(11) **DETAIL** 6
SCALE: 1/1

SIZE	CAGE CODE	AREA	TYPE	CL	BDMG		REV
D	01MF3	540	9999	43	061	DWG-635466	3
SCALE: NOTED						SHEET 4	

SD8032

NOTES:

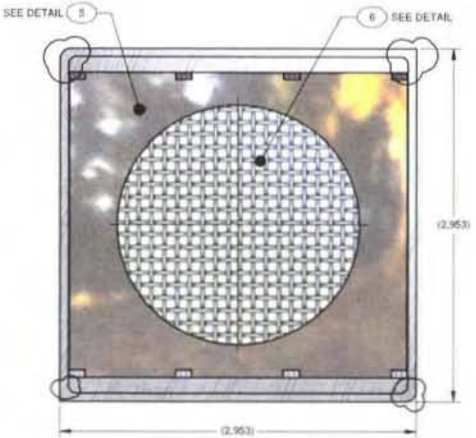
1. REMOVE ALL BURRS AND SHARP EDGES.
2. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
3. WELD PER SPECIFICATION SPC-382 APPENDIX B USING ITEM 8 (WELD FILLER METAL).
4. ALL CORNER AND FILLET RADII .030 UNLESS OTHERWISE NOTED.
5. REMOVED
6. TOLERANCE:
   .XX ±.01
   .XXX ±.005

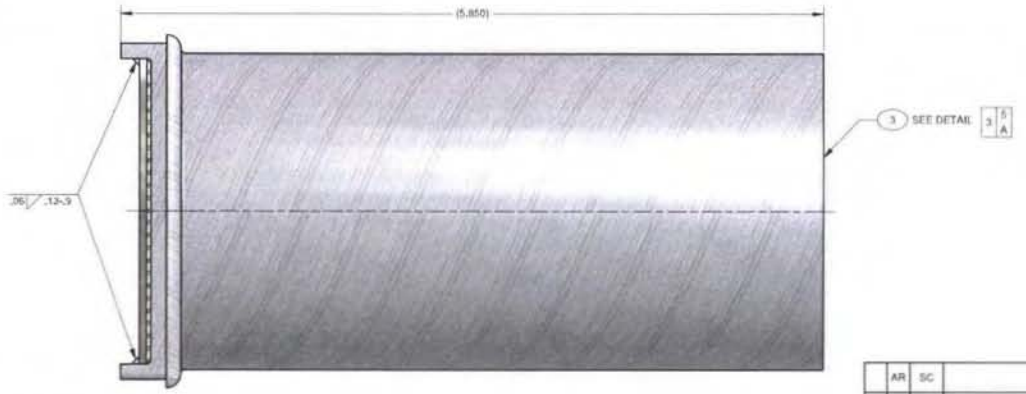
3	2	1	SHEET	REV STATUS OF SHEETS
2	2	2	REV	

REVISIONS

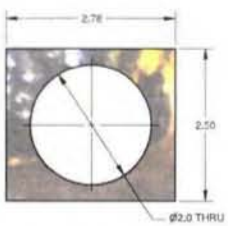
REV	DESCRIPTION	EFFECTIVE DATE
1	REVISED TITLE; REVISED ITEM 3; ADDED NOTE 4; ADDED SHEET 2 & 3; ADDED ITEM 2; SEE DAR-509420	1/18/07
2	TOTALLY REVISED SHEET 2. REMOVED CHAMFER FROM DETAIL 2. ADDED RADIUS DIMENSION TO ITEM 3. REMOVED NOTE 5. SEE DAR-514751	4/4/07

-1 **ISOMETRIC** REFERENCE ONLY
SCALE: NONE

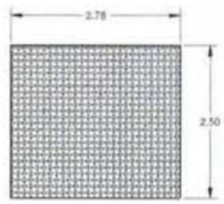




-1 **ASSEMBLY**
SCALE: 2/1



5 **DETAIL**
SCALE: 1/1



6 **DETAIL**
SCALE: 1/1

QTY REQD -1		SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL/SPECIFICATION OR VENDOR NAME	ITEM NO.
	AR	SC		WELD FILLER METAL	ER 4043 AWS A5.10	8
						7
	1	LSC	-6	MESH	WOVEN WIRE CLOTH, 16 X 16 MESH PER INCH, AL ASTM E 2016	6
	1	SC	-5	BACKING PLATE	SHEET, .063 THK AL 6061-T6 ASTM B209	5
						4
	1	SC	-3	NOZZLE PRELIMINARY MACHINED	BAR ASTM B211 OR B221 AL 6061-T6 OR AL 6061-T651	3
		SC	-2	SOURCE DRIVE TOP	PLATE AL 6061-T651 ASTM B209	2
		SC	-1	SOURCE DRIVE NOZZLE ASSEMBLY		1

PARTS LIST

DASH NO.	NEXT ASSY
-3	635462
-2	635457
-1	635462

APPLICATION

DIMENSIONING AND SYMBOLOGY ARE AMERICAN NATIONAL STANDARD
UNLESS OTHERWISE SPECIFIED
SURFACE ROUGHNESS
DIMENSIONS AND TOLERANCES ARE IN INCHES
DO NOT SCALE DRAWING

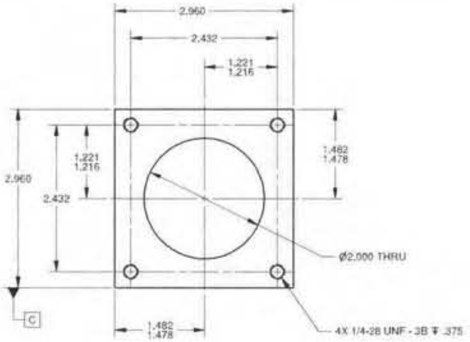
SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE 1/24/07

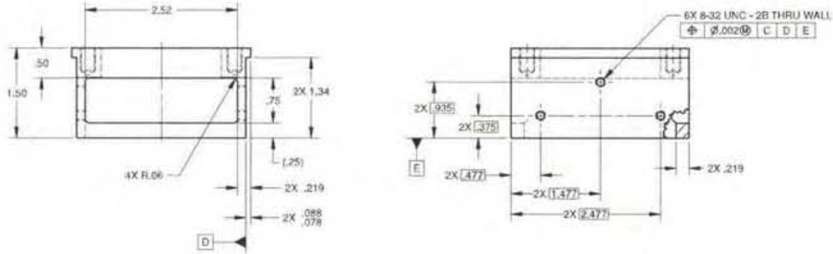
INL Idaho National Laboratory

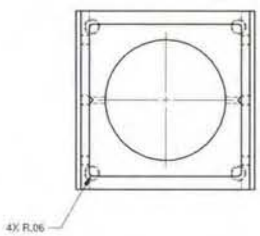
PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
SOURCE DRIVE NOZZLE ASSEMBLY
AND SOURCE DRIVE TOP

SIZE D | CAGE CODE 01MF3 | DWG-635467 | REV 2
SCALE: NOTED | SHEET 1 OF 3

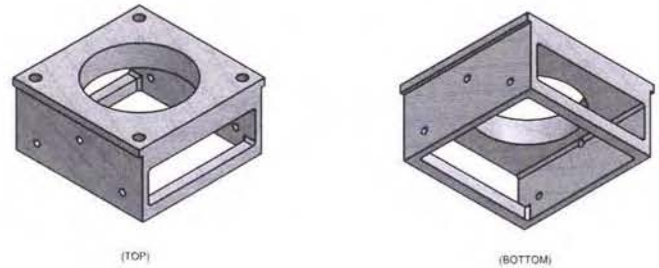
SD6032







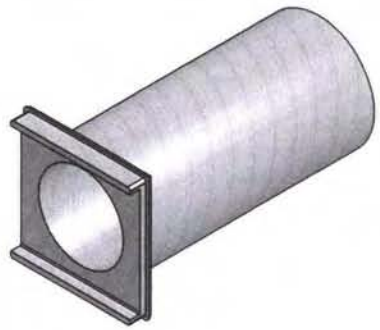
② **DETAIL** △6
SCALE: 1/1



-2 ISOMETRIC REFERENCE ONLY
SCALE: NONE

SIZE	CAGE CODE	AREA	TYPE	CL	DRBG		REV
D	01MF3	540	9999	43	061	DWG-635467	2
SCALE: NOTED						SHEET 2	

SD8032



-1 ISOMETRIC REFERENCE ONLY
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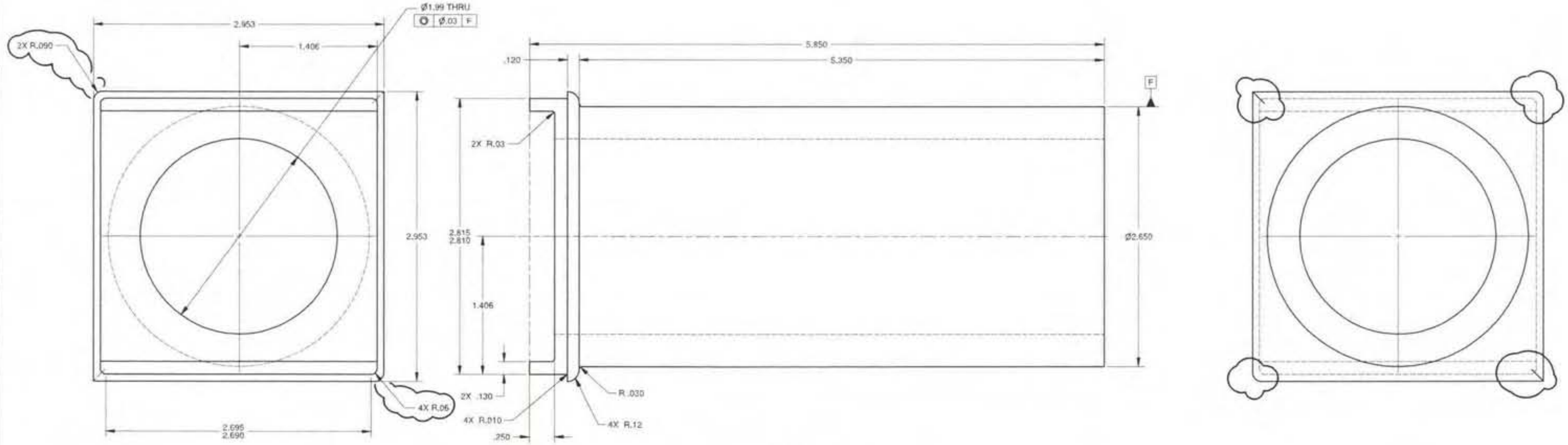


③ **DETAIL** 4
SCALE: 2/1

SIZE	CAGE CODE	AREA	TYPE	CL	DRWG	DWG-635467	REV
D	01MF3	540	9999	43	261		2
SCALE: NOTED						SHEET 3	

SD8032

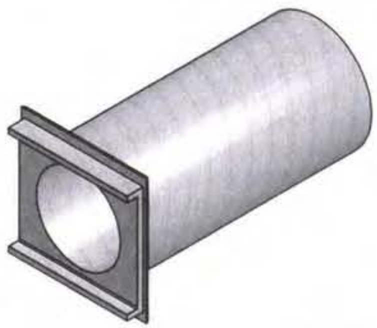
NOTES:

1. REMOVE ALL BURRS AND SHARP EDGES.
2. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
3. ALL CORNER AND FILLET RADII .030 UNLESS OTHERWISE NOTED.
4. REMOVED
5. TOLERANCE:
   .XX ±.01
   .XXX ±.005

2	1	SHEET	REV STATUS OF SHEETS
2	2	REV	

REVISIONS

REV	DESCRIPTION	EFFECTIVE DATE
1	REVISED TO FIT INTO NEW FUEL ELEMENT CONTAINER DESIGN; ADDED SHEET 2; ADDED DETAIL 2; REVISED TITLE; SEE DAR-509420	1/24/07
2	TOTALLY REVISED SHEET 2; REMOVED NOTE 4. SEE DAR-514751	4/4/07



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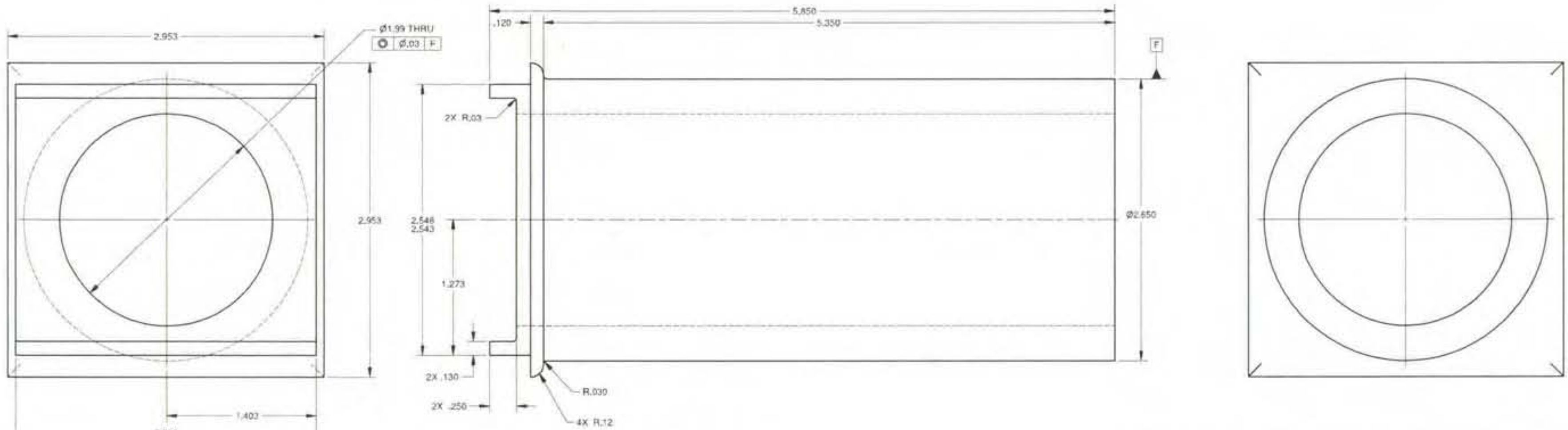
SCALE: NONE



1 **DETAIL** 3
SCALE: 2/1

QTY REQD		SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL/SPECIFICATION OR VENDOR NAME	ITEM NO.
		SC	-2	FISSION CHAMBER TOP	BAR AL 6061-T651 ASTM B211 OR B221	2
		SC	-1	NOZZLE PRELIMINARY MACHINED	BAR ASTM B211 OR B221 AL 6061-T6 OR AL 6061-T651	1

PARTS LIST

DASH NO.	NEXT ASSY
-2	635457
-1	635459
-1	635458

APPLICATION

DIMENSIONING AND SYMBOLOGY ARE AMERICAN NATIONAL STANDARD UNLESS OTHERWISE SPECIFIED
~~SURFACE ROUGHNESS~~
DIMENSIONS AND TOLERANCES ARE IN INCHES
TOLERANCES
FRACTIONS
ANGULAR
DO NOT SCALE DRAWING

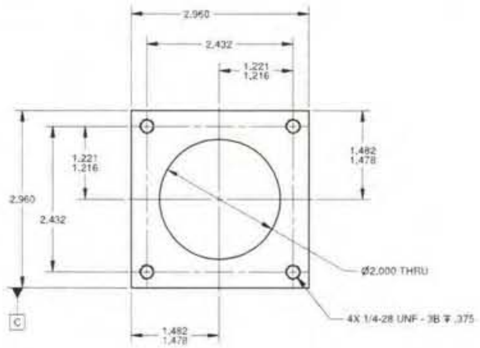
SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE: 6/1/06

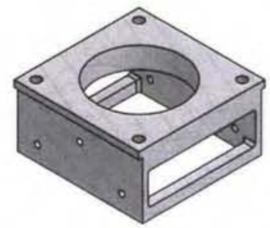
INL Idaho National Laboratory

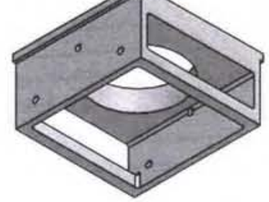
PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
NOZZLE PRELIMINARY MACHINED
AND FISSION CHAMBER TOP

SIZE	CAGE CODE	INDEX CODE NUMBER AREA	TYPE	CL	NUMBER	DWG-	REV
D	01MF3	540	9999	43	081	635468	2

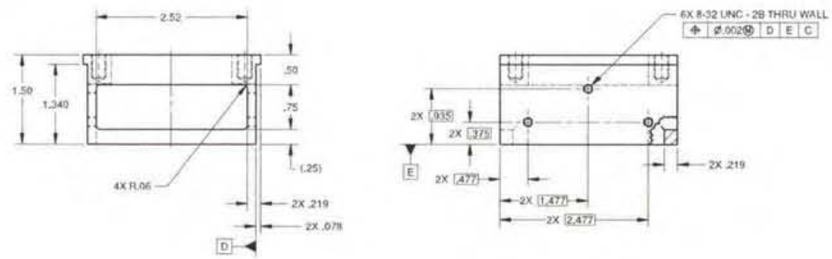
SCALE: NOTED
SHEET 1 OF 2
SD8032

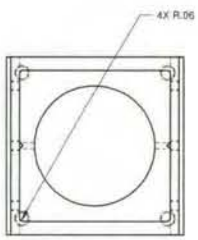






-2 ISOMETRIC
SCALE: NONE





(2) **DETAIL** (5)
SCALE: 1/1

SIZE	CAGE CODE	INDEX CODE NUMBER					REV
D	01MF3	540	9999	43	081	DWG-635468	2
SCALE: NOTED						SHEET 2	

SD8032

NOTES:

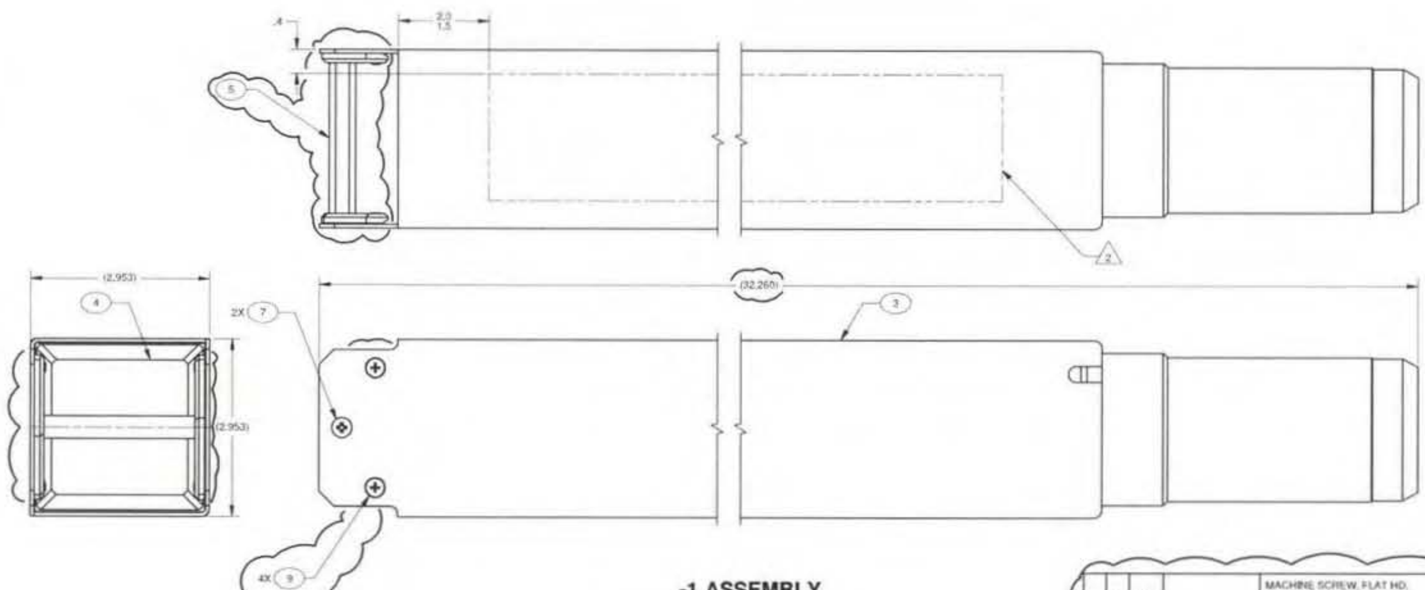
1. PROCURE, FABRICATE, AND INSPECT PER SPECIFICATION SPC-382.
2. MARK ASSEMBLY IDENTIFICATION PER SPECIFICATION SPC-382.
3. REMOVE ALL BURRS AND SHARP EDGES.

2	1	SHEET	REV STATUS OF SHEETS
2	2	REV	

REVISIONS

REV	DESCRIPTION	EFFECTIVE DATE
1	REVISED DESCRIPTION OF ITEM 3; ADDED SHEET 2; ADDED ITEMS 2, 6, & 8; SEE DAR-509420	1/24/07
2	ADDED ITEM 9, CHANGED MATERIAL ITEM 7 CHANGED ITEM 5, SEE DAR-514751	4/4/07

-1 SECTIONED ISOMETRIC REFERENCE ONLY
SCALE: NONE



-1 ASSEMBLY
SCALE: 1/1

QTY REQD -2	QTY REQD -1	SAFETY CAT.	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL/SPECIFICATION OR VENDOR NAME	ITEM NO.
	4	CG		MACHINE SCREW, FLAT HD, PHILLIPS, 100° 8-32 UNC-2A X 1/4 LG	316 SST OR 304 SST	9
1			635467-2	SOURCE DRIVE TOP		8
6	2	CG		MACHINE SCREW, FLAT HD, PHILLIPS, 100° 8-32 UNC-2A X 1/2 LG	316 SST OR 304 SST	7
1			635462-3	SOURCE DRIVE CONTAINER ASSEMBLY		6
	1		635464-2	BAIL ASSEMBLY		5
	1		635465-1	GRAPHITE BLOCK		4
	1		635462-1	GRAPHITE CONTAINER ASSEMBLY		3
		SC	-2	SOURCE DRIVE SUPPORT ASSEMBLY		2
		SG	-1	GRAPHITE REFLECTOR ASSEMBLY		1

PARTS LIST

DASH NO.	NEXT ASSY
-2	FINAL
-1	FINAL

APPLICATION

SUBCONTRACT NO.
REQUESTER: T. VINNOLA
DESIGN: D. MORRELL
DRAWN: N. OLDHAM
PROJECT NO.
SPOL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE DAR NO. 505051
EFFECTIVE DATE: 6/1/06

INL Idaho National Laboratory

PURDUE UNIVERSITY
TEST RESEARCH AND TRAINING REACTOR
GRAPHITE REFLECTOR ASSEMBLY AND
SOURCE DRIVE ASSEMBLY

SIZE D | CAGE CODE 01MF3 | INDEX CODE NUMBER: AREA 540, TYPE 9995, CL 43, ORIG 061 | DWG-635454 | REV 2

SCALE: NOTED | SHEET 1 OF 2

SD5032

OPERATOR REQUALIFICATION PROGRAM
for the
PUR-1 REACTOR FACILITY

This program is designed to comply with the intent of 10 CFR 55, Appendix A, concerning the continued training and requalification of operators for the PUR-1 reactor. It will be mandatory for all operators licensed on the PUR-1 reactor to participate in the program.

The requalification program consists of the following parts:

A. INSTRUCTION

A series of eight meetings will be held over a two year period, during which all topics listed below in part A.1.b will be covered.

1. Each meeting will consist of:
   a. A review of reactor operations and modifications, if any.
   b. A lecture of one or more of the following topics:
      i. Theory and principles of operations.
      ii. General and specific plant operating characteristics.
      iii. Plant instrumentation and control systems.
      iv. Plant protection systems.
      v. Engineered safety systems.
      vi. Normal, abnormal, and emergency operating procedures.
      vii. Radiation control and safety.
      viii. Technical specifications.
      ix. Applicable portions of Title 10, Chapter I, Code of Federal Regulations.

2. The lectures will be given by the reactor operators, senior operators, health physics personnel, faculty members of the School of Nuclear Engineering, or other qualified personnel as determined by the Laboratory Director.

B. PROGRAM EVALUATION

Completion of the biennial operator requalification program will consist of a written examination and a demonstration of operator proficiency in reactor operation.

1. Written examination:

   a. One of the senior operators will be exempt from taking the examination. This senior operator will make up and administer the examination to all other operators and senior operators. The senior operator may receive assistance for making up questions on the topics in part A.1.b from the instructor for each topic. The senior operator exemption will rotate through the entire senior operator roster.

b. The written examination for requalifying licensees will contain representative questions measuring the knowledge, skills and abilities needed to perform licensed duties. These will be identified from the licensed operator's duties performed, information in the Safety Analysis Report, operating procedures, facility license and amendments, License Events Reports, and any other information requested from the facility licensee by the NRC.

   The representative questions for the operator's examination will sample the following topics:

   i. Fundamentals of reactor theory including the fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.
   ii. General design features of the core, fuel assemblies, control rods, core instrumentation, and coolant flow.
   iii. Mechanical components and design features of the reactor coolant system.
   iv. Auxiliary systems that affect the facility.
   v. Facility operating characteristics during steady state and transient conditions.
   vi. Design, components, and functions of reactivity control mechanisms and instrumentation.
   vii. Design, components, and functions of control and safety systems including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
   viii. Components, capability, and functions of emergency systems.
   ix. Shielding, isolation, and containment design features, including access limitations.
   x. Administrative, normal, abnormal, and emergency operating procedures.
   xi. Purpose and operation of radiation monitoring systems, including alarms and survey equipment.
   xii. Radiological safety principles and procedures.
   xiii. Procedures and equipment available for handling and disposal of radioactive materials.

c. Representative questions for the senior operators examination will sample the topics in the operators list and in addition will sample the following list:

   i. Conditions and limitations in the facility license.

ii. Facility operating limitations in the technical specifications and their bases.
iii. Licensee procedures required to obtain authority for design and operating changes.
iv. Radiation hazards that may arise during normal and abnormal situations including maintenance activities and various contamination conditions.
v. Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
vi. Procedures and limitations involved in initial core loading, alterations in core configuration, and determination of various internal and external effects on core reactivity.
vii. Fuel handling facilities and procedures.

d. Any person who scores less than 70%, overall, on the examination will be relieved from licensed duties and enrolled in an accelerated program until such time as they can satisfactorily pass an examination covering the material. The course content and duration will depend upon the individual's deficiencies.

2. Operator proficiency:

a. The exempt senior operator will also administer an operator proficiency examination to all other operators and senior operators.

b. The content of the operating test will be identified from duties of the licensed operator/senior operator and reference documents listed in Part B.1.b

c. The operations test for the requalifying licensee will demonstrate an understanding of and the ability to accomplish a representative sample of the following items:

i. Perform the prestartup procedures.
ii. Manipulate the console controls as required to operate the facility between shutdown and designed power levels.
iii. Identify annunciators and condition-indicating signals and perform appropriate remedial actions where appropriate.
iv. Identify the instrumentation systems and the significance of facility instrument readings.
v. Observe and safely control the operating behavior characteristics of the facility.
vi. Perform control manipulations required to obtain desired operating results during normal, abnormal, and emergency situations.
vii. Safely operate the facilities auxiliary and emergency systems.

viii. Demonstrate or describe the use and function of the facilities radiation monitoring systems, including fixed radiation monitors and alarms, portable survey instruments, and personal monitoring equipment.
ix. Demonstrate knowledge of significant radiation hazards and the ability to perform procedures to reduce elevated levels of radiation and to maintain exposures as low as reasonably achievable (ALARA).
x. Demonstrate knowledge of the emergency plan, including, as appropriate, the operator's or senior operator's responsibility to decide whether the plan should be executed and the duties assigned under the plan.
xi. Demonstrate the knowledge and ability, as appropriate to the assigned position, to assume the responsibilities associated with the safe operation of the facility.
xii. Demonstrate the ability to follow all procedures in a way that conforms to all license limits, amendments, and applicable conditions and specifications.

d. Any person who can not demonstrate proficient operation of the reactor will be relieved of his licensed duties until such time as proficient operation could be demonstrated. Proficient operation may be established by performing a minimum of six hours of supervised reactor operations and demonstrating proficiency of section B.2.

C. ON THE JOB TRAINING

1. Each licensed operator in the requalification program may at the option of the exempt senior operator, be required to make 8 reactor startups, shutdowns, or power level changes during the two year period covered by the program.

2. Each licensed operator at the facility will manipulate the plant controls, and each licensed senior operator will either manipulate the plant controls or direct the activities of individuals during plant control manipulations during the term of the operators/senior operator's license. Manipulations by operators/senior operators must consist of the following activities:

   a. Completed annually.
      i. Plant shutdown.
      ii. Significant power changes (>10%).
      iii. Loss of coolant. (Not considered credible)
      iv. Loss of electrical power.
      v. Loss of coolant flow. (Not considered credible)
   b. Completed on a two-year cycle.
      i. Loss of protective system channel.
      ii. Mispositioned control rod or rods.

iii. Inability to drive control rods.
iv. Conditions requiring use of emergency boration
v. Fuel cladding failure or high activity in reactor coolant.
vi. Failure of servo system.
vii. Reactor trip.
viii. Failure of nuclear instrumentation.

Note: When the control panel of the facility is used for training, the action taken or to be taken for the emergency or abnormal condition may be discussed; actual manipulation of the controls is not required. 10 CFR 55.59 (c) (4) (iv).

3. Each licensed operator at the facility will perform the function of the license held. An SRO is credited with performing the function any time the operator is on call, instructing classes, student, or student operator in training, inside the reactor room with the key on, or maintaining custody of the key. Additionally unstructured activities such as participation in facility-related design and safety review groups, Emergency Plan, emergency drill, Committee on Reactor Operations (CORO) participation, experimental activities, related technical presentations, performing security related functions, and performing maintenance and calibration activities contribute to training in all parts of the program except parts B.2, C, D and E. A statement to the file is sufficient to document the training and/or time accounting.

D. LITERATURE REVIEW

Each reactor operator and senior operator will annually review the contents of the operating manual, technical specifications, and the emergency procedures. A statement to this fact will be kept in the requalification file.

E. RECORDS

Records will be maintained to document each instructor, each topic discussed, each licensed operator's and senior operator's participation in the requalification program. The records will contain copies of each written exam, answer sheets, results of evaluation, and the biennial operator proficiency demonstration. Documentation of additional training and test required for individuals exhibiting deficiencies will also be included in the files. All records of the requalification program will be retained by the as determined by the Laboratory Director until the licenses of the participants are renewed.

F. EXEMPTIONS

During intervals when the licensed operations crew consists only of senior operators who are instructors for topics in part A.1.b., the requalification program will be modified to exempt those senior operators from parts A and B.1. Parts B.2, C, D, and E will remain in effect.

When the licensed operations crew increases to include those who do not instruct in the program, the program will revert to its initial content. Operators may place a statement into the file stating that they have done a literature review and/or instructed the topics in section A and B.1 in lieu of meetings and exams.

During intervals when the licensed operations crew consists of only one senior operator this operator will be exempt from parts A and B, part C would be documented in the console log book and as stated in C.3, parts D and E will remain in effect.

In any of the above requalification activities exclusive of operations, additional methods may be used to accomplish the training requirement. These may include mail, electronic classroom or other methods may be used for training, meetings, testing or other required communication(s).