Purdue University Responses to PURDUE UNIVERSITY – REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL APPLICATION (TAC NO. ME1594), RESPONSES TO LETTER DATED August 29, 2014.

1. **RAI 18 in NRC letter dated July 6, 2011, stated:**

> **TS 4.3: TS 4.3(c) should reference the minimum 13 foot depth as specified in the LCO [limiting condition for operation] (TS 3.3(c)) for primary coolant and provided in the bases for TS 4.3. Please update the TS to include the numerical minimum depth and surveillance interval for this surveillance or justify why an alternative measure related to the height of the skimmer trough in TS 4.3(c) is more appropriate for specifying the minimum performance level of TS 3.3(c). Additionally, prescribe the frequency, scope and minimum water level of this surveillance when the reactor is secured or shutdown or justify why a minimum level is not required.**

The response to RAI 18 by letter dated January 4, 2012 (ADAMS Accession No. ML12006A193), proposed to modify TS 4.3(c) to specify that reactor pool water will be at a height of the 13 foot over the top of the core whenever the reactor is operated. However, the response to RAI 17 by letter dated January 30, 2012 (ADAMS Accession No. ML12031A223), proposed to modify TS 4.3(c) to specify the reactor pool water will be at or above the height of the skimmer. Clarify TS 4.3(c), ensuring that RAI 18 is answered clearly and completely. Include a basis for any TS changes proposed.

<u>Response:</u>

References to the skimmer trough have been removed from the Technical Specifications.

TS 4.3(c) will be modified to state:

> (c) The reactor pool water will be at a height of 13 feet over the top of the core whenever the reactor is operated. The reactor pool water height shall be visually inspected weekly, not to exceed ten days, and water will be added as necessary to reach the specification.

And the basis for TS 4.3(c) will be revised to state:

> When the reactor pool water is at a height of 13 feet above the core, adequate shielding during operations is assured. Experience has shown that approximately 35-40 gallons of water will evaporate weekly and weekly water make-up is sufficient to maintain the reactor pool water height.

A revised copy of the Technical Specifications, Amendment 13 (draft) has been attached.

2. **RAI 29 in NRC letter dated July 6, 2011, stated:**

TS 6.1.11, TS 6.1.14: ANSI/ANS-15.1-2007, Section 6.1.3(3) provides guidance for events requiring the presence at the facility of the senior reactor operator. Please update PUR-1 TS 6.1.11 and 6.1.14 for compliance with the requirements in ANSI/ANS-15.1-2007, Section 6.1.3(3) and 10 CFR 50.54(m)(1) or provide an explanation describing your reason(s) for not incorporating the changes.

The response to RAI 29 by letter dated January 30, 2012, proposed a modification of TSs 6.1.11 and 6.1.14 that does not meet the requirements of 10 CFR 50.54(m)(1) which state:

> **A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shutdown or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.**

The response proposed a change to TS 6.1.2 (Staffing), Item (3), stating:

(3) Events requiring the presence at the facility of an senior reactor operator [SRO] are:

(a) Initial startup and approach to power following a core change. The presence of an SRO at the reactor facility is unnecessary for the initial daily start-up, provided the core remains unchanged from the previous run;

(b) All fuel or control-rod relocations within the core region;

(c) Recovery from an unplanned or unscheduled shutdown except in instances which result in the following

(i) A verified electrical power failure ...;

(ii) Accidental manipulation of equipment in a manner which does not affect the safety of the reactor;

(iii) A verified practice of the evacuation of the building initiated by persons exclusive of reactor operations personnel.

Provide an explanation on how the proposed changes (a) and (c) are in compliance with the requirements of 10 CFR 50.54(m)(1).

Response:

According to 10 CFR 50.54(m)(1), a senior operator is required for *initial* start-up and approach to power. TS 6.1.2(3)(a) requires the SRO to be present for the *initial* startup and approach to power any time the core has been changed. After this, as the core has not changed and so the expected critical rod heights are known, a senior operator is not required to be present for subsequent operations, including start-up. If the core configuration is changed, a senior operator is again required for the *initial* start-up and approach to power.

As 10 CFR 50.54.(m)(1) does not recognize any exceptions for the required presence of a senior operator for recovery from an unplanned or unscheduled shutdown, TS 6.1.2(3)(c) will be revised to state:

(c) Recovery from an unplanned or unscheduled shutdown.

3. RAI 44 in NRC letter dated July 8, 2011, stated:

In your RAI response concerning decommissioning cost, dated June 4, 2010, you reference an "approved cost estimate for decommissioning under Purdue University's Radioactive Materials License" as the basis for the provided cost estimate. Please describe and explain the relationship between the decommissioning cost estimate for PUR-1 and cost estimate for decommissioning under Purdue University's Radioactive Materials License in determining the cost estimate for decommissioning PUR-1.

Provide a response to RAI 44, since we have not yet received one.

<u>Response:</u>

A rewritten Chapter 15 is attached and will be included in the Safety Analysis Report.

4. RAI 45 in NRC letter dated July 8, 2011, stated:

Pursuant to 10 CFR 55.59(a)(2), each licensee shall: "Pass a comprehensive requalification written examination and an annual operating test." In your Requalification Plan, Section B you state that "completion of the biennial requalification program will consist of a written examination and a demonstration of operator proficiency in reactor operation."

A. Explain how the facility ensures that operator proficiency examinations are performed annually during the biennial requalification cycle in compliance with 10 CFR 55.59(a)(2) or update your plan accordingly.
B. As required by 10 CFR 55.53(h), licensees are required to complete a requalification program as described by 10 CFR 55.59. The regulation in 10 CFR 55.59(a) states that each license shall:
(1) Successfully complete a requalification program developed by the facility licensee that has been approved by the Commission. This program shall be conducted for a continuous period not to exceed 24 months in duration.
(2) Pass a comprehensive requalification written examination and an annual operating test.

Section F of the PUR-1 Requalification Plan states:

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 logbook and as stated in C.3, parts D and E will remain in effect.

In any of the 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).

The response to RAI 45 by letter dated January 31, 2012 (ADAMS Accession No. ML14234A109, redacted version) indicated that the PUR-1 facility previously had an exemption and intends to request one again. Either (a) explain how this section, in its current form, meets the requirements of 10 CFR 55.33, "Disposition of an initial application" and 10 CFR 55.59; (b) delete this section of the requalification plan; or (c) submit an exemption for these requirements in accordance with 10 CFR 55.11, "Specific exemptions."

Response:

The requalification program has been revised to clarify that the operating test and demonstration of proficiency is an annual requirement.

Section F of the PUR-1 Requalification Plan will be deleted. Purdue University will not be seeking an exemption for the requalification program.

A revised requalification program has been attached.

5. RAI 63 in NRC letter dated July 14, 2011, states:

> **Major inconsistences are noted throughout the SAR [safety analysis report] related to calculation assumptions for initial and requested maximum licensed power under the PUR-1 license renewal. For example, SAR Section 13.2.2, p. 13-11, references current licensed power of 1 kW [kilowatt] for a reactivity insertion with scram. Please clarify the desired maximum licensed power level requested and ensure this power level, including any uncertainty in reactor power, is consistently applied in the safety analyses for the license. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.**

Additional clarification to RAI 63 is needed. Provide responses to the questions below:

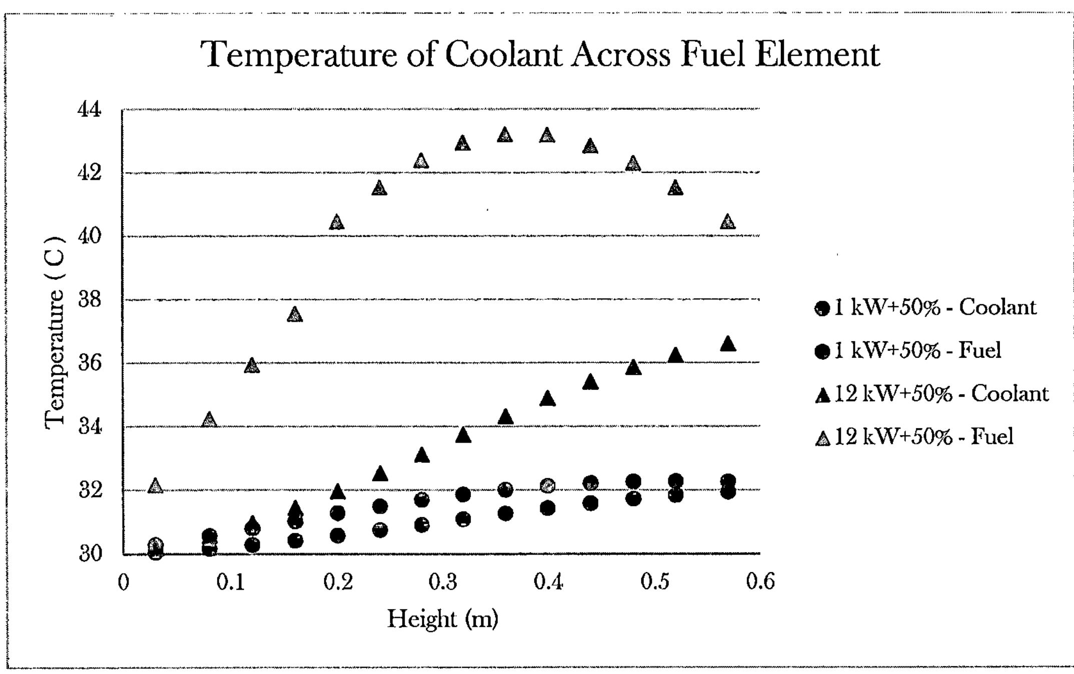
(a) Provide an answer to RAI 63 or indicate if the answer to RAI 63 is provided in the responses to RAIs 62 and 65 in your letter dated January 31, 2012.

(b) Table 4-21 in the revised PUR-1 SAR, Section 4.6 indicates that for 1 kW and 12 kW, the maximum fuel temperatures for the limiting fuel plate (31.92 and 39.1 degrees Celsius (C)) are lower than the maximum clad temperatures (43.42 and 43.4 degrees C). Discuss the physical phenomena that would result in these temperature values.

Response

(a) All inconsistencies regarding the power level have been addressed throughout the SAR and updated to 12 kW and 12kW+50% power uncertainty where appropriate.

(b) The axial temperature rise along a fuel element is plotted below. Due to the high thermal conductivity of the metallic fuel, small fuel meat thickness, and the lack of a gap between the fuel and the cladding of this plate design, the cladding and fuel exist at essentially the same temperature. The temperature of the wall never falls more than 0.02 $°C$ below that of the fuel.



An updated Table 4-21 (from the SAR) is shown below. Note that when the NATCON modelling tool is moved up to 98.6 kW, the power level for the onset of nucleate boiling (ONB), the maximum coolant temperature is 44.8 $°C$.

	Present Power	Uprate Power	ONB Power
Power Level	1 kW+50%	12 kW+50%	98.6 kW
Max. Fuel Temp. (°C)	32.28	43.20	112.61
Max. Clad Temp. (°C)	31.28	43.19	112.50
Coolant Inlet Temp. (°C)	30.0	30.0	30.0
Coolant Outlet Temp. (°C)	31.6	35.4	44.8
Margin to incipient boiling (°C)	78.49	68.12	0
Coolant Velocity (mm/s)	5.41	19.16	56.20
Coolant Mass Flow Rate (kg/m^{2}s)	5.39	19.04	55.69

6. RAI 70 in NRC letter dated July 14, 2011, stated:

> **NUREG-1537, Section 11 provides guidance for radiation protection provisions at the facility. In Section 4.4 of the SAR, it is stated that the radiation level above the reactor pool surface is about 1 mrem [millirem]/hr and that the radiation level along the outside lateral surface of the concrete biological shield is about 0.1 mrem/hr, when the core is operating at 1 kW. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential radiation levels and the potential radiation effects on facility staff. As part of evaluation, please indicate if the radiation levels bound those that would be encountered during fuel handling and maintenance operations. Additionally, include an evaluation of the safety analysis for potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).**

The response to RAI 70 by letter dated January 31, 2012, did not address the expected radiation dose levels at the requested increased licensed power level of 12 kW. The updated safety analysis you indicate bounding dose levels for facility staff and members of the public who may be located in nearby or adjacent, accessible public areas during the maximum operation power level for an extended period (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.) to demonstrate compliance with 10 CFR Part 20. Include all analyses, assumptions, and conclusions and indicate if the

radiation levels bound those that would be encountered during fuel handling and maintenance operations.

Response

The dose rate in the air above the reactor pool is given by

$$\dot{D} = \Phi(z)\mu_{air}$$

Where Φ is the flux of photons at a given energy, z is the height above the core, and μ_{air} is the mass absorption coefficient for photons of the given energy. The flux of photons is the primary question in determining the dose rate. The flux is essentially the attenuated dose at a distance $r\ [cm]$ from the core, multiplied by a buildup factor.

$$\Phi(z) = \int_0^\infty \Phi(z,E)dE = \int_0^\infty \frac{S_v}{4\pi r^2}\eta(E)B(\mu r,E)e^{-\mu r}\,dE$$

Where S_v is the photon source rate, $\eta(E)$ is the flux of photons with energy between E and E+dE, B is the buildup factor, μ is the attenuation coefficient, r is the distance form the core, and E is the energy of the photons. The photon attenuation coefficient is a function of energy and for water at 2 MeV, $\mu = 0.0493 \left[\frac{1}{cm}\right]$ (Attenuation data taken from the National Institute of Standards and Technology). For a shielding thickness of $395\ [cm]$ (the height of the water above the core), the quantity μR is 19.5 and represents the number of mean free paths the photon must travel through to be emitted through the top of the pool. Note that this is the dose directly above the reactor and would be further reduced at the edges of the pool as there is more attenuation.

The buildup factor $B(\mu r, E)$ has multiple forms one of which was reported by J.J. Taylor and has the form of a summed exponential

$$B(\mu r,E) = A(E)e^{-\alpha_1(E)*\mu R} + [1 - A(E)]e^{-\alpha_2(E)*\mu R}$$

The coefficients $A(E), \alpha_1(E)$, and $\alpha_2(E)$ have no physical meaning and are evaluated in tables reported by multiple sources. For a 2 $[MeV]$ photon,

$$B(\mu r,E) = 12.612 * e^{-0.0532*19.5} + [1 - 12.612]e^{-0.01932*19.5} = 27.7$$

The number of fissions in the reactor is bounded by a $12\ kW$ operating power plus 50% of power uncertainty. The number of fissions at full power is

$$18\ kW = 18\ kW * \frac{1000\ W}{1\ kW} * \frac{1\ \frac{J}{s}}{1\ W} * \frac{6.2415 \times 10^{12}\ MeV}{1\ Joule} * \frac{1\ fission}{200\ MeV} = 5.617 \times 10^{14}\ \frac{fissions}{sec}$$

As an example of the calculation, at $2\,[MeV]$, there are $1.8\left[\frac{\gamma}{fission}\right]$. The $2\,[MeV]$ flux at the surface of the pool is therefore

$$\Phi(395\,[cm], 2\,[MeV]) = \frac{5.617 \times 10^{14}\left[\frac{fissions}{sec}\right]}{4\pi * 395^2\,[cm^2]} * 1.8\left[\frac{\gamma}{fission}\right] * 27.7 * e^{-19.5}$$

$$\Phi(395, 2) = 47.5\left[\frac{\gamma}{cm^2 sec}\right]$$

Summing this expression across available energy values will give the total photons emitted from the reactor pool surface. This value is $1732\left[\frac{\gamma}{cm^2 sec}\right]$ over all energy groups (0-0.5 MeV, 0.5-2 MeV, etc.). The equivalent dose rate in air is then $3.25\left[\frac{mR}{hr}\right]$.

Energy (MeV)	Dose Rate (mRem/hr)
0.5	1.338×10^{-6}
2	0.141
4	1.681
6	1.183
8	0.239
Total: 3.245 mRem/hr	

These dose levels bound those that would be encountered during fuel handling operations as fuel remains at the bottom of the pool until it has cooled to levels which are acceptable within bounds of the TS.

For a dose to a member of the public in an unrestricted area, the unshielded dose reduces as the inverse square law. The closest such a person could get to the reactor pool is at minimum 5 meters. Treating the dose rate at the pool surface above the core as a point source, the source strength would be reduced as

$$I = \frac{S}{4\pi r^2} = \frac{3.245}{4\pi * 5^2} = 0.010\frac{mR}{m^2 hr}$$

Where S is the dose rate at the surface of the pool, and r is the distance from the pool surface. A very large person may have a frontal area of $1\,m^2$ which would give a dose rate of $0.010\,mR/hr$ which is less than that of the regulations in 10 CFR 20. Note that this does not take into account extended benefits from the shielding by air, walls, and the ceiling.

7. RAI 71 in NRC letter dated July 14, 2011 stated:

> **The requirements of 10 CFR 20.1101 states that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities, in order to limit the total effective dose equivalent to facility workers (annual occupational dose less than 5 rem [roentgen equivalent man]) and the total effective dose equivalent to individual members of the public (annual public dose less than 100 mrem). Please provide... a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total annual production of argon-41 from PUR-1 normal operations. In addition, please evaluate and discuss the potential maximum dose to a facility worker and to a member of the public (i.e., classrooms, hallways adjacent rooms, nearest dormitories, offices, etc.) due to this bounding yearly production and release of argon-41 from the facility.**

Your response to RAI 71 by letter dated April 10, 2013 (ADAMS Accession No. ML 13101A044), did not provide an analysis to determine the maximum effective dose to the maximally exposed member of the public for the total annual production of argon-41 from PUR-1 maximum licensed power operations. Provide a bounding safety analysis (with assumptions and conclusions) for the member of the public residing outside the facility perimeters.

Response

The buildup of ^{41}Ar is due to the absorption of a thermal neutron by ^{40}Ar. Gasses are naturally present in fluids like water and the amount is dependent on temperature as well as the partial pressure of the surrounding volume. Henry's Law dictates the amount of various gasses which are dissolved in fluid and carries a Henry's constant of $1.4 \times 10^{-5} \left[\frac{mol}{m^3 Pa}\right]$ at standard temperature and pressure conditions for natural argon. (Sander, 2014). Because solubility decreases with room temperature, this is a conservative estimate as the temperature of the pool for the rest of the analysis has been at or above 20 °C. With a pressure of 1 atm at the top of the pool, the number of Ar atoms dissolved per cubic centimeter is

$$1.4 \times 10^{-5} \left[\frac{mol}{m^3 Pa}\right] * 101325\, Pa * 6.022 \times 10^{23} \left[\frac{atoms}{mol}\right] * \frac{1}{1 \times 10^6} \left[\frac{m^3}{cm^3}\right] = 8.543 \times 10^{17} \left[\frac{atoms}{cm^3}\right]$$

^{40}Ar makes up 99% of all natural Argon, so the number of ^{40}Ar atoms per cubic centimeter would be corrected as

$$8.543 \times 10^{17} \left[\frac{atoms}{cm^3}\right] * .99 \left[\frac{^{40}Ar}{Ar_{nat}}\right] = 8.457 \times 10^{17} \left[\frac{atoms}{cm^3}\right]$$

The amount of time that the coolant spends within the core will be its irradiation time. The NATCON code, used in response to RAI #5 above, predicted a mass flow rate of $6.86\left[\frac{grams}{channel*sec}\right]$ or $6.86\left[\frac{cm^3}{channel*sec}\right]$. The 13 standard fuel elements have 15 channels and the 3 control elements have 11 which yields $228\,channels$ and a total volumetric flow rate of $1564.08\left[\frac{cm^3}{sec}\right]$. The recirculation time of this portion of the pool water through the core will be given by

$$\mathrm{T}_{circ} = \frac{V_{pool}}{\dot{V}_{core}}$$

The pool has a radius of $4\,[ft]$ or $121.92\,[cm]$ and a height of $17\,[ft]$ or $518\,[cm]$. The volume of $2.42 \times 10^7\,[cm^3]$ gives a recirculation period of

$$\mathrm{T}_{circ} = \frac{2.42 \times 10^7\,[cm^3]}{1564.08\left[\frac{cm^3}{sec}\right]} = 1.547 \times 10^4\,[sec] \cong 4.3\,[hours]$$

Using the MCNP6 model, the thermal neutron flux at 18 kW (12 kW + 50%) is predicted as being $\Phi = 2.66 \times 10^{11}\left[\frac{neutrons}{cm^2sec}\right]$ within the core volume. This value is conservative as the thermal neutron flux is usually considered linear with power increase which would suggest a flux of $\Phi = 10^{11}\left[\frac{neutrons}{cm^2sec}\right]$.

The saturation activity of ${}^{41}Ar$ will be that which would be normally produced and decayed while in the reactor volume, reduced by that which decays while circulating throughout the pool volume.

$$A_{sat} = \frac{N\sigma_{th}\Phi\left(1 - e^{-\lambda t}\right)}{\left(1 - e^{-\lambda(t+\mathrm{T}_{circ})}\right)}$$

Here, N is the number of atoms found within the core at any given time, σ_{th} is the thermal neutron absorption cross section for ${}^{40}Ar$ ($\sigma_{th} = 6.1 \times 10^{-25}\,cm^2$), λ is the decay constant ($1.0566 \times 10^{-4}\,[sec^{-1}]$, and t is the transit time through the core. With a core height of $60.96\,[cm]$ and a coolant velocity of $1.916\left[\frac{cm}{sec}\right]$, the time in the core is

$$t = \frac{60.96\,[cm]}{1.916\left[\frac{cm}{sec}\right]} = 31.82\,[sec]$$

Then A_{sat} is

$$A_{sat} = \frac{8.457 \times 10^{17} \left[\frac{\#}{cm^3}\right] 6.1 \times 10^{-25}[\mathrm{cm}^2]\, 2.66 \times 10^{11} \left[\frac{\#}{cm^2 sec}\right] \left(1 - e^{-(1.0566\times10^{-4}\,[sec^{-1}]*31.82\,[sec]}\right)}{\left(1 - e^{-(1.0566\times10^{-4}\,[sec^{-1}]*(31.82\,[sec]\,+1.547\times10^{4}\,[sec])}\right)}$$

$$A_{sat} = 571 \left[\frac{decays}{sec * cm^3}\right]$$

The number of ^{41}Ar atoms is

$$N_{Ar-41} = \frac{A_{sat}}{\lambda} = \frac{571 \left[\frac{decays}{sec * cm^3}\right]}{1.0566 \times 10^{-4} \left[\frac{1}{sec}\right]} = 5.411 \times 10^6 \left[\frac{atoms}{cm^3}\right]$$

The exchange of a gas in water with atmosphere can be modeled as

$$S = 0.93\ \mathrm{B} N_{Ar-41} A_{surf}$$

where B is an exchange coefficient reported as $5.7 \times 10^{-3} \left[\frac{cm}{sec}\right]$ and A_{surf} is the area of the surface of the pool as calculated from a radius of $121.92\ [cm]$. (Dorsey, 1940)

$$S = 0.93 *\ 5.7 \times 10^{-3} \left[\frac{cm}{sec}\right] * 5.411 \times 10^6 \left[\frac{atoms}{cm^3}\right] * \pi(121.92)^2\ [cm^2]$$

$$S = 1.34 \times 10^9 \left[\frac{atoms}{sec}\right]$$

Multiplying the source value obtained above with the decay constant would yield the activity emitted from the pool surface per second.

$$S = 1.34 \times 10^9 \left[\frac{atoms}{sec}\right] * 1.0566 \times 10^{-4} \left[\frac{1}{sec}\right] = 1.41 \times 10^5 \left[\frac{Bq}{sec}\right] * \frac{1}{3.7 \times 10^4} \left[\frac{\mu Ci}{Bq}\right]$$

$$S = 3.81 \left[\frac{\mu Ci}{sec}\right]$$

The time radioactive air remains in the reactor room is a factor of the pumping rate of ventilation systems and the decay of the argon. The half-lives can be combined into an effective half-life and would be

$$\frac{1}{\tau_{eff}} = \frac{1}{\tau_{Ar-41}} + \frac{1}{\tau_{air}}$$

$$\Rightarrow \tau_{eff} = \frac{\tau_{Ar-41} * \tau_{air}}{\tau_{Ar-41} + \tau_{air}}$$

If the reactor room has a volume of $4.24 \times 10^8\ [cm^3]$ and the fan removes air at a rate of $2 \times 10^5\ \left[\frac{cm^3}{sec}\right]$, the air has a lifetime of

$$\tau_{air} = \frac{V_{total}}{\dot{V}_{removal}} = \frac{4.24 \times 10^8\ [cm^3]}{2 \times 10^5\ \left[\frac{cm^3}{sec}\right]} = 2120\ [sec] = 35.3\ [min]$$

Using this as the half-life of air in the room, which accounts for mixing and parts of the air staying over time, the effective half-life of ^{41}Ar in the room is

$$\tau_{eff} = \frac{109.34\ [min] * 35.3\ [min]}{109.34\ [min] + 35.3\ [min]} = 26.7\ [min] = 1.602 \times 10^3\ [sec]$$

and the decay constant is

$$\lambda_{eff} = \frac{ln(2)}{\tau_{eff}} = 4.326 \times 10^{-4}\ \left[\frac{1}{sec}\right]$$

Considering the room now as the entire source term, the activity at saturation is

$$A_{room}(t) = \frac{S_{pool}}{\lambda V}$$

$$= \frac{3.81\ \left[\frac{\mu Ci}{sec}\right]}{4.326 \times 10^{-4}\ \left[\frac{1}{sec}\right] * 4.24 \times 10^8\ [cm^3]}$$

$$\boxed{A_{room,sat} = 2.08 \times 10^{-5}\ \left[\frac{\mu Ci}{cm^3}\right]}$$

Referencing FGR-11, the dose conversion factor for someone submerged in argon is

$$DCF_{Ar-41} = 8.029 \times 10^5\ \left[\frac{mRem/hr}{\mu Ci/cm^3}\right]$$

yielding a dose rate to a worker of

$$\dot{D} = 2.08 \times 10^{-5} * 8.029 \times 10^5 = 16.7\ \left[\frac{mRem}{hr}\right]$$

This is an incredibly conservative estimate for the steady state derived air concentration of the ^{41}Ar effluent. It assumes the pool water has reached a saturation of ^{41}Ar, which has then diffused completely into the reactor facility, as it is continually evacuated. To simply make the second pass of water through the reactor core would take more than four hours. An extremely long run time would be 20 hours at which time the pool would start to *approach* the saturation point, a time far less than that of the saturation time for the entire reactor facility.

The reactor facility operates at a negative pressure and air is expelled to the outside of the building through an exhaust fan. This is 15 meters above the ground. Following a similar analysis to that in RAI 12b regarding the MHA, the dispersion factor is $6.18 \times 10^{-3} [s/m^3]$. With an exhaust rate of $0.2\ [m^3/s]$

$$C_i = A\ (\chi/Q)\dot{V} = 2.08 \times 10^{-5} * 6.18 \times 10^{-3} * .2 = 2.571 \times 10^{-8} \left[\frac{\mu Ci}{cm^3}\right]$$

Multiplying this by the dose conversion factor for someone perfectly under the plume yields

$$\dot{D} = 8.029 \times 10^5 * 2.571 \times 10^{-8} = 0.021 \left[\frac{mRem}{hr}\right]$$

This dose rate is less than that cited in 10 CFR 20.1301(a)(2) of $2\ [mRem/hr]$.

8. In letter dated April 10, 2013, Purdue University provided an updated Section 13 of the SAR that included responses to RAIs 74-76, 80, 83, 85-91, and 93-95. State whether the updated SAR Section 13 also intended to provide answer to RAIs 79, 81, 82, 84, and 99 or provide answers:

(a) RAI 79 in NRC letter dated July 14, 2011, stated:

> **NUREG-1537, Section 13 provides guidance to identify the limiting event for each accident group and to perform quantitative analysis for that event. Please identify the categories of PUR-1 experiments that are performed and provide an evaluation of a safety analysis using the guidance of NUREG-1537, Section 13.1.6 for potential experiment malfunctions and their consequences.**

Response

The primary usage of the PUR-1 is training of nuclear students and educating the public about nuclear issues. Experiments performed include out of core dose measurements, fuel irradiations in the irradiation facilities, and other non-fueled irradiations. Each of the risks from these experiments has been evaluated in their respective SAR section or in responses to RAIs.

(b) RAI 81 in NRC letter dated July 14, 2011, stated:

> **NUREG-1537, Part 1, Section 13.1.1 provides guidance to identify Maximum Hypothetical Accidents (MHA) for non-power reactors. The**

MHA is to be selected so potential consequences of the postulated MHA scenario exceed and bound all credible accidents. NUREG-1537, Part 2, Chapter 13, p. 13-5 suggests that for a low-powered MTR fueled reactor, the MHA may be one of the following two events: cladding is stripped from one face of a fuel plate while suspended in air, or a fueled experiment fails in air. SAR, Section 13.1 states that "the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-1." Please provide ... a safety analysis of an MHA that considers the failure of one fuel plate in air would have lower consequences than the failure of a fueled experiment by justifying the MHA accident involving the fueled experiment capsule is more bounding that the failure of a fuel plate.

Response

The maximum hypothetical accident has been clarified to be that of having the cladding stripped from one face of a fuel plate while suspended in air in accordance with the suggestion of NUREG-1537, Part 1, Section 13.1.1.

(c) RAI 82 in NRC letter dated July 14, 2011, stated:

NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The PUR-1 MHA accident analysis for "Failure of a Fueled Experiment" is stated to be based upon a 1 W power deposition in the fueled experiment as consequence of the reactor operating at 1 kW. Please provide ... a safety analysis that provides the details of the energy deposition determination in the fueled sample with the reactor operating at the maximum requested licensed reactor power including the power level measurement uncertainty of 50% stated in SAR, Section 13.1.2.

Response

The PUR-1 reactor is equipped with irradiation facilities that can be loaded with a fueled experiment in order to test changes in material properties. 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 the general public.

Excess reactivity of the LEU core in PUR-1 was determined to be 0.00468 (0.46%) $\Delta k/k$ in a 190 fuel plate core (16 dummies), including a reactivity bias of $0.32\%\,\Delta k/k$. The Technical Specification limit of $0.6\%\Delta k/k$ is used to determine the maximum fueled experiment that can be utilized within the experimental facility. The main concern in the failure of a fueled experiment is the release of fission products. A limit of $0.5\ [Ci]$ of radio-iodine is specified in the PUR-1 TS. This is half of the radio-iodine that is postulated to be released in the MHA analyzed. This TS limit ensures that restrictions from 10 CFR 20 are met and the complete cladding failure of one face of a fuel plate remains the maximum hypothetical accident. Extended safety measures must be implemented to further ensure doses from a potential experimental failure remain far below those from the MHA.

a. A heating analysis must be done and approved by the reactor operating committee to verify that the maximum temperature experienced by the fueled experiment is less than or equal to half of the melting temperature.
b. Experimental analysis must include assumptions that include a loss of cooling within the experimental facility

The heat production in a sample experiment was found using a combination of the F7 tally from MCNP6. The result of an F7 tally over a cell in is units of

$$F7\ Tally\ \# = \left[\frac{MeV}{gram * Source\ Particle}\right]$$

By taking the ratio of the F7 tally for the fueled experiment with that of the entire fuel inventory, the proportion of reactor heat generation in the experiment is found. This can be scaled by the licensed reactor power to give the power generated in the fueled experiment.

$$P_{Experiment} = P_{reactor}\frac{(F7)_{Exp} * Mass_{Exp}}{(F7)_{Total} * 190\ plates * 12.5\frac{grams}{plate}}$$

As an example of this calculation, for the requested 12 kW license upgrade plus 50% power uncertainty, suppose a sample of 3% enriched uranium is placed in the experimental facility with a mass of 1.19 grams.

$$P_{experiment} = 18000\ Watts * \frac{8.134 \times 10^{-4}\left[\frac{MeV}{g * SP}\right] * 1.19\ [grams]}{3.74858 \times 10^{-3}\left[\frac{MeV}{g * SP}\right] * 190\ plates * 12.5\frac{g}{plate}}$$

$$P_{Exp} = 1.95\ Watts$$

Where the values for the F7 tallies came from the MCNP6 model of the PUR-1 core.

This mass would yield a radioiodine production of $0.46\ [Ci]$.

Following an identical analysis as the MHA of a fuel plate breach but with the conservative assumption that all of the fission products are released, the dose is shown in the tables below. The first table shows the dose from intake of radioiodine by an occupational worker if he were to remain in the plume for the specified amount of time. Realistic evacuation times are on the order of one minute which would lead to an exposure of approximately $\sim 0.1\ mrem$.

Time [hours]	Total [mRem] From Iodine Inhalation
0.01	0.055
0.1	0.545
1	4.697
10	24.704
100	68.457
1000	108.403
10000	109.990

The dose rate from submersion in the radionuclides is shown below.

Submersion Dose for Noble Gasses (Kr, Xe)	
Time [hours]	Total [mRem/hr]
0.01	0.0850
0.1	0.0816
1	0.0649
10	0.0278
100	0.0034
1000	2.44×10^{-5}
10000	2.48×10^{-16}

Again using the conservative assumptions for dose to a member of the public in a non-restricted area, the dose after submersion in the exhaust plume for an hour with the fan errantly still running is $9.03\ [mRem]$. This dose is below the limits set forth in 10 CFR 20. These dose rates fall below those of the MHA accident of having one face of a fuel plate completely exposed, making the failure of a fuel experiment not qualify for the MHA.

For other possible fueled experiments, the TS 3.5 specify that for singly encapsulated experiments

f. 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.

and for doubly encapsulated experiments

g. 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.

(d) RAI 84 in NRC letter dated July 14, 2011, stated:

NUREG-1537 states that the format and content of the TS follow ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 3.8.2 provides guidance for double encapsulation of experiments involving fissionable, explosive, reactive, or corrosive materials. Please provide ... a safety analysis for the MHA experiment of 1.1 g of U-235 with single encapsulation is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 3.8.2.

Response

Please reference RAI 14c) response above for a safety analysis for the failure of a fueled experiment of 1.19 g of U-235.

(e) RAI 99 in NRC letter dated July 14, 2011, stated:

SAR, Section 13.2.1, page 13-9 states "*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.*" Please provide ... a safety analysis that establishes the basis of 1.1 gm of U-235 for failure of a fueled experiment.

Response

Please reference RAI 14c) response above for a safety analysis for the failure of a fueled experiment of 1.19 g of U-235.

9. **In a letter dated April 10, 2013, Purdue University provided an update to the SAR, Section 13 containing an analysis for the designated MHA based on the failure of a fuel plate and not the malfunction of a fueled experiment. Provide an analysis determining whether the consequences of the failure of a potential experiment containing fissile material are bounded by the MHA as defined by PUR-1. In addition, discuss the basis for limiting the maximum allowable fissile content of fueled experiments and how the limit is to be controlled either by a TS or by other acceptable means.**

<u>Response</u>

Comments following the safety analysis of potential accidents from experiments clarify that the stripping of cladding from one face of a fuel plate is the bounding MHA.

10. **NUREG-1537, Part 2, Section 13, suggests that the definition of the maximum hypothetical accident (MHA) should be based on either a fuel plate or a fueled experiment failure, whichever leads to higher consequences. In the updated PUR-1 SAR, Section 13.1.1 and 13.1.6, it is stated that the failure of a fueled experiment containing fissile material is the designated MHA. However, in the updated PUR-1 SAR, Section 13.2.1, an analysis is proved for the designated MHA that is the failure of a fuel plate. Explain how this is consistent, or correct the updated PUR-1 SAR, Section 13, with regard to the MHA.**

<u>Response</u>

The PUR-1 SAR, Sections 13.1.1 and 13.1.6 have been updated to be consistent with PUR-1 SAR Section 13.2.1.

11. **NUREG-1537, Part 2, Section 13, defines the MHA as the failure of the cladding of one face of one fuel plate while suspended in air. PUR-1 SAR Section 13.2.1 defines the MHA as the failure of one face of one fuel plate submerged in the reactor pool resulting in a potentially non-conservative amount of radioactive iodine release into the reactor air volume. Discuss whether the assumption of radioactive fission product release into the pool water is a conservative assumption. Discuss whether the assumption bounds a failure of the fuel plate cladding while the fuel element is suspended in air, releasing fission products directly into the reactor air volume.**

Response

PUR-1 SAR Section 13.2.1 has been clarified to indicate analysis is done for the MHA of the failure of the cladding of one face of one fuel plate while suspended in air following the guideline of NUREG-1537, Part 2, Section 13. This complete analysis is done in the subsequent RAI 12 response.

12. Provide an MHA safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total radioactive fission product release after the failure of one side of one fuel plate. Discuss methodological assumptions associated with the following analytical steps:

(a) Derivation of fission product atmospheric dispersion factor, x/Q using either the methodology suggested in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments At Nuclear Power Plants," Revision 1, issued February 1983, or another equivalent method.

(b) Dose conversion calculation using the Environmental Protection Agency's Federal Guidance Report (FGR)-11 and FGR-12 dose conversion coefficients or another equivalent methodology to account for inhalation/ingestion and submersion exposures.

Response

This section will estimate the total amount of radioiodine released from the failed fuel plate. No credit is taken for the reduction in activity resulting from radioactive decay during the time of the release, i.e. an instantaneous release of the radioiodine that can escape the fuel is assumed. Complete and perfect mixing of the available radioiodine inventory with the reactor facility volume is also assumed.

The activity via the production rate of the *i*th radioiodine isotope in a plate is determined by the following:

$$A_i = \lambda_i N_i = \dot{R}_{fiss} F_i$$

where $\dot{R}$ is the rate of fissions in the plate of interest, F_i is the fraction fission yield for each radioiodine, λ_i is the decay constant for each radioiodine and N_i is the saturation number of atoms of each specific radioiodine. The constants F_i, λ_i, and the results of calculations for $\lambda_i N_i$ and N_i for plate 1348, are shown in Table 13-2 assuming $12\ kW + 50\%$ operating power.

It is assumed that not all of the iodine produced will be released from the plate. As suggested by NUREG/CR-2079, only the fission fragment gases within recoil range of the surface of the fuel (1.37×10^{-3} $[cm]$ for aluminum matrix fuels) will escape. The thickness of the fuel meat in a PUR-1 plate is 0.0508 cm. The total volume of the fuel meat is

$$V_{tot} = 60.01 * 5.96 * 0.0508 = 18.170\ [cm^3]$$

and the volume released from recoil of the fission gasses is

$$V_{release} = 60.01 * 5.96 * 1.37 \times 10^{-3} = 0.490\ [cm^3]$$

The fractional fission product gas release is therefore

$$F_{release} = \frac{V_{release}}{V_{tot}} = \frac{0.490}{18.170} = 0.027$$

The number of fissions per second in Plate 1348 with a fission power of 157 $[Watts]$ is given by

$$157\ W = 157\ W * \frac{1\frac{J}{s}}{1\ W} * \frac{6.2415 \times 10^{12}\ MeV}{1\ Joule} * \frac{1\ fission}{200\ MeV} = 4.90 \times 10^{12}\ \frac{fissions}{sec}$$

The activity of I-135 is then the product of these number of fissions and the fractional yield per fission of this element which is 6.28%.

$$Activity_{I-135} = 4.90 \times 10^{12} \left[\frac{fissions}{second}\right] * 0.0628 \left[\frac{decays}{fission}\right] * \frac{1}{3.7 \times 10^{10}} \left[\frac{Ci}{decay/second}\right] * 0.027\ \{Fractional\ Release\}$$

$$Activity_{I-135} = 0.224\ [Ci]$$

The calculated values for each of the five iodine isotopes of concern are shown in below.

Radioiodines released from Plate 1348 into Facility Air.

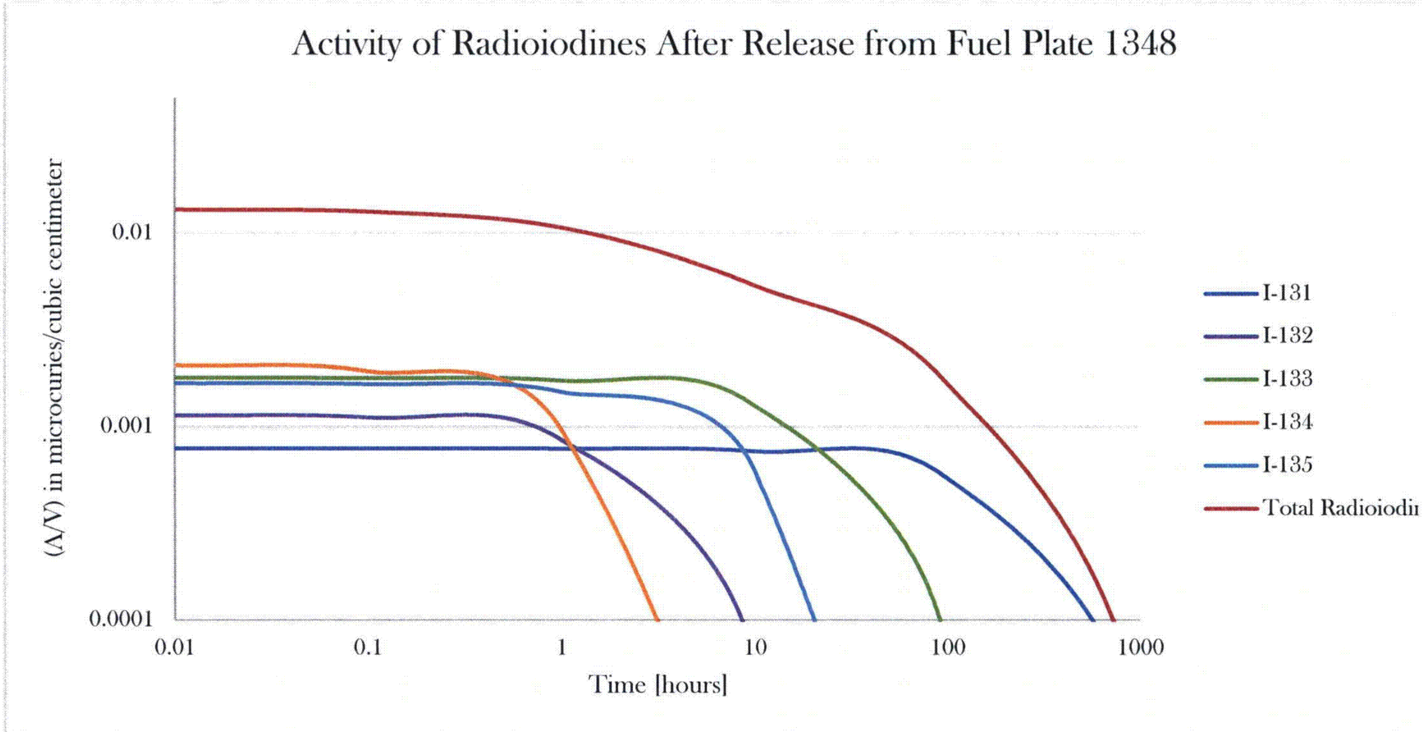
Isotope	Fractional Yield	Half-life (sec)	Decay Constant	Activity [Ci]
I-131	0.0289	6.95×10^5	9.98×10^{-7}	0.103
I-132	0.0431	8.21×10^3	8.44×10^{-5}	0.154
I-133	0.067	7.49×10^5	9.26×10^{-6}	0.239
I-134	0.0783	3.16×10^3	2.20×10^{-4}	0.280
I-135	0.0628	2.37×10^4	2.93×10^{-5}	0.224
			Total	1 Ci

The activity produced from the fuel plate breach is then dispersed throughout the immediate reactor facility air. The facility has a volume of $424\ m^3$ giving the activity per volume of air is given as

$$A/V = \frac{Activity}{Volume}$$

$$\Rightarrow (A/V)_{I-135} = \frac{0.224\ [Ci]}{424 \times 10^6\ [cm^3]} * \frac{10^6\ [\mu Ci]}{1\ [Ci]} = 5.283 \times 10^{-4} \left[\frac{\mu Ci}{cm^3}\right]$$

Over time, this level of activity decreases according to the laws of radioactive decay. The activity for each radioiodine is shown in the figure below.



As a facility worker breathes in the radioiodine, they will ingest some of the air borne material. Assuming that an individual breathes in 347 $\left[\frac{cm^3}{sec}\right]$, the dose from continued breathing of the radioactive air is given by

$$D_i = \int_0^t \dot{B} * C(t)_i\, dt = \{B * (A/V)_i\} * \int_0^t e^{-\lambda_i t}\, dt = \dot{B}(A/V)_i * \left(1 - e^{-\lambda_i t}\right)$$

where $\dot{D}_i$ is the dose rate from inhalation of each radioiodine, $\dot{B}$ is the breathing rate, $(A/V)_i$ is the activity as a function of time of each radioiodine, λ_i is the respective decay constant, and t is the time duration of inhalation. The final effective dose is this value multiplied by a dose conversion factor to account for the different energies of decay and other factors.

$$D_{eff} = \sum_{i=131}^{135} D_i K_i$$

For this analysis, the dose is expressed in units of dose (rads) to the thyroid gland of a person breathing the radioiodine-bearing air. Although the concentration of iodine in the building air is continuously reduced by various processes such as radioactive decay, purging of the building air by the exhaust fan, and plating out of the iodine on surfaces, we will assume only a reduction in concentration resulting from radioactive decay. This implies that the concentrations of iodine in the air outside the building, taking no credit for dissipation in the air outside the building, will be the same as those in the building. Estimates of the thyroid dose rates are shown below.

Integrated thyroid dose estimates for several exposure periods following release of Plate 1348 radioiodines into the reactor air.

Integral Dose in mRem for Several Exposure Periods after Release

Isotope	36 sec	6 min	1 hour	10 hours	4.2 days	42 days
I-131	1.42×10^{-2}	0.142	1.42	14.0	119.	385
I-132	6.47×10^{-2}	0.638	5.59	20.3	21.3	21.3
I-133	8.26×10^{-2}	0.824	8.12	70.2	239	248
I-134	0.143	1.38	9.89	18.1	18.1	18.1
I-135	7.59×10^{-2}	0.756	7.21	46.9	72.0	72.0
	0.380	3.74	32.22	170	470	744 $[mRem]$

As can be seen from these results, about two hours of continuous exposure to released radioiodine in the reactor room air would be required to attain a thyroid dose equivalent of $\sim 100\ mRem$. Even this exposure would be extremely unlikely, since it is difficult to conceive of a credible combination of accident conditions and personnel occupancy which will result in such doses being achieved. The imposition of such limited cloud dispersion effects required to approach these estimates is not realistic. It is more likely that dispersive effects will result in much lower doses. For example, even the building blower exhaust at $424\ ft^3/min$ flow rate will cause a concentration reduction.

MHA considering All Gaseous Fission Products

A similar analysis can be done for all gaseous radionuclides released from a single ruptured fuel plate. The same analysis for iodine as was done in the preceding sections applies, as well as all noble gases (fission products) available in the plate released to the building air. No credit for decay of the radioisotopes during release and dispersion is taken. This assumption, as before, leads to conservative results in that the estimates obtained are higher than those that would actually occur in this postulated accident.

	Dose Rate in mRem/hr for Several Exposure Periods After Release					
Isotope	36 sec	6 min	1 hour	10 hours	4.2 days	42 days
Kr-85m	3.78×10^{-6}	3.73×10^{-6}	3.25×10^{-6}	8.07×10^{-7}	0	0
Kr-87	3.56×10^{-2}	3.39×10^{-2}	2.08×10^{-2}	1.53×10^{-4}	0	0
Kr-88	0.126	0.123	9.87×10^{-2}	1.10×10^{-2}	0	0
Xe-131m	5.91×10^{-5}	5.91×10^{-5}	5.89×10^{-5}	5.77×10^{-5}	4.64×10^{-5}	5.22×10^{-6}
Xe-133m	1×10^{-3}	1×10^{-3}	9.89×10^{-4}	8.79×10^{-4}	2.68×10^{-4}	0
Xe-133	4.01×10^{-2}	4.01×10^{-2}	3.99×10^{-2}	3.79×10^{-2}	2.31×10^{-2}	1.62×10^{-4}
Xe-135	0.301	0.299	0.280	0.141	1.48×10^{-4}	0
Xe-135m	7.95×10^{-2}	6.22×10^{-2}	5.39×10^{-3}	0	0	0
Total	0.583	0.560	0.445	0.191	0.0236	1.68×10^{-4}

In the event of a maximum hypothetical accident such as the one discussed, reactor facility workers would be required to identify that an alarm has occurred, notify others in the immediate vicinity to evacuate, and disable the ventilation system to attempt to confine the fission products. This process would take approximately one minute in a very conservative setting. The dose received during this time due to the noble gasses listed above would be $0.01\ mRem$. The dose contribution from the radioiodines would be much greater at $\sim 0.75\ mRem$ but still well below the dose limits for a facility worker.

Doses to persons outside the building will come from submersion in a cloud of released radionuclides and from radiation emitted from the reactor building. The submersion dose results from the diluted radionuclide stream from the exhaust fan or from natural flow of air through the building that exits at the roofline (if the exhaust fan was properly disabled). An analysis for the activity concentration release from the building can be performed using the equation below,

which includes release from the exhaust fan. The concentration of activity of each radionuclide is given as

$$C_i = A_i(\chi/Q)\dot{V} \quad \left[\frac{Ci}{m^3}\right]$$

where A_i is the fractional release of activity of the radionuclide from the fuel plate as dispersed through the reactor facility air $[Ci/m^3]$, (χ/Q) is the atmospheric dispersion factor as discussed in NUREG/CR-2260, $\dot{V}$ is the volumetric release rate from the fan and t is the time after release. The dispersion factor is given by

$$(\chi/Q) = \frac{1}{u_{10}\left(\pi\sigma_y\sigma_z + \frac{A}{2}\right)}$$

where u_{10} is the velocity of the wind at a height of 10 meters above the release point, σ_y is the lateral plume spread based on the Pasquill-Gifford correlation, and σ_z is the vertical plume spread (both in meters). Both of the plume spreads are dependent on the atmospheric stability which is conservatively assumed to be highly stable (as given by Regulatory Guide 1.23). Assuming that the plume spread in the lateral and vertical direction are the same, there is a wind speed of $1\,m/s$, an exclusion zone of 10 meters (and extrapolating values from NUREG/CR-2260), giving values for the plume spread of $\sigma_y = \sigma_z = 2.38\,[meters]$ gives a dispersion factor of

$$(\chi/Q) = \frac{1}{1\left[\frac{m}{s}\right]\left(\pi * (2.38)^2\,[m^2] + \frac{288\,[m^2]}{2}\right)} = 6.18 \times 10^{-3} \left[\frac{s}{m^3}\right]$$

No credit is taken in the above expression for horizontal plume meandering or the fact that air from the basement area is exhausted at a minimum height of 50 feet. Additionally, the average wind speed is greater than $3\,m/s$ resulting in a conservative factor of at least 3.

For I-135, with an activity of $5.29 \times 10^{-4} \left[\frac{\mu Ci}{cm^3}\right]$ the downwind activity concentration would be

$$C_i = A_i(\chi/Q)\dot{V} = A_i * 0.00618\left[\frac{s}{m^3}\right]0.2\left[\frac{m^3}{sec}\right] = 0.00124 * A_i\left[\frac{Ci}{m^3}\right]$$

$$C_i = 0.00124 * 5.29 \times 10^{-4}\left[\frac{\mu Ci}{cm^3}\right]$$

$$C_i = 6.538 \times 10^{-7}\left[\frac{\mu Ci}{m^3}\right]$$

The table below shows the downwind concentration and activity for each radio nuclide.

Isotope	Accident Activity [Ci]	Downwind Concentration $[\mu Ci/m^3]$
I-131	0.103	0.300
I-132	0.154	0.449
I-133	0.239	0.698
I-134	0.280	0.815
I-135	0.224	0.654
Kr-85m	0.046	0.134
Kr-87	0.091	0.267
Kr-88	0.127	0.370
Xe-131m	0.001	0.004
Xe-133m	0.007	0.020
Xe-133	0.239	0.698
Xe-135	0.234	0.681
Xe-135m	0.039	0.115
		Total: 5.20 $[\mu Ci/m^3]$

The submersion dose rate is this concentration of activity multiplied for the dose conversion factor for submersion.

$$\dot{D}_{i,submersion} = (DCF)_i * C_i$$

where the dose conversion factor is in units of $\left[\frac{mRem/hr}{\mu Ci/cm^3}\right]$ and is given in FGR-11. The dose rate for inhalation of the radioiodine is similar but includes the rate of air intake by the average individual which is approximately 347 cm^3/sec.

$$\dot{D}_{i,thyroid} = (DCF)_{i,thyroid} * \dot{BR} * C_i$$

The total dose rate is therefore the sum of the inhalation and thyroid dose rate over all isotopes.

$$\dot{D}_{total} = \sum_i \dot{D}_{i,submersion} + \sum_i \dot{D}_{i,thyroid}$$

$$\dot{D}_{total} = 0.69 + 18.91 = 19.6 \left[\frac{mRem}{hr}\right]$$

This is the dose rate to a member of the public if they were standing directly in the plume downwind of a maximum hypothetical accident when the fan in the reactor facility failed to depower as considered on a very calm day. In the worst case scenario, an individual would station themselves directly outside of the reactor facility underneath the exhaust. If it took 15 minutes to setup the exclusion zone, a person could receive at most 4.9 mRem which is still below the dose requirements to the general public. Expanding the exclusion zone as well as

ensuring the facility ventilation is deactivated only serves to lessen the dose to the public. If it is assumed that the fan does turn off properly, and the leakage of air through the top of the facility is reduced to 0.001 $[m^3/sec]$, the dose rate falls to 0.1 $[mRem/hr]$.

13. Provide an analysis and discuss the potential maximum radiological dose estimate due to the MHA at the following suggested locations:

(a) Facility worker – located inside the restricted area considering any evacuation procedure and potential residence time for staff exposed to fission product inhalation/ingestion and direct gamma ray radiation. The exhaust system operational status should be consistent with conservative assumptions.

Response

Reference response to RAI 12 above for a discussion of the dose to a facility worker during a postulated maximum hypothetical accident.

(b) Members of the public – located in adjacent, publicly accessible areas inside the reactor building (i.e. classrooms, hallways, adjacent rooms) potentially exposed to fission product inhalation and/or gamma ray radiation, taking into account any procedural process for evacuation, including the emergency plan. The exhaust system operational status should be consistent with conservative assumptions.

Response

Reference response to RAI 12 above for a discussion of the dose to members of the public in adjacent publically accessible areas inside the reactor building during a postulated maximum hypothetical accident.

(c) Members of the public – located outside the reactor building (maximally exposed location, nearest dormitories, offices, etc.) exposed to fission product inhalation/ingestion released from the reactor building and gamma ray radiation. The exhaust system operation status should be consistent with conservative assumptions.

Response

Reference response to RAI 12 above for a discussion of the dose to member of the public outside the reactor building during a postulated maximum hypothetical accident.

14. 10 CFR Part 20, "Standards for Protection against Radiation," provides the regulatory framework and NUREG-1537, Part 1, Section 13.1.3 provides the guidance for licensees to systematically analyze and discuss credible accidents in each accident

category. Section 13.1.3 of the updated PUR-1 SAR, describes the loss of coolant accident (LOCA) scenario. The updated PUR-1 SAR does not include an estimate for radiation levels in the reactor floor and the roof areas, due to the unshielded reactor core, after a postulated large LOCA event. The SAR should provide the consequent maximum dose rates at various locations on the reactor floor and outside on the reactor building roof. In accordance with 10 CFR Part 20, provide the accumulated doses to reactor building occupants and the maximally exposed member of the public, considering evacuation procedure and potential residence time for staff. In addition, provide an estimate when facility staff may enter the reactor building to start recovery operations.

Response

The maximum possible conceivable LOCA accident in the PUR-1 reactor would be that of having a large hole or crack develop in the reactor pool liner and the water draining from there. The only other option would be that of pumping or evaporating water from the top of the pool which would not be a bounding case. Modelling the reactor pool as a suspended tank with a hole or crack in the bottom, the Bernoulli equation shows

$$\rho g z_{surf} + p_{atm} = \frac{1}{2}\rho v_{out}^2 + \rho g h + p_{atm}$$

$$v_{out} = \sqrt{2g(z_{surf} - h)}$$

where ρ is the density of water, g is the gravitational constant, p_{atm} is the atmospheric pressure, v_{out} is the velocity of the water going out of the hole, z is the height of the surface of the water, and h is the height of the hole. If the hole is defined to be at the bottom of the tank (bounding assumption)

$$v_{out} = \sqrt{2g(z_{surf})}$$

The loss of mass through the hole is given by

$$\frac{dm}{dt} = -\rho A_{hole} V$$

and the total mass in the tank is

$$m_{tot} = \rho A_{tank} z_{surf}$$

yielding

$$\frac{d(\rho A_{tank} z_{surf})}{dt} = -\rho A_{hole} v_{out}$$

$$\frac{d(z_{surf}(t))}{dt} = -\frac{A_{hole}}{A_{tank}} \sqrt{2g\left(z_{surf}(t)\right)}$$

If there was a crack at the bottom of the tank which formed and was 5 $[cm]$ wide that ran the diameter of the pool, its area would be

$$A_{hole} = 0.05\,[m] * 2.44[m] = 0.122\,[m^2]$$

This area of a hole is equivalent to an instantaneous hole in the bottom of the reactor pool with a diameter of $\sim 40\,[cm]$

$$A_{hole} = \pi\left(\frac{d}{2}\right)^2 = \pi\left(\frac{.4}{2}\right)^2 \cong 0.12\,[m^2]$$

$$\frac{d(z_{surf}(t))}{dt} = -\frac{0.122}{\pi\left(\frac{2.44}{2}\right)^2} \sqrt{2 * 9.81 * \left(z_{surf}(t)\right)} = -0.116\sqrt{z_{surf}(t)}$$

$$\frac{d(z_{surf}(t))}{\sqrt{z_{surf}(t)}} = -0.116\,dt$$

Integrating yields

$$t = 17.24\sqrt{z_{surf}}$$

In order for the tank to drain half of its water, the surface of the tank would go down 2.59 $[meters]$. The time to do this would be

$$t = 27.7\ seconds$$

The total time to completely drain the tank is 40 seconds. This time frame is adequate for an operator to become aware of the situation, initiate a scram and begin an evacuation if necessary. In a case where the operator did not initiate the scram, the pool-top radiation monitor will do so on a high-alarm at 50 $\left[\frac{mRem}{hr}\right]$. If for some reason a scram was not initiated, the reactor would become subcritical upon the loss of the water moderator. The potential source term would therefore not be the reactor at full power (12 kW plus 50%), but rather the fission product decays still present in the core.

Following the same analysis as the shielding of the reactor at full power as in RAI 6 above, using air as the only attenuator to the top of the pool, and using a source of 1.69×10^{13} fissions per second (assuming the source term is 3% of full reactor power after the scram), the dose rate at the original height of the pool surface is 295 $\left[\frac{Rem}{hr}\right]$. Assuming a facility worker errantly looked

over the pool edge immediately after the scram with no shielding, he would receive a dose of less than 5 Rem to his upper body in one minute. Upon such an event, the reactor room would be immediately evacuated, reducing the worker dose further. To others not directly above the pool the ground and concrete tank would reduce the dose still further.

The member of the public of potential concern in the event of a sudden LOCA is someone present in the classroom above the reactor room. The ceiling is 19 feet above the top of the reactor pool and the floor of the upper room is approximately 4 inches of ordinary concrete. Extending the dose calculation to include the additional air and the concrete gives a potential dose rate to a member of the public of 6.6 $\left[\frac{Rem}{hr}\right]$. Again, in such an event the building would be immediately evacuated such that the actual dose received by a member of the public would be significantly reduced.

TECHNICAL SPECIFICATIONS

FOR THE

PURDUE UNIVERSITY REACTOR, PUR-1

DOCKET NUMBER 50-182

FACILITY LICENSE NO. R-87

West Lafayette, IN 47907

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1. DEFINITIONS

The following frequently used terms are to aid in the uniform interpretation of these specifications:

1.1 Channel – A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.2 Channel Calibration - A channel calibration is an adjustment of the channel such that its output corresponds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and is deemed to include a channel test.

1.3 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison of the channel with other independent channels or methods of measuring the same variable.

1.4 Channel Test - A channel test is the introduction of a simulated signal into a channel to verify that it is operable.

1.5 Confinement – Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.6 Containment – Containment is an enclosure of the facility designed to (1) be at a negative internal pressure to ensure in-leakage, (2) control the release of effluents to the environment, and (3) mitigate the consequences of certain analyzed accidents or events.

1.7 Core Configuration – The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and regulating/control rods occupying the core grid.

1.8 Core Experiment - A core experiment is one placed in the core, in the graphite reflector, or within six inches (measured horizontally) of the reflector. This includes any experiment in the pool directly above or below the core.

1.9 Direct Supervision – In visual and audible contact.

1.10 Excess reactivity – Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff} = 1) at reference core conditions or at a specified set of conditions.

1.11 Experiment – Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of its design to carry our experiments is not normally considered an experiment.

1.12 Experimental Facility - Experimental facilities are:

a. those regions specifically designated as locations for experiments or

b. systems designed to permit or enhance the passage of a beam of radiation to another location.

1.13 Experiment With Movable Parts (Secured or Nonsecured) - An experiment with movable parts is an experiment that contains parts that are intended to be moved while the reactor is operating.

1.14 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Tenth Ed. (2000), or is given an Identification of Reactivity (Stability) index of 2, 3 or 4 by the National Fire Protection Association in its publication 704, "Identification System for Fire Hazards of Materials."

1.15 Fueled Experiment - A fueled experiment is any experiment planned for irradiation of uranium 233, uranium 235, plutonium 239, or plutonium 241.

1.16 License – The written authorization, by the responsible authority, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

1.17 Licensed – See licensee.

1.18 Licensee – An individual or organization holding a license.

1.19 Measured Value - The measured value is the value of a parameter as it appears at the output of a channel.

1.20 Movable Experiment - A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.21 New Experiment - A new experiment is one whose nuclear characteristics have not been experimentally determined.

1.22 Non-secured Experiment – See Unsecured Experiment.

1.23 Operable - A system or component is operable when it is capable of performing its intended function in a normal manner.

1.24 Operating - A system or component is operating when it is performing its intended function.

1.25 Pool Experiment - A pool experiment is one positioned within the pool more than six inches (measured horizontally) from the graphite reflector.

1.26 Protective action – Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

1.27 Reactivity worth of an experiment – The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted or removed from its intended position.

1.28 Reactor Facility - The reactor facility is that portion of the ground floor of the Duncan Annex of the Electrical Engineering Building occupied by the School of Nuclear Engineering used for activities associated with the reactor.

1.29 Reactor Operating – The reactor is operating whenever it is not secured or shut down.

1.30 Reactor Operator – An individual who is licensed to manipulate the controls of the reactor.

1.31 Reactor Safety System - The reactor safety system is that combination of measuring channels and associated circuitry which forms the automatic protective system of the reactor, or provides information which requires manual protective action to be initiated.

1.32 Reactor Secured – A reactor is secured when

a. *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

b. *Or* the following conditions exist:

1. Reactor shutdown

2. Electrical power to the control rod circuits is switched off and the switch key is in proper custody.

3. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;

1.33 Reactor Shutdown - That subcritical condition of the reactor where the negative reactivity, with or without experiments in place, is equal to or greater than the shutdown margin.

1.34 Readily Available on Call - Readily available on call shall mean the licensed senior operator shall insure that he is within a reasonable driving time (1/2 hour) from the reactor building, and the operator on duty is currently informed, and can contact him by phone.

1.35 Removable Experiment - A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.36 Responsible authority: A governmental or other entity with the authority to issue licenses, charters, permits, or certificates.

1.37 Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.38 Senior Reactor Operator – An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.39 Shall, should, and may – The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

1.40 Shutdown Margin – The shutdown margin, relative to the cold xenon-free condition with the most reactive shim rod fully withdrawn, and the regulating rod fully withdrawn.

1.41 Surveillance and Test Intervals - These are intervals established for periodic surveillance and test actions. Established intervals shall be maintained on the average. Maximum intervals are allowed to provide operational flexibility, not to reduce frequency.

1.42 Reference core condition: The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.30 dollar).

1.43 Rod, control: A control rod is a device fabricated from neutron-absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.44 Rod, regulating: The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability and may have a fueled follower. Its position may be varied manually or by a servo-controller.

1.45 Rod, Shim-Safety: The control rods used in PUR-1 as described in the definition for Rod, control.

1.46 Tried Experiment - A tried experiment is:

a. An experiment previously performed in this facility, or

b. An experiment of approximately the same nuclear characteristics as an experiment previously tried.

1.47 True Value -The true value of a parameter is its exact value at any instant.

1.48 Unscheduled Shutdown – An unscheduled shutdown is defined as any unplanned shutdown of the reactor by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

1.49 Unsecured Experiment - Any experiment, experimental facility, or component of an experiment is considered to be unsecured when it is not secured as defined in part 1.36 of this section.

2. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit

Safety limits for nuclear reactors are limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier is the fuel cladding.

Applicability - This specification applies to the temperature of the reactor fuel and cladding under any condition of operation.

Objective - *The objective is to ensure fuel cladding integrity.*

Specification – The fuel and cladding temperatures shall not exceed 530°C (986°F).

Basis – In the Purdue University Reactor, the first and principal barrier protecting against release of radioactivity is the cladding of the fuel plates. The 6061 aluminum alloy cladding of the LEU fuel plates 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 condition of operation.

2.2 Limiting Safety System Setting

Applicability - This specification applies to the reactor power level safety system setting for steady state operation.

Objective - The objective is to assure that the safety limit is not exceeded.

Specification – The measured value of the power level scram shall be no higher than 12.0 kW.

Basis - The LSSS has been chosen to assure that the automatic reactor protective system will be actuated in such a manner as to prevent the safety limit from being exceeded during the most severe expected abnormal condition.

The function of the LSSS is to prevent the temperature of the reactor fuel and cladding from reaching the safety limit under any condition of operation. During steady-state operation, a power level of 94.2 kW is required to initiate the onset of nucleate boiling. This is far higher than the maximum power of 18 kW, which allows for 50% instrument uncertainties in measuring power level.

For the transients that were analyzed, the temperature of the fuel and cladding reach maximum temperatures of 49°C, assuming reactor trip at 18 kW after failure of the first trip. This temperature is far below the safety limit of 530°C.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability - These specifications apply to the reactivity conditions of the reactor, and the reactivity worths of control rods and experiments

Objective - The objective is to assure that the reactor can be shut down at all times, that the safety limits will not be exceeded, and that operation is within the limits analyzed in the SAR.

Specification - The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin, relative to the cold xenon-free condition with the most reactive shim rod fully withdrawn, and the regulating rod fully withdrawn shall be at least 0.01 $\Delta k/k$.

b. The reactor shall be subcritical by more than 0.03 $\Delta k/k$ during core loading changes.

c. No shim-safety rod shall be removed from the core if the shutdown margin is less than 0.01 $\Delta k/k$ with the remaining shim-safety rod fully withdrawn.

d. The reactor shall be shutdown if the maximum positive reactivity of the core and any installed experiment exceeds 0.006 $\Delta k/k$.

e. The reactivity worth of each experiment shall be limited as follows:

Experiment	Maximum Reactivity Worth
Movable	.003 $\Delta k/k$
Unsecured	.003 $\Delta k/k$
Secured	.004 $\Delta k/k$

f. The total worth of all movable and unsecured experiments shall not exceed 0.003 $\Delta k/k$.

g. The total worth of all secured experiments shall not exceed 0.005 $\Delta k/k$.

Bases - The shutdown margin required by Specification 3.1.a assures that the reactor can be shut down from any operating condition and will remain shut down even if the control rod of the highest reactivity worth should be in the fully withdrawn position.

Specifications 3.1.b and 3.1.c provide assurance that the core will remain subcritical during loading changes and shim-safety rod maintenance or inspection.

Specification 3.1.d limits the allowable excess reactivity to the value assumed in the HSR. This limit assures that the consequences of reactivity transients will not be increased relative to transients previously reviewed, and assures reactor periods of sufficient length so that the reactor may be shutdown without exceeding the safety limit.

Specification 3.1.e limits the reactivity worth of secured experiments to values of reactivity which, if introduced as a positive step change, are calculated not to cause fuel melting. This specification also limits the reactivity worth of unsecured and movable experiments to values of reactivity which, if introduced as a positive step change, would not cause the violation of a safety limit. The manipulation of experiments worth up to 0.003 $\Delta k/k$ will result in reactor periods longer than 9 seconds. These periods can be readily compensated for by the action of the safety system without exceeding any safety limits.

A limitation of 0.003 $\Delta k/k$ for the total reactivity worth of all movable and unsecured experiments provides assurance that a common failure affecting all such experiments cannot result in an accident of greater consequences than the maximum credible accident analyzed in the HSR.

Specification 3.1.g along with 3.1.a assures that the reactor is capable of being shut down in the event of a positive reactivity insertion caused by the flooding of an experiment.

3.2 Reactor Safety System

Applicability - This specification applies to the reactor safety system and other safety-related instrumentation.

Objective - The objective is to specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

Specification - The reactor shall not be made critical unless the following conditions are met:

a. The reactor safety channels and safety-related instrumentation are operable in accordance with Tables I and II including the minimum number of channels and the indicated maximum or minimum set points.

b. Both shim-safety rods and the regulating rod shall be operable.

c. The time from the initiation of a scram condition in the scram circuit until the shim-safety rod reaches the rod lower limit switch shall not exceed one second.

TABLE I. SAFETY CHANNELS REQUIRED FOR OPERATION

Channel	Minimum Number Required	Setpoint	Function
Log count rate and period	1(a)	2 cps 12 sec. period 7 sec. period	2 cps rod withdrawal interlock Setback Slow Scram
Log N and period	1(b)	12 sec. period 7 sec. period 7 sec period 120% power	Setback Slow Scram Fast Scram Slow Scram
Linear	1	110% range 120% range	Setback Slow Scram
Safety	1(b)	110% power 120% power	Setback Fast Scram
Manual Scram (console) (hallway)	 1 1		 Slow Scram Slow Scram
(a) Not required after Log N-Period channel comes on scale. (b) Required to be operable but not on scale at startup.			

TABLE II. SAFETY-RELATED CHANNELS (AREA RADIATION MONITORS)

Channel	Minimum Number Required(c)	Setpoint	Function
Pool top monitor	1	50 mR/hr or2x full power background	Slow Scram
Water process	1	7 ½ mR/hr	Slow Scram
Console Monitor	1	7 ½ mR/hr	Slow Scram
Continuous air sampler	1	Stated on sampler	Air sampling
(c) For periods of one week or for the duration of a reactor run, a radiation monitor may be replaced by a gamma sensitive instrument which has its own alarm and is observable by the reactor operator.			

<u>Bases</u> - The neutron flux level scrams provide redundant automatic protective action to prevent exceeding the safety limit on reactor power, and the period scram conservatively limits the rate of rise of the reactor power to periods which are manually controllable without reaching excessive power levels or fuel temperatures.

The rod withdrawal interlock on the Log Count Rate Channel assures that the operator has a measuring channel operating and indicating neutron flux levels during the approach to criticality.

The manual scram button and the "reactor on" key switch provide two methods for the reactor operator to manually shut down the reactor if an unsafe or abnormal condition should occur and the automatic reactor protection does not function.

The use of the area radiation monitors (Table II) will assure that areas of the Purdue University Reactor (PUR-1) facility in which a potential high radiation area exists are monitored. These fixed monitors initiate a scram whenever the preset alarm point is exceeded to avoid high radiation conditions.

Specifications 3.2.b and 3.2.c assure that the safety system response will be consistent with the assumptions used in evaluating the reactor's capability to withstand the maximum credible accident.

In specification 3.2.c. the rod lower limit switches are positioned to measure, as close as possible, the fully inserted position.

3.3 Primary Coolant Conditions

Applicability - This specification applies to the limiting conditions for reactor operation for the primary coolant.

Objective - The objective is to assure a compatible environment, adequate shielding, and a continuous coolant path for the reactor core.

Specification -

a. The primary coolant pH shall be maintained at an average over one year of 5.5 ± 1.0.

b. The primary coolant resistivity shall be maintained at a value greater than 330,000 ohm-cm.

c. The primary coolant shall be maintained at least 13 feet above the core.

Bases - Experience at the PUR-1 and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in specification 3.3.a and 3.3.b will minimize the amount and severity of corrosion of the aluminum components of the primary coolant system and the fuel element cladding.

The height of water in specification 3.3.c is enough to furnish adequate shielding as well as to guarantee a continuous coolant path.

3.4 Confinement

Applicability - This specification applies to the integrity of the reactor room.

Objective - The objective is to limit and control the release of airborne radioactive material from the reactor room.

Specification -

a. During reactor operation the following conditions will be met:

   1. The reactor room will be maintained at a negative pressure of at least 0.05 inches of water with the operation of the room exhaust fan.

   2. All exterior doors in the reactor room shall remain closed except as required for personnel, equipment, or materials access.

b. All inlet and exhaust air ducts and the sewer vent shall contain a HEPA filter or its equivalent.

c. Dampers in the ventilation system inlet and outlet ducts are capable of being closed.

d. The air conditioner can be shut off by the reactor operator.

Bases - The PUR- 1 does not rely on a containment building to reduce the levels of airborne radioactive material released to the environment in the event of the design basis accident. However, in the event of such an accident, a significant fraction of the airborne material will be confined within the reactor room, and the specifications stated above will further reduce the release to the environment.

3.5 Limitations on Experiments

Applicability- This specification applies to experiments installed in the reactor and its experimental facilities.

Objective - The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure, and to assure the safe operation of the reactor.

Specification - The reactor will not be operated unless the following conditions are met:

a. All experiments shall be constructed of material which will be corrosion resistant for the duration of their residence in the pool.

b. All experiments and experimental procedures must received approval by the Committee on Reactor Operations.

c. Known explosive materials shall not be placed in the reactor pool.

d. Cooling shall be provided to prevent the surface temperature of an experiment from exceeding 100°C.

e. No experiment shall be placed in the reactor or pool that interferes with the safe operation of the reactor.

f. 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.

g. 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.

<u>Bases</u> - Specification 3.5.a through 3.5.e are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and serve as a guide for the review and approval of new experiments by the facility personnel and the Committee on Reactor Operations.

Specification 3.5.f and 3.5.g conform to the criteria set forth in Regulatory Guide 2.2 issued in November, 1973.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactivity Limits

Applicability - This specification applies to the surveillance requirements for reactivity limits.

Objective - The objective is to assure that the reactivity limits of Specification 3.1 are not exceeded.

Specification -

a. The shim-safety rod reactivity worths shall be measured and the shutdown margin calculated biennially with no interval to exceed 2½ years, which may be deferred with CORO approval during any reactor shutdown, and whenever a core configuration is loaded for which shim-safety rod worths have not been measured. In the case of a deferred measurement, the measurement must be performed prior to resuming reactor operations.

b. The shim-safety rods shall be visually inspected biennially with no interval to exceed 2½ years, which may be deferred with CORO approval during any reactor shutdown. If the rod is found to be deteriorated, it shall be replaced with a rod of approximately equivalent or greater worth, meeting the limiting conditions of operation specified in 3.1. In the case of a deferred measurement, the measurement must be performed prior to resuming reactor operations.

c. The reactivity worth of experiments placed in the PUR-1 shall be measured during the first startup subsequent to the experiment's insertion and shall be verified if core configuration changes cause increases in experiment reactivity worth which may cause the experiment worth to exceed the values specified in Specification 3.1

Bases - Specification 4.1.a will assure that shim-safety rod reactivity worths are not degraded or changed by core manipulations which cause these rods to operate in regions where their effectiveness is reduced.

The boron stainless steel shim-safety rods have been in use at the PUR-1 since 1962, and over this period of time, no cracks or other evidence of deterioration have been observed. Based on this performance and the experience of other facilities using similar shim-safety rods, the specified inspection times are considered adequate to assure that the control rods will not fail.

4.2 Reactor Safety System

Applicability - This specification applies to the surveillance of the reactor safety system.

Objective - The objective is to assure that the reactor safety system is operable as required by Specification 3.2

Specification -

a. A channel test of each of the reactor safety system channels listed in Table III shall be performed prior to each reactor startup following a shutdown in excess of 8 hours or if they have been repaired or de-energized.

TABLE III.
SAFETY SYSTEM CHANNELS TESTED AFTER PROLONGED SHUTDOWN

Log Count Rate (startup channel)

Log N-Period

Linear Level

Safety Channel

b. A channel check of each of the reactor safety system measuring channels in use or on scale shall be performed approximately every four hours when the reactor is in operation.

c. A channel calibration of the reactor safety channels shall be performed at the following average intervals:

1. An electronic calibration will be performed annually, with no interval to exceed 15 months. This may be deferred with CORO approval during periods of reactor shutdown, but must be performed prior to startup.

2. A power calibration by foil activation will be performed annually, with no interval to exceed 15 months. This may be deferred with CORO approval during periods of reactor shutdown, but must be performed prior to startup.

d. The operation of the radiation monitoring equipment shall be verified daily during periods when the reactor is in operation. Calibration of these monitors shall be performed annually, with no interval to exceed 15 months. This may be deferred with CORO approval during periods of reactor shutdown, but must be performed prior to startup.

e. Shim-safety rod drop times will be measured annually, with no interval to exceed 15 months. These drop times shall also be measured prior to operation following maintenance which could affect the drop time or cause movement of the shim-safety rod control assembly. This may be deferred with CORO approval during periods of reactor shutdown, but must be performed prior to startup.

Bases - A test of the safety system channels prior to each startup will assure their operability, and annual calibration will detect any long-term drift that is not detected by normal intercomparison of channels. The channel check of the neutron flux level channel will assure that changes in core-to-detector geometry or operating conditions will not cause undetected changes in the response of the measuring channels.

Area monitors will give a clear indication when they are not operating correctly. In addition, the operator routinely records the readings of these monitors and will be aware of any reading which indicates loss of function.

The area monitoring system employed at the PUR-1 has exhibited very good stability over its lifetime, and annual calibration is considered adequate to correct long-term drift.

The measured drop times of the shim-safety rods have been consistent since the PUR-1 was built. An annual check of this parameter is considered adequate to detect operation with materially changed drop times. Binding or rubbing caused by rod misalignment could result from maintenance; therefore, drop times will be checked after such maintenance.

4.3 Primary Coolant System

Applicability - This specification applies to the average surveillance schedules of the primary coolant system.

Objective - The objective is to assure high quality pool water, adequate shielding, and to detect the release of fission products from fuel elements.

Specification -

a. The pH of the primary coolant shall be recorded monthly, not to exceed six weeks. This cannot be deferred during reactor shutdown.

b. The conductivity of the primary coolant shall be recorded monthly, not to exceed six weeks. This cannot be deferred during reactor shutdown.

c. The reactor pool water will be at a height of 13 feet over the top of the core whenever the reactor is operated. The reactor pool water height shall be visually inspected weekly, not to exceed ten days, and water will be added as necessary to reach the specification.

d. The primary coolant shall be sampled monthly, not to exceed six weeks, and analyzed for gross alpha and beta activity. This cannot be deferred during reactor shutdown.

Bases - Monthly surveillance of pool water quality provides assurance that pH and conductivity changes will be detected before significant corrosive damage could occur.

When the reactor pool water is at a height of 13 feet above the core, adequate shielding during operations is assured. Experience has shown that approximately 35-40 gallons of water will evaporate weekly and weekly water make-up is sufficient to maintain the reactor pool water height.

Analysis of the reactor water for gross alpha and beta activity assures against undetected leaking fuel assemblies.

4.4 Confinement

Applicability - This specification applies to the surveillance requirements for maintaining the integrity of the reactor room and fuel clad.

Objective - The objective is to assure that the integrity of the reactor room and the fuel clad is maintained, by specifying average surveillance intervals.

Specification -

a. The negative pressure of the reactor room will be recorded weekly.

b. Operation of the inlet and outlet dampers shall be checked semiannually, with no interval to exceed 7 1/2 months.

c. Operation of the air conditioner shall be checked semiannually, with no interval to exceed 7 1/2 months.

d. Representative fuel assemblies shall be inspected annually, with no interval to exceed 15 months.

Bases - Specification a, b, and c check the integrity of the reactor room, and d the integrity of the fuel clad. Based upon past experience these intervals have been shown to be adequate for ensuring the operation of the systems affecting the integrity of the reactor room and fuel clad.

4.5 Experiments

Applicability - This specification applies to the surveillance of limitations on experiments.

Objective – To assure compliance with the provision of the utilization license, the Technical Specifications, and 10 CFR Parts 20 and 50.

Specification – No experiments will be performed unless:

a. It is a tried experiment.

b. The experiment has been properly reviewed and approved according to Section 6 of the technical specifications.

Bases - The basis for this specification is to ensure the safety of the reactor and associated components, personnel, and the public by verification of proper review and approval of experiments as specified in Section 6 of these technical specifications.

5. DESIGN FEATURES

5.1 Site Description

5.1.1 The reactor is located on the ground floor of the Duncan Annex of the Electrical Engineering Building, Purdue University, West Lafayette, Indiana.

5.1.2 The School of Nuclear Engineering controls approximately 5000 square feet of the Duncan Annex ground floor, which includes the reactor room. Access to the Nuclear Engineering controlled area is restricted except when classes are held there.

5.1.3 The licensed areas include the reactor room, and a fuel storage room. Both of these areas are restricted to authorized personnel, or those escorted by authorized personnel.

5.1.4 The reactor room remains locked at all times except for the entry or exit of authorized personnel or those escorted by authorized personnel, equipment, or materials.

5.1.5 The PUR-1 reactor room is a closed room designed to restrict leakage.

5.1.6 The minimum free volume of the reactor room is approximately 15,000 cubic feet.

5.1.7 The ventilation system is designed to exhaust air or other gases from the reactor room through an exhaust vent at a minimum of 50 feet above the ground.

5.1.8 Openings into the reactor room consist of the following:

a. Three personnel doors

b. One door to a storage room with no outside access.

c. Air intake

d. Air exhaust

e. Sewer vent

5.2 Reactor Coolant System

5.2.1 Primary Cooling System – The PUR-1 primary cooling system is a pool containing approximately 6,400 gallons of water.

5.2.2 Process Water System – The process water system is assembled in one unit and contains a pump, filter, demineralizer, valves, flow meters, and a heat exchanger (see 5.2.4). The demineralizer contains a removable *cartridge that is monitored continuously for radioactivity buildup.* This system limits, by the use of filters and ion-exchange resin, the aluminum

corrosion rate, corrosion product buildup, and neutron activation of impurities in the coolant.

5.2.3 Primary Coolant Makeup Water System – Makeup water for the pool is taken batchwise from the Purdue University water line and is passed through the demineralizer enroute to the pool. A vacuum breaker excludes any possibility of siphoning pool water into the supply line. The pool makeup water system, in addition to the demineralizer, also includes a normally closed manual shutoff and throttle valve and a check valve.

5.2.4 Primary Coolant Chiller System – The chiller is designed with three loops. Pool water passes through the primary loop, a Freon refrigerant is in the secondary loop, and water from the building water supply is used to remove heat, which is then discharged to the building sewer system. The heat-removal capacity of the heat exchanger is 10.5 kW. It was designed to maintain the reactor pool temperature at 75°F during continuous operation at 10 kW.

5.3 Reactor Core and Fuel

5.3.1 The fuel assemblies shall be MTR type consisting of aluminum clad plates enriched up to 20% in the U-235 isotope.

5.3.2 A standard fuel assembly shall consist of up to 14 fuel plates containing a maximum of 180 grams of U-235.

5.3.3 A control fuel assembly shall consist of up to 8 fuel plates containing a maximum of 103 grams of U-235.

5.3.4 Partially loaded fuel assemblies in which some of the fuel plates are replaced by aluminum plates containing no uranium may be used.

5.3.5 The core configuration shall consist of 13 standard fuel assemblies as described in 5.3.2, and 3 control fuel assemblies as described in 5.3.3, and two shim-safety rods and one regulating rod.

5.3.6 Representative fuel assemblies shall be inspected annually, with no interval to exceed 15 months.

5.4 Fuel Storage

5.4.1 All reactor fuel assemblies shall be stored in a geometric array where k_{eff} is less than 0.8 for all conditions of moderation and reflection.

5.4.2 Irradiated fuel assemblies and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel integrity is maintained per the Safety Analysis Report.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The reactor facility shall be an integral part of the School of Nuclear Engineering of the Schools of Engineering at Purdue University as shown in Figure 6.1 and listed below

a. The Dean of the College of Engineering (Level 1) will be the individual responsible for the facility's licenses or charter.

b. The Laboratory Director (Level 2) or the designated alternate shall be responsible for reactor facility operation.

c. The Reactor Supervisor (Level 3) shall be responsible for the day-to-day safe operation of the PUR-1. The Reactor Supervisor shall 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.

d. 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 Level 1 Licensee, the Dean of the College of Engineering. In all matters pertaining to radiation safety they shall be responsible to the Radiation Safety Committee.

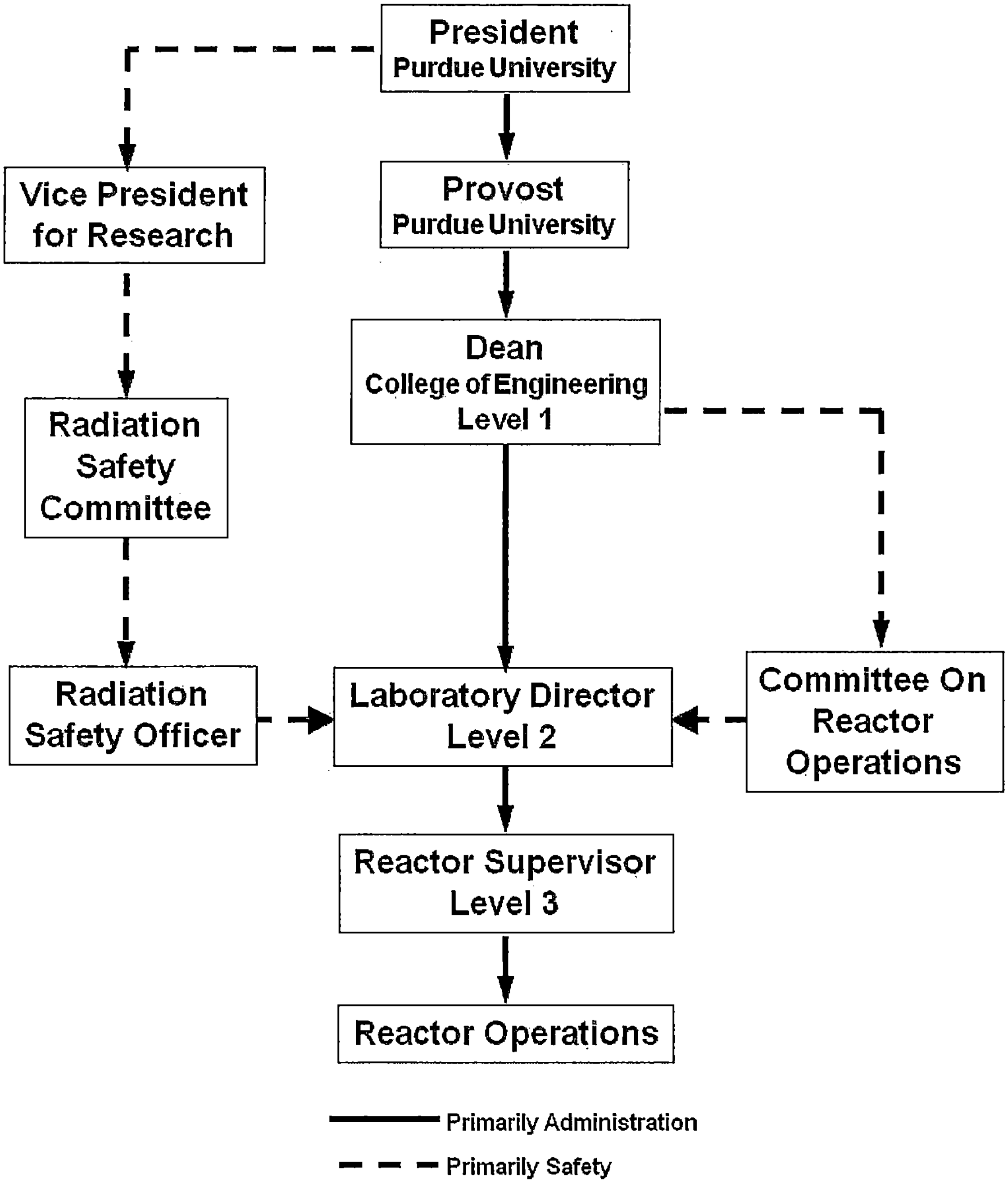

FIGURE 6.1: PUR-1 Organization

6.1.2 Staffing

(1) The minimum staffing when the reactor is not secured shall be

(a) A licensed reactor operator in the reactor room,

(b) The minimum crew for operating the reactor shall consist of 2 (two) persons, one of whom must be an NRC licensed member of the PUR-1 operations staff, the second crew member must be instructed as to how to shut down the reactor in the event of an emergency.

(c) A designated senior reactor operator (unless the operations staff consists of only one senior reactor operator, and that individual is operating the reactor) shall be present or readily available on call at any time that the reactor is operating. "Readily Available on Call means an individual who

(i) Has been specifically designates and the designation is known to the operator on duty,

(ii) Can be rapidly contacted by phone or other method by the operator on duty,

(iii) Is capable of getting to the reactor facility within a reasonable time under normal conditions

(2) No licensed reactor operator or senior reactor operator shall be required within the licensed facility if the reactor is secure.

(3) Events requiring the presence at the facility of a senior reactor operator are

(a) Initial startup and approach to power following a core change. The presence of an SRO at the reactor facility is unnecessary for the initial daily start up, provided the core remains unchanged from the previous run;

(b) All fuel or control-rod relocations within the core region;

(c) Recovery from an unplanned or unscheduled shutdown.

6.1.3 Minimum Qualifications of Reactor Personnel The minimum qualifications should be consistent with the American National Standard for the Selection and Training of Personnel for Research Reactors, ANSI/ANS 15.4, and include the following:

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

b. 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.

c. 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)

d. 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.

e. 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.

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

6.1.4 A Radiation Safety Officer who is organizationally independent of the PUR-1 operations group shall advise the Laboratory Director and/or Reactor Supervisor in matters concerning radiological safety. Minimum qualifications for the Radiation Safety Officer (RSO) is a bachelor's degree or the equivalent in a science or engineering subject, including some formal training in radiation protection. The RSO should have at least five years of professional experience in applied radiation protection. A master's degree may be considered equivalent to one year of professional experience, and a doctor's degree equivalent to two years of professional experience where course work related to radiation protection is involved. At least three years of this professional experience should be in applied radiation protection work in a nuclear facility dealing with radiological problems.

6.1.5 The Reactor Supervisor or his designated alternate shall be responsible for the facility retraining and replacement training program.

6.2 Review and Audit

6.2.1 A Committee on Reactor Operations (CORO) shall report to the LEVEL 1 Licensee on matters of administration and safety. CORO will advise the Laboratory Director and/or the Reactor Supervisor on those areas of responsibility specified in Sections 6.2.5 and 6.2.6. The minimum qualifications for persons on the CORO shall be five years of professional work experience in the discipline or specific field they represent. A baccalaureate degree may fulfill four years of experience.

6.2.2 The CORO shall have at least 7 (seven) members of whom no more than a minority shall be directly concerned with the administration or direct use of the reactor. These members shall include the following:

a. The Chairman, a responsible, senior technical person, knowledgeable in the field of reactor technology, who does not have line responsibility for day-to-day operation of the reactor.

b. A senior radiation safety officer.

c. A senior member of the Purdue University safety and security organization.

d. The Laboratory Director

e. The Reactor Supervisor.

f. Two senior scientific staff members.

6.2.3 The CORO shall meet no less than once per calendar year, or more frequently as circumstances warrant, consistent with effective monitoring of facility activities. A sub-committee may be assembled by the CORO as the need arises.

Sub committees may be formed as needed, which may consist of a minimum of 3 (three) members, only one of which may have line responsibility for day-to-day operations of the reactor. These sub-committees may perform the functions of the whole committee as necessary provided the review/audit functions are maintained. Actions by the subcommittee must be approved by the whole CORO.

A quorum of CORO shall consist of not less than a majority of the full Committee and shall include the chairman or his designated alternate. No more than one-half of the voting members present shall be members of the reactor operations staff.

6.2.4 The CORO shall review and approve:

a. Safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments that may be conducted without prior NRC approval under the provision of Section 50.59, 10 CFR, to ascertain whether such actions would constitute an unreviewed safety question, or would require a change in Technical Specifications.

b. Proposed changes to procedures, equipment or systems that change the original intent or use, or those that might involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which are significantly different from previous approved tests or experiments, and those that might involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that might affect nuclear safety.

g. Events which have been reported within 24 hours to the NRC.

h. Audit reports.

6.2.5 Audits

Audits of 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 one individual who need not be an identified CORO member. These audits shall examine the operating records and encompass:

a. The conformance of facility operation to the Technical Specifications and applicable license conditions, to be done annually with no interval to exceed 15 months.

b. The performance training and qualifications of the licensed facility staff, to be done annually with no interval to exceed 15 months.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety, to be done annually with no interval to exceed 15 months.

d. The Facility Emergency Plan and implementing procedures, to be done biennially with no interval to exceed 2 1/2 years.

e. The Facility Security Plan and implementing procedures, to be done biennially with no interval to exceed 2 1/2 years.

f. Any other area of facility operation considered appropriate by the CORO or the Reactor Supervisor, to be done annually with no interval to exceed 15 months.

6.2.6 Records

Records of CORO activities shall be prepared and distributed as indicated below.

a. Minutes of each CORO meeting shall be prepared and forwarded to the Reactor Supervisor within 30 days following each meeting.

b. Reports of reviews encompassed by section 6.2.4 e, f, and g above, shall be prepared and forwarded to the Reactor Supervisor within 30 days following completion of the review.

c. Audit reports encompassed by Section 6.2.5 above, shall be forwarded to the CORO Chairman and to the management responsible for the areas audited within 30 days after completion of the audit.

6.3 Operating Procedures

Written procedures, including applicable check lists reviewed and approved by the CORO, shall be in effect and followed for the following operations:

6.3.1 Startup, operation, and shutdown of the reactor.

6.3.2 Installation and removal of fuel elements and control rods.

6.3.3 Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes.

6.3.4 Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

6.3.5 Maintenance procedures which could have an effect on reactor safety.

6.3.6 Experiment installation, operation, and removal.

6.3.7 Implementation of the Security Plan and Emergency Plan.

6.3.8 Calibration and preventive maintenance procedures on required instruments, systems, or components.

Non-routine operations which require the sequential performance of a series of subtasks shall be carried out with the written procedure at the console. To assure adherence to the documentation of the procedure, each step will be entered in the log book by the operator on duty as it is completed.

Substantive changes to the above procedures shall be made only with the approval of the CORO. The Reactor Supervisor or Laboratory Director may make changes to procedures which do not change the intent of the original procedure or impact nuclear safety. All such changes to the procedures shall be documented and subsequently reviewed by CORO.

6.4 Operating Records

6.4.1 The following records and logs shall be prepared and retained for at least five years:

a. Normal facility operation and maintenance.

b. Principal maintenance operations

c. Reportable occurrences.

d. Tests, checks, and measurements documenting compliance with surveillance requirements.

e. Facility radiation and contamination surveys.

f. Records of experiments performed.

g. Fuel inventories, receipts and shipments

h. Approved changes of operating procedures.

i. Records of meeting and audit reports of the Committee on Reactor Operations.

6.4.2 Record of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed.

6.4.3 The following records and logs shall be prepared and retained for the life of the facility:

a. Gaseous and liquid waste released to the environs.

b. Offsite environmental monitoring surveys.

c. Radiation exposures for all PUR-1 personnel.

d. Updated, corrected, and as-built facility drawings.

e. Annual operating reports.

f. Reviews of instances where the safety limit was exceeded.

g. Reviews of failure of the automatic safety system that protects the limiting safety system settings (LSSSs).

h. Reviews of instances where limiting conditions of operation were not met.

6.5 <u>Required Actions</u>

The following actions shall be taken relating to the types of events listed in Secs. 6.6.1 and 6.6.2.

6.5.1 The following actions shall be taken in the event the Safety Limit is violated:

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

(2) 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.

(3) 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.

(4) 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.

6.5.2 The following actions are to be taken in the event of reportable occurrence as defined in 6.6.2.

(1) Reactor conditions shall be returned to normal, or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates;

(2) Occurrence shall be reported to Level 2 or designated alternates and to chartering or licensing authorities as required;

(3) Occurrence shall be reviewed by the review group at its next scheduled meeting.

6.6 Reporting Requirements

The following information shall be submitted to the USNRC in addition to the reports required by Title 10, Code of Federal Regulations.

6.6.1 Annual Operating Reports - a report covering the previous year shall be submitted to the Director of the Office of Nuclear Reactor Regulation with a copy to the NRC Regional Administrator by March 31 of each year. It shall include the following:

a. Changes in plant design and operation

1. changes in facility design

2. performance characteristics (e.g. equipment and fuel performance).

3. changes in operating procedures which relate to the safety of facility operations

4. results of surveillance tests and inspections required by these technical specifications

5. a brief summary of those changes, tests, and experiments which required authorization from the Commission pursuant to 10 CFR 50.59(a)

b. Power Generation - A tabulation of the thermal output of the facility during the reporting period.

c. Shutdowns - A listing of unscheduled shutdowns which have occurred during the reporting period, tabulated according to cause, and a brief

discussion of the corrective and preventive actions taken to prevent recurrence.

d. Maintenance - A discussion of corrective maintenance (excluding preventive maintenance) performed during the reporting period on safety-related systems and components.

e. Changes, Tests, and Experiments - A brief description and a summary of the safety analysis and evaluation for those changes, tests, and experiments which were carried out without prior Commission approval, pursuant to the requirements of 10 CFR Part 50.59(b).

f. Radioactive Effluent Releases - A summary of the nature, amount, and maximum concentrations of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

g. Occupational Personnel Radiation Exposure - A summary of radiation exposures greater than 25% of the appropriate limits of 10 CFR 20 received during the reporting period by facility personnel (faculty, students, or experimenters).

6.6.2 Non-Routine Reports

a. Special Reports

Special reports are used to report unplanned events as well as planned major facility and administrative changes. The following schedule shall be incorporated in the specifications:

(1) There shall be a report not later than the following working day by telephone and confirmed in writing by facsimile or similar conveyance to licensing authorities, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

(a) operation with actual safety system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications,

(b) operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken as permitted in Sec. 3,

(c) a reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.

(d) an unanticipated or uncontrolled change in reactivity greater than 0.6% $\Delta k/k$,

(e) abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary,

(f) an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations;

(2) There shall be a written report within 30 days to the chartering or licensing authorities of the following:

(a) Permanent changes in the facility organization involving Level 1 or 2 personnel

(b) significant changes in the transient or accident analysis as described in the Safety Analysis Report.

b. Unusual Events

A written report shall be forwarded within 30 days to the Director, Office of Nuclear Reactor Regulation with a copy to the in the event of:

1. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the SAR or the bases for the Technical Specifications.

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications, in the SAR.

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.

4. Discovery of an inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unusual condition with regard to reactor operations.

Cost Analysis for Decommissioning of Purdue University Reactor 1 (PUR-1)

The following analysis for decommissioning of the Purdue University Reactor 1 (PUR-1) is based on the analysis done by the Department of Defense (DOD) [1] for the AFRRI TRIGA reactor facility and the University of Utah Safety Analysis Report (SAR) [2]. The cost analysis reflects decommissioning through the decontamination of the reactor site, referred to as DECON. DECON costs include the removal of equipment, structures, and portions of the facility that contain radioactive contaminants. The removal of spent nuclear fuels and demolition of any uncontaminated areas of the site are considered ancillary costs.

The cost of decommissioning is divided into three major categories:

- Waste disposal costs
- Labor costs
- Energy costs

Detailed data is provided for each of the major categories of costs based on the report by DOD [1] and taking into account differences in design. The amounts are adjusted to 2015 dollars using the Bureau of Labor Statistics Consumer Price Index (CPI) taken from reference [3].

Waste Disposal Costs

The amount of structural material that has been exposed to neutron irradiation in the reactor building and the cost for transportation are provided in Table 1. The cost of crates and shipping are obtained from [1] which is developed based on data provided in NUREG/CR-1756 [4]. For the purposes of this report, the conservative scenario of shipment to a destination in Washington State has been considered which cost \$0.12/kg of low level waste in 1989 dollars (equivalent to \$0.23/kg in 2015 dollars). The cost per volume for disposing of radioactive waste in a depository was obtained from [1], which is based on Barnwell rates of \$2,825/m^3 in 1989 dollars (equivalent to \$5,348/m^3 in 2015 dollars). Plywood 3.5 m^3 crates are used for removing the waste which cost \$400 each in 1981 dollars (equivalent to \$1,033 in 2015 dollars).

Table 1 Waste disposal costs in 2015 dollars

Material	Volume	Crates	Density	Mass	Shipping	Total Cost*
	m^3	No.	kg/m^3	kg	USD	USD
Contaminated concrete	10	3	2,400	24,000	\$5,520	\$62,099
Contaminated sand	60	18	1,442	86,490	\$19,893	\$359,367
Contaminated aluminum	5	2	2,700	13,500	\$3,105	\$31,911
Contaminated stainless steel	5	2	8,050	40,250	\$9,258	\$38,064
Total					**\$37,776**	**\$491,441**

Total Cost = (cost/crate)(# of crates) + shipping costs + disposal costs

The volumes are rounded up in order to estimate the costs conservatively. The shipping costs are adjusted to 2015 dollars based on [1] and the highest value (stainless steel) is used for all the materials to be conservative in estimating the cost.

Labor Costs

The labor costs are obtained from [1] which is based on NUREG/CR-1756. The PUR-1 is smaller than the AFRRI TRIGA facility; however, the numbers are unchanged to provide a conservative estimate of the labor costs. The amounts are adjusted based on CPI from 1981 dollars to 2015 dollars.

Table 2 Decommissioning labor costs (for DECON) in 2015 dollars

	Work years	**Rate (2015 dollars)**	**Cost**
Management and support staff			
Decomm superintendent	2	$230,100	$460,200
Decomm engineer	2	$196,200	$392,400
Secretary	2	$62,500	$125,000
Clerk	0.5	$62,500	$31,300
Health physicist	2	$121,100	$242,200
Radioactive shipment specialist	0.5	$101,500	$50,800
Procurement specialist	0.5	$101,500	$50,800
Contract and accounting specialist	0.8	$121,600	$97,300
Security supervisor	0.625	$144,300	$90,200
Security patrol officer	3.6	$65,600	$236,200
QA engineer	0.7	$121,100	$84,800
Control room operator	1	$88,600	$88,600
Consultant	1	$258,200	$258,200
Decomm workers			
Shift engineer	1	$134,800	$134,800
Craftsman	2	$82,900	$165,800
Crew leader	0.5	$114,600	$57,300
Utility operator	0.342	$82,900	$28,400
Laborer	6	$79,800	$478,800
Health physics technician	3	$77,500	$232,500
Total			**$3,305,600**

Energy Costs

The energy costs are also obtained from [1] which is based on NUREG/CR-1756 and the energy cost per kWh is obtained from the U.S. Department of Energy Information Administration Electric Power Monthly report [5]. The average retail price of electricity in the state of Indiana for all sectors for December 2014 (YTD) was 8.97 cents per kWh.

Table 3 Energy costs

Equipment	Energy use (kWh)	Cost ($)
General system	9,000	$807
HV AV	20,000	$1,794
Lighting	23,000	$2,063
Control room	5,200	$466
Fire protection	600	$54
Security	5,600	$502
Communications	900	$81
Domestic water	36,300	$3,256
Reactor water	23,400	$2,099
Compressed air	15,000	$1,346
Building heating	302,600	$27,143
Decommissioning equipment	20,000	$1,794
Total (USD)		**$41,406**

Total Decommissioning Cost and Inflation Adjustment Methodology

The total cost for the reactor decommissioning based on the costs detailed above is provided in Table 4. The cost of spent fuel removal, shipment, and site demolition costs are also estimated in table 4.

Table 4 Total cost of decommissioning PUR-1 in 2015 dollars, DECON methodology

Category	Cost
DECON	
Waste disposal	$491,441
Labor	$3,305,600
Energy	$41,406
Contingency fund (25% of decommissioning costs)	$959,611
Ancillary	
Spent fuel removal and shipment	$387,330
Site demolition	$645,550
Total Cost	**$5,830,938**

The estimated cost of decommissioning the Purdue University Reactor 1 (PUR-1) is reflected in 2015 dollars using CPI as the basis for adjustment to 2015 dollar values. A contingency fund equal to 25% of decommissioning costs is added to the total cost as required by NUREG-1756. Ancillary costs were also obtained from [1].

References

[1] M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145

[2] The University of Utah Reactor (UUTR) Safety Analysis Report, 2005

[3] United States Department of Labor, Bureau of Labor Statistics CPI Inflation Calculator [http://www.bls.gov/data/inflation_calculator.htm], April 2015

[4] U.S. Nuclear Regulatory Commission, NUREG/CR-1756 "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors", 1983

[5] U.S. Department of Energy Information Administration Electric Power Monthly, February 2015 Report with data for December 2014 [http://www.eia.gov/electricity/monthly/current_year/february2015.pdf]

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 will consist 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, university radiation control officers, or faculty members of the School of Nuclear Engineering.

B. PROGRAM EVALUATION

Completion of the biennial operator requalification program will consist of a written examination and an annual 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.

c. The representative questions for the operators 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.

d. 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.

e. 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 annual 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 excessive levels of radiation and to guard against personal exposure.

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 act as a member of the operations crew so that all procedures, the limits to the license and its amendments are not violated.

d. Any person who cannot 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 per 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 senor 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 training coordinator until the licenses of the participants are renewed.

In any of the above requalification, exclusive of operations, mail, electronic classroom or other methods may be used for training, meetings, testing or other required communication/s.

SAFETY ANALYSIS REPORT

for the

PURDUE UNIVERSITY

PUR-1 REACTOR

LICENSE NUMBER R-87

DOCKET NUMBER 50-182

Prepared by:

J. H. Jenkins, E. C. Merritt, June 30, 2008

Additional Revisions by:

C. Townsend, R. Bean July 23, 2015

West Lafayette, IN 47907

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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 12 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 original design power level for PUR-1 was 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 12 kW. 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 design, pool type reactor, previously licensed for operation in 1962, 1968 and 1988 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 tool, which is stored securely when not in use. Security of the fuel handling tool is under administrative control of the licensed senior operators.

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 typically 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

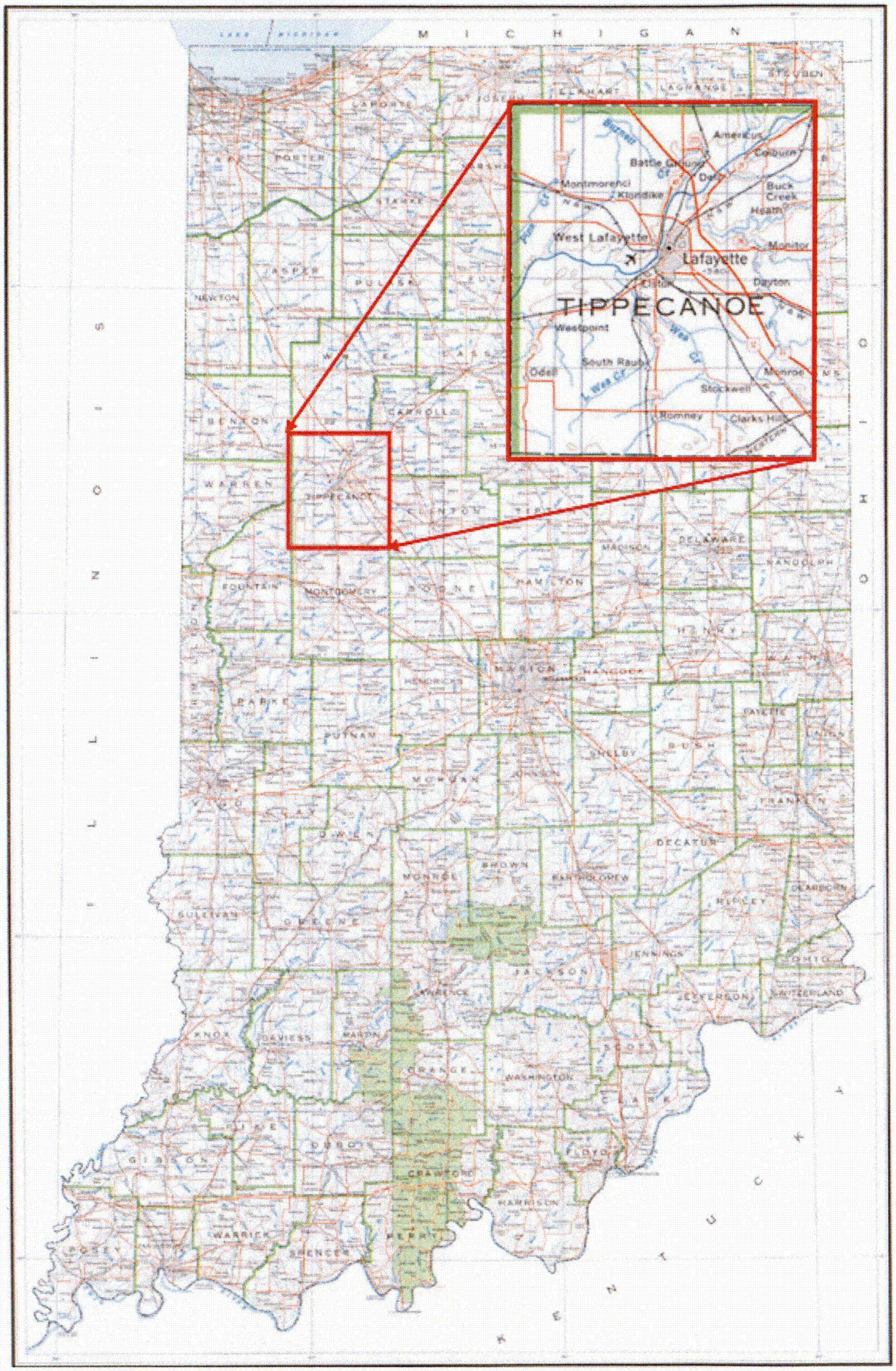


Figure 2-1: State of Indiana, showing location of Tippecanoe County.[2]

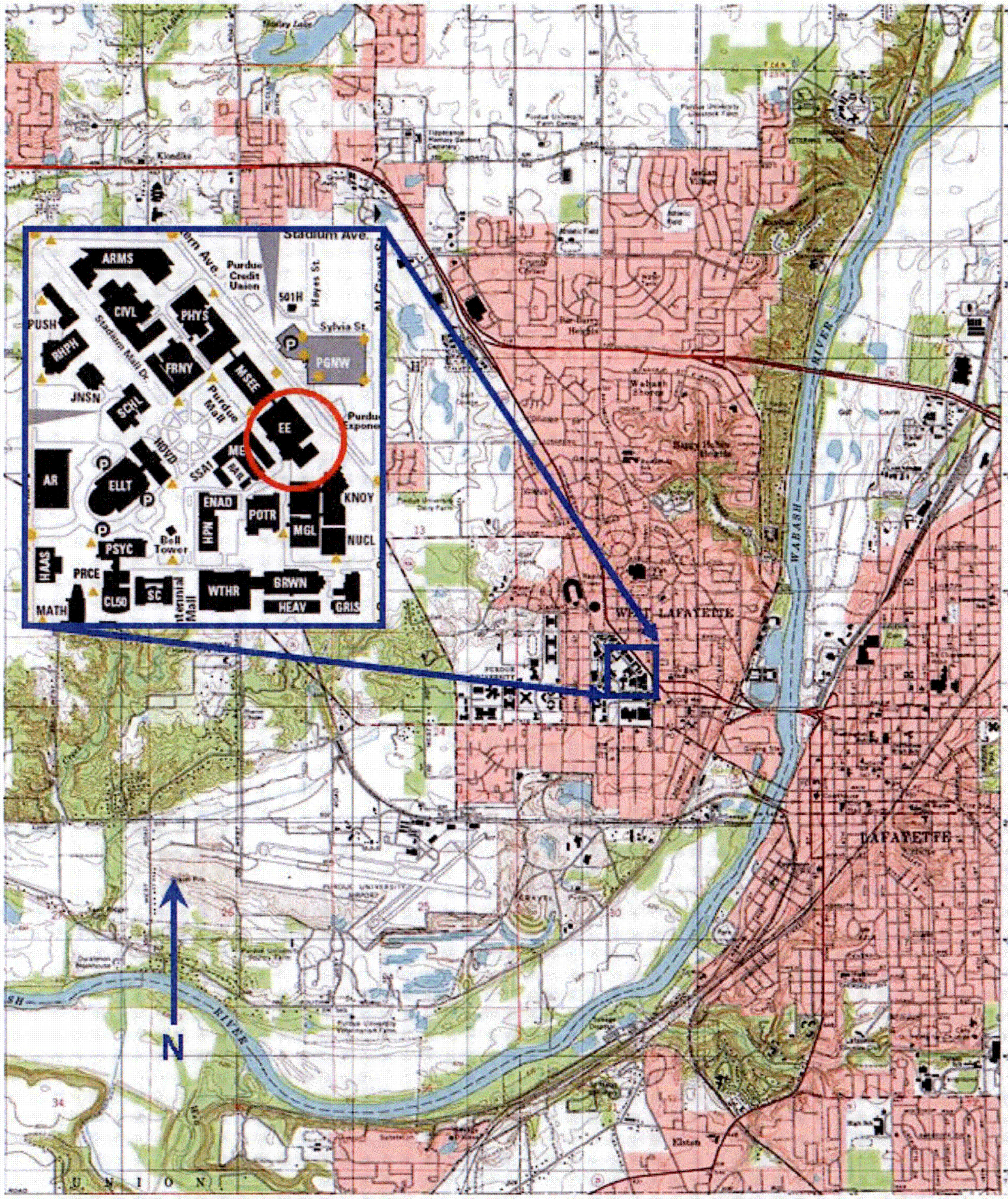


Figure 2-2: Map of West Lafayette, Indiana, showing inset picture of the location of the Electrical Engineering Building on the Purdue Campus.[2]

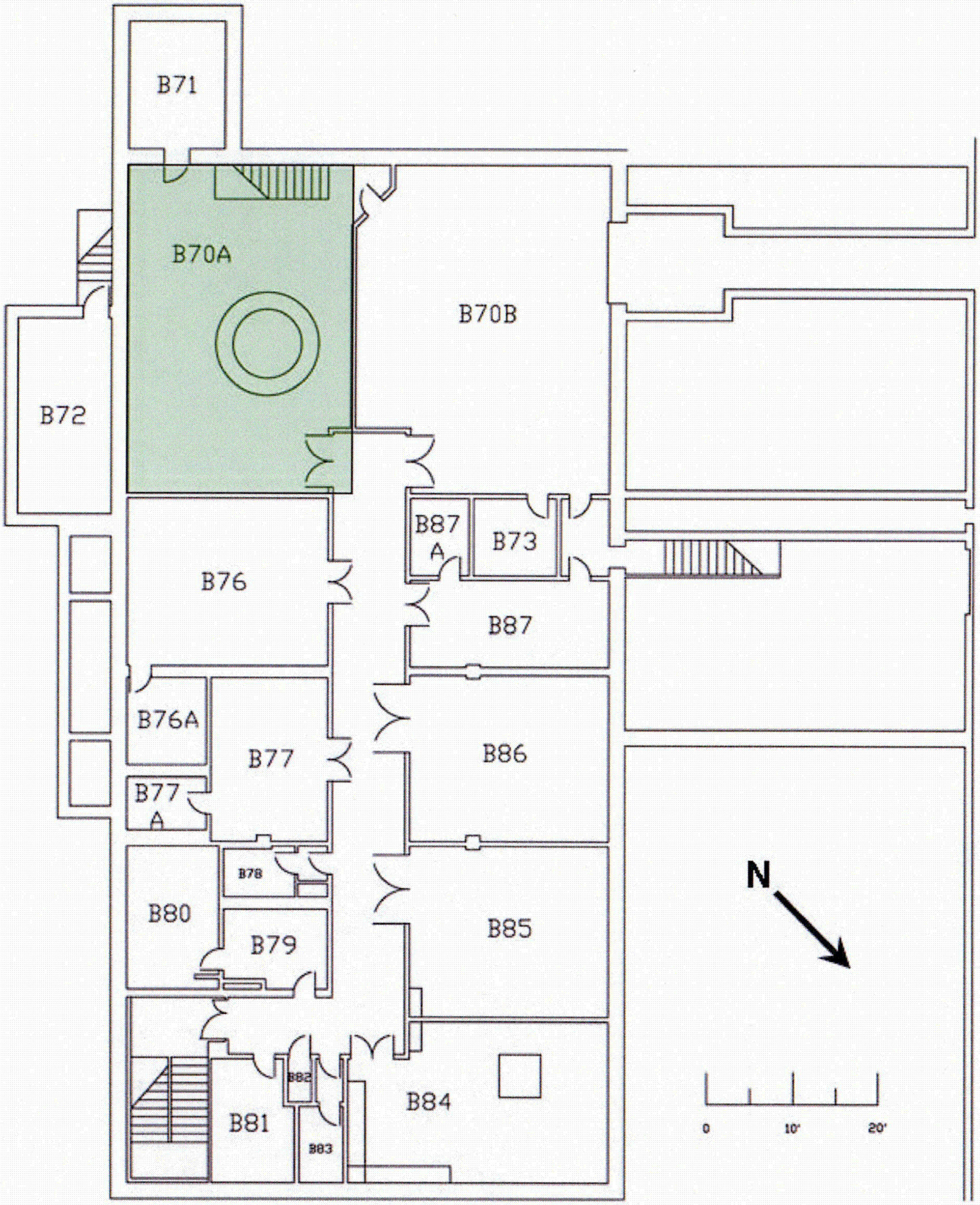


Figure 2-3: Map of the Duncan Annex of the Electrical Engineering Building

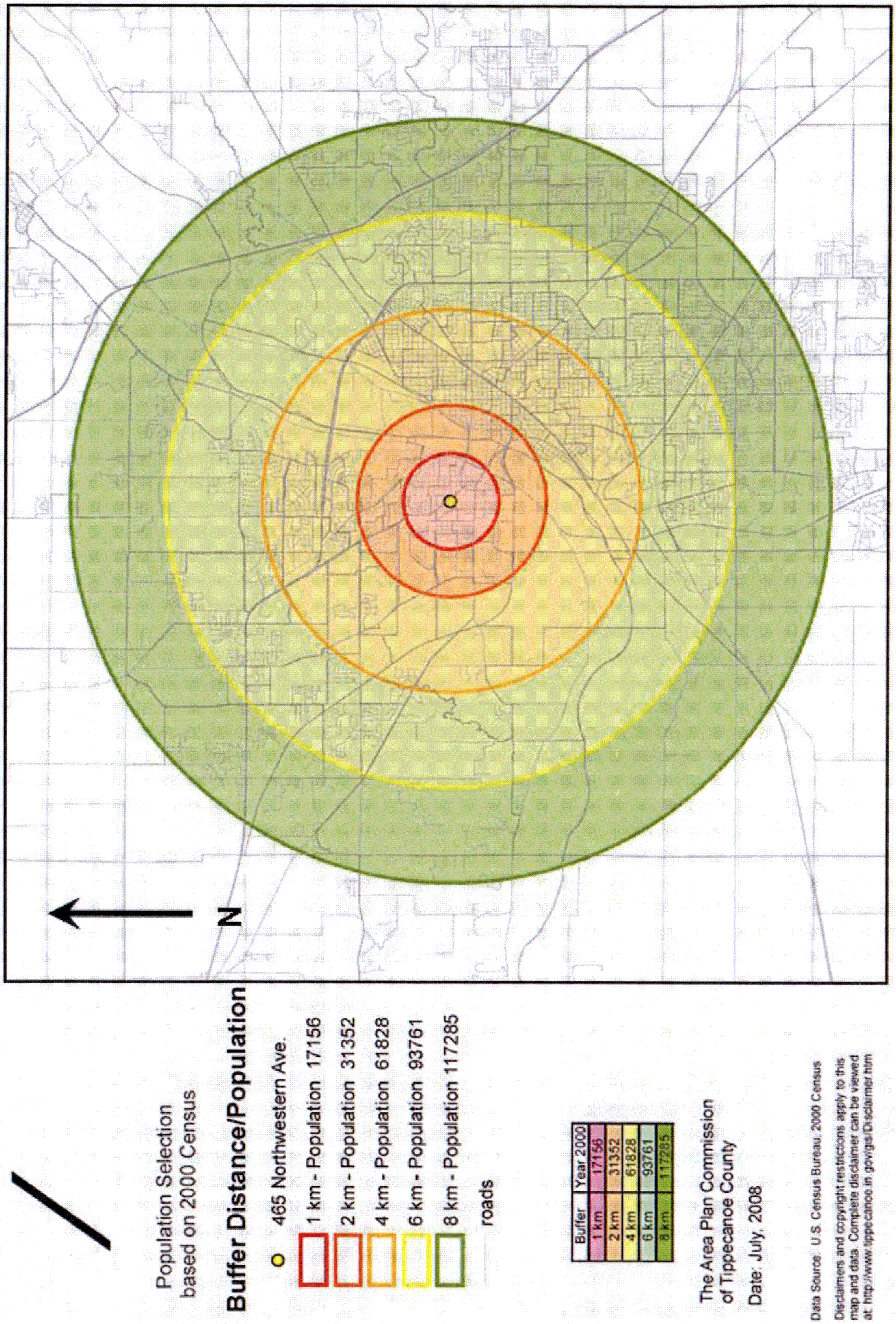
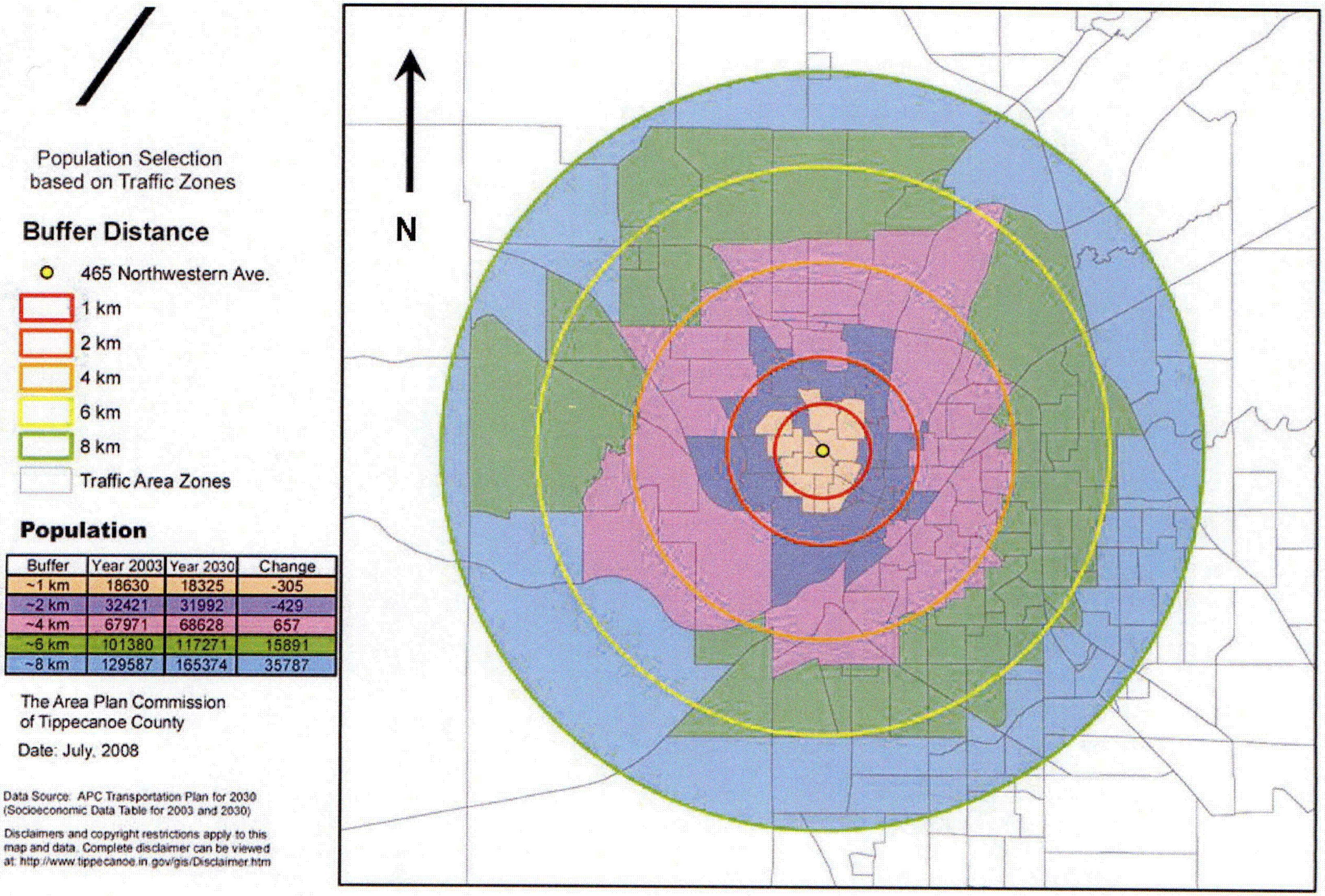


Figure 2-4: Population of area surrounding PUR-1 reactor.[3]



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

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

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.

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 the reactor below ground level, 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 13 feet below the water surface (and 9 feet below floor level) 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.

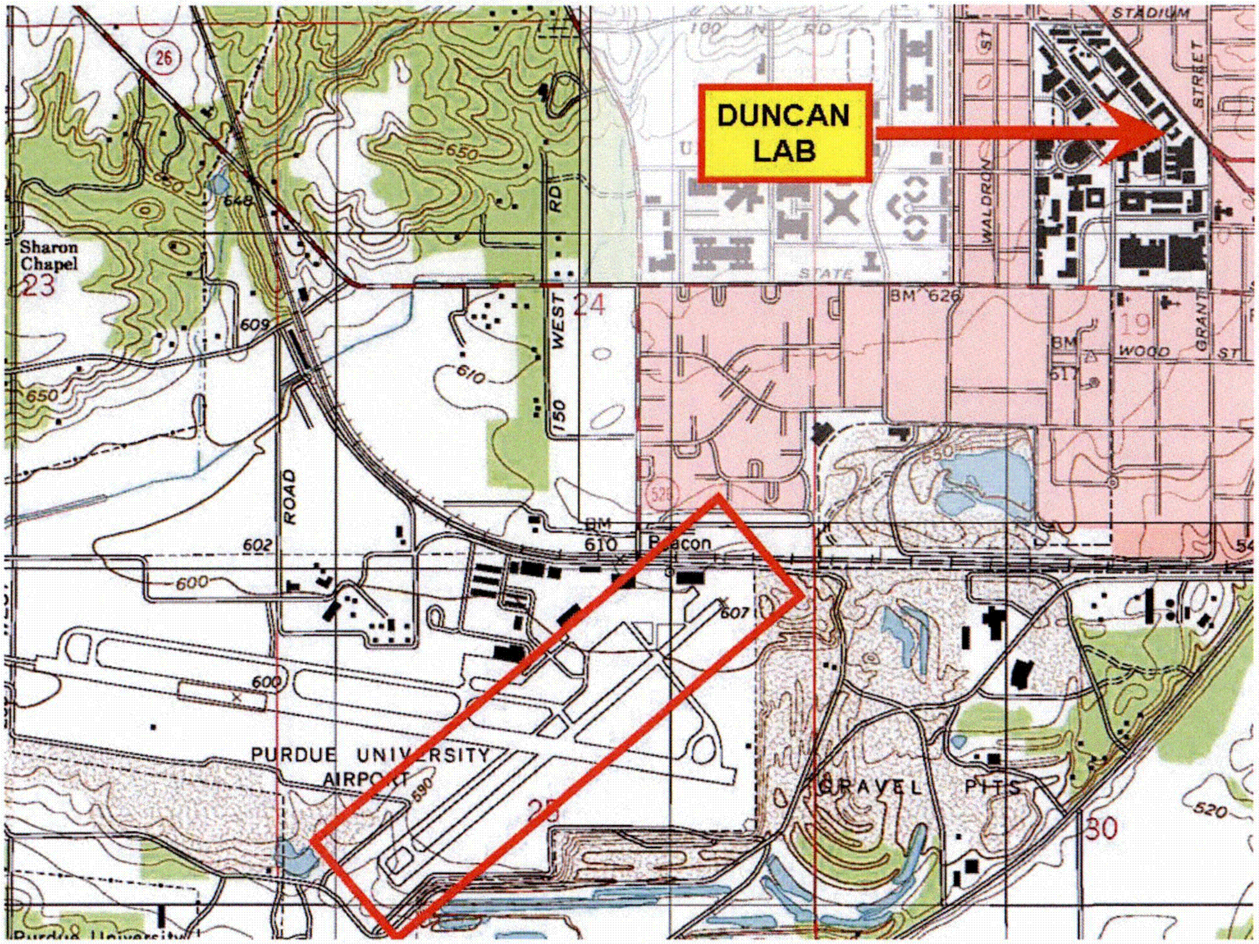


Figure 2-6: Map showing the location of the Purdue University airport in relation to the location of PUR-1.[2]

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

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
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,

* 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.

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 Indiana 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. 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 Wabash Valley and on the upland in the western part of the county. 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. 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

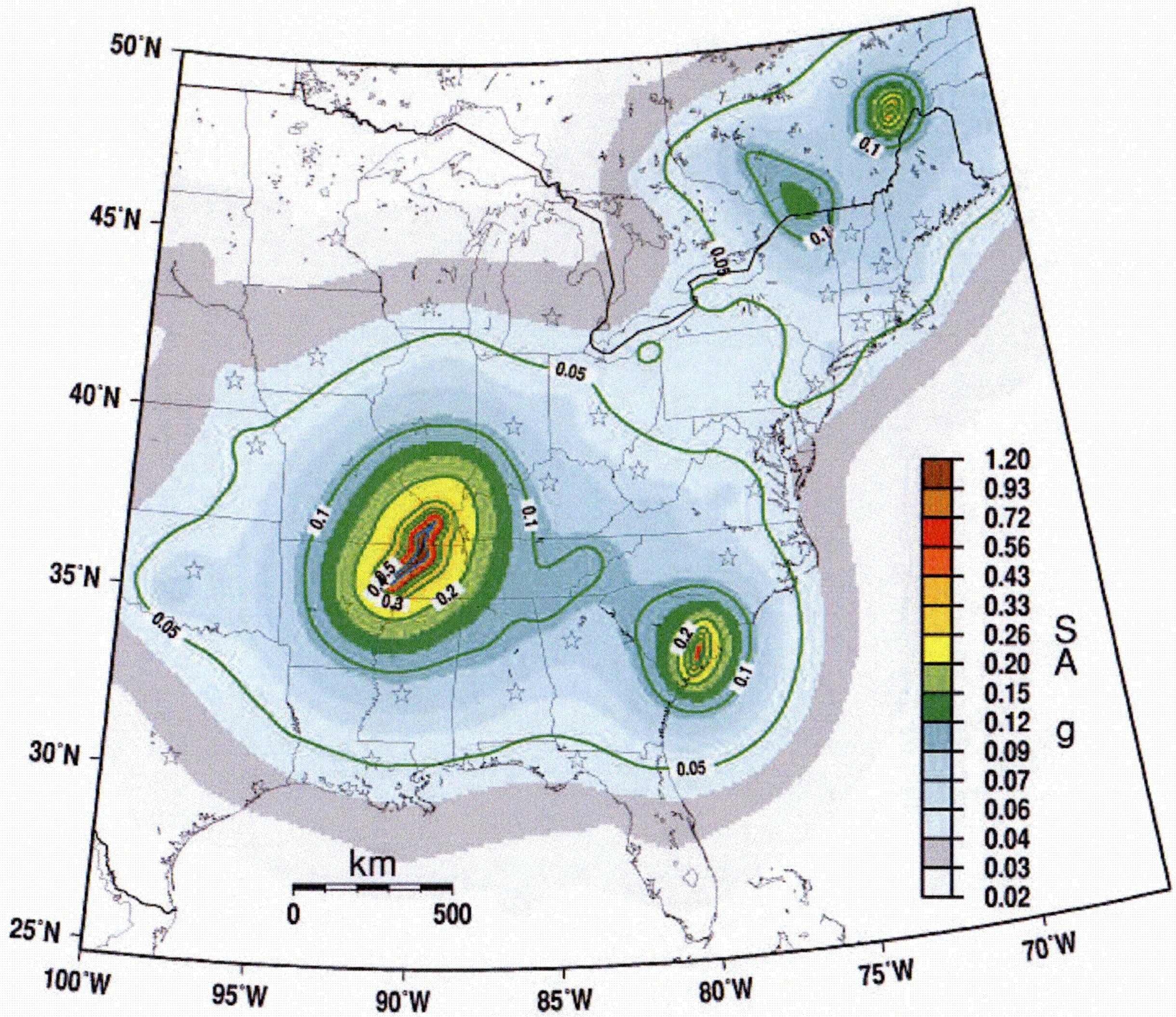


Figure 2-7: Map of the 1-Hz spectral acceleration for 2% probability of exceedance in 50 years for the Central and Eastern United States in standard gravity (g).[13]

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

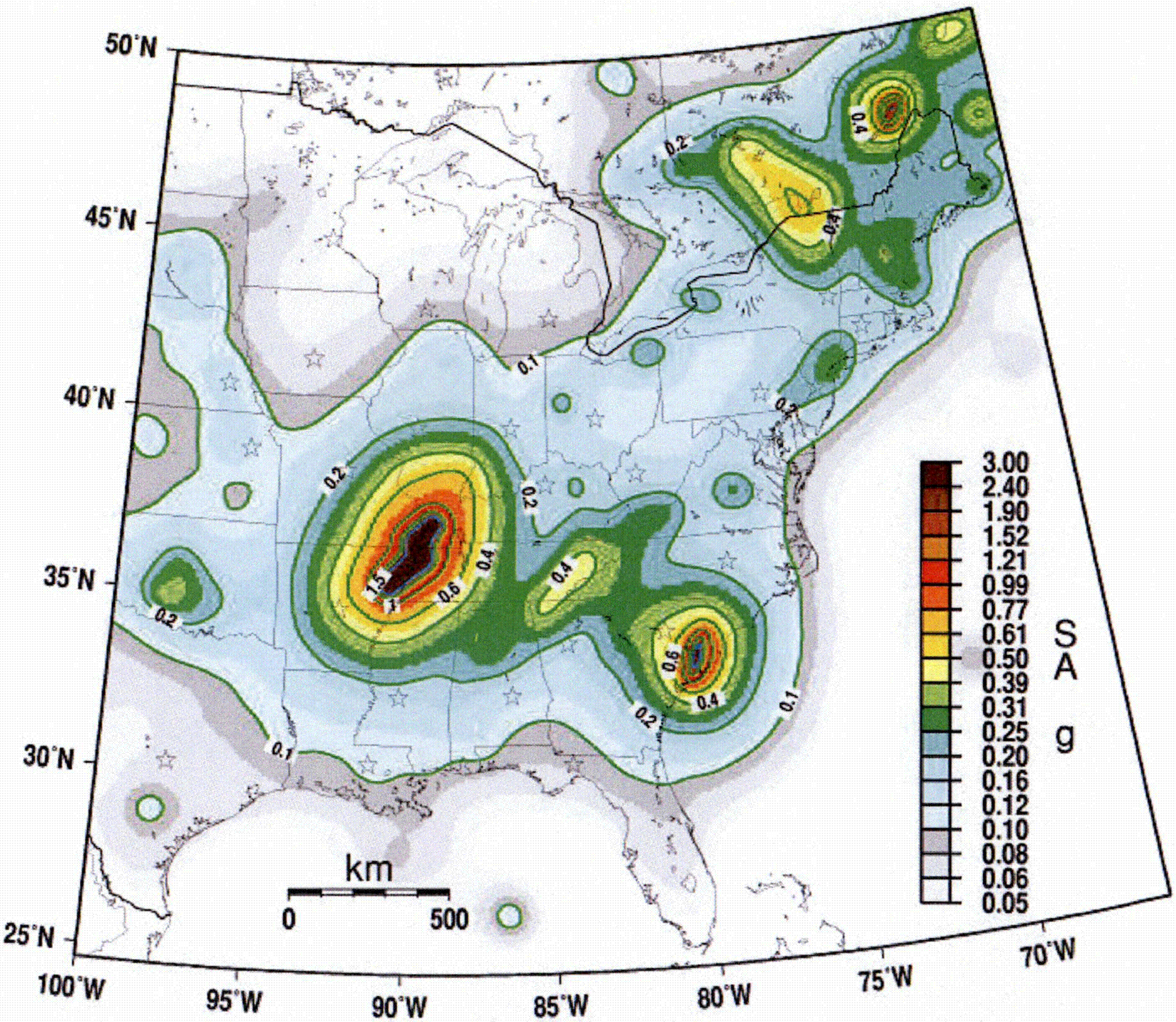


Figure 2-8: Map of the 5-hertz spectral acceleration (SA) for 2% probability of exceedance in 50 years in the Central and Eastern United States in standard gravity (g).

PGA with 2%/50 yr PE, 2008

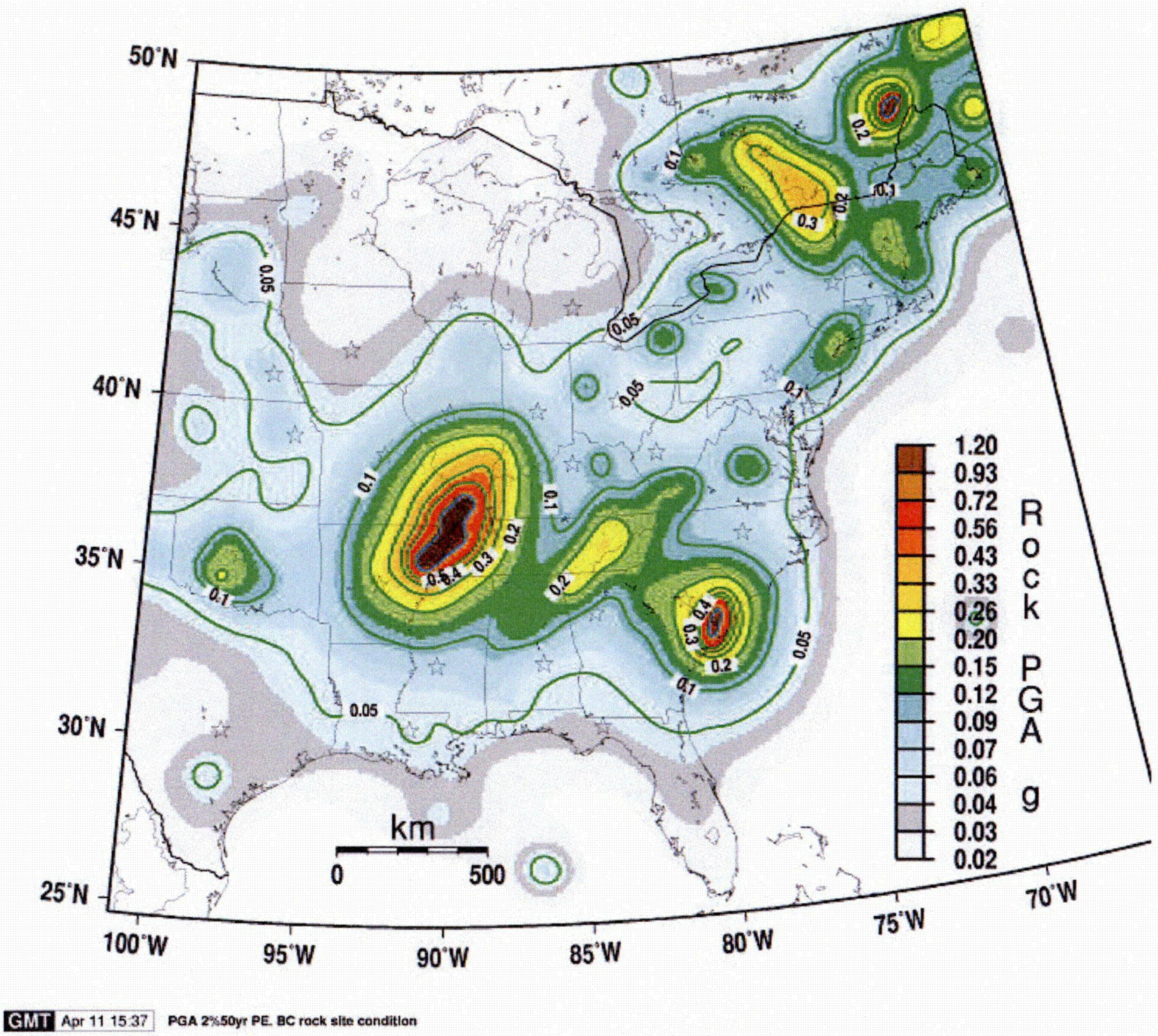


Figure 2-9: Map of peak ground acceleration (PGA) for 2% probability of exceedance in 50 years in the Central and Eastern united States in standard gravity (g).

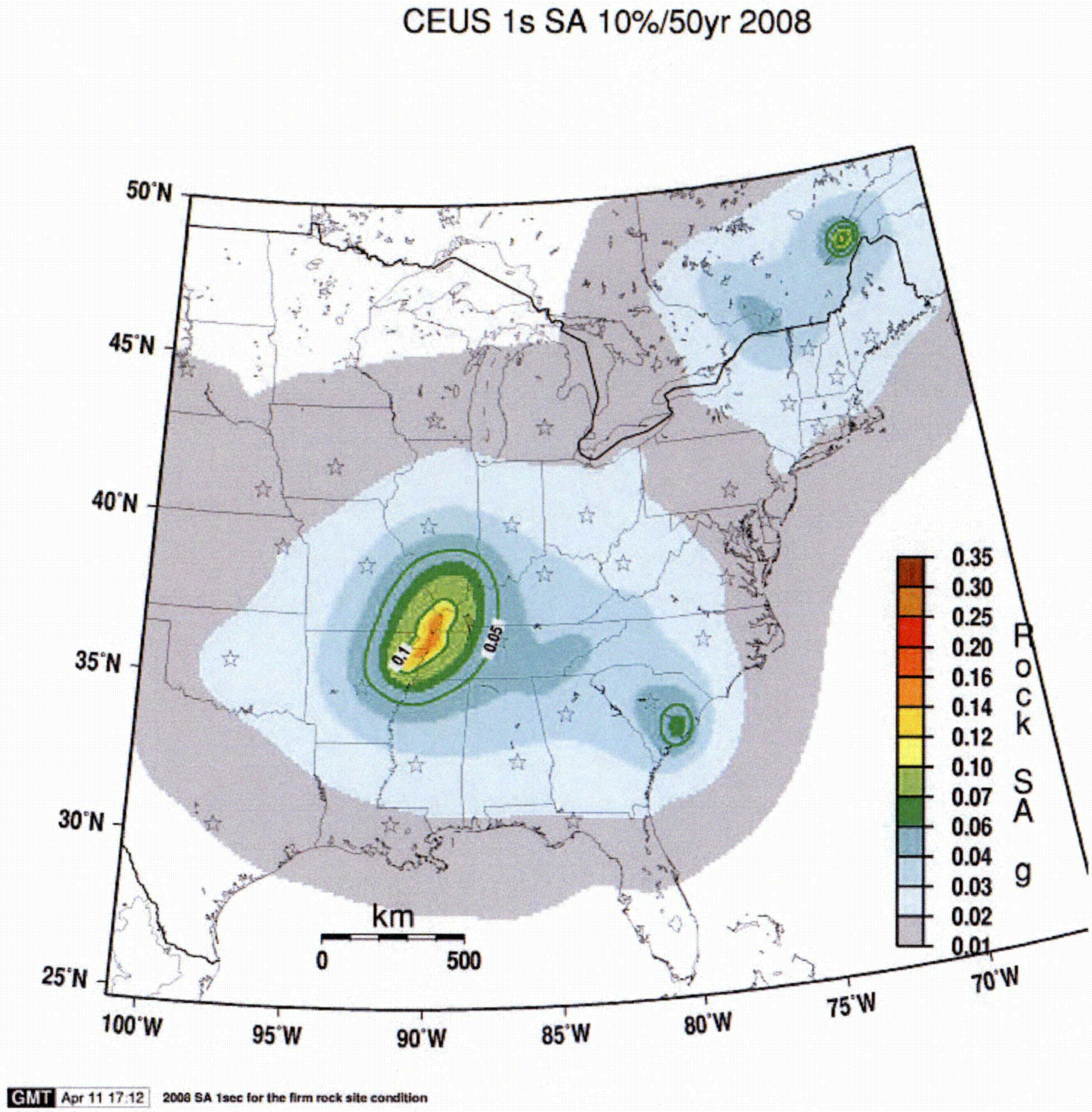


Figure 2-10: Map of 1-hertz spectral acceleration (SA) for 10% exceedance in 50 years in the Central and Eastern United States in standard gravity (g).

CEUS PGA 10%/50 years, 2008

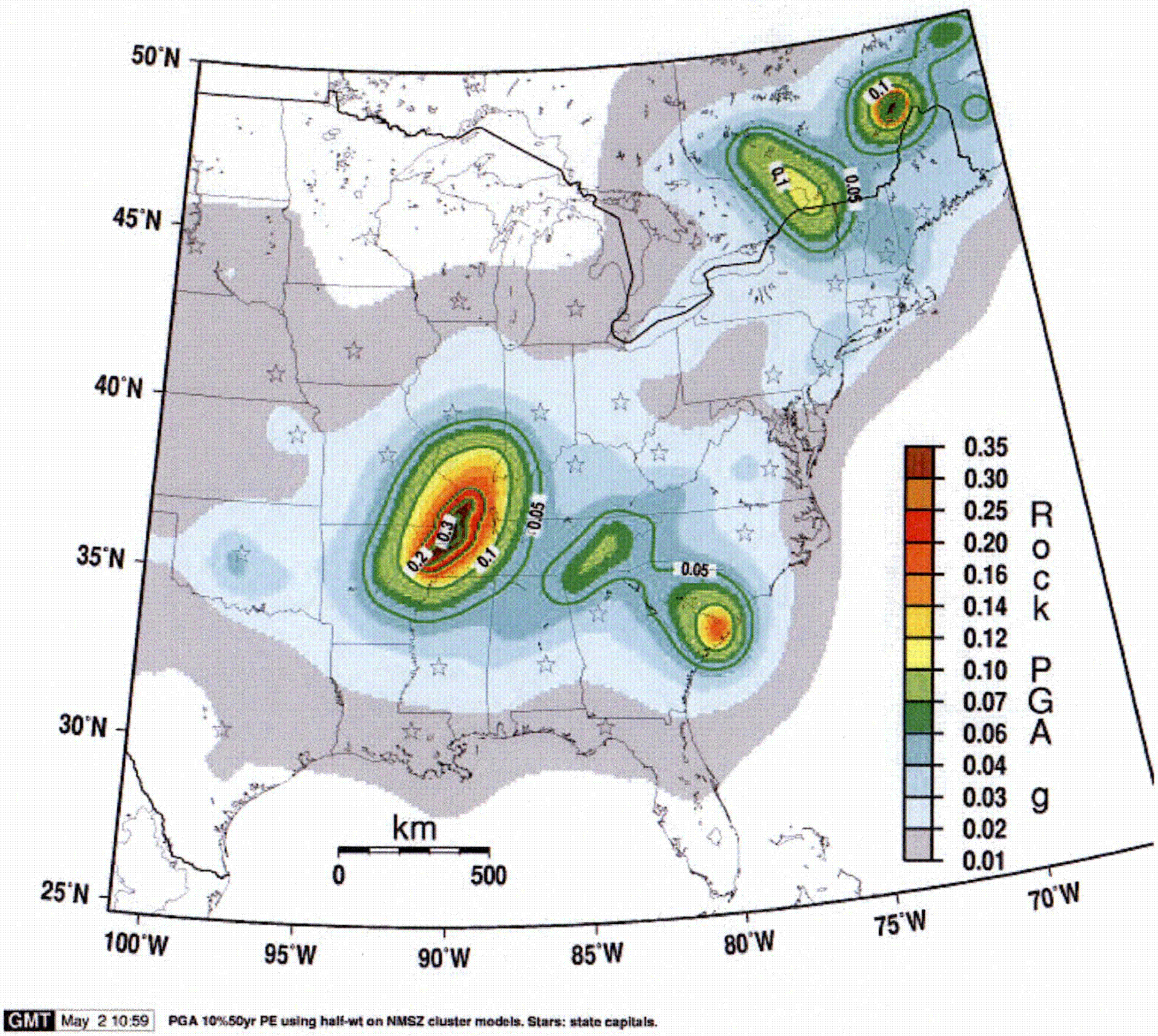


Figure 2-11: Map of peak ground acceleration (PGA) for 10% probability in 50 years in the Central and Eastern United States in standard gravity (g).

2.6 References

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

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[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)

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[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. The top of the reactor pool is approximately 10 feet below ground level.

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 feet 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. Additionally, due to the fact that that reactor and instrument console are well below ground level, damage to the reactor or controls is also very unlikely.

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 four feet 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 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.

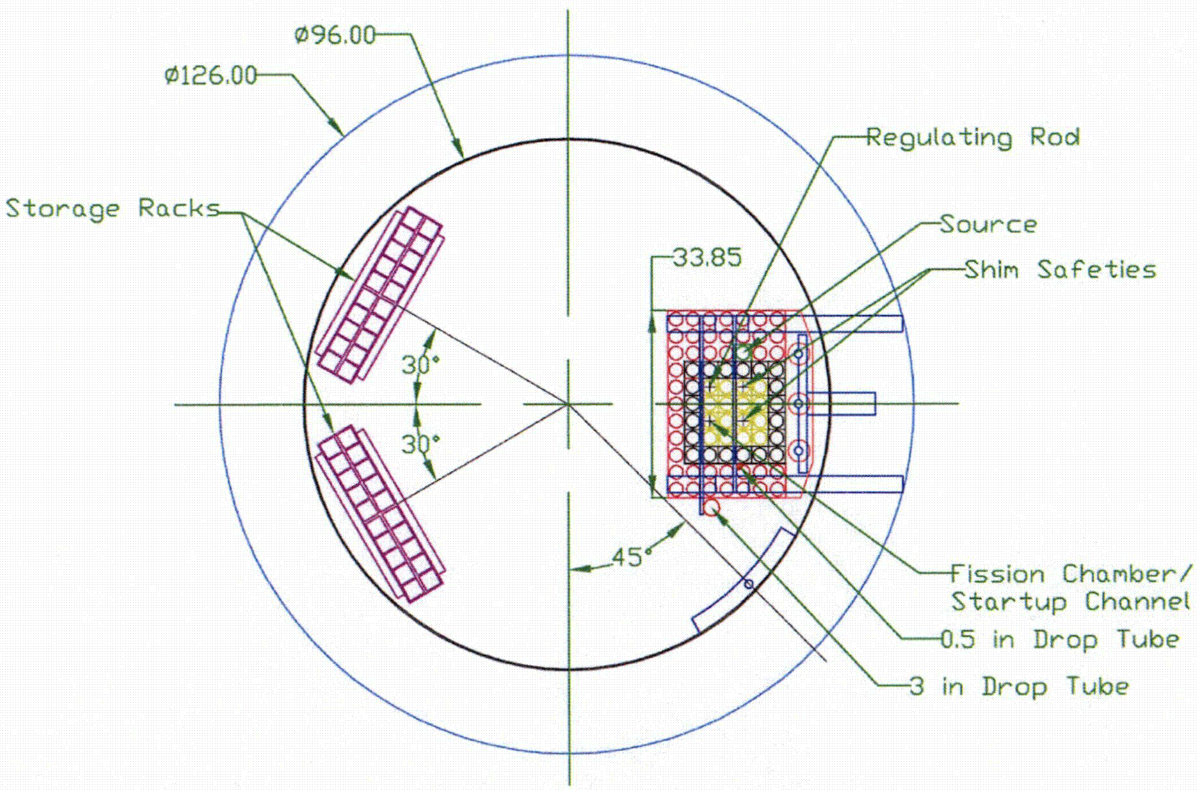


Figure 4-1: PUR-1 Pool Layout

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. Each fuel element consists of several thin metal plates assembled into a unit about 7 cm by 7 cm with an active fuel length of approximately 60 cm. 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)	12.5

Fuel Meat Dimensions Width (mm) Thickness (mm) Height (mm)	 59.6 0.508 600.1
Fuel Plate Dimensions Width (mm) Thickness (mm) Height (mm)	 70.2 1.27 638.6
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)	206
Fuel plates in core (current)	190
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
Control rod worth (%Δk/k) 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 $\left(\frac{\%\Delta k}{k \cdot {}^{o}C}\right)$[3]	-9.05E-3±9%
Fuel temperature coefficient $\left(\frac{\%\Delta k}{k \cdot {}^{o}C}\right)$[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

[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.

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.

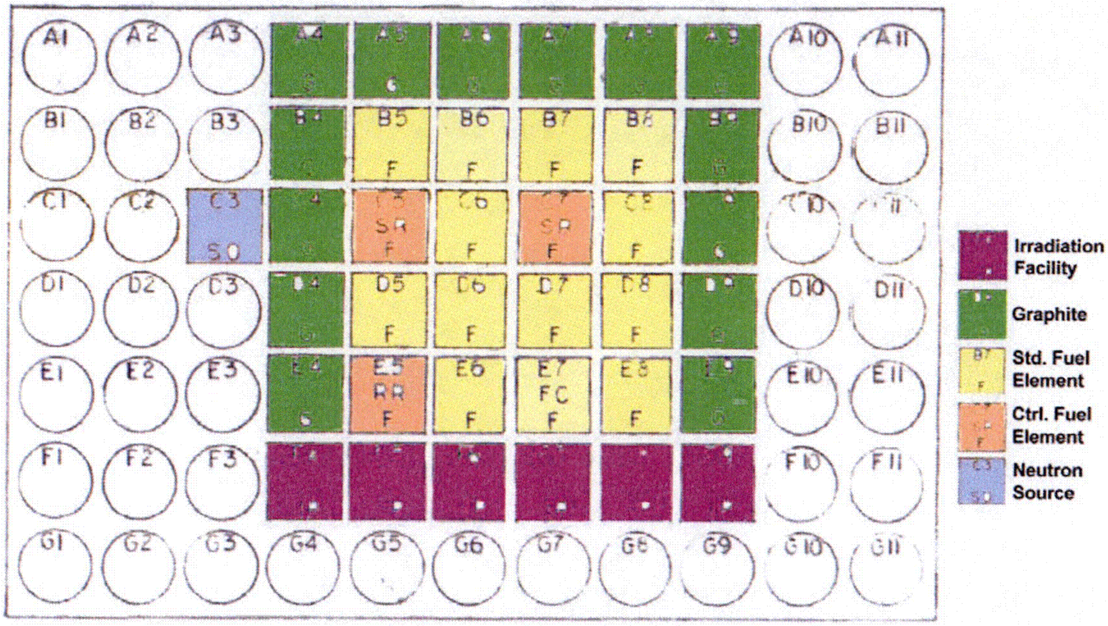


Figure 4-2: PUR-1 Grid Plate

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	U_3Si_2-Al
Enrichment	19.75%
Mass ^{235}U per fuel plate	12.5 g
Fuel Plate Dimensions	
Width (mm)	70.2
Thickness (mm)	1.27
Height (mm)	638.6
Fuel Meat Dimensions	
Width (mm)	59.6[2]
Thickness (mm)	0.508
Height (mm)	600.1[3]
Cladding Type	6061 Al
Cladding:	
Along width (mm)	3.63 (min)
Along thickness (mm)	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.

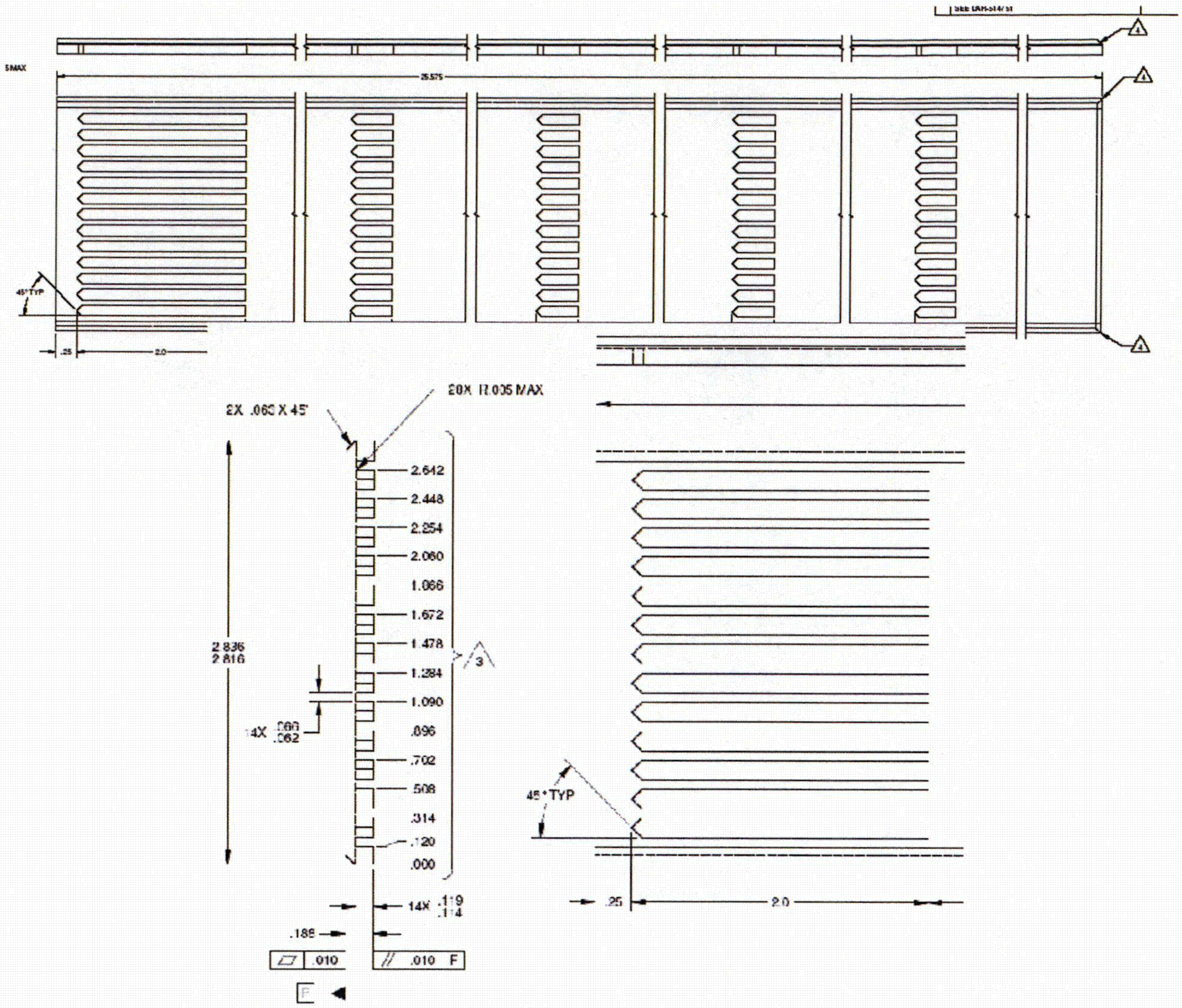


Figure 4-3: Standard assembly can detail, showing wall spacers.

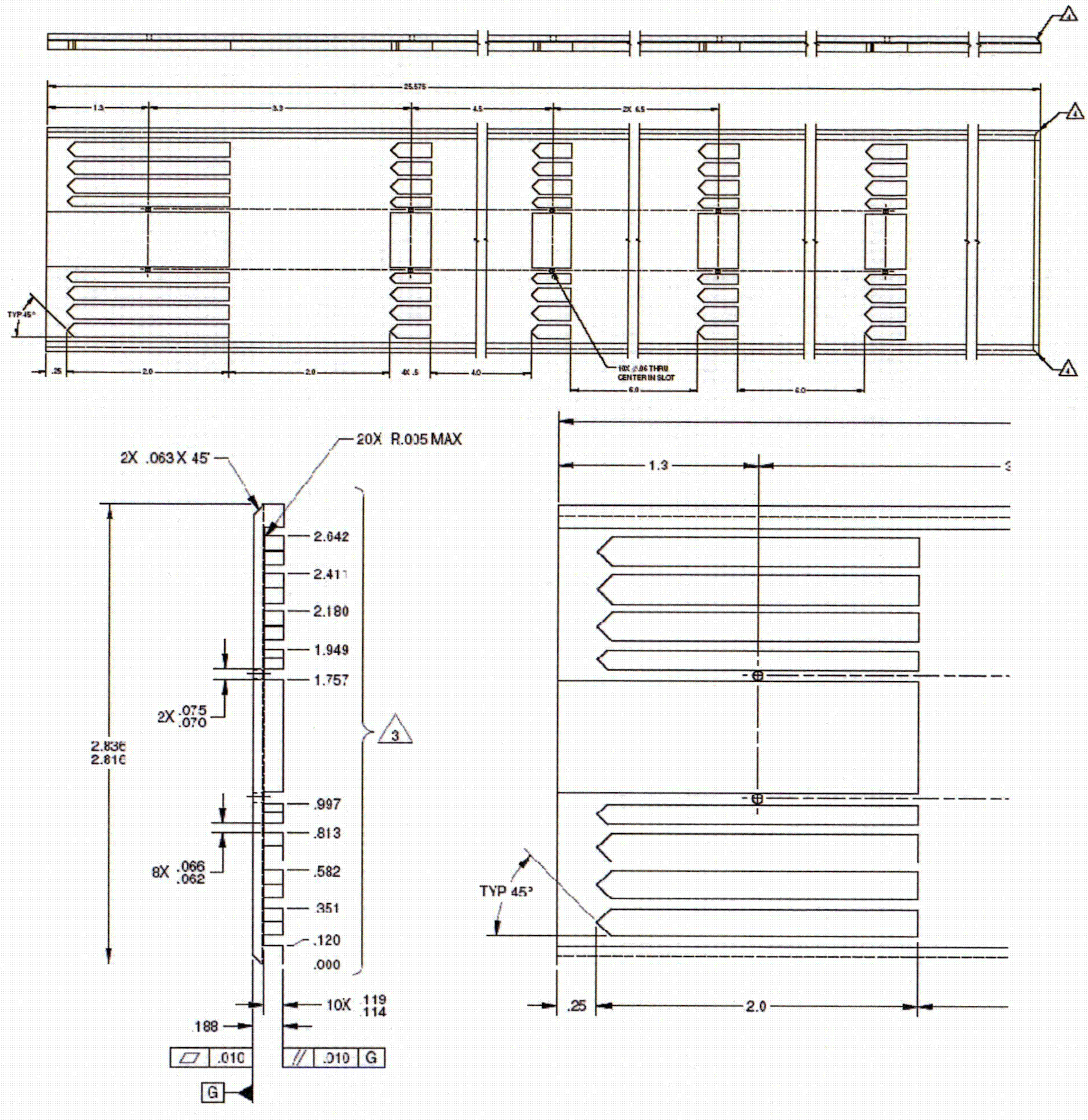


Figure 4-4: Control assembly can detail, showing wall spacers.

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.

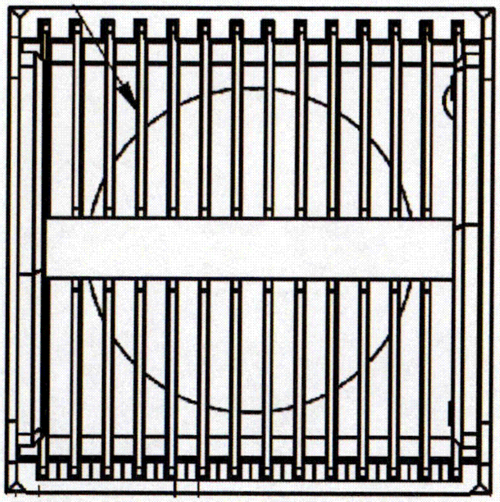


Figure 4-5: Standard fuel assembly.

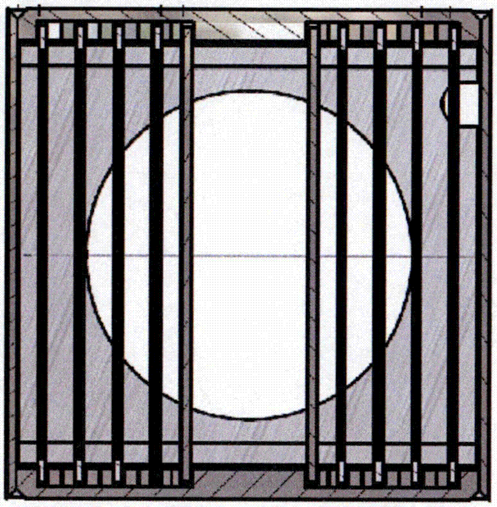


Figure 4-6: Control fuel assembly.

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 notches machined into the end of the plates, as shown in Figure 4-7 and Figure 4-8.

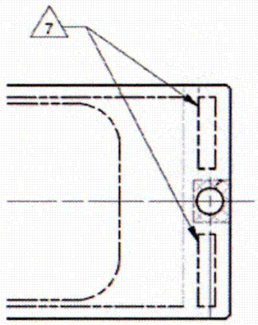


Figure 4-7: Plates are identified by unique serial numbers as shown here.

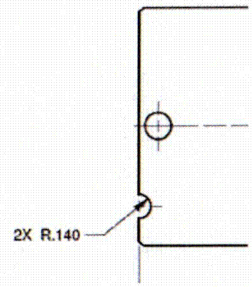


Figure 4-8: Dummy plates are differentiated from the fuel plates by a notch machined into the end of the plate.

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	17.7 in/min
Shim Safety Rod	4.4 in/min
Scram	Less than 1 second (from signal to complete insertion).
Size	
Regulating rod (inches)	½ x 2 ¼
Shim safety rods (inches)	½ x 2 ¼ x 25 ½

Maximum rate of reactivity change Regulating rod Shim safety rods	 $1.1\text{x}10^{-4}\ \frac{\Delta k}{k \cdot s}$ 4.2.2.1. $9.9\text{x}10^{-3}\ \frac{\Delta k}{k \cdot s}$
Average rate of reactivity change 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

The reactor pool is built below floor level except for the three foot wall that serves as a biological shield for the operators and experimenters. 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.

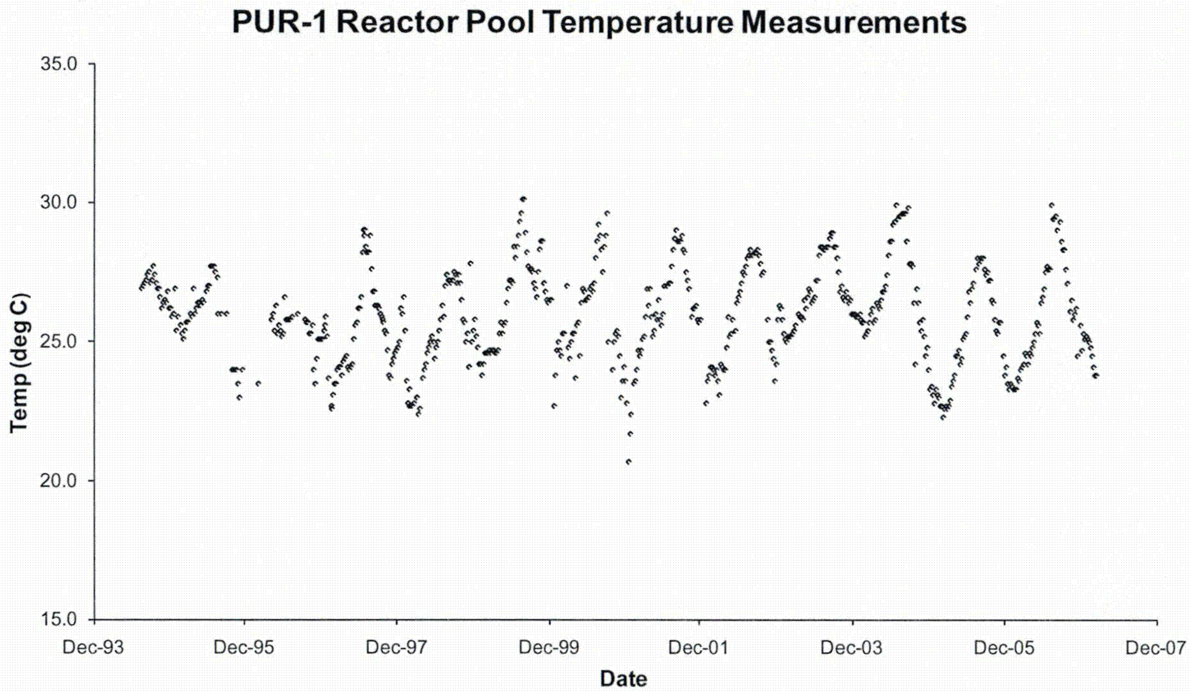


Figure 4-9: PUR-1 Reactor pool temperature measurements from 1994-2007.

4.4 Biological Shield

The biological shield consists of light water, magnetite sand, earth, and structural concrete. The reactor tank sits in a pit 14 feet deep with only the top 3 feet 4 inches of the tank above floor level. The tank sits on a concrete pad and is enclosed below ground level by 2 feet of barytes sand between the tank and earth. A monolithic mass of ordinary concrete 15 inches thick is provided from the floor level to the top of the tank. 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.

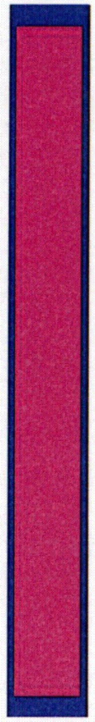
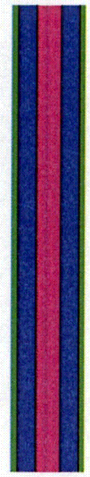
4.5 Nuclear Design

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.			

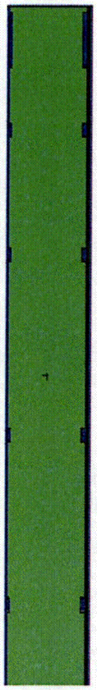


Figure 4-13: Model representation of standard assembly plate spacing detail showing wall spacers.

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.

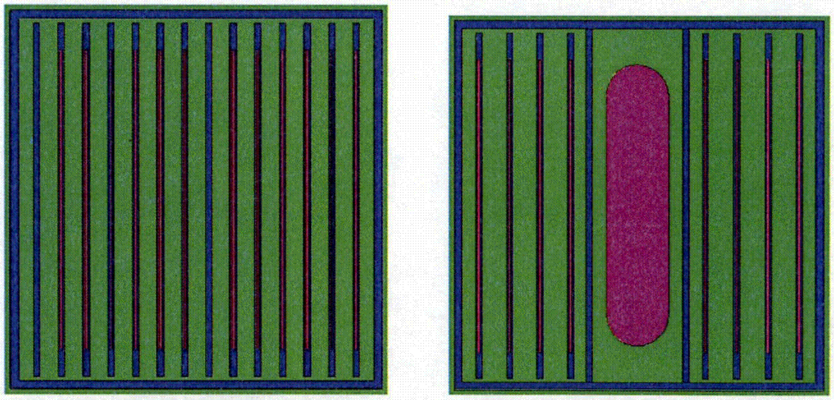


Figure 4-14: Comparison of standard and control assemblies in the model.

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 assembly shown in the figure above, but reflects the fact that the rod is hollow, and filled with water.

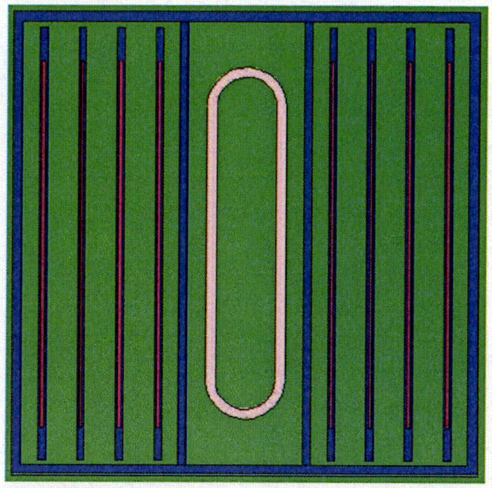


Figure 4-15: Model of regulating rod assembly.

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 on the left side of the core.

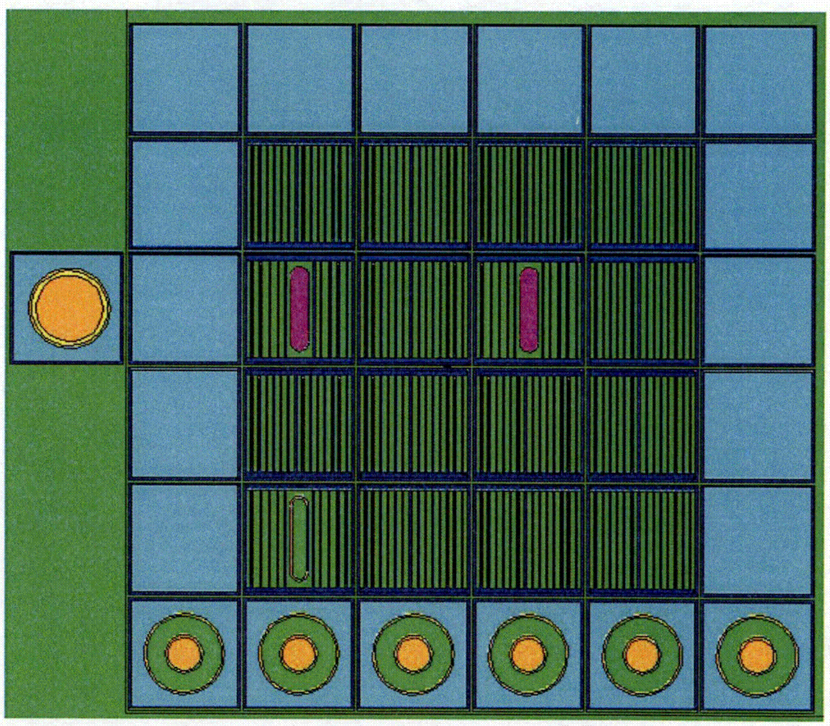


Figure 4-16: Representation of LEU core load

4.5.2 Normal Operating Conditions

The PUR-1 LEU fuel plates are fabricated with U_3Si_2 dispersion fuel. Each fuel plate has a nominal loading of 12.5 g U-235. A fresh core with 191 fuel plates and 15 "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).

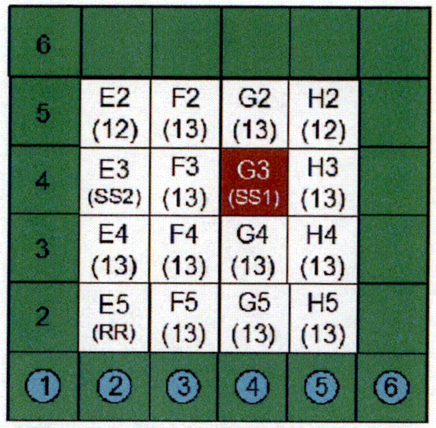
Table 4-7 compares the heating by assembly and in non-fueled components for the LEU core with the rods banked at 53.5 cm. The banked rod critical configuration was found to have the highest peak power density in the analysis of the HEU core during the HEU-LEU conversion analyses. Elements 3-4, 4-3, and 4-4 are noted as having the highest average plate powers, and are therefore of interest for thermal-hydraulics analysis. Figure 4-17 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.



Purdue LEU Core Layout (191 plates)

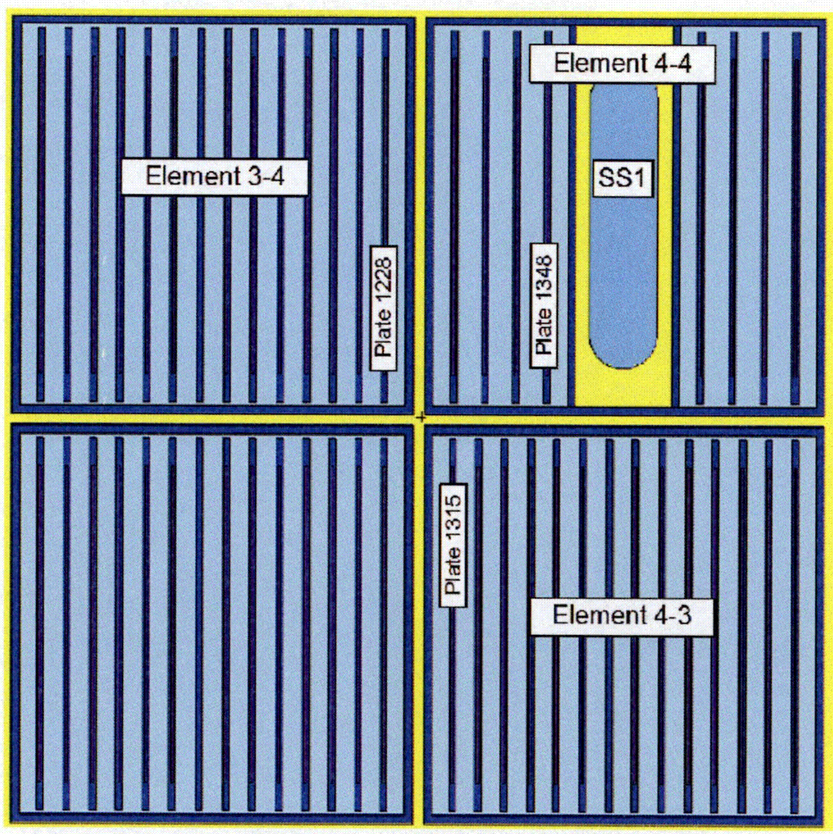


Figure 4-17: PUR-1 Core Layout and Bundle Drawings.

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

	Assembly Power (W) 1000 W	**Average Plate Power (W)**	**Assembly Power (W) 12 kW**	**Average Plate Power (W)**
2-2 (RR)	38.19	4.77	458.28	57.24
2-3	61.85	4.76	742.2	57.12
2-4 (SS2)	45.65	5.71	547.8	68.52
2-5	45.93	3.83	551.16	45.96
3-2	62.56	4.81	750.72	57.72
3-3	78.09	6.01	937.08	72.12
3-4	78.42	6.03	941.04	72.36
3-5	61.20	4.71	734.4	56.52
4-2	63.52	4.89	762.24	58.68
4-3	80.78	6.21	969.36	74.52
4-4 (SS1)	60.09	7.51	721.08	90.12
4-5	62.29	4.79	747.48	57.48
5-2	52.32	4.03	627.84	48.36
5-3	63.97	4.92	767.64	59.04
5-4	63.49	4.88	761.88	58.56
5-5	47.64	3.97	571.68	47.64
Inter-assembly water	3.78		45.36	

Graphite reflector	9.52		114.24	
Grid plate	2.27		27.24	
Water reflector (pool)	18.00		216	
SS1	0.23		2.76	
SS2	0.17		2.04	
RR	0.03		0.36	
Total	1000.0		12000	

Table 4-8: Plate Power (W) Computed from Heating Tallies in Bundles 4-4, 3-3 and 3-4 in PUR-1 Core with 190 Fuel Plates.

Bundle 4-4	**Power (W)**
Plate 1345 Meat	70
Plate 1346 Meat	70.9
Plate 1347 Meat	74
Plate 1348 Meat	80.7
Plate 1355 Meat	77.4
Plate 1356 Meat	69.8
Plate 1357 Meat	65.9
Plate 1358 Meat	64.3
Clad	3.6
Water	21.9
Can	2.4
Total	600.9
Bundle 3-4	**Power (W)**
Plate 1215 Meat	58
Plate 1216 Meat	55.6
Plate 1217 Meat	54.9
Plate 1218 Meat	55
Plate 1219 Meat	56.1
Plate 1220 Meat	58.4
Plate 1222 Meat	59.1
Plate 1223 Meat	57.6
Plate 1224 Meat	57.3
Plate 1225 Meat	57.6
Plate 1226 Meat	59.2
Plate 1227 Meat	61.2
Plate 1228 Meat	65.1
Clad	4.8
Water	21.9
Can	2.4
Total	784.2
Bundle 4-3	**Power (W)**
Plate 1315 Meat	64.1
Plate 1316 Meat	61.6
Plate 1317 Meat	60.7
Plate 1318 Meat	60.4
Plate 1319 Meat	60.9
Plate 1320 Meat	62.8
Plate 1322 Meat	62.1
Plate 1323 Meat	59.2
Plate 1324 Meat	57.6
Plate 1325 Meat	56.8
Plate 1326 Meat	56.5
Plate 1327 Meat	56.8
Plate 1328 Meat	58.4
Clad	4.9
Water	21.9
Can	2.4
Total	807.1

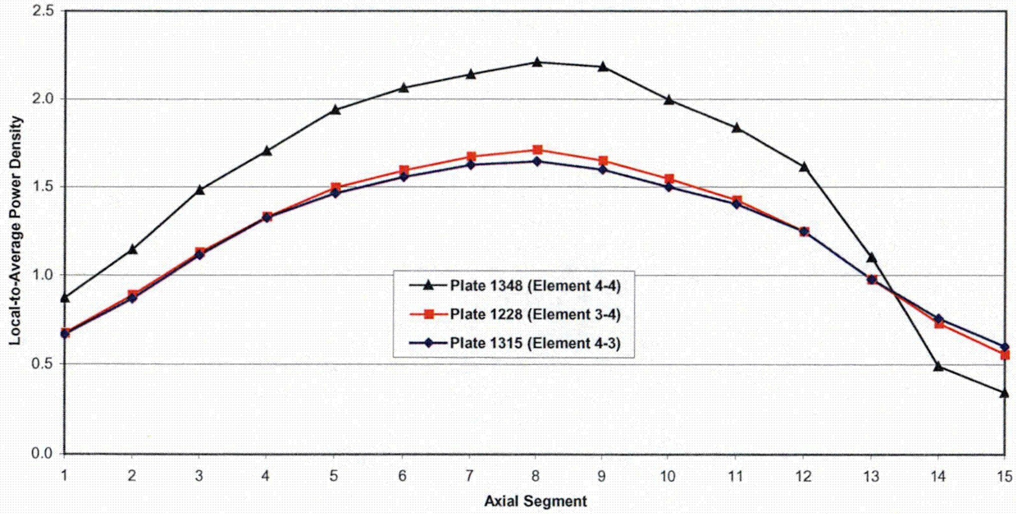


Figure 4-18: Axial Power Profiles in Plates 1348, 1228 and 1315 for Banked Rod Critical Configuration in PUR-1

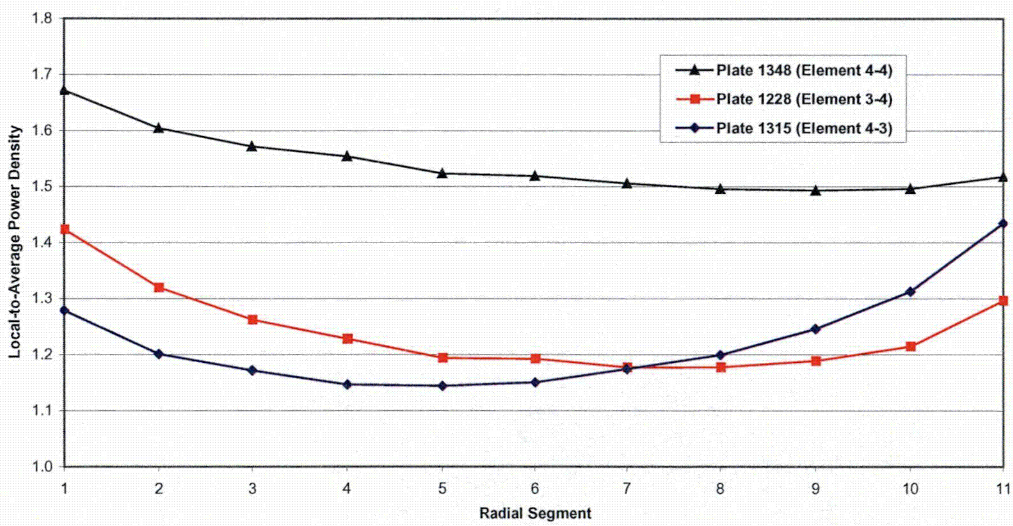


Figure 4-19: Radial Power Profiles in Plates 1345, 1228 and 1315 for Banked Rod Critical Configuration in PUR-1

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	Power (W at 12 kW)
1	1.88595	5.88645	3.04	0.99%	36.48
2	5.88645	9.88695	4	0.86%	48
3	9.88695	13.88745	5.17	0.76%	62.04
4	13.88745	17.88795	5.95	0.70%	71.4
5	17.88795	21.88845	6.76	0.66%	81.12
6	21.88845	25.88895	7.2	0.64%	86.4
7	25.88895	29.88945	7.47	0.63%	89.64
8	29.88945	33.88995	7.71	0.62%	92.52
9	33.88995	37.89045	7.62	0.63%	91.44
10	37.89045	41.89095	6.97	0.65%	83.64
11	41.89095	45.89145	6.41	0.68%	76.92
12	45.89145	49.89195	5.63	0.74%	67.56
13	49.89195	53.89245	3.85	0.86%	46.2
14	53.89245	57.89295	1.72	1.25%	20.64
15	57.89295	61.89345	1.2	1.48%	14.4
Total			80.7	**0.21%**	968.4
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	Power (W at 12 kW)
1	1.88595	5.88645	2.36	1.07%	28.32
2	5.88645	9.88695	3.11	0.94%	37.32
3	9.88695	13.88745	3.95	0.83%	47.4
4	13.88745	17.88795	4.65	0.77%	55.8
5	17.88795	21.88845	5.23	0.73%	62.76
6	21.88845	25.88895	5.57	0.70%	66.84
7	25.88895	29.88945	5.84	0.70%	70.08
8	29.88945	33.88995	5.98	0.69%	71.76
9	33.88995	37.89045	5.76	0.69%	69.12
10	37.89045	41.89095	5.4	0.71%	64.8
11	41.89095	45.89145	4.98	0.74%	59.76
12	45.89145	49.89195	4.36	0.80%	52.32
13	49.89195	53.89245	3.42	0.89%	41.04
14	53.89245	57.89295	2.55	1.05%	30.6
15	57.89295	61.89345	1.95	1.17%	23.4
Total			65.11	**0.22%**	781.32
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power	σ	

			Density		
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	Power (W at 12 kW)
1	1.88595	5.88645	2.33	1.07%	27.96
2	5.88645	9.88695	3.03	0.96%	36.36
3	9.88695	13.88745	3.89	0.85%	46.68
4	13.88745	17.88795	4.63	0.79%	55.56
5	17.88795	21.88845	5.11	0.75%	61.32
6	21.88845	25.88895	5.43	0.72%	65.16
7	25.88895	29.88945	5.67	0.70%	68.04
8	29.88945	33.88995	5.74	0.70%	68.88
9	33.88995	37.89045	5.58	0.70%	66.96
10	37.89045	41.89095	5.23	0.73%	62.76
11	41.89095	45.89145	4.9	0.76%	58.8
12	45.89145	49.89195	4.36	0.81%	52.32
13	49.89195	53.89245	3.42	0.90%	41.04
14	53.89245	57.89295	2.65	1.02%	31.8

15	57.89295	61.89345	2.1	1.13%	25.2
Total			64.07	0.23%	768.84
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 at 1 kW. Assuming a linearity of neutron flux with power, the maximum and average fluxes at 12 kW should be 2.41E11 and 1.66E11 n/(cm2*s), respectively

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% Δk/k) for 25 million histories; the uncertainty was reduced to ±12 pcm (0. 012% Δk/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% Δk/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%) Δk/k in a 190 fuel plate core (16 dummies), including a reactivity bias of 0.32% Δk/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 measured and 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

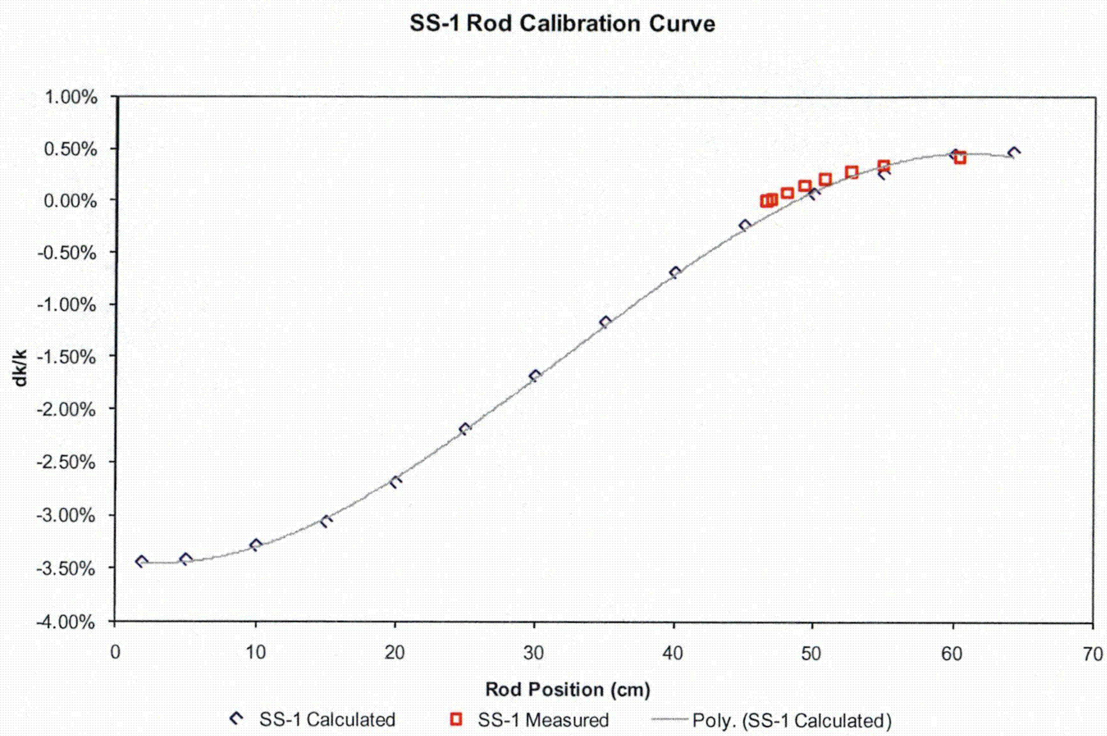


Figure 4-20: Calibration curve for SS-1 rod with calculated and measured values.

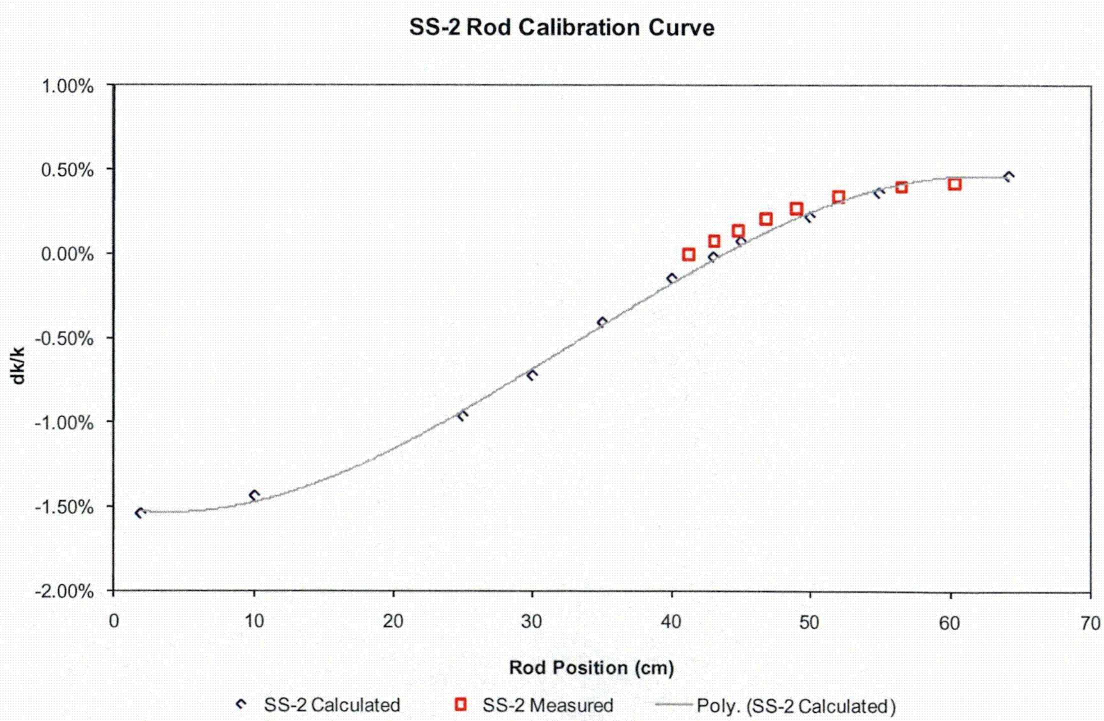


Figure 4-21: Calibration curve for SS-2 rod with calculated and measured values.

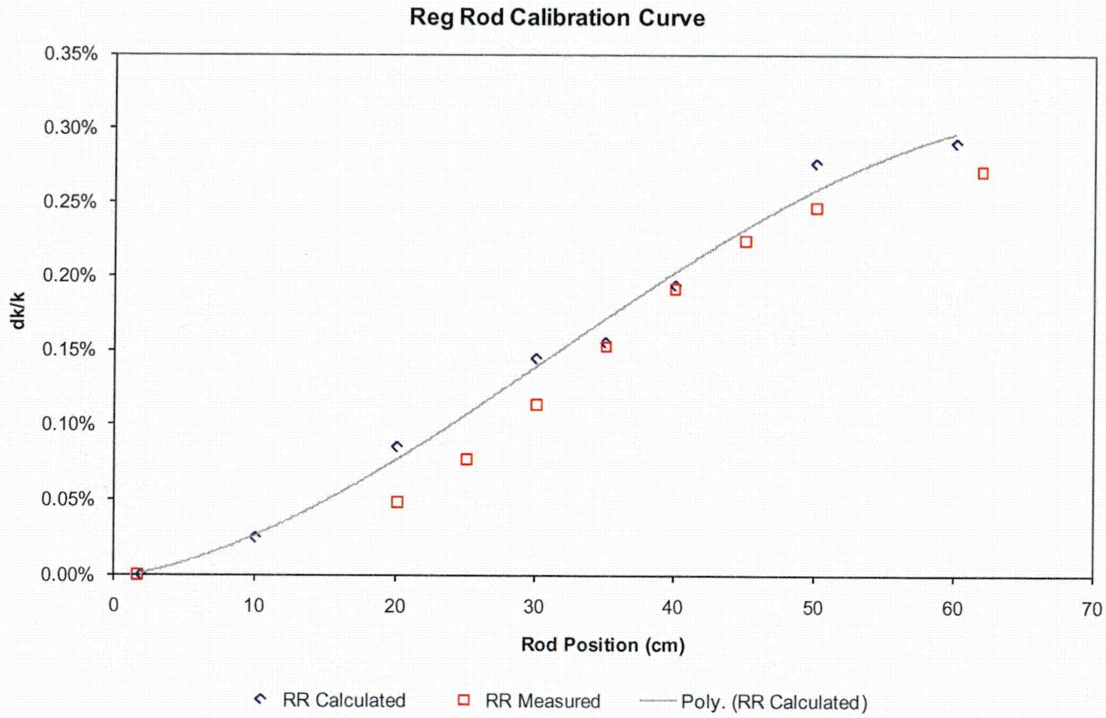


Figure 4-22: Calibration curve for RR with calculated and measured values.

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%
K_{excess} (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 = 7.870x10^{-3} ± 1.79x10^{-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

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

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 reactor physics parameters and reactivity coefficients is presented in Tables 4-15 and 4-16.

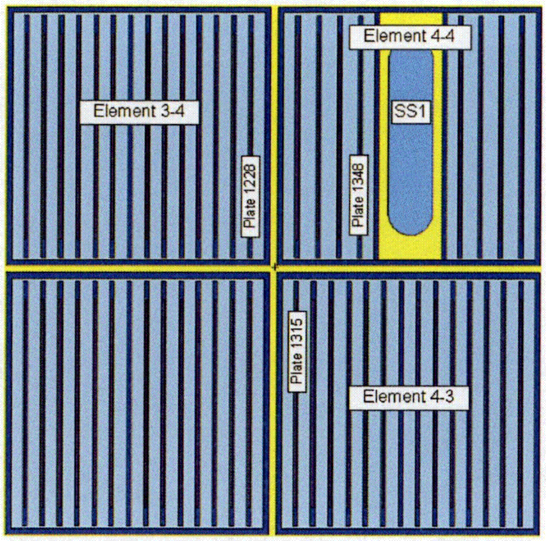


Figure 4-23: Figure showing water regions for perturbation models

Table 4-15: Other core physics parameters.

	LEU (calculated)
α_{fuel} $\left(\frac{\%\Delta k}{k \cdot {}^{o}C}\right)$	-8.05E-04
$\alpha_{moderator}$ $\left(\frac{\%\Delta k}{k \cdot {}^{o}C}\right)$	-9.05E-03
α_{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	$\Delta\rho$/°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	$\Delta\rho$/°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.

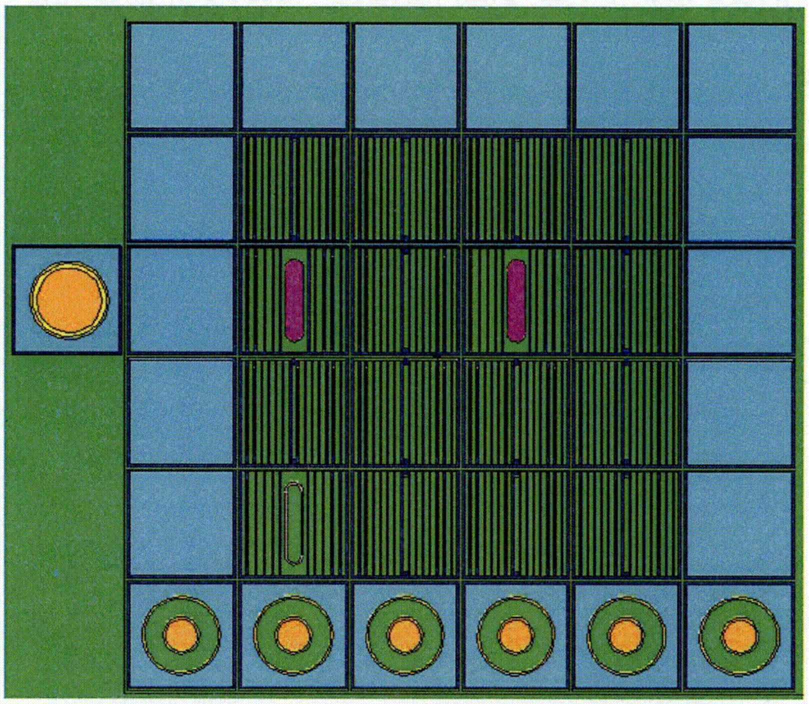


Figure 4-24: PUR-1 Core Layout with LEU Fuel.

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 present 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 $2.38\times10^{-3} \pm 2.11\times10^{-3}$ % $\Delta k/k/°C$. 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 $\Delta\rho/°C$. 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 $\Delta\rho/°C$ 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.

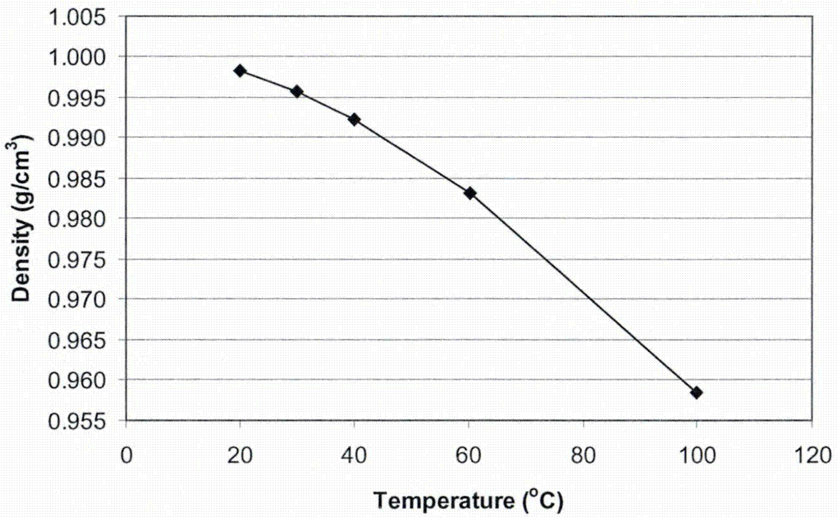


Figure 4-25: Density of sub-cooled water at 1.5 atmospheres.

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 °C were employed. Furthermore, $S(\alpha,\beta)$ which had previously been evaluated for light water at 20, 30, 60, 100, and 150 °C were also used.

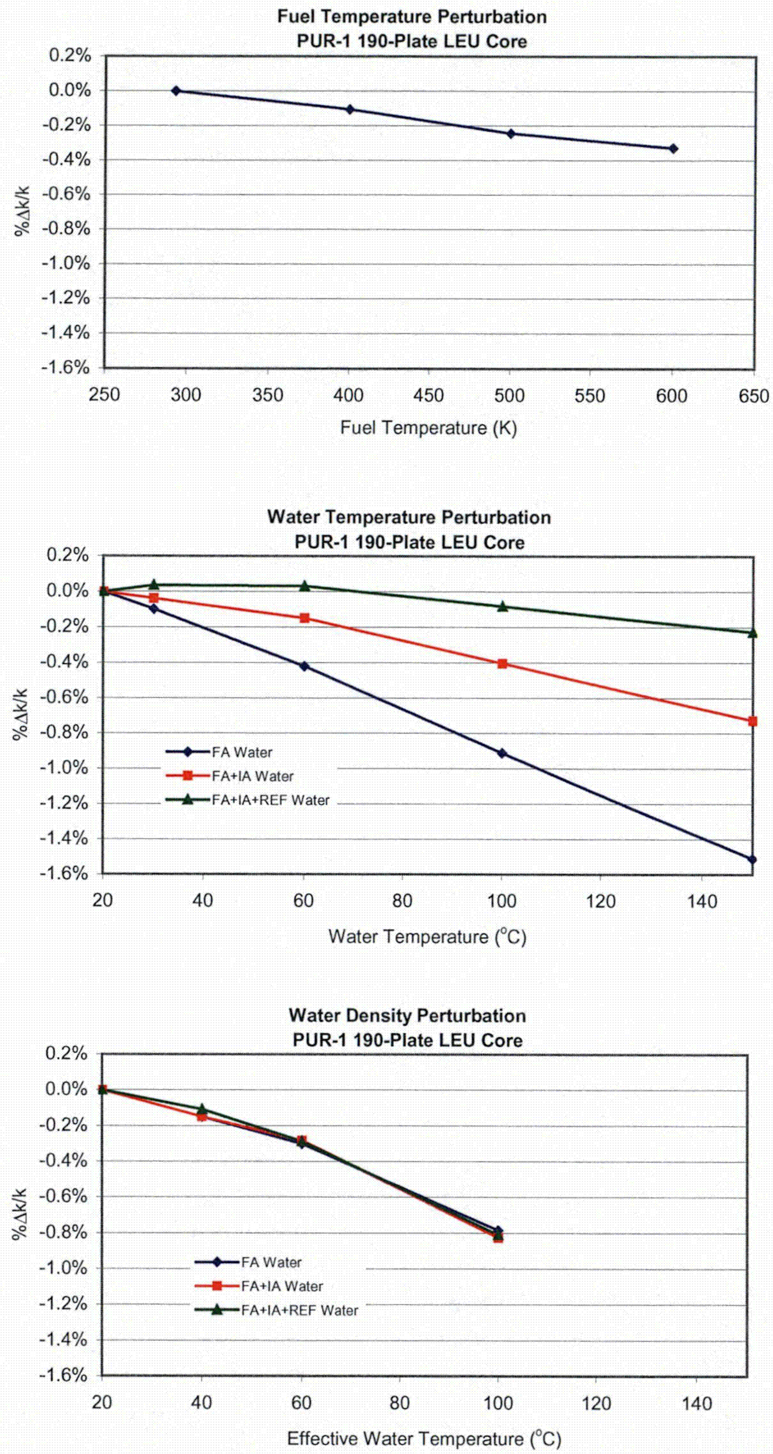


Figure 4-26: Effect of Fuel Temperature, Water Temperature and Water Density Perturbations on Core Reactivity.

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 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.. Cross section views of the two different types of assemblies, standard and control, are shown in Figure 4-27 and Figure 4-28.

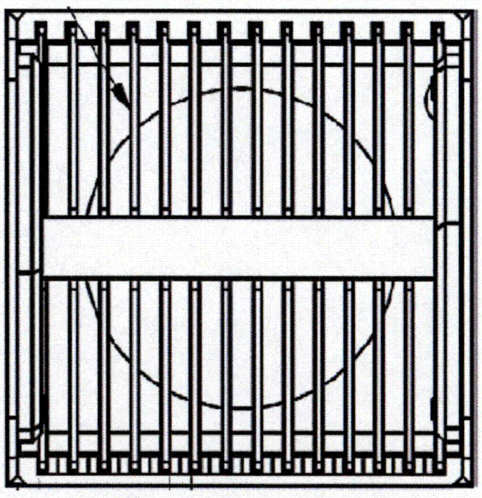


Figure 4-27: Standard LEU fuel assembly.

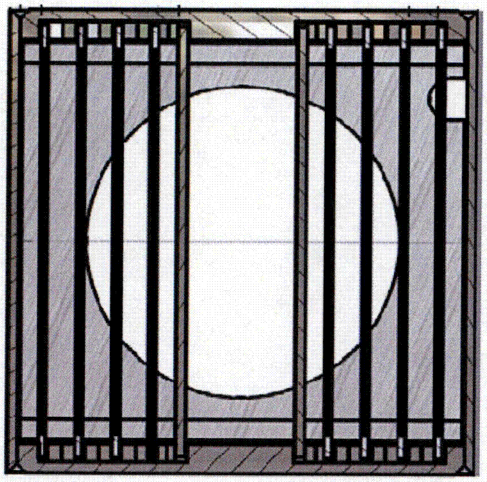


Figure 4-28: Control LEU fuel assembly.

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 Assemblies	144±15	181±15	127±8	127±8

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. Plate 1348 (see Table 4-9) was the limiting plate.

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 are shown in

Table 4-18: Model Dimensions for the Thermal-Hydraulic Models. And the hot channel factors are listed in

Table 4-19. The hot channel factors were calculated using equations that include 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 the hot channel factors shown in

Table 4-19, for a total of 190 fuel plates, a channel thickness of 181 mil, a radial power factor of 1.5414 (=the ratio of 8.07 kW power in plate to 5.2356 kW at 1 kW per average plate), and 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.

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

Number of Axial Nodes	14	
Number of Plates	190	
Thermal Conductivity (W/m*K)		
Fuel Meat	80	
Clad	180	
Pool Temperature (°C)	30	
	Inches	mm
Fuel Meat		
Height	23.625	600.08
Width	2.345	59.563
Thickness	0.020	0.508
Channel		
Height	25.110	637.79
Width	2.832	71.933
Channel Thickness	0.197	5.004
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 Plate 1348 NATCON Analysis

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**		**1.334**	**1.251**	**1.226**
Power Measurement F_Q	Systematic	0.500	**1.500**		
Flow friction factor F_W	Systematic	0.048	**1.048**		
Heat Transfer Coeff. F_H	Systematic	0.200	**1.200**		

Calculations were done to determine the core power distribution and the ONB power for the PUR-1 based on the limiting plate, 1348. The ONB power calculation given below includes the impact of the power density variation along the width of the plate itself. The resulting ONB power was found to be 98.55 kW The 20 mil uncertainty used is conservative compared to the 15 mil uncertainty given in Table 4-17 for the evaluated core design. These results are Table 4-19. 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.

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 margin to incipient boiling shown in Table 4-21 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:

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_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.. This was done to identify the limiting channel. Then the limiting channel was evaluated under nominal operating. 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.

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. Table 4-20 provides a summary of the ONB powers for each of the cases analyzed. For the PUR-1 core, the limiting channel/plate was plate 1348 with the plate-to-plate channel, which had an ONB power of 98.6 kW.

Table 4-20: ONB Powers for the high power plates.

	Plate 1348	Plate 1228	Plate 1315
ONB Power (kW)	98.6	153.4	158.9

Using NATCON, and the thermal-hydraulics parameters for the limiting plate (1348), the nominal operating conditions were also calculated. All hot channel factors are included in these calculations. These results are shown in Table 4-21.

Table 4-21: Operating Conditions for PUR-1 as Determined by NATCON for Limiting Plate 1348

	Present Power	**Uprate Power**	**ONB Power**
Power Level	1 kW	12 kW	98.6 kW
Max. Fuel Temp. (°C)	31.92	39.1	112.6
Max. Clad Temp. (°C)	43.42	43.4	112.5
Coolant Inlet Temp. (°C)	30.0	30.0	30.0
Coolant Outlet Temp. (°C)	31.7	35.8	46.3
Margin to incipient boiling (°C)	78.3	67.9	0
Coolant Velocity (mm/s)	5.18	18.5	54.0
Coolant Mass Flow Rate (kg/m^2s)	5.16	18.41	53.4

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 4-22 shows the maximum power, the LSSS and operating power for PUR-1.

Table 4-22: Key Power Levels for Reactor Operation and LSSS for PUR-1

	1 kW	12 kW Proposed
Maximum Power Level Including 50% Uncertainty	1 kW	12
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 4-21). NATCON was used to determine the minimum power for ONB for core in the limiting channels, as well as the thermal-hydraulic parameters at these calculated powers. The results of these calculations are shown in Table 4-21.

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 98.6 kW 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% $\Delta k/k$ of all moveable and non-secured experiments, and (2) Slow insertion of reactivity at the maximum allowed rate of 0.04% $\Delta k/(k^*s)$ 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. The ONB power was found to be 79.3 kW, indicating a large margin compared to the proposed PUR-1 operating power of 12 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	Centrifugal
Capacity At 100°F/110 ft Head	30 gpm
Ion Exchanger	Mixed bed (replaceable cartridge type)
Rated Capacity	15, 000 grains as $CaCO_3$
Flow Rate	0-20 gpm
Design Pressure	100 psi
Heat Exchanger	Shell and tube (water chiller)
Cooling Capacity	36,000 Btu/hr (10,550 W)
Design Pressure	150 psi
Tube	225 psi
Shell	
Filter	Cartridge Type
Maximum Flow	80 gpm
Pressure Drop	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 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.