North Carolina State University is a land-grant university and a constituent institution of The University of North Carolina

Nuclear Reactor Program

NC STATE UNIVERSITY

Department of Nuclear Engineering
Campus Box 7909
Raleigh, NC 27695-7909

919.515.2321
919.513.1276 (fax)
URL: www.ne.ncsu.edu/NRP reactor_program.html

20 November 2007

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC

Re: Technical Specifications Amendment 17
License No. R-120
Docket No. 50-297

Attached please find a response to the letter dated 23 October 2007 titled "Request for Additional Information Regarding Amendment 17 of Operating License R-120 for the North Carolina State University PULSTAR Reactor (TAC NO.MD4280). A response to each individual question is provided.

As a result of the questions and responses, changes were made to the Technical Specifications and Attachment 1. Details of the changes made to the specifications are explained in the response to the request for additional information. Please note that Attachment 2 is provided and replaces the Emergency Plan Appendix B. If the Technical Specification amendment request is approved, the Emergency Plan will be revised within 30 days of the effective date of the amendment. Attachments 3, 4, and 5 provide reference material in support of the response to the request for additional information. Attachment 6 provides the basis for changes made to specification 3.8 and will replace the existing fueled experiment analysis in the Safety Analysis Report if the Technical Specifications are approved.

If you have any questions regarding this amendment or require additional information, please contact Gerald Wicks at 919-515-4601 or wicks@ncsu.edu .

I declare under penalty of perjury that the forgoing is true and correct. Executed on 20 November 2007.

Sincerely,

Ayman Hawari

Ayman I. Hawari, Ph.D.
Director, Nuclear Reactor Program
North Carolina State University

A020

NRR

Enclosures: Response to Request for Additional Information Dated 23 October 2007 with Attachments 1 through 6
Technical Specification Amendment 17

cc: Daniel E. Hughes, US NRC

RAI Number and Response

1. The ventilation exhaust flow rate with the new ventilation system is given as 1870 cfm.

   a) Attachment B-1 shows a flow of 270 cfm into the exhaust plenum from an unidentified source. Is this source external to the reactor bay? If so, how are the fuel failure event durations, as calculated on page 49 of Appendix B, affected by the decreased reactor bay exhaust flow rate, and how does this affect off-site doses and AEC fractions? How are occupational accident doses affected by a longer reactor bay air exchange time?

   b) Attachment B-1 shows a flow of 200 cfm into the exhaust plenum from the BT&TC exhaust fan. Is this fan required to be running during accident scenarios? If not, how are the fuel failure event durations, as calculated on page 49 of Appendix B, affected by the decreased reactor bay exhaust flow rate, and how does this affect off-site doses and AEC fractions? How are occupational accident doses affected by a longer reactor bay air exchange time?

The ventilation system description provided in the amendment request was in error. Corrections are summarized below:

> In the reactor building ventilation system, there is only one exhaust duct and only one exhaust fan in operation at a given time. For normal ventilation, the nominal fixed exhaust rate is 1870 cfm. In confinement mode, the nominal fixed exhaust rate is 600 cfm. The (Pneumatic) PN and (Beam Tube & Thermal Column) BT&TC fans are used to move air these systems or areas into the exhaust duct without entering the reactor bay. The PN and BT&TC fans do not increase the ventilation system exhaust rate. 270 cfm is the air return flow rate from the ventilation room.

Responses to specific questions follow:

The 270 cfm is the air return flow rate from the ventilation. This air flow has no affect on the fuel failure event duration, off-site doses, or AEC fractions calculated in Attachment 2. Attachment 2 used exhaust flow rates of 1870 cfm for normal ventilation and 600 cfm for confinement.

The 200 cfm is the input into the normal ventilation from the BT & TC if it is on. This fan is routinely used while the reactor is in operation and turned off upon reactor shutdown or initiation of confinement. If an accident occurs, this fan would have no effect on the accident analyses. Calculations given in Attachment 2 were performed at a normal ventilation exhaust rate of 1870 cfm and confinement exhaust rate of 600 cfm. Therefore, the calculations are not affected.

Regarding reactor bay dose rate to occupational workers and emergency responders, the following points are noteworthy:

i The fuel handling accident is considered to be a short duration, or puff, release.

ii Airborne monitors would initiate the reactor bay evacuation horns and personnel would evacuate the area. The fuel failure accident was considered in determining the monitor set points. Monitor set points are conservatively set. Appropriate response to the reactor evacuation horns by personnel is covered in training and would be enforced by the Reactor Health Physicist and reactor staff.

iii Entries into airborne radioactivity areas would be infrequent, short in duration, and made by respirator qualified personnel using appropriate respiratory protective devices. The main reason for personnel to enter an airborne radioactivity area in the reactor building would be for monitoring and initiating the Confinement system, if not already operating, to reduce off-site dose.

iv The concentration of airborne activity for the fuel handling accident are reduced by a factor of at least 22,000 over a 24 hour period by the ventilation system:

0.75 h per air change in normal ventilation 32 air changes in 24 hours
2.35 h per air change in confinement ~ 10 air changes in 24 hours
$1 / [\exp(-32)] = 7.9\text{ E}13$ and $1 / [\exp(-10)] = 2.20\text{ E}4$

The ventilation removal rate constants are:
$k = 1/ 0.75\text{ h} = 1.33$ per hour in normal ventilation
$k = 1/ 2.35\text{ h} = 0.426$ per hour in confinement

By assuming an individual is present for 1 air exchange at the initial (or peak) concentration, C(0), is equivalent to placing the individual in the area for the 24 hour period exposed to the 24 h average concentration, i.e. the same integrated concentration-hour value (uCi-h/ml) is obtained:

$$\text{uCi-h/ml} = C(0) \int^{T} \exp(-kt)\, dt = [C(0) / k]\, [1 - \exp(-kT)]$$

And if T = 24 h for either confinement or normal ventilation:

$$\begin{aligned} \text{uCi-h/ml} = C(0) / k \quad &= (1 \text{ air exchange period})\, [C(0)] \\ &= 0.75\text{ h } C(0) \text{ for normal ventilation} \\ &= 2.35\text{ h } C(0) \text{ for confinement} \end{aligned}$$

The uCi-h/ml value for confinement is higher than that for normal ventilation because the airborne radioactivity is removed at a slower rate.

The projected total dose for a fuel handling accident to emergency personnel inside the reactor building was calculated in Section 13 of the SAR to be below occupational limits. The methods used and parameters used in this calculation have not been changed as noted above. Section 13 of the SAR is attached.

Similar calculations for Ar-41 for the worse case postulated accident meets 10 CFR 20 limits:

Ar-41 effective dose
= (2.35h)(3 E-5 uCi/ml)(1000 mrem / rem)(1100 rem/h per uCi/ml) = 78 mrem

The limiting dose in the above calculations is obtained in the confinement mode of ventilation. The confinement system was not changed by the ventilation system design modification.

The above calculated doses from postulated accidents do not correct for decay or energy spatial equilibrium for the deep dose-equivalent (i.e. external gamma radiation). Corrections for room dimensions would reduce the external doses calculated by a factor of 6 or more for partial energy spatial equilibrium:

$H(R) = H(\infty)\ [\ 1 - \exp(-u_{en}R)]$, as given in publication ICRP Report 30

where, $H(\infty)$ is the dose-equivalent rate for an infinite cloud
u_{en} = linear energy absorption coefficient for air
R = effective radius of the room
$R = [V/\pi]^{1/3} = [2.4\ E9\ cm^3 / \pi\ ]^{1/3} = 914\ cm$

This gives the correction factor: $H(R) / H(\infty) = [\ 1 - \exp(-u_{en}*914)]$

Nuclide	E(Mev)	$[\ 1 - \exp(-u_{en}R)]$	$1/[\ 1 - \exp(-u_{en}R)]$
Xe-133	0.03	0.149	6.7
Xe-133	0.08	0.026	38
Ar-41	1.293	0.029	34

The correction for partial energy spatial equilibrium is not made in the calculations given in Section 13 of the SAR.

External dose rates would be measured by the emergency response team and personnel dose tracking for each entry is part of the Emergency Procedures. Based on the above discussion, 10 CFR 20 limits are not exceeded.

2. The "worse case," 24-hour average Ar-41 concentration in the reactor bay is given as 2.7E-5 μCi/ml on page 44 of Appendix B. This gives an AEC fraction of 2700. Section 5.1 of Revision 8 of the Emergency Plan gives an AEC fraction of 2500 as the Action Level for Notification of Unusual Events for noble gas releases. Section 5.0 of the descriptions of the changes to the Emergency Plan states, "...from Appendix B, it can be concluded that airborne effluent from an experimental failure and postulated fuel failure continue to remain below the "Notification of Unusual Event" classification EAL." Please justify the inconsistency of this statement.

The 24 h average AEC fraction associated with the maximum concentration calculated in Appendix B used an atmospheric dispersion value, X/Q, of 0.017 s m^{-3} as follows:

46 AEC fraction = [2.7E-5 uCi/ml over 24 h/1E-8 uCi/ml per AEC] [0.017]

The value of 0.017 was rounded off to 0.02 to give a EAL value of 2500 AEC fractions for the "Notification of Unusual Event". The EAL of 2500 AEC fractions is a result of the round-off and is slightly conservative. The actual AEC fraction is 2941 (50 AEC / 0.017).

It is also noted that the Ar-41 AEC value of 1 E-8 uCi/ml was rounded off to one significant figure from the value reported in publication EPA 520/1-88-020. The reported value in EPA 520/1-88-020 converts to 1.42 E-8 uCi/ml (2.17 E-10 Sv/h per Bq/m^3). Appendix B of 10 CFR 20 is based on publication EPA 520/1-88-020.

$$\frac{0.1 \text{ rem/y}}{(2.17\text{ E-10 Sv/h per Bq/m}^3)](8760 \text{ h/y})(100 \text{ rem/Sv})(3.7\text{E4 Bq/uCi})(1\text{E6 ml/m}^3)}$$

$$= 1.42\text{ E-8 uCi/ml or} \sim 1\text{ E-8 uCi/ml}$$

Using the EPA 520/1-88-020 value gives a lower AEC fraction:

32 AEC fraction = [2.7E-5 uCi/ml over 24 h/1.42 E-8 uCi/ml per AEC] [0.017]

The 24 h average calculation provided in the Appendix B had errors. Corrections were made regarding the 2nd and 3rd release scenarios for Ar-41. Peak and average concentrations were calculated. Appendix B is now Attachment 2. The highest 24 h average concentration is 3 E-5 uCi for the 3rd scenario which assumes 24 hours of Pneumatic (PN) system operation followed by release of Ar-41 at the saturation activity into the reactor bay from a 10.4 cubic feet beam tube. This is the largest beam tube in the reactor facility. This Ar-41 concentration is close to that which was previously used.

It is noted that the X/Q value used was for Class F weather stability at a wind speed of 1 mph in the AEC fraction calculation. Upon review of ANSI/ANS-15.7-1977, Class F weather stability at a wind speed of 1 m/s may be used. 1 m/s equates to approximately 2.24 mph.

With this wind speed correction, the 24 h average AEC fraction is reduced by a factor of 2.24 and is calculated as follows:

$$26 \text{ AEC fraction} = \frac{[\text{ 3E-5 uCi/ml/1E-8 uCi/ml per AEC}](0.017 \text{ at 1 mph})}{(2.24 \text{ mph})}$$

Attachment 2 has been revised for a wind speed of 1 m/s. The X/Q value of 0.017 at 1 mph gives a X/Q value of 7.6 E-3 s m^{-3} for a wind speed of 1 m/s (or 2.24 mph). Use of the 1 m/s wind speed is well below average wind speeds given in the FSAR and in the weather data provided in the US EPA Comply Code for Raleigh, NC. This change is made because it is more realistic and prevents premature or inadvertent activation of the Emergency Plan. With this change, the Notification of Unusual Event EAL for airborne effluent is not exceeded.

Using the X/Q value of 7.6 E-3 s m^{-3} at a wind speed of 1 m/s has resulted in a new basis for the airborne radiation monitors set points. TS 3.5 has been modified based on this change as described in the revised Attachment 1.

In conclusion, differences in the Ar-41 concentration vs. the EAL and TS 3.5 set points given in Appendix B previously were due to rounding off of the X/Q value. The EAL of 50 AEC fractions was met. Upon re-analysis of the Ar-41 source term, a slightly higher 24 h average concentration was determined as shown in Attachment 2. Upon using the weather stability and wind speed recommended in ANSI/ANS-15.7-1977, the X/Q value is decreased by a factor of 2.24. These changes have resulted in a calculated 24 h AEC fraction of 26 for the Ar-41 source term. Therefore, the statement that the release is within the "Notification of Unusual Event" EAL is correct.

3. The analysis in Appendix B treats noble gases differently from other radionuclides. AEC values given in 10 CFR 20, Appendix B are equivalent to the concentration of a specific nuclide that would result in an annual total effective dose equivalent of 0.05 rem, and ANSI/ANS-15.16 and NUREG-0849 do not differentiate between noble gasses and all other radionuclides. Please justify treating these categories of nuclides differently.

Noble gases are an external hazard, so the hazard is independent of age and the AEC in 10 CFR 20 is based on 100 mrem/y. For radionuclides other than noble gases, an inhalation hazard exists and the annual effective dose limit of 50 mrem applies. The 50 mrem limit applies to adults and includes an age dependent factor of 2 for children thereby making 50 mrem to an adult equivalent to 100 mrem to a child. The note provided in the original Attachment 1 gave this explanation. Refer to NUREG 0849 Appendix I errata sheet and NRC Information Notice 97-34and the discussion given in Appendix B of 10 CFR 20 (Attachments 3 and 4).

In the original Attachment 1, set points provided in the original amendment were based on EAL criteria and were correct. EAL based set points were determined based on a wind speed of 1 mph for an accident.

However, EAL based set points have been revised using a wind speed of 1 m/s (2.24 mph) as recommended in ANSI/ANS-15.7-1977 are no longer limiting. Therefore, TS 3.5 has been changed to 5000 AEC for both the stack gas and particulate channels based on 10 CFR 20 Appendix B limits. Refer to the revised Attachment 1 for details.

4. Revision 7 of the Emergency Plan cited a "worse case" Ar-41 concentration in the reactor bay of 2.0E-4 μCi/ml. Please explain the reason for the significant decrease in the "worse case" Ar-41 concentration cited in Revision 8 (2.7E-5 μCi/ml). Please explain the basis for the PN tube volume length of 61 cm.

The value of 2 E-4 uCi/ml for Ar-41 given in Revision 7 of the Emergency Plan was based on the peak activity concentration in the reactor exhaust from operation of the PN system reported in Revision 9 of the SAR. This estimate was applied to the reactor exhaust duct, not the reactor bay, and assumed continuous PN system operation for 24 hours. The PN system empties directly into the exhaust duct rather than the reactor bay. Actual measured data for Ar-41 exhausted from the stack indicates a peak Ar-41 concentration of approximately 1.5 E-6 uCi/ml prior to the ventilation system modification. With the ventilation system modification, the estimate is 5 times higher and recent measured data is in agreement with this estimate, i.e. the Ar-41 concentration from PN operation is now approximately 8 E-6 uCi/ml.

The discrepancy in peak Ar-41 concentration from operation of the PN system was noticed in Revision 8 of the Emergency Plan. As stated in the response to question 2, the Ar-41 source term has been re-evaluated and new calculations are given in Attachment 2. The accident scenarios evaluated assume either a PN tube rupture or a beam tube opening with release of activity into the reactor bay and prior PN system operation. The highest 24 hour average concentration is 3 E-5 uCi/ml and is associated with the opening of an empty beam tube containing air saturated with Ar-41 preceded by continuous operation of PN system at the measured peak Ar-41 concentration of approximately 8 E-6 uCi/ml.

Regarding the PN tube length, the active region of the reactor fuel is 24 inches, or 61 cm. It was conservatively assumed that the peak thermal neutron flux of 1 E13 $cm^{-2}s^{-1}$ extends over the entire 61 cm length. Actually, the flux profile decreases appreciably at the top and bottom of the reactor fuel and is orders of magnitude lower at distances of a few cm away from the reactor fuel region. For the current reactor core configuration, the peak PN thermal neutron flux is approximately 8 E 12 $cm^{-2}s^{-1}$. Absolute flux measurements in the PN tube at different locations have not been measured since the PN shuttle (sample holder or "rabbit") goes to a fixed location and is approximately 15 cm in length.

A similar experimental facility, the rotating exposure ports, have measured thermal neutron fluxes ranging from ~ 35% to ~85% of the peak thermal neutron flux over the length of active fuel, with < 35% of the peak thermal neutron flux in the upper regions of the active fuel where the control rods are present.

This assumption of a constant, steady thermal neutron flux is considered to be conservative since it does not take credit for the actual flux profile.

Note that the saturation activity of Ar-41 calculated for the PN tube is a factor of 23 times lower than that associated with the largest beam tube (4.2 E4 uCi vs 9.9 E5 uCi), or about 4% of that for the largest beam tube. This makes the beam tube more of a concern than a bolus of air from the PN system containing Ar-41 at the saturation activity.

5. Calculations of the "worse case" Ar-41 concentrations on page 44 of Appendix B use a value of 2.25E9 ml for the volume of the reactor building, while calculations of event durations on page 49 of Appendix B use a value of 2.4E9 ml for the volume of the reactor building. Please justify the use of the two different values, or make the value consistent throughout the analysis.

2.25 E9 ml is the estimated free air space in the reactor building while 2.4 E9 ml is the estimated total volume of the reactor building in Appendix B. The difference is due to equipment and shielding. The free air space of 2.25 E9 ml was used in Appendix B to determine concentrations. The value of 2.4 E9 ml was used in Appendix B to determine exhaust times (and time for an air exchange).

TS 5.2.a requires a minimum free air volume of 2.25 E9 ml.

The 24 hour average concentrations and associated total doses would not be changed by using 2.25 E9 ml or 2.4 E9 ml to determine the concentrations and air exchange times since the accidental releases occur over a short time and several air changes occur over 24 hours. This results in essentially all of the activity being exhausted and all of the dose being received within 24 hours from the reactor building.

Therefore, use of the two volumes is inconsistent, but conservative; i.e. concentrations are larger for a smaller volume and air exchange time is longer for a larger volume. As this is conservative, the next revision to the Emergency Plan (which includes Appendix B) will use the same volume given in the SAR.

6. Specification 1.2.9.a, Tried Experiment

Your justification says this was changed to make it consistent with ANSI/ANS-15.1-1990, however this definition is not in that standard. Please explain.

The comment is correct. No definition for a tried experiment is given in ANSI/ANS-15.1-1990. However, the definition for a tried experiment has not been changed from the one currently given in the approved revision of the Technical Specifications (Specification 1.9.a). Therefore, this item has is removed from the change description.

7. Specifications 1.2.19 and 1.2.27

In these two definitions for Reactor Operator and Senior Reactor Operator the reference, 10 CFR 50.55, is wrong. Please correct.

The reference has been corrected in the revised Technical Specification amendment.

8. Figure 2.2-1, Power-Flow Safety Limit Curve

In the proposed TS this figure does not show the Operating Envelope as the same figure did in the previous version of TS. Also, in the proposed version of the figure the labeling does not specify the Safety Limit Specifications that the pool level shall be 14 feet or greater and the pool temperature shall not be greater than 120 °F. Since Specification 2.2.1.a. references the operating envelop in the figure the fact that it is missing may cause confusion. In addition, the title for the figure contains a typo with the word "full" rather than "flow." Please clarify your intentions with regard to the labeling on this figure.

This comment applies to Figure 2.1-1. The changes noted in this statement have been made to Figure 2.1-1 in the revised Technical Specification amendment.

9. Specification 3.2.f

Apparently the version of the SAR, as supplemented, that was part of the license renewal application used 2900 pcm since NUREG 1572, "Safety Evaluation Report related to the renewal of the operating license for the research reactor at North Carolina State University," refers to this value in section 13.9. To what version of your SAR are you referring in the justification of this change?

Secured experiments have a limit of 1590 pcm (1.59% $\Delta k/k$) as reported in the SAR Section 13 rather than 1600 pcm. The value of 1.59% $\Delta k/k$ was rounded off to 1.6% $\Delta k/k$, or 1600 pcm, when the Technical Specifications were revised in 1997. The change to Specification 3.2f corrects this minor difference to agree with the SAR Section 13.2.2.1. Applicable SAR Sections 13 dated August, 1996 are given in Attachment 5 indicating that the secured experiment reactivity limit is 1.59% $\Delta k/k$. Adding the movable and non-secured experiments reactivity limit to 1590 pcm gives 2890 pcm rather than 2900 pcm.

10. Specification 3.3-1 g

Provide discussion and assurance that the risk to the health and safety of the public will not significantly increase with the removal of the Manual SCRAM from the Specification for the Pool Water Temperature Monitoring Switch.

This specification was changed to eliminate a required manual SCRAM to a situation that may be effectively managed without shut down of the reactor. The alarm function remains unchanged making the Reactor Operator aware of the situation and prompting a response. Response to this alarm by the Reactor Operator is controlled by procedures and training. For example, the secondary cooling system may be adjusted, restored, or initiated to maintain a pool temperature not in excess of 117 F. If the situation is not resolved or uncertain, appropriate action is to shut down the reactor since the specifications on pool temperature monitoring and temperature limits (LSSS specification 2.2.1) for operation of the reactor would not be met. If pool temperature reaches the set point of $\leq$ 117 F, there is an automatic SCRAM under TS 3.3-h. The manual SCRAM button (specification 3.3-j) remains making a manual SCRAM available to the Reactor Operator for this situation.

Because the alarm is not changed and a response is required to meet pool temperature monitoring and pool temperature limits and there is an automatic SCRAM at the pool temperature limit, the risk and safety of the public are not affected by the proposed change.

11. Specification 3.3.l

Provide discussion and assurance that the risk to the health and safety of the public will not significantly increase with the removal of the Manual SCRAM from the Specification for the Over-the-Pool Radiation Monitor.

This specification was changed to eliminate a required manual SCRAM to a situation that may be managed without shut down of the reactor. The alarm function remains unchanged making the Reactor Operator aware of the situation and prompting a response. The Over-the-Pool radiation monitor may alarm for a number of reasons other than a high radiation level originating from the operating reactor. For example, this alarm may be temporarily due to an external radiation source placed to close to the radiation monitor (e.g. calibration source), sample or component removal from the reactor pool, or mechanical shock or brief electrical disruption to the monitor. If the monitor response is due to the operating reactor or rapid loss of coolant or if the monitor operational status is in question, the Reactor Operator would shut down the reactor or initiate a manual SCRAM. The manual SCRAM button (specification 3.3-j) remains making a manual SCRAM available to the Reactor Operator for this situation. Radiation safety procedures would be followed to control radiation exposure to occupational workers and members of the public. Members of the public are not allowed to enter radiation fields in excess of 2 mrem per hour and would be evacuated from the immediate area and possibly from the reactor building if such radiation fields were present. Initiation of the facility Emergency Plan may occur in response to this alarm if it is determined that the Emergency Action Level has been exceeded or is imminent.

Because the alarm is not changed and a response is required to meet radiation safety and emergency plan requirements, the risk and safety of the public are not affected by the proposed change.

12. Specification 3.7.e(iii)

Justify that this change will not reduce the oversight of experiments involving the use of explosive materials.

All new experiments are approved by the Radiation Safety Committee (RSC) and Reactor Safety and Audit Committee (RSAC) in accordance with Specification 6.2.3. Additionally, Specification 6.5 requires review and approval by RSC and RSAC for untried experiments and members of the reactor staff for tried experiments.

Chemical and physical hazards in any laboratory at the University must meet requirements given in the University Environmental Health and Safety Manual (EHSM), which is based on 29 CFR (OSHA) and other applicable codes, regulations, and commitments (e.g. insurance).

Explosive materials are not allowed in the reactor. Use of a container or encapsulation is required for potentially explosive materials. Potentially explosive materials are not defined, but it is reasonably concluded that placement of an explosive material in a container or capsule would be insufficient. The explosive material must be rendered to make an explosion highly unlikely in some way to be considered "potentially" explosive. For example, by combining the explosive material with another in a matrix or mixture or compound, ensuring that an ignition or heat source is not present, and use of double encapsulation may satisfy the limitation on experiments involving potentially explosive materials.

Reactor procedures require identification of materials used in experiments and any associated hazards. The experimenter and reactor staff would collaborate in this matter. The Reactor Operator, Designated Senior Reactor Operator, Manager of Engineering and Operations, and Reactor Health Physicist are all involved in determining that the experiment is in compliance with the facility Technical Specifications and University laboratory safety requirements.

Oversight of experiments with potentially explosive materials is provided by the involvement of safety committees, university safety personnel, and reactor staff in accordance with the Technical Specifications, facility procedures, and University policy.

13. Specification 3.7.e(v)

The change in the wording of specification is not included in TS change analysis and justification. Provide justification for the change. In particular, justify how this change does not reduce the oversight of experiments involving the use of cryogenic, flammable, or toxic materials.

In this specification, the phrase "approved by the Radiation Safety Committee" was replaced with "approved as specified in Specification 6.2.3". Specification 6.2.3 applies to the review and approval of experiments by the Radiation Safety Committee (RSC) and Reactor Safety and Audit Committee (RSAC). Additionally, Specification 6.5 requires review and approval by RSC and RSAC for untried experiments and members of the reactor staff for tried experiments.

Chemical and physical hazards in any laboratory at the University must meet requirements given in the University Environmental Health and Safety Manual (EHSM), which is based on 29 CFR (OSHA) and other applicable codes, regulations, and commitments (e.g. insurance).

Reactor procedures require identification of materials used in experiments and any associated hazards. The experimenter and reactor staff would collaborate in this matter. The Reactor Operator, Designated Senior Reactor Operator, Manager of Engineering and Operations, and Reactor Health Physicist are all involved in determining that the experiment is in compliance with the facility Technical Specifications and University laboratory safety requirements.

Oversight of experiments with cryogenic liquids, flammable materials, or highly toxic materials is provided by the involvement of safety committees, university safety personnel, and reactor staff in accordance with the Technical Specifications, facility procedures, and University policy.

14. Specification 3.8

You have stated that the changes being requested are consistent with the definition of fueled experiments given in TS 1.2.9.e, i.e. any fissionable material, and therefore do not limit the type of fuel used assuming it is allowed by the R-120 license.

The definition is not a specification in the sense that it places or doesn't place limits on an action or actions. The specification on fueled experiments is TS 3.8. The TSs and the regulations are the requirements placed on experiments. The basis is not part of the TSs, however, the analyses in the SAR, from which the TSs are derived, provide the benchmark against which the 10 CFR 50.59(c)(2) criteria are assessed.

You have stated that using the MHA consequences (maximum off-site dose given in the FSAR for the fuel handling accident is well below 1 mrem) is very restrictive when compared to 10 CFR Part 20 limits.

However, without a less restrictive analysis in the SAR the MHA methodology and consequences are what must be used as the criteria in doing a 10 CFR 50.56 review. Stating that credible failures of fueled experiments are not allowed to exceed 10 CFR 20 limits in the TS does not release you from doing the 10 CFR 50.59 review.

You have also stated that credible failures of fueled experiments depend on many factors and each case needs to be analyzed, e.g. in-pool irradiation vs. out-of-pool irradiation, fuel chemical formulation, fuel enrichment, type of encapsulation used and other barriers to the escape of fission products (coatings, fuel shape and size, cladding, gap inventory, fraction of equilibrium).

However, 10 CFR 50.36(b) states that the TSs are derived from the analyses and evaluations made in the SAR. A TS that uses the term "credible failure" must be derived from a specific experiment and analysis in the SAR. The staff cannot make a determination of the acceptability of the credibility of an experiment failure without evaluating a fully described experiment and its detailed analysis in the SAR.

TS 3.8 has been modified to limit the fissionable material to U-235 and includes limiting doses to less than10 mrem total effective dose-equivalent for members of the public and less than 500 mrem total effective dose-equivalent and less than 5000 mrem total organ dose-equivalent for occupationally exposed workers. The public dose is based on 10 CFR 20.1101(d) and the occupational dose limit is based on 10 CFR 20.1502. The occupational doses are therefore restricted to 10% of the applicable limits. The proposed limit of 10% of the public dose limt, or 10 mrem, is below the dose associated with emergency declaration of a "Notification of Unusual Event".

The specification 3.8.e.v from the original amendment request has been removed. ALARA is required by reactor license conditions, reactor health physics procedures, and University policies on experiments. Refer to Attachment 6 for details on the fueled experiment analysis.

The proposed change includes a review of the fueled experiment before it is conducted. This review would include a 10 CFR 50.59 screening and, if necessary, a 10 CFR 50.59 evaluation and SAR updates. Specifications 6.2.3 and 6.5 apply regarding the review.

15. Specification 6.1.1

The advice and liaison lines have been removed with the proposed Figure 6.1-1. Provide discussion of the changes in the organizational structure. Include changes in responsibility, authority, and lines of communications, e.g. what are the interface inter relationships between the Reactor Health Physicist, the Nuclear reactor Program personnel, the Reactor Safety Committee, and the Reactor Safety and Audit Committee. Discuss the change from the Reactor Safety Committee reporting to the Reactor Safety and Audit Committee which in turn reports to the Chancellor and the proposed structure where they both report to the Chancellor independently. Discuss how the proposed changes will affect safety, independence, and oversight.

Advice and liaison lines in Figure 6.1-1 have been added. The notes for Figure 6.1-1 have been changed to indicate direct communication and advice and liaison as follows:

NOTES: Line of direct communication ———

Line of advice and liaison — — —

The last note indicates that communication occurs between several parties regarding reactor operations, experiments, radiation safety, and regulatory compliance.

Also, RSC and RSAC membership by positions knowledgeable of reactor operations and radiation safety have been added to Figure 6.1-1.

As stated in the proposed Specification 6.1:

- The NRP Director works through the Manager of Engineering and Operations to monitor daily operations and with the Reactor Health Physicist to monitor radiation safety practices and regulatory compliance.
- The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility.
- Reactor Health Physicist (RHP) reports to the Head, Department of Nuclear Engineering and serves both the NRP and Department of Nuclear Engineering.

- Communication on reactor operations, experiments, radiation safety, and regulatory compliance occurs between the NRP, RHP, Reactor Safety and Audit Committee, Radiation Safety Committee, and campus Radiation Safety Division as described in these Technical Specifications and facility procedures.

Changes to the line organization include a more active role in the administration of the facility by the Nuclear Reactor Program (NRP) Director and consolidates the roles of the Associate Director and Reactor Operations Manager positions in a new position titled "Manager of Engineering and Operations" (MEO). No responsibilities or functions are lost by the proposed changes.

Responsibilities, authority, and lines of communication are not changed by the proposed Specification 6.1.

As stated in the proposed Specification 6.2:

- The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at the University are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all experiments involving the potential release of radioactive material conducted at the University and provides oversight of the University Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

- RSAC has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, Technical Specifications, and all applicable regulations. RSAC performs an annual audit of the operations and performance of the NRP.

  RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety and individuals from the line organization shown in Figure 6.1-1.

  Specification 6.2.1a has been revised as follows:

  RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety. RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

  The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center serve on RSAC.

- A quorum of RSC or RSAC shall consist of a majority of the full committee membership and shall include the committee Chair or a designated alternate for the committee Chair. Members from the line organization shown in Figure 6.1-1 shall not constitute a majority of the RSC or RSAC quorum.

Distribution of RSC summaries and meeting minutes shall include the RSAC Chair and Director of the Nuclear Reactor Program.

- A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to the Dean of the College of Engineering, the Nuclear Engineering Department Head, the Director of the Nuclear Reactor Program, the RSC Chair, Director of Environmental Health and Safety, Manager of Engineering and Operations, and the RSAC Chair prior to the next scheduled RSAC meeting.

  Deficiencies uncovered that affect reactor safety shall be immediately reported to the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC.

  A summary of the annual audit made by the RSAC, including any recommendations, is forwarded to the RSC.

  Changes to the campus Radiation Safety Committee (RSC) and Reactor Safety and Audit Committee (RSAC) are proposed regarding membership, items reviewed, and RSC meeting frequency.

With this change, RSC reviews experiments involving the potential release of radioactive material and changes to the facility license except those containing safeguards information. RSAC continues to review all items and provides reports to the RSC.

  All items currently listed in Technical Specifications (TS) continue to be reviewed by either RSAC or both RSAC and RSC with this TS amendment.

RSC composition and qualifications were reworded. RSC membership reflects the on-campus user community of radiation devices and radioactive materials and university radiation safety administration. The RSC description has been simplified while ensuring that minimal qualifications and membership are maintained to permit a thorough and appropriate review of items associated with the nuclear reactor. RSC membership has been revised to require membership by the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

RSAC continues to be comprised of persons knowledgeable in reactor operations and continues to report its activities to the RSC. Areas of expertise in the RSAC membership description have been expanded to include mechanical engineering.

  Communications from the RSAC to the RSC continue as before with the proposed TS, with the exception that some items (e.g. design and procedure changes) no longer require RSC approval.

  As stated in proposed Specification 6.3:

- The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6.1-1.

The proposed changes in Specification 6.1 are summarized as follows:

- Responsibilities as currently stated in TS are maintained. The MEO and Director, NRP have been assigned duties performed by the former Associate Director, NRP position.
- Titles and qualifications for personnel in organizational levels 1 through 4 and the RHP were changed and are consistent with ANSIANS-15.1-1990. No areas of responsibility have been lost.
- RSAC is a stand-alone committee and continues to report its actions to RSC and have communications with RSC.

None of the proposed changes affect safety, independence, or oversight of NRP activities by the RSAC, RSC and RHP. RSAC, RSC, and the RHP continue to have independence in conducting reviews of the NRP.

16. Specification 6.1.3.a

    Verify that the reference is 10 CFR 55 rather than10 CFR 50.55.

Specification 6.1.3.a. has no reference to 10 CFR. This specification is similar to the current specification 6.1.2 a with minor wording changes. There is no change in the intent or meaning of the minimum staffing required when the reactor is not secure.

17. Specification 6.2

    One change to Figure 6.1-1 is that it now shows the RSAC no longer reports to the RSC. This is reinforced by the removal of the statement that the RSC performs final review of the actions of the RSAC. Discuss the reason for that change and the safety implication.

ANSI/ANS-15.1-1990 states that a method for the independent review and audit of the safety aspects of reactor facility operations shall be established to advise management. ANSI/ANS-15.1-1990 states a minimum of three members on the review and audit group or three members for review and one member for auditing if separate groups are used. ANSI/ANS-15.1-1990 states the review and audit group shall report to Level 1 management (Level 1 is the individual responsible for the facility license or charter).

RSC membership reflects the on-campus user community of radiation devices and radioactive materials and non-faculty members who are knowledgeable in nuclear science or radiation safety. The RSC description has been simplified while ensuring that minimal qualifications and membership are maintained to permit a thorough and appropriate review of items associated with their area of responsibility. Positions knowledgeable about reactor operations and radiation safety have been included in the RSC membership. However, not all RSC members may not meet the ANSI/ANS-15.1-1990 requirement for collectively representing a broad spectrum of expertise in the appropriate reactor technology.

RSAC continues to be comprised of persons knowledgeable in reactor operations consistent with ANSI/ANS-15.1-1990. RSAC continues to report its activities to the RSC. Areas of expertise in the RSAC membership description have been expanded to include mechanical engineering.

Based on ANSI/ANS-15.1-1990 and the expertise of the committee members, RSAC was made into a separate, independent committee that reports directly to the University Chancellor (license holder). This direct communication with the Chancellor may be beneficial in addressing the needs of the reactor facility since communications are no longer underneath the RSC. The current TS states that items are reviewed and approved by RSC or by referral to RSAC thereby recognizing that RSAC is a separate and independent safety review committee with specific knowledge about nuclear reactor operations.

In the proposed TS 6.2.3, only RSAC review and approval is required for:

- Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and Technical Specification requirements.
- All new procedures and major revisions having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.
- All new experiments or classes of experiments that could affect reactivity.
- Proposed changes to the facility license containing safeguards information.

All other items in the proposed TS 6.2.3 require review and approval by both RSC and RSAC.

In the proposed TS, the annual audit is performed by RSAC and RSC receives the report and any recommendations. Deficiencies uncovered that affect reactor safety are immediately reported to the RSC, as well.

From the above discussion and TS changes it is concluded that:

- RSAC alone may fulfill the requirement for a review and audit committee as described in ANSI/ANS-15.1-1990.
- Direct communication by RSAC to the Chancellor may be beneficial in addressing the needs of the reactor facility since communications are no longer underneath the RSC.
- All items currently listed in TS continue to be reviewed by either RSAC alone or both RSAC and RSC with the exception of licensing items containing safeguards information (SGI).
- In the proposed TS, only RSAC reviews SGI. This arrangement satisfies the review requirement while limiting access to SGI, thereby protecting SGI.
- RSC reviews those items affecting the facility license and experiments involving the potential release of radioactive materials. RSAC continues to review all items and provides reports to the RSC. This arrangement is consistent with ANSI/ANS-15.1-1990.

- RSC receives the audit report information from the RSAC.
- In the proposed TS, RSAC informs RSC of its actions and RSC and RSAC are allowed to communicate directly with each other. Furthermore, RSC may require additional actions by the RSAC or NRP. RSAC, RSC, and the RHP continue to be independent of the NRP as shown in Figure 6.1-1.

Therefore, the proposed TS changes to Specification 6.2 regarding the review and approval function are considered to be consistent with safe operating practices at research reactors.

18. Specification 6.2.1 a

    Is there still a requirement for a permanent member on the RSC from the Radiation Safety Division of the Environmental Health and Safety Center? Discuss the reason for the changes and the safety implications of them.

    Specification 6.2.1a has been revised to require membership by the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP as follows:

    RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety. RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

The reason for change to the membership of the RSC in the proposed TS was in recognition that the RSC is defined in the University Radiation Safety Manual and University Faculty Handbook. In the proposed TS, a general description of RSC membership was made. The University Radiation Safety Manual indicates that while RSC serves as the primary body for the institution in all matters related to the use of radioactive materials and radiation-producing devices while RSAC provides specialized oversight of the nuclear reactor and has the primary function to assure that the reactor is operating in compliance with the reactor license and applicable regulations.

RSAC has a member from the Radiation Safety Division of the Environmental Health and Safety Center. The last paragraph of specification 6.2.1 was reworded as follows:

> The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center are permanent members of RSAC.

It is concluded that the RSC membership as described in the proposed TS is capable of performing a thorough and appropriate review of those items assigned to the RSC in the proposed TS.

19. Specification 6.2.1 b

The NRP Director is listed as being a faculty member. The minimum qualifications that are listed in TS 6.1.1 do not specify that the position is a faculty position. Clarify your intention?

The purpose of TS 6.1.1 is to define the qualifications and responsibilities of the NRP Direction regarding the NRP organization and the purpose of TS 6.2.1 is to specify requirements of the RSAC membership. However for consistency, the description in TS 6.1.1 has been changed to specify that the NRP Director is a faculty member.

20. Specification 6.2.2 a

The term "University Management" is not defined. Figure 6.1-1 shows the committees reporting to the Chancellor. The existing TSs state that the office of the Vice Chancellor for Finance and Business and the Provost. Considering TS 6.1.2, which specifies how responsibility may be delegated, clarify the proposed change, discuss the purpose of the change, and justify it.

The current TS state the current position at the University who is responsible for making committee appointments. The University Administration, or management, is currently and will continue to be responsible for this function since safety committees are considered to be an administrative function. Responsibilities and titles within the University administration may change without NRP knowledge. Therefore, the general term "University Management" is used. To avoid any confusion, "University Management" has been added to the TS definitions and indicates that this includes the Chancellor or Office of the Chancellor or other University Administrator(s) having authority designated by the Chancellor or as specified in University policies.

21. Specification 6.2.2

It is not clear from the proposed TS that there is a requirement to have an established charter for the RSC and RSAC as recommended in the ANSI/ANS-15.1-1990 guidance. What is the purpose of TS 6.2.2? Please discuss.

The current specification 6.2.2 titled "RSC and RSAC Composition and Qualifications" included committee rules. Therefore, the old specification was changed to have one specification on "composition and qualifications" and another specification on "charter and rules" following the format given in ANSI/ANS-15.1.1990. However as noted in the comment, there is no need for a charter. To avoid confusion, the word "charter" has been deleted from the title of Specification 6.2.2.

22. Specification 6.2.2 b

TS 6.2 states that the RSC is informed of the actions of the RSAC and may require additional actions by RSAC and the NRP. With the meeting schedule dictated by the State broad scope license (TS 6.2.2.b) and the possibility of mismatched meeting schedules how is it ensured that the RSC is informed and responds in a timely manner to the actions of the RSAC?

Meeting of RSC and RSAC are coordinated currently and will continue to be so to allow timely review and approval of items. In the proposed specifications, it is anticipated that fewer items will be sent to RSC. In the event that there is a mismatched meeting schedule or untimely delay in the review and approval of items, the proposed and current specification state that RSC and RSAC may meet upon calls of the respective Chairs. In summary, minimal meeting frequency of the two committees are given in this specification with recognition that additional meetings may occur if there is a need.

23. Specification 6.2.3

There is no statement in TS 6.2.3 that the Manager of Engineering and Operations shall receive a summary of RSAC meeting minutes as mentioned in your justification. Is that still your intention?

The Manager of Engineering and Operations has been added to the RSAC meeting minutes distribution.

24. Specification 6.2.3

The requirement in TS 6.2.3 of the existing TSs is that: "Recommendations of the annual audit made by RSAC are forwarded to the RSC for concurrence before being implemented." That requirement no longer appears in the proposed TS 6.2.4. Justify its removal.

The last part of the phrase in the current statement "...for concurrence before being implemented" is the reason this statement was removed. Recommendations are advice or suggestions from a knowledgeable source and are not findings or deficiencies. The NRP would consider the recommendations and may implement the recommendation if it is practical and beneficial and without delay if review and approval is not needed or if it only involves a minor procedure change. A finding or deficiency requires action on the part of the NRP and possibly RSAC and RSC. Specifications 6.2.3 and 6.2.4 discuss the audit recommendations:

The proposed TS 6.2.3 wording has been revised as follows:

> A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to the Dean of the College of Engineering, the Nuclear Engineering Department Head, the Director of the Nuclear Reactor Program, the RSC Chair, Director of Environmental Health and Safety, RSAC Chair, and the Manager of Engineering and Operations prior to the next scheduled RSAC meeting.

The proposed TS 6.2.4 wording has been revised as follows:

> The annual audit made by the RSAC, including any recommendations, is provided to the RSC.

In the proposed TS, the RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP). Furthermore, RSAC in the proposed TS is an independent review and audit group and has been given the responsibility of performing the audit. RSAC may therefore make its own recommendations and decide if follow up is needed.

> If deficiencies are uncovered that affect reactor safety, the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC are immediately informed and the NRP would take necessary action.

Therefore, the removal of the statement "Recommendations of the annual audit made by RSAC are forwarded to the RSC for concurrence before being implemented" is justified because in the proposed TS recommendations are shared with the RSC and the RSC may determine what if any action is needed by the NRP, RSAC, or RSC in response to any recommendations made in the audit.

25. Specification 6.2.3 a

The regulation 10 CFR 50.59 applies to: ...changes in the facility as described in the final safety analysis report (as updated), ...changes in the procedures as described in the final safety analysis report (as updated), ...tests or experiments not described in the final safety analysis report (as updated). It is not clear that the RSC will perform or review the results of all of those 10 CFR 50.59 reviews that involve production and release of radioactive material, and radiation protection. Please clarify the scope of their responsibility. Does the committee also review changes to the Emergency Plan? Who makes the determination as to what is reviewed by the RSC, RSAC, or both when it is not explicitly stated in the TS and or charter? Discuss the interrelationships of the facility, the RSC, and the RSAC.

RSAC reviews all changes to the facility and has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, Technical Specifications, and all applicable regulations.

The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at the University are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all experiments involving the potential release of radioactive material conducted at the University and provides oversight of the University Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

Facility procedures on the review and approval process have been written, reviewed, and approved. These procedures contain the forms, questions, and an extensive list of changes that would require a review. The list is not all-inclusive. Personnel in the organizational structure shown in Figure 6.1-1 may identify other items in need of review. A 10 CFR 50.59 screening or evaluation may not always be necessary for each change, e.g. the annual audit in TS 6.2.4. The NRP staff would initiate the change and review documents, including required 10 CFR 50.59 documentation. Upon review by the NRP staff and RHP, the change information and documents are submitted to RSAC.

RSAC reviews the change and the associated 10 CFR 50.59 documentation. If the review indicates that a license amendment, revision to licensing document (including the R-120 license, Emergency Plan, and Quality Assurance Program for Radioactive Material Shipments), or a TS amendment is needed, the change and the associated 10 CFR 50.59 documentation are forwarded to RSC for review.

In the proposed TS, only RSAC reviews SGI. This arrangement satisfies the review requirement while limiting access to SGI, thereby protecting SGI.

If the change and the associated 10 CFR 50.59 documentation indicate that the change is in compliance with the existing license and TS, RSC is informed of the RSAC review and approval actions. RSC may may require further action by the NRP and/or RSAC.

Specifications 6.2.3 and 6.5 address experiments. If the review involves an experiment, the experiment must be classified as tried or untried based on Specification 6.5. If the experiment change is classified as tried and does not increase radiation dose, radioactivity production, or release of radioactive material, then the experiment may be approved by RSAC alone. If the experiment change is classified as tried and increases radiation dose, radioactivity production, or release of radioactive material, then the experiment would be reviewed by both RSAC and RSC. If the experiment change is classified as untried, then the experiment would be reviewed by both RSAC and RSC.

The RHP and a member from the Radiation Safety Division are permanent members on RSAC and have extensive knowledge about the University and Reactor Radiation Protection Programs. If there is a non-conservative discrepancy in the reactor radiation protection program, the change would be discussed and a determination made by the RHP, Radiation Safety Officer, and possibly others (NRP Director, RSAC, RSC) about reviewing the change as is, revising the change to agree with the University radiation protection program requirements or vice-versa, or canceling the change request.

If an item is in need of review and it is not explicitly clear if both RSAC and RSC need to do a review, then the matter would be discussed at RSAC and forwarded to RSC if there was any doubt.

In summary, the review and approval functions of the RSC and RSAC were changed to be consistent with ANSI/ANS-15.1-1990 and the respective areas of expertise and responsibility of the two committees. The changes clarify items reviewed by the two committees. All items currently listed in TS continue to be reviewed by either RSAC or both RSAC and RSC with this TS amendment.

26 Specification 6.2.3.a

The existing TS 6.2.3 lists the items that "...shall be reviewed and approved by the RSC or by referral to the RSAC, as needed:.." This implies a dominate role for the RSC as shown and stated more explicitly in the existing Figure 6.1-1 and TS 6.2.1, and TS 6.2.4. Discuss how the interaction of the two committees will change if the TSs are approved as proposed. Discuss how it is assured that safety and oversight will not be diminished.

RSAC has been recognized as the review and audit group for the reactor facility in the current and proposed TS. RSAC is currently a subcommittee of RSC. RSAC was created because it was recognized there are unique processes, hazards and concerns associated a research reactor that are not encountered in a typical university research laboratory. Having RSAC report to RSC and having RSC make final approval of RSAC actions is done for maintenance of licenses and policies under one committee.

In the proposed TS, the argument presented is that RSAC fulfills all of the review and audit components stated in ANSI/ANS-15.1.1990 and that RSAC is capable of functioning as a separate, independent committee with the purpose of evaluating compliance of the reactor facility with the reactor license and applicable regulations.

In the proposed TS, RSC action is required if there is a release of radioactivity, untried experiment, response to reportable events or violations or operational abnormalities, and any licensing change except those containing safeguards information,. In the proposed TS, RSC is informed of all RSAC actions and RSC may then require RSAC or the NRP to take additional actions. In the proposed TS, RSC would not vote on procedure changes or design changes or tests or experiments that do not affect the license, experiments affecting reactivity or experiments in which the release of radioactivity is decreased, annual audit, and safeguards information. (NOTE: It is assumed that a decrease in the release of radioactivity is acceptable to the RSC.)

As a result, the unique processes, hazards and concerns at the reactor are still evaluated by RSAC and RSC is still informed of all activities with the exception of safeguards information. RSAC still formally communicates with RSC. RSC may still require RSAC to take additional actions. RSC still maintains final review and approval of experiments involving the release of radioactivity, untried experiments, and licensing matters except those containing safeguards information.

In summary, the review and approval functions of the RSC and RSAC were changed to be consistent with ANSI/ANS-15.1-1990 and the respective areas of expertise and responsibility of the two committees. All items currently listed in TS continue to be reviewed by either RSAC or both RSAC and RSC with this TS amendment. Therefore, oversight is not diminished.

27. Specification 6.2.3.b

The meaning of the proposed TS 6.2.3.b.i is not clear as written. Please reword as desired to clarify its meaning and scope.

The word "or" after "safety significance" was deleted. The revised TS 6.2.3 b.i. has been reworded as follows:

b. The following items shall be reviewed and approved by the RSAC:

i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and Technical Specification requirements.

This specification combined the previous specification 6.2.3 a and b. The intent is to determine if the proposed change meets the facility license and TS requirements.

The determinations are documented using a 10 CFR 50.59 screening process and evaluation if needed. RSC would review any changes that require a change to the facility license or if there is an increase in released radioactivity. Refer to the answer to question 25 for the details on how these changes are processed by RSAC and RSC.

28. Specification 6.2.3.b

In the wording of the proposed TS 6.2.3 and TS 6.2.3.b in particular, there appears to be a screening or triage process where changes, new procedures, new experiments, "major" revisions are determined to have safety significance or not, affect reactivity or not, release radioactivity or not, and go to the RSC, RSAC, or both or neither. Describe this process and what oversight is there on this process?

The proposed TS 6.2.3 is based on the current TS 6.2.3 and statements made in ANSI/ANS-15.1-1990. The phrases "major revisions" ,"safety significance" , "affect reactitivity", "release of radioactivity" are present in both the current TS and ANSI/ANS-15.1-1990. Similarly for "changes to equipment, systems, tests, experiments or procedures". Facility procedures require all changes to be originated by the NRP. Input from other groups is considered, which would obviously include RSAC and RSC. Therefore, the process currently in place for making these determinations remains in place. Once initiated, the change is reviewed as discussed in the answers to questions 25 and 26.

29. Specification 6.2.4

In your proposed TS 6.2.4 the audit responsibility is to be changed from the RSC to the RSAC. Discuss this change and how it is assured that safety and oversight will not be diminished. Also see questions under TS 6.2.1, TS 6.2.3, and TS 6.2.3.a concerning the responsibilities of the two committees.

Current TS 6.2.4 states "The RSAC, under the authority of the RSC, shall be responsible for this audit function." The proposed TS 6.2.4 states "The RSAC shall be responsible for this audit function" is made. RSAC meets the requirements for a review and audit group as described in ANSI/ANS-15.1-1990. The current TS statement is made since RSC has final approval action on the audit. In the proposed TS, the audit is provided to RSC and RSC may take any action they deem appropriate regarding the audit.

The question of "how is it assured that safety and oversight will not be diminished" asked in this an other questions in the NRC request for additional information is slightly different than the question asked by the NRP in determining what level of review is needed to ensure safety and oversight. The NRP relied on ANSI/ANS-15.1.1990 "The Development of Technical Specifications for Research Reactors" for information to answer this question.

ANSI/ANS-15.1-1990 indicates that a review and audit group is needed and that this group may be performed by a single group or separate groups. Composition and qualifications of the group are provided in ANSI/ANS-15.1-1990 along with specifics on rules, function, and audit responsibilities. Based on the standard, the NRP concluded that RSAC meets the requirement for a review and audit group.

However, it was recognized that the RSC is necessary to maintain consistency and final approval for licensing matters, regulatory compliance issues, and University policies on the use of radioactive materials. RSC membership was revised to include positions knowledgeable in reactor operations and radiation safety. TS are proposed with limited independence for the RSAC and continue to require approval of licensing and policy level matters affecting the nuclear reactor facility. Final authority is divided between RSAC and RSC, with RSC having the ability to require additional actions by RSAC and the Nuclear Reactor Program (NRP). Therefore, it is concluded that the level of review maintained in the proposed TS ensures that safety and oversight is maintained.

30. Specification 6.2.4

The proposed specification states that a "summary" of the audit made by the RSAC will be forwarded to the RSC and not the audit report as stated in the justification section of your application. Is there a difference between the report and the summary of the audit? How is it assured that the RSC has prompt and full review of the audit and all other actions of the RSAC?

Specification 6.2.4 has been reworded as follows:

> The annual audit report made by the RSAC, including any recommendations, is provided to the RSC.

The statement about the summary of the audit report was deleted.

Meetings of RSC and RSAC are coordinated currently and will continue to be so to allow timely review and approval of items. In the proposed specifications, it is anticipated that fewer items will be sent to RSC. In the event that there is a mismatched meeting schedule or untimely delay in the review and approval of items, the proposed and current specification state that RSC and RSAC may meet upon calls of the respective Chairs. In summary, minimal meeting frequency of the two committees are given in this specification 6.2.2 with recognition that additional meetings may occur if there is a need.

31. Specification 6.2.4.c

In TS 6.2.4.c there is a missing the word, "of" after the word "methods."

TS 6.2.4c has been revised as noted in the comment as follows:

> c. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed fifteen (15) months.

This change corrects a grammatical error.

32. Specification 6.7

The proposed TS 6.7, as submitted, does not contain the change in the submission time for the annual operating report as you have stated in the justification section of the your application. Clarify your intention.

TS 6.7.4 has been changed as follows:

6.7.4. Annual Operating Report

An annual operating report for the previous calendar year is required to be submitted no later than March 31st of the present year to the Nuclear Regulatory Commission Document Control Desk. The annual report shall contain as a minimum, the following information:

The time to submit the annual operating report was extended from 60 days to 90 days. This extension allows for completion of radiation dosimeter and environmental sample analyses, which have taken up to 90 days after the end of the report period in the past. As a result, data that was not available was reported in the following annual operating report. This change will allow inclusion of all data relevant to a given period to be provided in one report.

33. Specification 6.8.3

Operator licenses are on a 6 year cycle and the training cycle is 2 years. This specification is not clear. Please clarify by suggesting a restatement of the TS.

The current specification 6.8 c and the proposed specification 6.8.3 have the same wording. To avoid confusion, specification 6.8.3 has been reworded indicating the record retention period is 6 years:

6.8.3 Records to be retained for at least one (1) license period of six (6) years:

Records of retraining and requalification of certified operating personnel shall be maintained at all times the individual is employed, or until the certification is renewed.

ATTACHMENT 1: TS 3.5 Process Radiation Monitor Set Point Changes

The reactor building ventilation system was modified for air conditioning by an approved design change in 2006. As a result, several changes were made including re-location of ventilation system equipment and air intake, stack sampling equipment, and air exhaust connection to the reactor stack. Other changes included a decrease in the normal ventilation exhaust rate and re-circulation of air in the normal ventilation mode of operation. The confinement ventilation mode was not changed. Ventilation equipment and the exhaust radiation monitors were moved from the Mechanical Equipment Room (at the basement level) to the third floor above the Control Room (inside the reactor building). Diagrams of the new ventilation system and radiation monitor locations are given on the following pages.

The normal ventilation system previously had an exhaust flow rate of 10,050 cfm. The new exhaust flow rate is 7445 cfm with 1870 cfm going to the stack and 5575 cfm being re-circulated to the reactor bay. The Pneumatic Transfer (PN) System exhaust fan will not be affected by the changes to the ventilation system and has a flow rate of 190 cfm. The confinement ventilation system remains at 600 cfm. The R-63 exhaust fan from the old reactor building remains at 12,500 cfm.

Because of the changes to the normal ventilation system, the concentration of contaminates exhausted from the PN system has increased and the air exchange rate for the reactor building has decreased.

In addition, the stack sampling system, stack sample pump, and location of the effluent radiation monitors were changed. The stack sampling system equipment was re-located to a lower background area above the Control Room. The previous sample pump was adjustable to 10 cfm while the new sample pump may operate up to 4 cfm. The stack gas, auxiliary monitor, and filter monitor detector responses are based on sample concentration. The stack particulate detector response is based on the sample activity collected by a filter. Sample flow rate of 2 cfm was used in determination of the particulate radiation monitor set point.

The filter radiation monitor was used in the exhaust duct upstream of the confinement filters and therefore indicated the concentration of radioactivity present in the reactor bay. In the new ventilation system, the filter radiation monitor has been placed in the recirculation air duct and continues to indicate radioactivity concentration present in the reactor bay. In confinement mode of operation, the re-circulation duct has no air flow so the filter monitor provides no useful data.

Activation of air in the Pneumatic Transfer (PN) System is the major source of Ar-41 released from routine reactor operations. Other sources of airborne radioactivity are associated with abnormal operating conditions, e.g. damaged fuel, failed experiment encapsulation, fuel handling accident, in which the radioactivity produced is fixed and dispersed within the reactor building. Therefore the concentration in the exhaust stack from these other sources is not affected by the ventilation system changes. However, for the PN system the activity produced is based on experimental conditions.

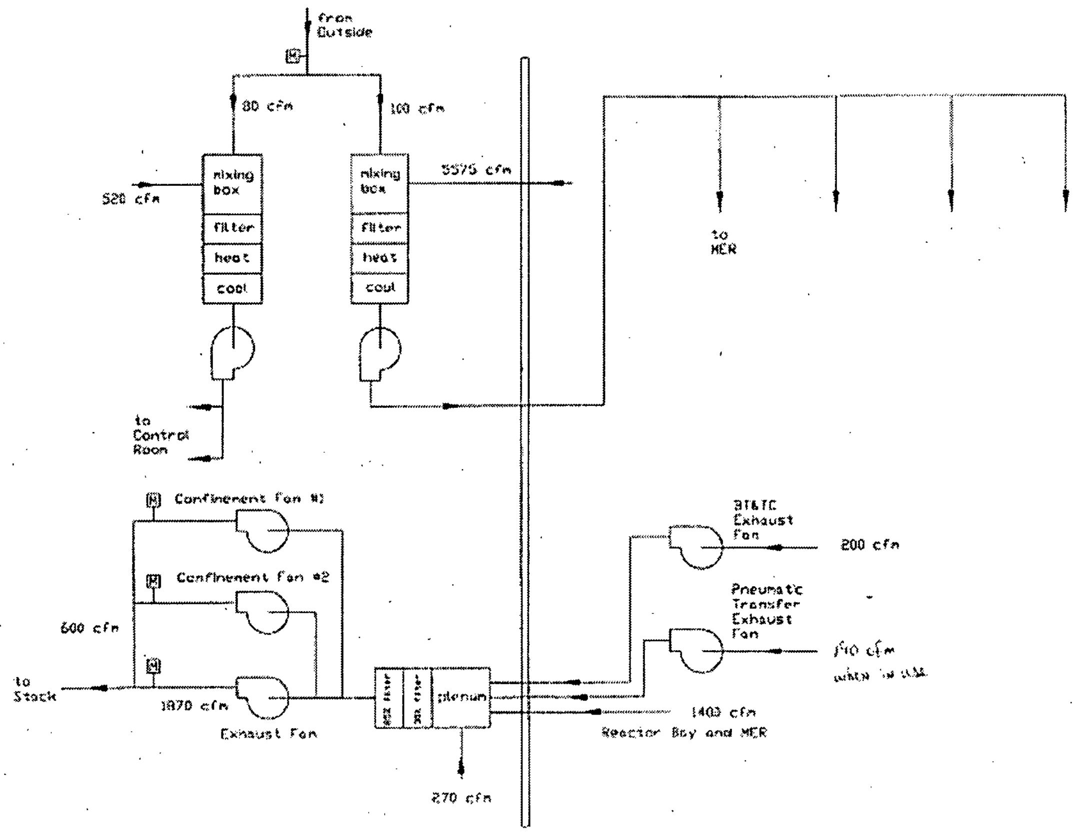
Changes made to the normal ventilation system exhaust pattern and flow rates have increased the concentration of radioactive material exhausted from the PN system by approximately a factor of 5:

Previous normal ventilation system data:
Total normal exhaust = 10,050 cfm

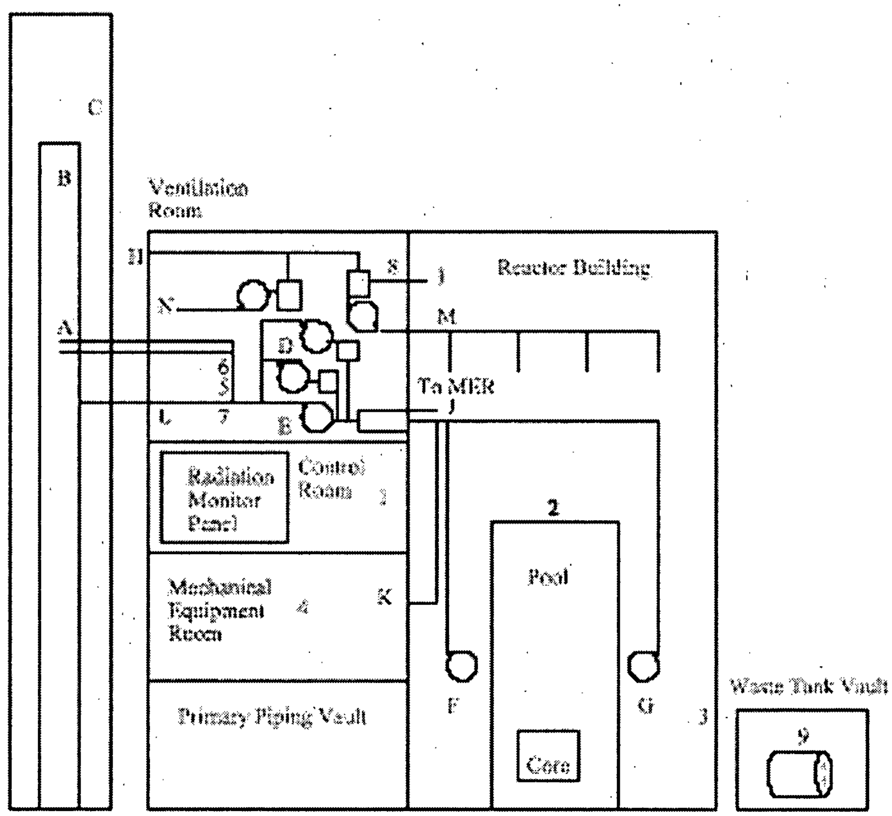
New normal ventilation system data:
Total normal exhaust = 1870 cfm

10,050 cfm / 1870 cfm = 5.37

Ventilation System Diagram:



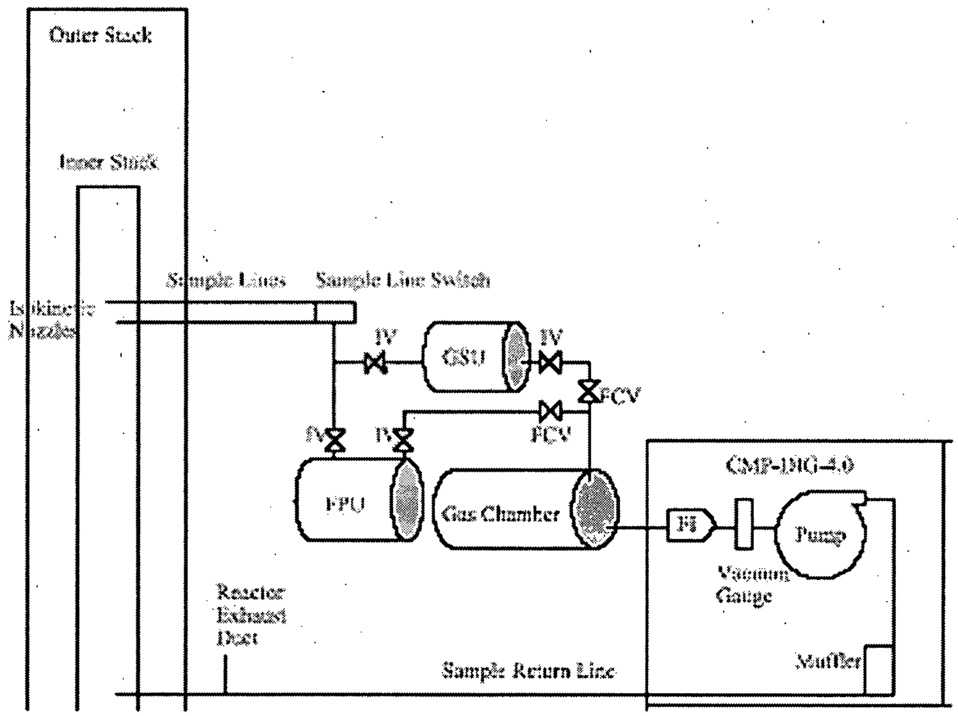
VENTILATION AND RADIATION MONITORING SYSTEM DIAGRAM



KEY:

1	Control Room	6	Stack Particulate
2	Over-the-pool	7	Auxiliary Monitor
3	West Wall	8	Filter Monitor
4	Demineralizer	9	Waste Tank Vault
5	Stack Gas		
A	Isokinetic stack sample probes	H	Air intake
B	Inner reactor stack	I	Normal ventilation re-circulation
C	Outer BEL stack	J	Reactor bay exhaust
D	Confinement fans and filters	K	MER exhaust
E	Normal Stack Exhaust fan and filters	L	Reactor stack exhaust
F	FN Blower	M	Reactor bay & MER supply
G	BT fan	N	Supply to Control Room

FIXED PARTICULATE FILTER UNIT and GRAB SAMPLING UNIT ALIGNMENT



Key: IV is Isolation Valve
FCV is Flow Control Valve
FPU is Fixed Particulate Unit
GSU is Grab Sampling Unit
FI is Flow Indicator (Rotometer, e.g.)

Previous set point calculations for effluent monitors given in TS at the time of re-licensing in 1997 were based on a factor of 1000. The factor of 1000 was based on an atmospheric dilution factor and airborne effluent concentration limit. By inspection of the atmospheric dilution factor (ADF) calculation for various locations, the minimum ADF for an occupied area occurs at the level of the stack height (30 m) during stable weather conditions (Class F). The upper levels of surrounding buildings from 150 m to 200 m are the locations of concern. ADF of 1/0.017, or 58.4 has been calculated and is listed in Appendix B of the Emergency Plan at a wind speed of 1 mph. At 1 m/s, the minimum ADF of 58.4 is 131, or approximately 1 E2.

$$ADF = 1 / [X/Q]$$

where, $[X/Q]_{x,y,z}$ is the atmospheric dispersion parameter for location (x,y,z)

$$\left(\frac{X}{Q}\right)_{xyz} = \left(\exp\left[-\frac{y^2}{2\sigma_y^2}\right]\right)\cdot\left(\exp\left[\frac{-(h-z)^2}{2\sigma_z^2}\right] + \exp\left[\frac{-(h+z)^2}{2\sigma_z^2}\right]\right)\cdot\left(2\pi U\sigma_y\sigma_z\right)^{-1}$$

where,

x is the downwind distance from the stack to receptor in m
y is the lateral distance from the plume centerline in m
z is the receptor elevation in m
σ_y is the lateral dispersion parameter in m
σ_z is the vertical dispersion parameter in m
h is the physical stack height in m, or 30 m
U is wind speed in m/s and χ/Q is in s/m^3

NOTE: Decay during transport is neglected. χ is in Ci/m^3 and Q is in Ci/s.

NUREG 0849 provides requirements for emergency classification. At the time of re-licensing, NUREG 0849 listed 10 Maximum Permissible Concentration (MPC) in airborne effluent when averaged over 24 hours for the lowest Emergency Action Level (EAL) associated with airborne releases. MPC was replaced in 1994 by the US Nuclear Regulatory Commission (NRC) in 10 CFR 20 with Effluent Concentration (EC). The methodology for deriving MPC and EC are different. Also, 10 CFR 20 changes made in 1994 included lowering the annual public dose limit from 500 mrem to 100 mrem. This discrepancy in NUREG 0849 was corrected in April, 1997 via an errata sheet. The lowest EAL associated with airborne releases in the corrected NUREG 0849 is 15 mrem or 24 hours at 100 EC for radionuclides other than noble gases or 24 hours at 50 EC for noble gases. However the reactor re-licensing documents and the Technical Specifications (TS) had been submitted prior to April, 1997. TS 3.5 set points submitted prior to 1997 for the gas and particulate channels were based on the ADF of 100 and the assumed emergency action level at 10 EC for airborne effluent, or 1000 AEC.

Since the time of re-licensing in 1997, two significant changes have occurred:

1. Constraint dose of 10 mrem per year was promulgated in the regulations. In response to this regulatory condition it is noted that radiation monitors respond essentially instantaneously and prolonged operation at abnormal levels are not typical. The dose to the public increases slowly and is monitored by periodic evaluation of the radiation monitor data. Compliance with the constraint dose level has always been met by this facility.

2. The facility ventilation system was modified to permit air conditioning and recycling of air with a lower reactor exhaust in normal mode of operation. Confinement mode of operation was not changed by this modification.

Until the time of the ventilation system modification, the TS limits were conservative and the constraint dose was not being approached, so no changes were necessary. However, with the ventilation system modifications, it was recognized that the concentration of airborne effluent would increase and have an associated higher dose rate to the public. It is therefore prudent to re-examine the TS set points based on emergency and routine operations at this time.

Public dose from routine operations and Emergency Action Levels (EAL) need to be considered in the basis for air effluent monitor set points. For the TS effluent monitors, the following three cases were evaluated for set point determination:

1. Effluent monitors provide an alarm essentially instantaneously for a given concentration or activity. The warning (alert) level will be based on abnormal levels but at a fraction of the alarm levels to allow for operator action to mitigate the release.

2. Effluent from routine operations at the 10 CFR 20 public dose limit of 100 mrem for the alarm level.

3. Airborne effluent based EAL for accidents at the alarm level.

Locations of interest are the site boundary ranging from 20 m to 50 m, surrounding buildings which range from approximately 50 m to 200 m, and the nearest residence at approximately 250 m. PULSTAR reactor stack height, h, is 30 m.

Using guidance given in ANSI/ANS-15.7-1977, Research Reactor Site Evaluation , the effective stack height for the PULSTAR reactor was determined to be only slightly higher than the actual stack height of 30 m at a wind speed of 1 m/s and with normal ventilation (1870 cfm) or confinement ventilation (600 cfm). Maximum effective H is 31.2 m vs. 30 m. As a result and for simplicity and conservatism, the effective stack height was not used.

The building height relative to the outer stack is approximately 2.5 (100 ft vs. 42 ft gives a ratio of 2.38) and therefore the above equation without modification is used. Modification of σ_y and σ_z as discussed in ANSI/ANS-15.7-1977 increases or has no affect on the parameters used in the above equation; e.g. if σ is 2 m, Σ is > 2.6 m and if σ is 100 m, Σ is 100 m:

$$\Sigma = [\sigma^2 + (0.5*A)/\pi]^{1/2}$$

where, A has a maximum value of 54 m for the BEL

Larger deviations would decrease the χ/Q value at a given receptor location. Therefore, using the above equation for X/Q without modification is conservative.

Using σ_y and σ_z data from Meteorology and Atomic Energy 1968 and the FSAR for various weather stability classes for distances from 100 m to 500 m indicates that:

- Stable weather conditions are most severe for elevated receptor locations equal to the stack height
- Unstable weather conditions are most severe for ground level locations
- Neutral weather conditions are most severe for elevations between the ground and stack height
- $(\chi/Q)_{x,0,0}^{max}$ varies with the weather stability class and occurs at downwind distances (x) greater than 100 m

For distances within 100 m, σ_y and σ_z data is not available to calculate χ/Q values. For locations within 100 m from the exhaust stack, the projected dose rate may be estimated using two line sources to represent the stack and an overhead plume. This assumption is valid because the plume from the exhaust stack does not reach the ground elevation or intersect a surrounding building at an elevated location for any weather stability class within 100 m from the exhaust stack. This assumption is conservative since it concentrates dispersed activity into a line and no credit is taken for shielding. Two locations at heights of 0 m and 20 m at a distance of 50 m have been evaluated in the Emergency Plan. The stack source term is located in the upper 20 m of the stack and constant until discharged. Effluent concentration is reduced by the Stack Dilution Factor (SDF) and wind speed (U). Dose rates from either the stack or the overhead line sources depend on total activity, receptor location (x,y,z), and source term.

These locations are illustrated below:

Overhead Plume - length of 100 m

Stack - length of 20 m • Elevated point at 20 m height and 50 m away

Stack height is 30 m • Ground point at 0 m height and 50 m away

The exposure rate, X', and line source equations used in the FSAR and NRP Calculation 91-001 are given below:

$$X' = \phi_{\gamma}(r)\, E \left(\frac{\mu_{en}}{\rho} \right) \frac{e}{W}$$

$$\phi_{\gamma}(r) = \frac{S_L}{4\pi r} \left[TAN^{-1}\left(\frac{l_1}{r} \right) + TAN^{-1}\left(\frac{l_2}{r} \right) \right]$$

where,

E represents the photon energy (J),

e the electron charge (C),

μ_{en}/ρ the mass energy absorption coefficient (m^2/kg), and

W is the mean energy expended in air per ion pair formed (J).

S is the source strength in disintegrations per second and

r is the distance to the receptor.

$\varphi(r)$, photon fluence at a point **r** normal to a line source of length **l** containing a total activity **S**

In this form, the X' will be given in C/kg/s and may be converted to R/s by the conversion factor 1 R = 2.58E-04 C/kg.

For routine operations, the following criteria apply:

a. Average weather conditions used in the Final Safety Analyzes Report (FSAR) and this calculation were taken from ANSI-15.7-1977 "Research Reactor Site Evaluation" and are given below:

Stability Class	Frequency	Wind Speed (m/s)
C	33.33%	3
D	33.33%	2
F	33.33%	2

b. Effluent monitors provide an alarm essentially instantaneously for a given effluent concentration. The warning level is based on abnormal levels but at a fraction of the alarm levels to allow for operator action to mitigate the release.

c. Sector averaging accounting for lateral dispersion in the y direction as a result of variation in wind direction for periods greater than or equal to 24 hours was used in the FSAR. Guidance on sector averaging is given in ANSI-15.7-1977 and the sector averaging X/Q equation from ANSI-15.7-1977 for ground level receptors is given below:

$$X/Q = 2.032 \exp[-h^2/2\sigma_z^2] \bullet [\mu\ \sigma_z\ x]^{-1}$$

$$ADF = 1 / [X/Q]$$

$$\text{where, } 2.032 = (16 \text{ sectors} / 2\ \pi\ x)[2 / \pi\ ]^{1/2}$$

Not using sector averaging is conservative and more realistic since sector averaging for locations near the reactor stack for weather stability class C are not recommended for distances under 1000 m. The factor of 2.032 used in ANSI-15.7-1977 assumes the relationship of $\pi x/n > 2\sigma_y$ is valid. This occurs at distances for x $\geq$1000 m for weather stability class C and at distances < 500 m for stability classes D and F.

Based on the data in the Emergency Plan, FSAR, and the above equation for X/Q without sector averaging and without accounting for wind frequency, an average ADF was calculated for various distances from 100 m to 1000 m.

X/Q values were evaluated for heights (z) of 0 m, 20 m, and 30 m without sector averaging and at 0 m with sector averaging. From this data, it is concluded that:

1. Elevated locations have higher X/Q values, or lower ADF values. The specific locations of concern are the upper floors of the DH Hill Library (x = 150 m), Poe Hall and Dabney Hall (x = 200 m).

2. Ground level locations have significantly lower X/Q values, or higher ADF values, for all weather stability classes.

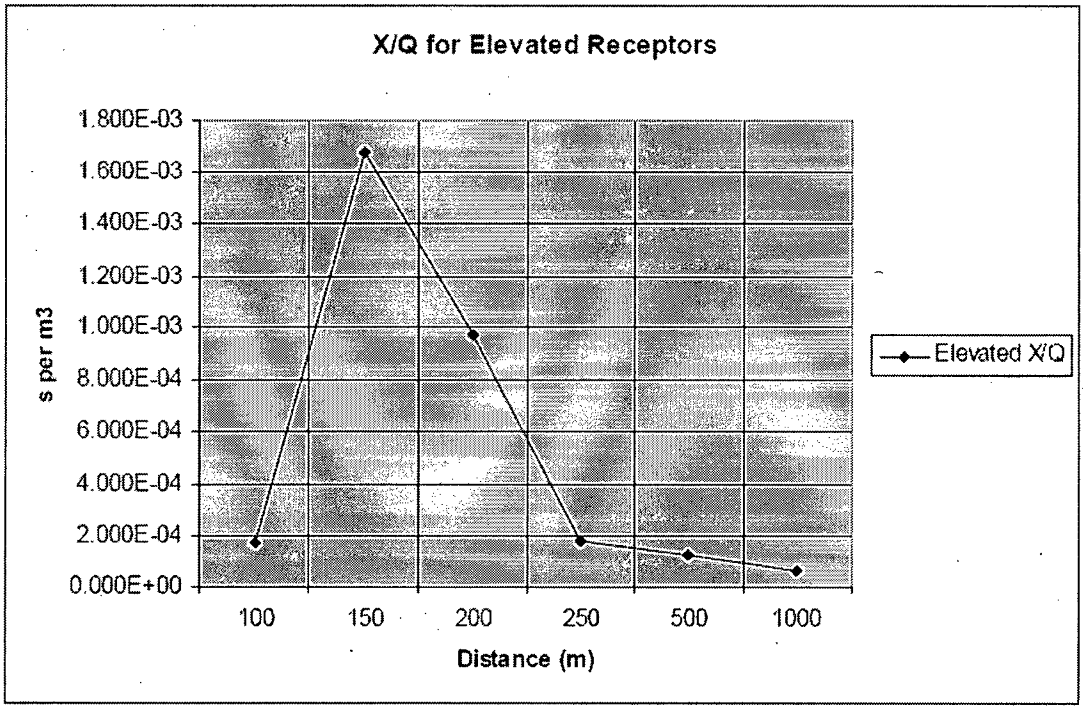
3. The maximum average X/Q of occurs at distance of 150 m and a height of 30 m, giving a minimum ADF in excess of 5 E2.

NOTES:

Elevated receptors within a distance of 150 m are at a height of $\leq$20 m, not 30 m.
Elevated receptors at 150 m to 200 m are at a height of approximately 30 m.
Elevated receptors from 200 m to 1000 m are at a height of $\leq$20 m, not 30 m.

Data for average X/Q and ADF at height of 20 m and 30 m:

x m	Ave X/Q	z m	ADF
100	1.712E-04	20	5.8 E3
150	1.677E-03	30	**6.0 E2**
200	9.692E-04	30	1.0 E3
250	1.804E-04	20	5.5 E3
500	1.203E-04	20	8.3 E3
1000	5.946E-05	20	1.7 E4



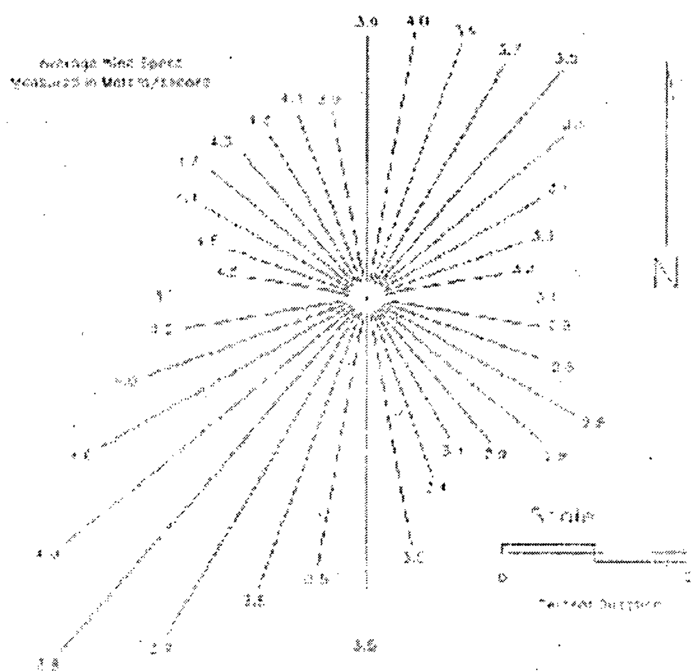
d. Weather data, such as wind speed and direction, at the RDU airport is applicable to NCSU and is available in Section 2 of the FSAR and data from the COMPLY code. Both references give a maximum wind frequency for 16 sectors at 0.1.

COMPLY data for Raleigh, NC (STAR DATA FILE) rdu0083.str

DIR-FROM	FREQUENCY	SPEED (m/s)
'N '	9.27E-02	4.17E+00
'NNE'	5.53E-02	4.07E+00
'NE '	6.36E-02	3.76E+00
'ENE'	4.69E-02	3.70E+00
'E '	6.64E-02	3.68E+00
'ESE'	5.51E-02	3.47E+00
'SE '	4.50E-02	3.24E+00
'SSE'	3.75E-02	3.46E+00
'S '	9.54E-02	3.67E+00
'SSW'	6.85E-02	3.80E+00
'SW '	9.14E-02	4.34E+00
'WSW'	6.79E-02	4.34E+00
'W '	7.23E-02	4.28E+00
'WNW'	4.19E-02	4.77E+00
'NW '	4.94E-02	4.83E+00
'NNW'	5.08E-02	4.53E+00
SUM OF FREQUENCIES =		1.00

FSAR Figure 2.9 is shown below and provides weather data at RDU for the years from 1990-1994. Data is provided for every 10 degrees, so 36 sectors are given. For any two adjacent sectors, the maximum frequency is approximately 0.1.

FIGURE 2-9



Average Wind Speed and Percent Direction
at Raleigh-Durham International Airport
1990 - 1994

If the wind frequency of 0.1 is taken into account, the average calculated X/Q is 10 times lower thereby making the minimum ADF 10 times higher. Therefore, the minimum ADF for distances at or beyond 100 m exceeds 5.9 E3 (i.e. 1/(0.1*1.68E-3)).

10 CFR 20 states that compliance with annual public dose limits may be met by (1) measurement or calculation to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit or (2) the annual average concentrations of radioactive material released in gaseous effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of appendix B to part 20, AND (ii) if an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 2 mrem per hour and 50 mrem per year.

Based on item (2) in the previous paragraph, the AEC for all airborne effluent is limited to 5.9 E3 AEC for the PULSTAR nuclear reactor for distances at or beyond 100 m from the reactor stack. 5.9 E3 AEC may therefore be used as the basis for an alarm set point realizing that the radiation monitors respond to an instantaneous level above the set point and that prolonged operation at elevated effluent monitor levels is not likely. Thus, exceeding the annual public dose limit is not likely if the alarm is set at level associated with the average annual concentration of airborne effluent for the location with the lowest ADF.

For distances less than 100 m from the stack, credit for the wind frequency is not appropriate due to the proximity of the receptor location to the stack. The ADF is back calculated to be 5 E3 based on 10 CFR 20 limits using data taken from Appendix B of the Emergency Plan:

RADIO-NUCLIDE	**ELEVA-TION**	**H_{shine} Normal Ventilation rem/h per μCi/ml at 1 m/s**	**H_{shine} Confinement Ventilation rem/h/ per μCi/ml at 1 m/s**	**H_{shine} Confinement Ventilation with Dilution rem/h per μCi/ml at 1 m/s**
Ar-41	0 m	3.8 E-2	1.3 E-2	1.5 E-3
Ar-41	20 m	9.8 E-2	3.2 E-2	2.5 E-3
Fuel Failure	0 m	7.1 E-3	2.4 E-3	2.8 E-4
Fuel Failure	20 m	1.9 E-2	6.3 E-3	4.4 E-4

The maximum value is 2.2 E-1 rem/h per uCi/ml for Ar-41 under calm winds. Calm winds have a wind speed of 1 mph, or ~0.5 m/s.

Ar-41 AEC is 1 E-8 uCi/ml per 10 CFR 20. At 1 AEC, the predicted annual dose rate is:

$$1.93\text{ E-2 } \frac{\text{mrem per y}}{\text{AEC}} \leq (2.2\text{ E2 mrem/h per uCi/ml})\left(\frac{1\text{ E-8 uCi/ml}}{\text{AEC}}\right)(8760\text{ h/y})$$

At the 10 CFR 20 annual public dose limit of 100 mrem, the AEC fraction is:

$$5.1\text{ E3 AEC} = (100\text{ mrem per y}) \,/\, \left(\frac{1.93\text{ E-2 mrem per y}}{\text{AEC}}\right)$$

EAL Based Set Points

Emergency classification for airborne effluent was analyzed as a result of the ventilation system changes in Attachment 3. From that analysis it is concluded that:

a. Emergency classification would be necessary at approximately 6,500 AEC for noble gases and approximately 13,000 AEC for radionuclides other than noble gases.

   NOTE: The EAL are based on Class F weather stability at a wind speed of 1 m/s as stated in ANSI/ANS-15.7-1977. EAL are provided in NUREG 0849 EAL. The limiting ADF for Class F weather stability and a wind speed of 1 m/s is 131 [1 / (0.017 / 2.24)].

b. Emergency classification is not necessary based on airborne releases in any mode of operation of the ventilation system for postulated accidental releases

c. Maximum dose to members of the public would approximate the constraint level of 10 mrem for postulated accidental releases in any mode of operation of the ventilation system

If Class F weather stability at a wind speed of 1 m/s is used as recommended in ANSI/ANS-15.7-1977, EAL based set points are calculated as follows:

At a wind speed of 1 m/s (or 2.24 mph), the associated EAL set points are as follows:

For noble gases: $6{,}588 \text{ AEC fractions} = \dfrac{50 \text{ AEC}}{(1.7 \text{ E-}2/2.24)} \sim 6{,}500 \text{ AEC fractions}$

For nuclides other than noble gases:

$$13{,}176 \text{ AEC fractions} = \frac{100 \text{ AEC}}{(1.7 \text{ E-}2/2.24)} \sim 13{,}000 \text{ AEC fractions}$$

where, 2.24 mph = 1 m/s
1.7 E-2 s/m^3 is the atmospheric dispersion parameter at a wind speed of 1mph

It is noted that the radiation monitors respond to an instantaneous level above the set point and that prolonged operation at elevated effluent monitor levels is not likely.

Conclusions

Based on the calculations and discussion given above and the Emergency Plan and FSAR, the following conclusions are made regarding the stack gas and stack particulate radiation monitoring channel set points:

- External public dose considerations for locations within 100 m of the reactor stack effluent to ≤ 5,000 AEC

- Demonstrating the annual average concentration of effluent for locations at or beyond 100 m from the reactor stack limit effluent to ≤ 5000 AEC

- Alert (warning) set points are based on abnormal levels that are a fraction of the alarm levels thereby allowing operator action(s) to mitigate the release. A value of ≤ 1000 AEC meets this criterion based on a review of historical data and calculated alarm set points.

- Ar-41 and Co-60 are the radionuclides of concern during routine operations and are the basis for set points for the stack gas and stack particulate radiation monitors, respectively. Set points may be lowered based on AEC values and detector efficiencies if the release of other radionuclides becomes a concern.

ATTACHMENT 2: EFFLUENT RADIONUCLIDE MEASUREMENTS

This attachment is provided to:

- Assist with the determination of Airborne Effluent Concentration (AEC) fraction
- Assist with off-site dose projections at or beyond the site boundary from airborne effluent
- Describe liquid effluent concentration and activity measurements

System Descriptions

PULSTAR Ventilation System and Radiation Monitoring System Diagrams are given in Attachment 1. Four channels are provided for the detection and measurement of airborne effluent in the radiation monitoring system. These four channels are:

CHANNEL NUMBER	CHANNEL NAME	DETECTOR TYPE
5	Stack Gas	GM
6	Stack Particulate	Plastic Scintillator
7	Auxiliary	GM
8	Filter	GM

Channels 5, 7, and 8 use the same type of GM detector. Channel 6 uses a plastic scintillator sensitive primarily to beta radiation. All of the channels provide readings in the Control Room on a digital ratemeter and chart recorder. All of the channels also provide annunciation and remote alarm indication in the Control Room if a radiation set point is exceeded. On alarm, channels 5, 6, and 7 initiate the automatic evacuation system and the Confinement Ventilation System.

Three alarm functions are provided by each channel; Fail, Warn, and Alarm.

- "Fail" indicates either a power failure or inoperative equipment.
- "Warn" indicates that abnormally high radiation levels are being detected.
- "Alarm" indicates that a radiation level associated with the Notification of Unusual Event emergency classification is being approached.

Sufficient margin is included in the "Warn" and "Alarm" set points to allow for Reactor Operator action(s) to mitigate radiological consequences of airborne effluent.

Detectors for Channels 5, 6, and 7 are located downstream of the Confinement System filters and the detector for Channel 8 is located upstream of the Normal Ventilation System Recirculation filters. Detectors for Channels 7 and 8 are located in the ventilation ducts. Detectors for Channels 5 and 6 analyze sampled air taken from the exhaust stack downstream of the Confinement System filters at a flow rate of 2 to 4 cfm. The sample flow is directed to a fixed particulate filter which is monitored by Channel 6 and then directed to a gas chamber monitored by Channel 5. The sample flow is then returned to the exhaust ventilation duct.

Roughing filters with nominal particulate removal efficiencies are located upstream of the normal and confinement exhaust fans. Activated charcoal and High Efficiency Particulate Absorbers (HEPA) filters are used in the confinement system. These filters have removal efficiencies of 99% for halogens and 99.97% for particulates.

Normal and confinement system exhaust rates from the reactor building are 1,870 cfm and 600 cfm, respectively. There are two confinement system filter trains, each is rated at 600 cfm but only one train is used at any one time. Both the normal and confinement system are capable of placing a negative pressure on the reactor building with respect to the atmosphere. Building penetrations are sealed and doors have gaskets and are kept closed, except for brief entries and exits, to ensure negative pressure is maintained. Furthermore, negative pressure is continuously monitored and indicated in the Control Room. Failure to maintain a negative pressure for more than five minutes activates an annunciator in the Control Room. Therefore the only release point of airborne effluent with confinement maintained is the PULSTAR exhaust stack.

The BEL South Wing (R-63) ventilation exhaust is rated at 12,500 cfm. This portion of the building exhaust is a source of clean process air. The PULSTAR reactor building exhaust stack is actually inside of the BEL South Wing (R-63) exhaust stack. The outer BEL exhaust stack is 10 feet higher than the inner PULSTAR exhaust stack. Thorough mixing of the clean air from the R-63 exhaust stack with the normal PULSTAR reactor bay exhaust is not assumed to occur because of the difference in exhaust velocities. However, when the PULSTAR Confinement System is in use, thorough mixing is assumed to occur because of the lower exhaust velocity from the PULSTAR Confinement System. If the R-63 exhaust fan is operating, the resulting stack dilution factor (SDF) is approximately 20.

$$\textbf{SDF} = \textbf{[12,500 + 600] cfm / 600 cfm} \approx \textbf{20}$$
$$\textbf{SDF} \equiv \textbf{1 for all other ventilation modes}$$

Airborne Radioactivity Source Terms

Source terms for possible airborne effluent releases depends on fuel integrity and other operational events. Evidence of fuel failure would be indicated by the presence of fission products in reactor coolant or air samples. If fuel failure is not evident, then Ar-41 and Co-60 are assumed to be present in the exhausted reactor building air.

Ar-41 is produced by neutron activation of stable Ar-40, which is a nuclide present in normal air, while the reactor is operating. Section 10 of the PULSTAR Final Safety Analysis Report (FSAR) indicates that Ar-41 production is associated with air in the reactor pneumatic sample system (PN) and experimental beam tubes and thermal column (BT & TC). Measures have been taken to minimize the amount of Ar-41 produced in these facilities.

Three Ar-41 release scenarios are considered and the 24 hour average concentrations are calculated below. However, all three are considered to be unlikely due to administrative controls on experiments, reactor operation, radiation safety, and exceeding radiation monitor alarm set points.

(1) A bolus of air from the PN system is removed with Ar-41 activity saturated followed by reactor operation at 1 MW with the PN system operating;

Saturation activity of Ar-41 in PN tube: $A(\infty) = [\sigma\phi N]$ [1 uCi / 3.7E4 dps] = 4.2 E4 uCi of Ar-41

where, Thermal neutron cross-section for Ar-40, σ is 0.5 E-24 cm^2
Peak thermal neutron flux density, ϕ is 1 E13 $cm^{-2}s^{-1}$
PN tube volume = $\pi(2.54\ cm)^2(61cm)$ = 1236 ml
Note: 61 cm is the active length of reactor fuel
N is number of Ar-40 atoms

$$N = \frac{(1236\ ml)(1.2\ E\text{-}3\ g/ml)(0.01g\ Ar/g\ of\ air)(0.996\ Ar\text{-}40)(6.022\ E23\ at/mol)}{28.964\ g/mol}$$
$$= 3.1\ E20 \text{ atoms of Ar-40}$$

PN bolus release time = 1236 ml / 28,317 ml per cubic foot / 190 cfm x 60 s/m = 0.015 s

PN bolus concentration or PN bolus [C] = 4.2E4 uCi/1236 ml = 33 uCi/ml

$$24 \text{ hour average [C]} = \frac{(33uCi/ml)(0.015s)+(8.3E\text{-}6\ uCi/ml)(24\ h)(3600s/h)}{(24h)(3600\ s/h)}$$
$$= \mathbf{1.4\ E\text{-}5\ uCi/ml}$$

where, 8.3 E-6 uCi/ml is based on historical data with the PN system in operation
8.3 E-6 uCi/ml = 776 cpm at stack gas monitor / 9.3 E7 cpm per uCi/ml
Stack gas detector Ar-41 detection efficiency is 9.3 E7 cpm per uCi/ml
776 cpm = 130 cpm x {(10,050 cfm) / (1870 cfm)} x 1 MW/0.9 MW
10,050 cfm is former normal ventilation system flow rate
1870 cfm is current (new) ventilation system exhaust flow rate
190 cfm is the PN exhaust blower flow rate

For confinement, the expected stack gas reading is 650 cpm giving approximately 7 E-6 uCi/ml in the reactor exhaust. This would be is reduced by the SDF to 3.5 E-7 uCi/ml at the stack discharge if BEL dilution is present.

(2) Same as the first scenario except the PN blower releases activity into the reactor bay via a PN blower rupture preceded by PN system usage for 24 hours at 1 MW;

Peak concentration C_p = 4.2 E4 uCi / 2.25 E9 ml = 1.9 E-5 uCi/ml

where, free air space in reactor building is 2.25 E9 ml

24 hour average concentration, $[C] = \dfrac{A(0)\,[1 - \exp(-k24h)]}{(k)(24h)(V)} + C_B$

where, Decays in occurring in time T, $D = \int^{T} A(0) \exp(-kt)\, dt$

$[C] = D / V / T$ with V = 2.25 E9 ml and T = 24 h

D in 24 h = $A(0)\,[1 - \exp(-k24h)] / k$

A(0) = 4.2 E4 uCi is the Ar-41 saturation activity in the PN tube

k is the total removal rate constant = air exchange rate and decay constant

$k = v + \lambda$

Confinement $v = \dfrac{(600 \text{ cfm})(28{,}317 \text{ ml per cubic foot})(60 \text{m/h})}{2.25 \text{ E9 ml}}$

= 0.43 per h, or 1 air change per 2.35 hours or 132 minutes

Normal v $= \dfrac{(1870 \text{ cfm})(28{,}317 \text{ ml per cubic foot})(60 \text{ m/h})}{2.25 \text{ E9 ml}}$

= 1.32 per h, or 1 air change per 0.75 hours or 45 minutes

$\lambda = (\ln 2 / 1.83 \text{ h}) = 0.379$ per hour

k = 0.805 per hour for Confinement and

k = 1.70 per hour for Normal ventilation

C_B = 8.3 E-6 uCi/ml from 24 hours of PN use

24 hour average [C] $= \dfrac{(4.2 \text{ E4 uCi})[1 - \exp[(-k)(24h)]}{(k)(2.25 \text{ E9})(24h)} + 8.3 \text{ E-6 uCi/ml}$

= 8.8 E-6 uCi/ml for Normal ventilation

= **9.3 E-6 uCi/ml** for Confinement

(3) A drained but shielded BT is opened immediately after the reactor is operated at 1 MW with the PN system used for 24 hours prior to reactor shut down releasing a bolus of air with Ar-41 activity saturated.

Saturation activity of Ar-41 in BT:

$A(\infty) = [\sigma\phi N][1 \text{ uCi}/3.7\text{E4 dps}] = 9.86\text{E5 uCi of Ar-41}$

where, σ is 0.5 E-24 cm^2

ϕ is 1 E12 $cm^{-2}s^{-1}$

N is number of Ar-40 atoms

$N = \dfrac{(2.95\text{E5ml})(1.2 \text{ E-3g/ml})(0.01\text{gAr/g of air})(0.996 \text{ Ar-40})(6.022 \text{ E23at/mol})}{28.964 \text{ g/mol}}$

N = 7.3 E22 atoms of Ar-40

Maximum BT tube volume = 2.95 E5 ml, or 10.42 cubic feet

Peak concentration C_p = 9.86 E5 uCi / 2.25 E9 ml = 4.4 E-4 uCi/ml

$$\text{24 hour average [C]} = \frac{(9.86\text{ E5 uCi})[1 - \exp[(-0.806)(24h)]]}{(0.806)(2.25e9\text{ ml})(24h)} + 8.3\text{ E-6 uCi/ml}$$

= **3.0 E-5 uCi/ml**

The third scenario 24 h average concentration for Ar-41 represents a worse case value suitable for accident analysis if no radiation monitoring system data is available.

Co-60 is the most predominant long-lived and radiologically significant activation product present in reactor coolant. Co-60 is produced by neutron activation of Co-59 contained in stainless steel. Several components used in the PULSTAR reactor and primary coolant system are made of stainless steel. However, it is unlikely that particulate activity would be released by routine operation due to the low volatility of particulates and corrosion control and encapsulation requirements.

If fuel failure is evident, then a mixture of fission products, primarily noble gases and halogens, may become airborne in the reactor building. Section 13 of the FSAR gives fission product activity in the reactor building and concentrations in the exhaust stack after passing through the Confinement System filters following a fuel handling accident. The relative distribution of those radionuclides listed in these tables is valid for any fuel failure incident. FSAR Section 13 does not take credit for dilution flow from the BEL South Wing (R-63) exhaust fans which discharge through the BEL outer exhaust stack. Noble gases are assumed to escape from the fuel and reactor coolant system readily. Halogens are assumed to partially escape from the fuel and reactor coolant system. Particulates are unlikely to escape from the fuel and reactor coolant system. Confinement System filters remove significant amounts of halogens and particulates. Therefore, releases from failed fuel are mostly noble gases with much lower amounts of halogens and negligible amounts of particulates.

Based on the above discussion, the following conclusions are made regarding airborne effluent:

- The only exhaust location is the PULSTAR stack
- If there is no evidence of fuel failure,
  - Ar-41 is assumed to be the gaseous radionuclide released
  - Any particulate effluent detected is assumed to be Co-60
- If fuel failure is evident,
  - Xe-133 is the major gaseous radionuclide released. However, Kr-88, Xe-133, and Xe-138 are all radiologically significant.
  - Any particulate effluent detected is assumed to be Cs-137 for determination of effluent concentration and Sr/Y-90 for dose calculations

- FSAR Section 13 lists those radionuclides and their concentrations released if a fuel handling accident occurs with the confinement system in use. This relative distribution is assumed to be valid for any fuel failure incident.

- FSAR Section 13 concentrations for halogens are 100 times higher if fuel failure occurs and normal ventilation is in use

- Concentrations at the stack exhaust point are lower by a factor of 20 if the BEL South Wing (R-63) ventilation fans and PULSTAR confinement system are both in use. Otherwise the concentration detected at the stack radiation monitor equals the stack exhaust concentration.

- Halogen to Noble Gas ratio of 3E-2 with normal ventilation and 3E-4 for confinement ventilation may be used based on FSAR Section 13. This ratio applies to radionclides of Iodine and Bromine.

For the failed fuel experiment, mass of U-235 and irradiation conditions are limited to a dose up to 10% of the applicable limits for workers and members of the public. Therefore, the public total effective dose-equivalent is limited to 10 mrem, which is less than the 15 mrem for declaration of an "Unusual Event". Refer to Attachment 6 for details on the fueled experiment dose calculations.

Determination of AEC Fraction

Emergency Action Level (EAL) determination is based on average concentrations at the site boundary released over a 24 hour period. Units are typically AEC fractions. AEC is the applicable concentration in μCi/ml for airborne effluent given in 10 CFR 20 Appendix B Table 2 Column 1. AEC fraction is the sum of the ratio of a radionuclide concentration [C] in μCi/ml to its AEC value:

$$\textbf{AEC Fraction} = \Sigma_i \, [\mathbf{C}]_i/\mathbf{AEC}_i$$

where, i is the i^{th} radionuclide
[C] is the effluent concentration

Airborne source concentrations [C] inside the exhaust stack are determined preferably by measurements from the radiation monitoring system since that is where the activity is at its highest concentration. Detector response to known concentrations of airborne radioactivity is necessary to determine [C].

The stack concentration is equal to the effluent concentration unless the ventilation system in the Confinement mode with BEL dilution flow. In Confinement with BEL dilution flow, effluent concentration is 20 times lower than the stack concentration due to SDF.

Effluent [C] = Stack [C] for normal ventilation
Effluent [C] = Stack [C] for confinement ventilation without BEL dilution
Effluent [C] = Stack [C]/ SDF = Stack [C] / 20 for confinement ventilation with BEL dilution

Normalized detection response for the different detector model types used by the radiation monitoring system have been determined by the vendor for various radionuclides and are summarized below:

RADIO-NUCLIDE	STACK GAS (cpm per µCi/ml)	AUXILIARY OR FILTER (cpm per µCi/ml)	STACK PARTICULATE (cpm per µCi/ml)
Ar-41	5.1 E7	4.0 E8	Not Applicable
Kr-85	3.5 E7	1.2 E8	
Xe-133	5.0 E6	4.7 E6	
Tc-99	Not Applicable	Not Applicable	3.4 E9* FT
Cs-137			8.2 E9*FT
Sr/Y-90			1.6 E10*FT (#)

Where, F is sample flow rate in cfm and T is sample time in minutes

NOTE: Stack Particulate channel efficiency in cpm/µCi/ml for a sample time of T minutes and at a sample flow rate of F cfm is approximately equal to EFT for long-lived radionuclides, where E is cpm/µCi. E is reported as 1.2 E5 for Tc-99, 2.9 E5 for Cs-137, and 5.8 E5 (# see note below) for Sr/Y-90. Values of E are taken from the primary calibration performed by the instrument vendor for the detector used.

Stack [C] based on actual radiation monitor readings and halogen to noble gas ratios are summarized below:

FUEL STATUS	RADIONUCLIDE	CONCENTRATION [C] NORMAL OR CONFINEMENT VENTILATION MODE (μCi/ml)
No failure	Particulate (Co-60)	Channel 6 net cpm divided by 3.4 E9*FT where, F is cfm and T is minutes
No failure	Noble gas (Ar-41)	Channel 5 net cpm divided by 5.1 E7, or Channel 8 net cpm divided by 4.0 E8
Failure	Particulate (Sr/Y-90)(#)	Channel 6 net cpm divided by 8.2 E9*FT where, F is cfm and T is minutes
	Halogens (I & Br)	Noble Gas [C] times 3E-2 for normal ventilation Noble Gas [C] times 3 E-4 for confinement
	Noble Gas (Xe-133)	Channel 5 net cpm divided by 5.0 E6, or Channel 8 net cpm divided by 4.7 E6

NOTE:(#) Sr/Y-90 has two betas per decay, while Cs-137 and Tc-99 have one beta per decay. The Sr/Y-90 E value is for the Sr-90 and Y-90 combined activities (which are in secular equilibrium), so the activity for each radionuclide (Sr-90 or Y-90) is determined by using half of the reported E value (or half the count rate).

Stack [C] values may be based on the following values if radiation monitor readings are not available:

RADIONUCLIDE	Stack [C] in NORMAL VENTILATION (μCi/ml)	Stack [C] in CONFINEMENT (μCi/ml)
Ar-41	3.0 E-5	3.0 E-5
Kr-88	1.6 E-6	1.6 E-6
Xe-133	2.3 E-5	2.3 E-5
Xe-138	8.1 E-7	8.1 E-7
I-131	3.4 E-7	3.4 E-9
I-133	4.1 E-7	4.1 E-9
Br-83	7.2 E-7	7.2 E-9
Br-84	6.0 E-9	6.0 E-11

All Fission Products	2.7 E-5	2.5 E-5

AEC values for various radionuclides are given below:

RADIONUCLIDE	AEC (μCi/ml)
Ar-41	1 E-08
Co-60	5 E-11
Kr-88	9 E-09
Xe-133	5 E-07
Xe-138	2 E-08
I-131	2 E-10
I-133	1 E-09
Br-83	9 E-08
Br-84	8 E-08
Sr-90/Y-90	6 E-12

NOTE: 10 CFR 20 Appendix B rounded off all AEC values to 1 significant figure.

In the event of fuel failure, AEC values may be simplified for noble gases, iodines, and bromines based on the FSAR fuel failure acitivity distribution. Particulate activity is assumed to be Sr-90/Y-90. These simplified FSAR based AEC values for fuel failure are:

$$\textbf{FSAR Fuel Failure AEC} = \Sigma_i \textbf{ Stack } [C]_i \bullet \textbf{AEC}_i / \Sigma_i \textbf{ Stack } [C]_i$$

RADIONUCLIDE	FSAR Fuel Failure AEC (μCi/ml)
Noble Gases	5 E-07
Iodines	6 E-10
Bromines	9 E-8
Particulates (Sr-90/Y-90)	6 E-12

AEC fraction for off-site locations averaged over 24 hours may be calculated as follows:

$$AECFraction = \frac{1}{24} \cdot \Sigma_j \Sigma_i [\frac{C_i}{AEC_i}]_j \cdot [\frac{T}{SDF}]_j \cdot [\frac{\chi}{Q}]_j$$

where, i is the i^{th} radionuclide and j is the j^{th} time interval
T is time in hours and $\Sigma_j T_j$ is not to exceed 24 h
[C] is concentration and SDF is stack dilution factor
χ/Q is the atmospheric dispersion parameter for the off-site location

EAL values given in NUREG 0849 are based on associated doses from integrated exposures to airborne radioactivity in AEC·h for a 24 hour period or less at the site boundary or off-site locations and are listed below:

EMERGENCY CLASS	NOBLE GAS	RADIONUCLIDES OTHER THAN NOBLE GAS
Notification of Unusual Event	50	100
Alert	250	500
Site Area Emergency	1250	2500

[C] values for airborne effluents at occupied locations are higher at distances of greater than 100 m than at the site boundary because of the stack height. Airborne effluent is present in an overhead plume for distances less than 100 m from the stack.

The closest building to the stack with a height of 30 m is 140 m from the stack. The most restrictive χ/Q value calculated for the PULSTAR reactor is 1.7 E-3 s/m^3 at a wind speed of 1 mph or 7.6 E-3 s/m^3 at a wind speed 1 m/s (2.24 mph) at a distance of ~ 150 m.

Substituting and simplifying the above AEC fraction equation for 24 hour averaging gives:

$$AECFraction = 3.2E-4 \cdot \Sigma_j \Sigma_i [\frac{C_i}{AEC_i}]_j \cdot [\frac{T}{SDF}]_j$$

where, (1/24)(7.6E-3) = 3.2 E-4

Assuming no radiation monitoring readings are available, the 24 hour average AEC fractions for Ar-41 and the fuel handling accident are conservatively determined below using the above defined source terms for Ar-41 and fuel failure, event durations of 0.75 h and 2.35 h for fuel handling accident in normal ventilation and confinement ventilation modes, and the above simplified 24 hour average AEC fraction equation:

RADIO-NUCLIDE	AEC Fraction NORMAL VENTILATION	AEC Fraction CONFINEMENT without BEL DILUTION	AEC Fraction CONFINEMENT with BEL DILUTION
Ar-41	21	21	1
Fuel Failure	5.8	2.2 E-1	1.1 E-2

NOTES: 24 hour event times are used for Ar-41 since Ar-41 releases are expected.
Event duration for fuel handling accident in normal ventilation is
T = 2.4 E9 ml/{ (1870 cfm)(28,317 ml/cubic foot)(60 m per h)} = 0.75 h
Event duration for fuel handling accident in confinement ventilation is
T = 2.4 E9 ml/{ (600 cfm)(28,317 ml/cubic foot)(60 m per h)} = 2.35 h
These event times are used for the fuel handling accident since the activity is a puff release. The concentration inside the reactor bay would be at a maximum initially and then decrease over 24 hours due to decay and several air exchanges.

Therefore, no emergency classification is associated with releases of Ar-41 or a fuel handling accident as postulated in the FSAR based on airborne effluent concentration at or beyond the site boundary since an AEC fraction of 50 is not exceeded in any ventilation mode. However, release of fission products to the reactor coolant system may result in either the "Notification of Unusual Event" or "Alert" emergency classification, depending on severity of fuel damage.

For the failed fuel experiment, mass of U-235 and irradiation conditions are limited to a dose up to 10% of the applicable limits for workers and members of the public. Therefore, the public total effective dose-equivalent is limited to 10 mrem, which is less than the 15 mrem for declaration of an "Unusual Event". Refer to Attachment 6 for details on the fueled experiment dose calculations.

Release Rate Determination

Total release rate, Q^{total}, for all time intervals may be calculated using concentrations and ventilation flow rates by the following equations:

$$Q^{total} = \Sigma_j [\Sigma_i Q_i]_j$$

$$Q_i = \{4.72 \text{ E-04} \bullet [C]_i \bullet F\} / SDF$$

where, Q_i is Ci/s for the i[th] radionuclide

$[C]_i$ is concentration in µCi/ml

F is flow rate in cfm and equals 1870 cfm for Normal ventilation and 600 cfm for Confinement ventilation

4.72E-4 is unit conversion constant equal to the following product;

4.72 E-4 = (28,317 ml / cubic foot)(1 minute / 60 s)(1 m^3 /1 E6 ml)

j[th] time interval

If no radiation monitoring readings are availabe, Q values for Ar-41 are based on the 24 hour average concentration and Q values for the fuel handling accident are based on FSAR data.

Q is determined using the above release rate equation:

RADIO-NUCLIDE	Q in NORMAL VENTILATION (Ci/s)	Q in CONFINEMENT without BEL DILUTION (Ci/s)	Q in CONFINEMENT with BEL DILUTION (Ci/s)
Ar-41	2.7 E-05	8.5 E-06	4.3 E-07
Noble Gases	2.2 E-05	7.1 E-06	3.6 E-07
Iodines	6.6 E-07	2.1 E-09	1.1 E-10
Bromines	6.5 E-07	2.1 E-09	1.1 E-10
All Fission Products	2.3 E-5	7.1 E-6	3.6 E-7

Off-site Dose Projections

Projection of off-site radiation dose from airborne effluent should include the use of appropriate atmospheric dilution, which is defined as the reciprocal of the atmospheric dispersion parameter χ/Q. χ/Q may be estimated using the following equation taken from Section 13 of the FSAR:

$$\left(\frac{\chi}{Q}\right)_{xyz} = \left(\exp\left[-\frac{y^2}{2\sigma_y^2}\right]\right)\cdot\left(\exp\left[\frac{-(h-z)^2}{2\sigma_z^2}\right] + \exp\left[\frac{-(h+z)^2}{2\sigma_z^2}\right]\right)\cdot\left(2\pi U\sigma_y\sigma_z\right)^{-1}$$

where, x is the downwind distance from the stack to receptor in m
y is the lateral distance from the plume centerline in m
z is the receptor elevation in m
σ_y is the lateral dispersion parameter in m
σ_z is the vertical dispersion parameter in m
h is the physical stack height in m, or 30 m
U is wind speed in m/s
χ/Q is in s/m^3

NOTE: Decay during transport is neglected since transit time for 1000 meters is approximately 17 minutes at a wind speed of 1 m/s and 38 minutes at a wind speed of 1 mph.

Using guidance given in ANSI/ANS-15.7-1977, the effective stack height for the PULSTAR reactor was determined to be only slightly higher than the actual stack height of 30 m at a wind speed of 1 m/s and with normal ventilation (1870 cfm) or confinement ventilation (600 cfm). Maximum effective H is 31.2 m vs. 30 m. As a result and for simplicity and conservatism, the effective stack height is not used in this calculation.

The building height relative to the outer stack is approximately 2.5 (100 ft vs. 42 ft gives a ration of 2.38) and therefore the above equation without modification is used. Modification of σ_y and σ_z as discussed in ANSI/ANS-15.7-1977 increases or has no affect on the plume dimension standard deviations used in the above equation; e.g. if σ is 2 m, Σ is > 2.6 m and if σ is 100 m, Σ is 100 m:

$$\Sigma = [\sigma^2 + (0.5*A)/\pi]^{1/2}$$

where, A has a maximum value of 54 m for the BEL

Larger deviations would decrease the χ/Q value at a given receptor location. Therefore, using the above equation is conservative.

Simplifying the $(\chi/Q)_{x,y,z}$ equation gives:

- For ground elevation at plume centerline: y = 0 and z = 0

$$\mathbf{(\chi/Q)_{x,0,0} = [\pi U\sigma_y\sigma_z]^{-1} \bullet \exp[-h^2/2\sigma_z^2]}$$

And if $h^2 = 2\sigma_z^2$, maximum ground level concentration is observed

$$\mathbf{(\chi/Q)_{x,0,0}^{max} = [\pi U\sigma_y\sigma_z e]^{-1}}$$

- For stack elevation at plume centerline: y = 0 and z = h

$$\mathbf{(\chi/Q)_{x,0,h} = [2\pi U\sigma_y\sigma_z]^{-1} \bullet \{1+ \exp[-2h^2/\sigma_z^2]}$$

Locations of interest are the site boundary which ranges from 20 m to 50 m, surrounding buildings which range from 40 m to 140 m, and the nearest residence at 240 m. PULSTAR reactor stack height, h, is 30 m.

Using σ_y and σ_z data from Meteorology and Atomic Energy 1968 and from NRP Calculation 91-001 in the above equations for various weather stability classes and distances from 100 m to 500 m indicates that:

- Stable weather conditions are most severe for elevated receptor locations equal to the stack height
- Unstable weather conditions are most severe for ground level locations
- Neutral weather conditions are most severe for elevations between the ground and stack height
- $(\chi/Q)_{x,0,0}^{max}$ occurs at downwind distances (x) greater than 100 m

NOTE: For distances within 100 m, σ_y and σ_z data is not available to calculate χ/Q values.
For locations within 100 m from the exhaust stack, the projected dose rate may be estimated using two line sources to represent the stack and an overhead plume.

Using the most restrictive χ/Q values will conservatively estimate radiation doses from airborne effluent to off-site locations. Restrictive χ/Q values for distances greater than or equal to 100 m are given below:

ELEVATION, z (m)	**Downwind Distance, x (m)**	**χ/Q at U of 1 mph (s/m^3)**
0 (Ground)	100	2.9 E-4
	150	4.0 E-4
	200	3.6 E-4
	250	3.5 E-4
	500	2.6 E-4
20	100	1.4 E-3
	150	1.5 E-3
	200	1.4 E-3
	250	1.4 E-3
	500	1.0 E-3
30 (Stack-height)	100	*
	150	1.7 E-2
	200	1.0 E-2
	250	7.1 E-3
	500	2.2 E-3

* NOTE: This location is not considered in this calculation since there are no structures 30 m high at 100 m distance.

Submersion or inhalation dose rate projections for distances greater than or equal to 100 m may be made using the following equation:

$$\mathbf{H_{x,y,z} = \{(4.72\ E\text{-}4)(\chi/Q)_{x,y,z}\ (\Sigma_i[C]_i\ )(DCF_i)\ (\ F\ )\}\ /\ (U \bullet SDF\ )}$$

where, $H_{x,y,z}$ is dose rate in rem/h at location x,y,z
$(\chi/Q)_{x,y,z}$ is the atmospheric dispersion parameter for 1 mph wind speed at location x,y,z in s/m^3
$[C]_i$ is the stack [C] of the i^{th} radionuclide in µCi/ml
DCF is dose rate conversion constant in rem/h per µCi/ml for the i^{th} radionuclide
F is stack exhaust flow rate in cfm
U is wind speed in mph and is assumed to be 2.24 mph if no data is available
SDF is stack dilution factor

DCF values are given below as listed in publication EPA 400-R-92-001 "Manual for Protective

Action Guides and Protective Actions for Nuclear Incidents" Tables 5-1 and 5-2 or as determined from 10 CFR 20 Appendix B Table 2 Column 1.

EPA 400-R-92-001 Tables 5-1 and 5-2 include dose from three pathways for the early phase (≤ 4 days) following an airborne effluent release; (1) external gamma radiation dose from the plume, (2) internal dose from inhalation, and (3) external gamma radiation dose from ground deposition. EPA-520/1-88-020 "US EPA Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors For Inhalation, Submersion, and Ingestion" was used for the inhalation data. Immersion (submersion) and ground deposition data were taken from DOE/EH-00070 "US DOE External Dose Rate Conversion Factors for Calculation of Dose the Public". Noble gases result in submersion dose only, while particulates and halogens have dose contribution from all three pathways with inhalation being the greatest dose contributor.

10 CFR 20 Appendix B DCF were taken from EPA-520/1-88-020 DCF values based on submersion dose for noble gas, inhalation dose for long-lived (> 1 day) radionuclides other than noble gas, and inhalation dose and submersion dose for short-lived (< 1day) radionuclides other than noble gas. 10 CFR 20 Appendix B Effective Dose DCF were converted for this calculation as follows:

$$\mathbf{EFFECTIVE\ DOSE\ DCF_i = H_y / (8760 \bullet AEC_i)}$$

where, H_y is 1E-1 rem per year for noble gases and H_y is 5E-2 rem per year for other radionuclides
8760 is the number of hours in a calendar year, i is the i^{th} radionuclide

<table>
<tr><th>RADIONUCLIDE</th><th>EFFECTIVE DOSE DCF
(rem/h per µCi/ml)</th><th>THYROID DOSE DCF
(rem/h per µCi/ml)</th></tr>
<tr><td>Ar-41</td><td>1.1 E 3</td><td rowspan="5">Not
Applicable</td></tr>
<tr><td>Co-60</td><td>2.7 E 5</td></tr>
<tr><td>Kr-88</td><td>1.3 E 3</td></tr>
<tr><td>Xe-133</td><td>2.0 E 1</td></tr>
<tr><td>Xe-138</td><td>7.2 E 2</td></tr>
<tr><td>I-131</td><td>5.3 E 4</td><td>1.3 E 6</td></tr>
<tr><td>I-133</td><td>1.5 E 4</td><td>2.2 E 5</td></tr>
<tr><td>Br-83</td><td>6.3 E 1</td><td rowspan="3">Not
Applicable</td></tr>
<tr><td>Br-84</td><td>7.1 E 1</td></tr>
<tr><td>Sr -90</td><td>1.6 E 6</td></tr>
</table>

If there is no evidence of fuel failure, then submersion dose rate projections should be based on Ar-41 for noble gas and inhalation dose rate projections should be based on Co-60 for particulates. If there is evidence of fuel failure, DCF values may be simplified to the values listed below based on the FSAR fuel failure activity distribution:

RADIONUCLIDE	**EFFECTIVE DOSE DCF (rem/h per μCi/ml)**	**THYROID DOSE DCF (rem/h per μCi/ml)**
Noble Gases	1.2 E 2	Not Applicable
Particulates	1.6 E 6	
Iodines	3.2 E 4	7.1 E 5
Bromines	7.1 E 1	Not Applicable

e.g. Simplified Iodine Effective Dose DCF =
[(3.4E-7)(5.3E4)+(4.1E-7)(1.5E4)] / [3.4 E-7 + 4.1 E-7] = 3.2 E4 rem/h per uCi/ml

Total thyroid dose rate projections, H_{thy}, are estimated by adding $H_{x,y,z}$ for the inhalation pathway dose from radioiodines and any external dose projections or survey measurements, H_{ext}:

$$\mathbf{H_{thy} = H_{x,y,z} + H_{ext}}$$

For locations within 100 m from the exhaust stack, the projected dose rate may be estimated using two line sources to represent the stack and an overhead plume. This assumption is valid because the plume from the exhaust stack does not reach the ground elevation or intersect a surrounding building at an elevated location for any weather stability class within 100 m from the exhaust stack. This assumption is conservative since it concentrates dispersed activity into a line and no credit is taken for shielding.

The stack source term is located in the upper 20 m of the stack and constant until discharged. Effluent concentration is reduced by the SDF and wind speed (U). Dose rates from either the stack or the overhead line sources depend on total activity, receptor location (x,y,z), and source term.

The following equations may be used to estimate dose rates within 100 m from the stack for a specific location (x,y,z):

$$\mathbf{H_{stack} = (H_{stack}\ per\ Ci) \bullet 4.05 \bullet [C]}$$

$$\mathbf{H_{line} = (H_{line}\ per\ Ci) \bullet 224 \bullet Q}$$

$$\mathbf{H_{shine} = H_{stack} + (H_{line} / U)}$$

where, H_{shine} is in rem/h

H_{stack} is in rem/h from stack line source

H_{line} is in rem/h from overhead line source

[C] is concentration in μCi/ml

Q is the release rate at the stack exhaust point in Ci/s

U is windspeed in mph and is assumed to be 2.24 mph if no data is available

SDF is Stack Dilution Factor

NOTES:

1. (224 • Q) gives the overhead line source activity in Ci and is equal to the line length of 100 m times the release rate in Ci/s divided by the wind speed of 0.447 m/s.

2. (4.05 • [C]) gives the stack line source activity in Ci an is equal to the stack volume of 4.05 E+6 ml times the concentration in μCi/ml times the conversion constant of 1 E-6 Ci per μCi. The stack line source was taken as being 20 m in length.

H_{shine} was solved for a reference stack [C] value of 1 μCi/ml of Ar-41 and the FSAR distribution from fuel failure at ground and elevated locations. It should be noted that both H_{stack} and H_{line} were solved without consideration of shielding from surrounding structures or the stack and do not correct for decay during transit. The line source equation used is provided in the FSAR and NRP Calculation 91-001.

Weather data, such as wind speed and direction, at the RDU airport is applicable to NCSU and is available by calling 515-8225. Alternately, weather data is also available from radio, television/cable, newspaper, or by direct observation.

The following table summarizes reference H_{shine} dose rates for stack [C] values of 1 μCi/ml and U of 2.24 mph (1m/s):

RADIO-NUCLIDE	ELEVATION	H_{shine} Normal Ventilation rem/h per μCi/ml at 1 m/s	H_{shine} Confinement Ventilation rem/h/ per μCi/ml at 1 m/s	H_{shine} Confinement Ventilation with Dilution rem/h per μCi/ml at 1 m/s
Ar-41	0 m	3.8 E-2	1.3 E-2	1.5 E-3
Ar-41	20 m	9.8 E-2	3.2 E-2	2.5 E-3
Fuel Failure	0 m	7.1 E-3	2.4 E-3	2.8 E-4
Fuel Failure	20 m	1.9 E-2	6.1 E-3	4.4 E-4

Calculation NRP 94-001 indicates that current stack gas and auxiliary radiation monitor set points correspond to 1 E-5 μCi/ml of Ar-41. Peak concentrations for Ar-41 accidents would cause annunciation in the Control Room and the confinement system is either automatically initiated or initiated by operator action. Therefore, the maximum concentration of Ar-41 for the normal ventilation mode is 1 E-5 uCi/ml and the maximum 24 h average accident concentration of 3 E-5 μCi/ml applies to the confinement ventilation mode. Either of these releases of Ar-41 result in a worse case dose rate of approximately 2 urem/h from Ar-41 within 100 m from the stack for normal and confinement ventilation modes, respectively. It should be noted that Ar-41 sources are controlled by various methods and prolonged Ar-41 releases at elevated levels are unlikely.

Total dose projection is given by the product of H_{xyz}, H_{thy}, or H_{shine} and event duration, T, for effective dose and thyroid dose, as applicable. These values may compared against the Protective Action Guide (PAG) values of 1 rem effective dose (H_{ede}) and 5 rem thyroid dose (H_{ode}) and 10 CFR 20 dose limit for individual members of the public. Dose rates in excess of the PAG values or 10 CFR 20 limits are not anticipated for postulated accidents at the PULSTAR reactor.

$$\mathbf{H_{ede} = H_{xyz} \bullet T} \qquad \text{for distances} \geq 100 \text{ m}$$

$$\mathbf{H_{ode} = H_{thy} \bullet T} \qquad \text{for distances} \geq 100 \text{ m}$$

$$\mathbf{H_{ede} = H_{ode} = H_{shine} \bullet T} \qquad \text{for distances} < 100\text{ m}$$

Worse Case Reference Off-site Dose Rate Projections

Off-site dose rate projections for locations of interest were made using the data and equations given above for a reference wind speed, reference concentration, and worse case χ/Q value. The reference wind speed used was 0.447 m/s (1 mph) and the reference concentration [C] used was 1 µCi/ml. At a wind speed of 1 m/s (2.24 mph), doses would be lower.

Dose projections for Ar-41 releases and the fuel handling accident described in the FSAR were calculated. All projected doses are well below PAG values and 10 CFR 20 limits. Worse case results for occupied areas are summarized below may be used if no other data is available.

WORSE CASE RELEASE	**NORMAL VENTILATION DOSE Rem**	**CONFINEMENT VENTILATION DOSE Rem**	**CONFINEMENT with BEL Dilution DOSE Rem**
Ar-41	1 E-2 (DDE)	3 E-4 (DDE)	2 E-5 (DDE)
Fuel Failure	3 E-4 (TEDE) 6 E-3 (thyroid)	4 E-5 (TEDE) 6 E-5 (thyroid)	2 E-6 (TEDE) 3 E-6 (thyroid)

ATTACHMENT 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

June 12, 1997

NRC INFORMATION NOTICE 97-34: DEFICIENCIES IN LICENSEE SUBMITTALS REGARDING TERMINOLOGY FOR RADIOLOGICAL EMERGENCY ACTION LEVELS IN ACCORDANCE WITH THE NEW PART 20

Addressees

All holders of operating licenses or construction permits for test and research reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of the issuance of a revised Appendix I to NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors." This revised appendix provides guidance on incorporating the new 10 CFR Part 20 regulations into the radiological emergency action levels (EALs). It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Part 20 was revised and the final rule was published in the Federal Register on May 21, 1991 (56 FR 23360), with an original implementation date of January 1, 1993. Subsequently, the implementation date was extended to January 1, 1994 (57 FR 38588). In response to these regulations, many licensees of test and research reactors have submitted to the NRC revised emergency plans that were intended to incorporate or reflect the terminology introduced in the new regulations. Title 10 of the Code of Federal Regulations, Part 20 is referenced in the section of the emergency plan involving radiological EALs.

During its review of these submittals, the NRC staff noted that many licensees incorrectly converted the wording of the EALs from the old terminology found in NUREG-0849. As an example, (1) some licensees inappropriately converted "maximum permissible concentrations" (MPCs) to "effluent concentrations" (ECs) and (2) some inaccurately used the term "deep dose equivalent." Several licensees merely substituted the EC term for the MPC term. Other licensees used various multipliers in an attempt to correct the differences between the two values. Still other licensees replaced the MPC term with "derived air concentration." These submittals indicate the need for additional guidance that could be used to assist in updating EALs.

9706090329

Discussion

If the errors in the revised EAL schemes are left uncorrected, they could potentially result in a licensee taking some conservative actions that are unwarranted or in the licensee delaying appropriate actions.

Guidance in the form of a revision to Appendix I to NUREG-0849 was issued in

April 1997. The revised appendix provides guidance on incorporating the current 10 CFR Part 20 regulations into the radiological EALs and, at the same time, corrects some omissions in prior guidance and clarifies some other EALs for the implementation of the new Part 20 regulations. Attachment 1 to this document is a copy of the revised Appendix I. NUREG-0849, which was published in 1983, is available from the following sources:

þ The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20037 (202-634-3380). (Document may be copied for a fee, currently $0.08 per page.)

þ The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328 (202-512-1800), for a fee of $3.75.

þ The National Technical Information Service, Springfield, VA 22161-0002 (703-487-4650), for a fee of $24.50 (paper) and $12.50 (microfiche), plus a $4.00 handling fee.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate regional office.

signed by S.H. Weiss for

Marylee M. Slosson, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contact: Lawrence K. Cohen
301-415-2923
E-mail: lkc@nrc.gov

Attachments:
1. Revised Appendix I of NUREG-0849.

APPENDIX 1
EMERGENCY CLASSES

ACTION LEVEL*	PURPOSE
Notification of Unusual Events (NOUE)	
Actual or projected radiological effluentþ at the site boundary which is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:	þ Ensure that the first step in any response later found to be necessary has been carried out. þ Bring the operating staff to a state of readiness.
(1) A deep dose equivalent of 0.15 milliSievert (mSv) [15 mrem] OR (2) A committed effective dose equivalent of 0.15 mSv [15 mrem] based on the following considerations:	þ Provide systematic handling of unusual events information and decisionmaking.
þ 100 EC þ 24 hr** = 2.4 þ 103 EC-hr ÷ 0.15 mSv [15 mrem] (for radionuclides other than noble gases)	
þ 50 EC þ 24 hr** = 1.2 þ 103 EC-hr ÷ 0.15 mSv [15 mrem] (for noble gases)þ	
Report or observation of a severe natural phenomenon affecting the reactor site	
Receipt of bomb threat affecting the reactor facility	
Fire within the reactor facility not extinguished within 15 minutes	
Alert	
Actual or projected radiological effluentþ at the site boundary which is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:	þ Ensure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required.
(1) A deep dose equivalent of 0.75 mSv [75mrem] OR (2) A committed effective dose equivalent of 0.75 mSv [75 mrem] based on the following considerations:	þ Provide current offsite authorities with status information.
þ 500 EC þ 24 hr** = 1.2 þ 104 EC-hr ÷ 0.75 mSv [75 mrem] (for radionuclides other than noble gases)	
þ 250 EC þ 24 hr** = 6 þ 103 EC-hr ÷ 0.75 mSv [75 mrem] (for noble gases)þþ	
Actual or projected radiation levelsþ at the site boundary of 0.2 mSv/hr deep dose equivalent [20 mrem/hr] for 1 hour or 1.0 mSv [100 mrem] to the thyroid (committed dose equivalent).	

APPENDIX I (contþd)

ACTION LEVEL*	PURPOSE

Site Area Emergency

Actual or projected radiological effluentþ at the site boundary which is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:

(1) A deep dose equivalent of 3.75 mSv [375 mrem]
OR
(2) A committed effective dose equivalent of 3.75 mSv [375 mrem] based on the following considerations:

þ 2500 EC þ 24 hr** = 6 þ 104 EC-hr ÷ 3.75 mSv [375 mrem] (for radionuclides other than noble gases)

þ 1250 EC þ 24 hr** = 3 þ 104 EC-hr ÷ 3.75 mSv [375 mrem] (for noble gases)þþ

Actual or projected radiation levelsþ at the site boundary of 1.0 mSv/hr [100 mrem/hr] deep dose equivalent for 1 hour or 5.0 mSv [500 mrem] to the thyroid (committed dose equivalent)

þ Ensure that response centers are manned.

þ Ensure that monitoring teams are dispatched.

þ Ensure that personnel required for evacuation of onsite areas are at duty stations

þ Provide consultation with offsite authorities.

þ Provide information for the public through offsite authorities.

General Emergency

Sustained actual or projected radiation levelsþ at the site boundary of 5.0 mSv/hr [500 mrem/hr] deep dose equivalent

Actual or projected doseþ at the site boundary in the plume exposure pathway of 10 mSv [1 rem] (total effective dose equivalent) or 50 mSv [5 rem] to the thyroid (committed dose equivalent)

þ Initiate predetermined protective actions for the public.

þ Provide continuous assessment of information from licensee and offsite organization measurements.

þ Initiate additional measures as indicated by actual or potential releases.

þ Provide consultation with offsite authorities.

þ Provide updates for the public through offsite authorities.

* The situation that may lead to an emergency class described in the subsections of Section 4.0 may be referenced as emergency action levels appropriate to the emergency class.

þ It is expected that licensees will determine the relationship of the EAL dose levels at the site boundary to instrumentation readings and/or safety analyses accident conditions for their specific facilities..

** Effluent concentration (EC) as listed in Title 10 of the Code of Federal Regulations, Part 20 (10 CFR Part 20),

þ Standards for the Protection Against Radiation,þ Appendix B, Table 2. If the exposure time is less than 24 hours, the EC multiplier can be increased portionately, provided that the values of 2.4 þ 103 and 1.2 þ 103 EC-hr are used to declare a NOUE; the proportional increase for an alert is 5 and for a site area emergency it is 25.

þþ Table 2 of Appendix B to 10 CFR Part 20 lists the concentration values that are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.5 mSv [50 mrem]. However, for noble gases where the submersion (external dose) is limiting, the concentration values would produce a total effective dose equivalent of 1 mSv [100 mrem].

ATTACHMENT 4

Appendix B to Part 20--Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

Table 2

The columns in Table 2 of this appendix captioned "Effluents," "Air," and "Water," are applicable to the assessment and control of dose to the public, particularly in the implementation of the provisions of § 20.1302. The concentration values given in Columns 1 and 2 of Table 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts).

Consideration of non-stochastic limits has not been included in deriving the air and water effluent concentration limits because non-stochastic effects are presumed not to occur at the dose levels established for individual members of the public. For radionuclides, where the non-stochastic limit was governing in deriving the occupational DAC, the stochastic ALI was used in deriving the corresponding airborne effluent limit in Table 2. For this reason, the DAC and airborne effluent limits are not always proportional as was the case in appendix B to §§ 20.1-20.601.

The air concentration values listed in Table 2, Column 1, were derived by one of two methods. For those radionuclides for which the stochastic limit is governing, the occupational stochastic inhalation ALI was divided by 2.4×10^{9}ml, relating the inhalation ALI to the DAC, as explained above, and then divided by a factor of 300. The factor of 300 includes the following components: a factor of 50 to relate the 5-rem annual occupational dose limit to the 0.1-rem limit for members of the public, a factor of 3 to adjust for the difference in exposure time and the inhalation rate for a worker and that for members of the public; and a factor of 2 to adjust the occupational values (derived for adults) so that they are applicable to other age groups.

For those radionuclides for which submersion (external dose) is limiting, the occupational DAC in Table 1, Column 3, was divided by 219. The factor of 219 is composed of a factor of 50, as described above, and a factor of 4.38 relating occupational exposure for 2,000 hours per year to full-time exposure (8,760 hours per year). Note that an additional factor of 2 for age considerations is not warranted in the submersion case.

The water concentrations were derived by taking the most restrictive occupational stochastic oral ingestion ALI and dividing by 7.3×10^{7}. The factor of 7.3×10^{7} (ml) includes the following components: the factors of 50 and 2 described above and a factor of 7.3×10^{5} (ml) which is the annual water intake of "Reference Man."

Note 2 of this appendix provides groupings of radionuclides which are applicable to unknown mixtures of radionuclides. These groupings (including occupational inhalation ALIs and DACs, air and water effluent concentrations and sewerage) require demonstrating that the most limiting radionuclides in successive classes are absent. The limit for the unknown mixture is defined when the presence of one of the listed radionuclides cannot be definitely excluded either from knowledge of the radionuclide composition of the source or from actual measurements.

ATTACHMENT 5: SAR Section 13 August,1996

13.2.2 Excursion Accidents

13.2.2.1 Fuel Loading Accident

The fuel loading accident is the MCA for excursion type accidents associated with the NCSU PULSTAR Reactor. The maximum worth of a fuel assembly for the existing 5 by 5 Graphite Reflected Core No. 3 has been measured to be 1.130%Δk/k (1130 pcm) (see Section 3.2.3.5.3.5). It is estimated that a twenty five assembly core of PULSTAR fuel is capable of absorbing a step input of more than 1.59%Δk/k (1590 pcm) without failure of fuel pin cladding. The 1.59%Δk/k (1590 pcm) reactivity insertion would yield an estimated 58 MW·s pulse (see Figure 3-20) which has been established as the Safety Limit for a pulse as detailed in Section 3.2.4.2.3.

To demonstrate this, it is assumed that erroneous handling of the fuel does occur under the following conditions:

(1) The core has been loaded in the optimum configuration.

(2) The reactor is critical.

(3) A fuel assembly is dropped from a height of two feet above the core and subsequently enters the optimum location.

The optimum position for the 5 by 5 Graphite Reflected Core No. 3 has a measured worth of 1.13%Δk/k (1130 pcm). To get the upper limit of this accident, it is assumed the reactivity is a step input. The resulting pulse has an estimated energy release of 40 MW·s (see Figure 3-20) and a maximum specific energy production of 325 watt·s/gram. This is less than the specific energy density of 400 watt·s/gram design criterion. A fuel loading accident therefore does not jeopardize the safety of operating personnel or the general public.

For future core arrangements, the maximum fuel assembly worth shall be measured during start-up testing and limited to 1.59%Δk/k (1590 pcm).

13.2.2.2 Start-up Accident

The following accident analysis was made to determine the results of the continuous rod withdrawal

13.3 Conclusions

A review of those accidents which could possibly occur, both in the non-excursion and excursion categories, indicates that the maximum credible accident is of the non-excursion category and consists of a loss of pool water due to a ruptured inlet or outlet pipe line. Even in this event, there is no core melting or loss of cladding integrity. The hazard associated is related only to the vertical radiation beam emanating from the unshielded shutdown core. Corrective measures can be taken to plug the leak and refill the tank without danger from the vertical radiation beam.

The maximum credible excursion accident is of even less severity than the postulated non-excursion accident. It consists a fuel loading accident whereby up to 1.59%Δk/k (1590 pcm) step input is added to the reactor core. This value reactivity step input is found to produce a lower specific energy release density than that experienced by the PULSTAR fuel pins used in the initial Buffalo PULSTAR pulse test core.

ATTACHMENT 6: FUELED EXPERIMENT ANALYSIS

INTRODUCTION

The fueled experiment accident analysis used to support the current Technical Specifications is bounded by the fuel handling accident. This bounding condition is too restrictive and not realistic.

Reactor fuel is irradiated for a substantial period which allows the buildup of long-lived radionuclides. The SAR analysis considers only a few radionuclides to be present. The current limitation on fueled experiments restricts the production of activity to the production rates for radioiodines and radiobromines given in the fuel pin annuli. As a result, noble gas production for the fueled experiment accident is lower than that for the fuel handling accident. The associated dose to workers and members of the public from a fueled experiment are therefore lower than that for the postulated fuel handling accident. Doses to workers for the postulated fuel handling accident results in doses of 50 mrem total effective dose-equivalent and 2000 mrem committed dose-equivalent to the thyroid assuming an individual were present during the entire event time with no respiratory protection. Doses to members of the public following a fuel handling accident are calculated to be approximately 0.02 mrem.

For a fueled experiment in a dry experimental facility, particulate fission products may be released during a mishandling accident. Most fueled experiments are typically short in duration, so shorter lived radionuclides would represent the hazard. Up to 10% of the applicable dose limits is a reasonable limitation since that is where actions require an official regulatory response, i.e. monitoring of occupational personnel and reporting of doses in excess of the constraint dose for members of the public.

It is therefore prudent to more realistically evaluate the source term and doses to occupational workers and members of the public in determination of limitations and conditions associated with a fueled experiment.

SOURCE TERM

Activity, A, for a given radionuclide is calculated as follows:

$$A = dN/dt = \text{formation rate} - \text{destruction rate}$$

$$dN/dt = \sigma\varphi\gamma N_{U\text{-}235} - \lambda N$$

where, σ, U-235 thermal neutron fission cross-section of 585 E-24 cm^{-2}
φ, the thermal neutron flux density in n cm^{-2} s^{-1}
γ, the fission product cumulative yield (i.e. chain yield)
$N_{U\text{-}235}$, the number of U-235 atoms present in the target
λ, the decay constant of the radionuclide
N, the number of radioactive atoms present

Solving for the activity at the end of irradiation time, t, or A(0), gives the following:

$$N = \sigma\varphi\gamma N_{U\text{-}235}\,[1 - \exp(-\lambda t)] / \lambda$$
$$A(0) = \lambda N = \sigma\varphi\gamma N_{U\text{-}235}\,[1 - \exp(-\lambda t)]$$

where, t is the irradiation time

The number of radionuclides evaluated was increased from the few listed for the fuel handling accident to approximately 500. This increase was a result of considering many short-lived fission products.

Thermal neutron fission yields for U-235 for the fission product chain were used to account for the decay of precursors leading to the production of the fission product. Corrections for decay to daughter nuclides were made.

For example;	Atomic mass number 92 has a chain yield of 0.0603	
	Kr-92 decays to Rb-92 decays to Sr-92 decays to Y-92	
	Kr-92 yield is reported as 0.0187, or	
	Kr-92 yield	= Chain Yield – yield for (Rb-92+ Sr-92 + Y-92)
		= 0.0603 – (0.0343+ 0+0.0073) = 0.0187
	Rb-92 yield	= 0.0187 = 0.0343 = 0.053
	Sr-92 yield	= 0.0187+0.0343+0 = 0.053
	Y-92 yield	= 0.0598 = 0.0187+0.0343+0.0073 = 0.0603

Including decay by precursors is conservative and for longer irradiation times is a good approximation to the correct cumulative yield.

CONCENTRATION and TIME INTEGRATED CALCULATIONS

After the source is produced, the source is assumed to be removed from the experiment without delay with all of the activity released to the reactor bay instantaneous resulting in uniform airborne activity distribution throughout the entire reactor bay. Initial concentration, C(0), in the reactor bay is given by the following:

$$C(0) = A(0) / V = A(0) / 2.25E9 \text{ ml}$$

where, V = 2.25 E9 ml is the reactor bay free air volume

For short-lived radionuclides, the time-integrated exposure and removal by radioactive decay and the ventilation system are taken into account as follows:

$$\int^{T} C(0) \exp(-kt)\, dt = [C(0) / k]\, [1 - \exp(-kT)]$$

where, $k = \lambda + v$, and
v is the confinement ventilation mode removal rate constant
$v = 1.18\text{ E-4 } s^{-1}$ at a 600 cfm exhaust rate

By assuming an individual is present for 1 air exchange at the initial (or peak) concentration, C(0), is equivalent to placing the individual in the area for the 24 hour period exposed to the 24 h average concentration, i.e. the same integrated concentration-hour value (uCi-h/ml) is obtained. This simplifies the above equation to the following:

If T = 24 h and if C(0) is in uCi/ml, then

uCi-h/ml = [C(0) / k] [1 - exp(-kT)] = C(0) / k [~ 0.99996] , or

uCi-h/ml = C (0) / k

Time integrated exposure in public areas is further reduced by removal of halogens and particulates by the confinement filters and by atmospheric dispersion as follows:

uCi-h.ml = [C(0) / k](1-R)(7.6 E-3)

where, R = 0.99 for halogens and 0.9997 for particulates
R = 0 for noble gases
7.6 E-3 is reciprocal of the atmospheric dilution factor (ADF) evaluated at the limiting location for Class F weather stability at a wind speed of 1 m/s
ADF = 1 s m^{-3} / 7.6 E-3 s m^{-3}

NOTE: 1 s m^{-3} is the dilution factor at the top of the stack, i.e. no dilution is present

Exposure times are taken as 24 hours for members of the public.

For occupational workers, an exposure time of 0.25 hours is assumed to allow time for the release and detection of the contaminated air followed by evacuation from the reactor bay. No credit for respiratory protection is assumed.

DOSE ASSESSMENT

Dose to occupational workers and members of the public is determined as follows:

Dose = (Time-Integrated Exposure)(DCF)

where, DCF = Dose Conversion Factor taken from 10 CFR 20 Appendix B

For H3, halogens and particulates:

$$\text{Effective DCF} = \frac{(0.05 \text{ rem} / 8760 \text{ h})}{[\text{10CFR20 Appendix B Table 2 air concentration in uCi/ml}]}$$

For noble gases:

$$\text{Effective DCF} = \frac{(0.1 \text{ rem} / 8760 \text{ h})}{[\text{10CFR20 Appendix B Table 2 air concentration in uCi/ml}]}$$

For radioiodines:

$$\text{Thyroid DCF} = \frac{(5 \text{ rem} / 2000 \text{ h})}{[\text{10CFR20 Appendix B Table 1 derived air concentration in uCi/ml}]}$$

Limiting values given in 10 CFR 20 Appendix B were used for the various inhalation classes.

MASS LIMITATION

The U-235 mass limits depend on flux, irradiation time, and the postulated accident dose. Dose to occupational workers is limited to 0.5 rem total effective dose-equivalent and 5 rem committed effective dose-equivalent to the thyroid. Dose to members of the public is limited to 0.01 rem (constraint dose) total effective dose-equivalent.

The U-235 mass was determined at a thermal neutron fluence rate of 1 E13 cm^{-2} s^{-1} for various irradiation times ranging from 1 minute to an infinite time. The thermal neutron flux of 1 E13 cm^{-2} s^{-1} is a typical maximum thermal neutron flux measured in reactor experimental facilities.

Calculation results are presented in different formats on the following pages. The mass is a function of fluence and irradiation time. The mass is calculated based on dose considerations only. Other factors may limit the mass to a lower value, such as license possession limits, maintaining a uniform flux in the experimental beam, reactivity and heat limits for experiments.

Input and output data from the calculations for 1 g of U-235 irradiated at a thermal neutron flux of 1E13 cm^{-2} s^{-1} are given on the following pages as an example. The lowest value of the output data is used as the limit, i.e. in this example 8.34 mg at the fluence of (60s)(1 E13 cm^{-2} s^{-1}). This limits a 1 minute irradiation to 5.0 E15 mg cm^{-2}.

Calculation data:

INPUT DATA	Bay dose	Thy dose	Pub dose
Rem limit	0.5	5	0.01
Flux = *1E+13*	8.34E+00	4.86E+02	1.74E+01
Irrad time s= *6.00E+01*			
U235 g= *1*			
Decay Time s= *0*			

e.g. if the test flux is 1 E 11 cm^{-2} s^{-1} for 1 minute, then the mass is limited to:

$$\frac{5.0\text{ E}15\text{ mg cm}^{-2}}{60\text{ E }11\text{cm}^{-2}} = 833\text{ mg}$$

Calculation results are also presented in a table and graph for select times. To obtain an experiment mass limit for an irradiation time, t, the graph or table may be used to determine the mass limit at the same irradiation time, t, at the fluence rate of rate of 1 E13 cm^{-2} s^{-1} divided by the test fluence rate. For example, if the irradiation time is 4 hours and the fluence rate is 1 E8 cm^{-2} s^{-1}, then the mass limit is as follows:

$$8130\text{ mg} = \frac{(0.0813\text{ mg})(1\text{ E}13\text{ cm}^{-2}\text{ s}^{-1})}{(1\text{ E}8\text{ cm}^{-2}\text{ s}^{-1})}$$

The data from the calculations for select times was also graphed and then separated into 4 groups based on time and fluence; (1) 10 – 60 minutes, (2) 1 – 10 hours, (3) 10 to 100 hours, and (4) 100 h to 1 y. A line of best fit was determined for each time group. Mass-fluence, (mg cm^{-2}) was plotted vs. time.

To determine the test mass, the line of best fit for the applicable time group is adjusted for the test flux as follows:

$$\text{Mass limit for experiment at the test irradiation time (t)} = \frac{(\text{mg cm}^{-2}\text{ evaluated at t})}{\text{Test thermal neutron fluence(in cm}^{-2})}$$

For each group, the line of best fit equations are as follows:

Group	Times	mg cm $^{-2}$ evaluated at 1 E13 cm^{-2} s^{-1}	
-	1 minute	8.3 E13	
1	10-60 minutes	$8\ E15 * [t/10]^{0.1537}$	t in minutes
2	1 – 10 hours	$1.06\ E16 * t^{0.0576}$	t in hours
3	10-100 hours	9 E14*(t/10) + 1E14	t in hours
4	100-600 h	$-8E13*(t/100)^3+1E15*(t/100)^2+9E15(t/100)+1e16$	t in hours

For 600 h to1y, the mass is limited to 4.19 E-3 mg at fluence rate of 1 E13 cm^{-2} s^{-1} .

These equations may be used to estimate the U-235 for an experiment at any time greater than 10 minutes.

Using the same example as above for an irradiation time of 4 hours at a fluence rate of is 1 E8 cm^{-2} s^{-1}, the mass for the experiment is calculated as follows:

7973 mg = $1.06\ E16 * 4^{0.0576}$ vs. the Tabular value of 8130 mg

The difference in this example is within 2%.

FISSION RATE and TOTAL ENERGY RELEASE

The fission rate (R_f) and total energy release rate (R_E) are calculated as follows:

$R_f = \sigma\varphi N_{U\text{-}235}$ in fissions per second
R_E = (200 MeV per fission)($\sigma\varphi N_{U\text{-}235}$) in MeV per second
R_E = (200 MeV per fission)($\sigma\varphi N_{U\text{-}235}$)(1 watt / 6.243 E12 MeV per s) in watts
Energy release = (R_E in watts)(Irradiation time in seconds), in Joules

Data is given in a table on the following pages for irradiation times ranging from 1 minute to 1year. For each of the 4 groups discussed above, the maximum fission rate and energy release are as follows:

Group	Time	Fission rate	milliwatts
-	1 minute	1.2 E11	4010
1	10 minutes	2.0 E10	639
2	1 hour	4.4 E9	140
3	10 hours	5.1 E8	16.3
4	100hours	8.6 E7	2.75

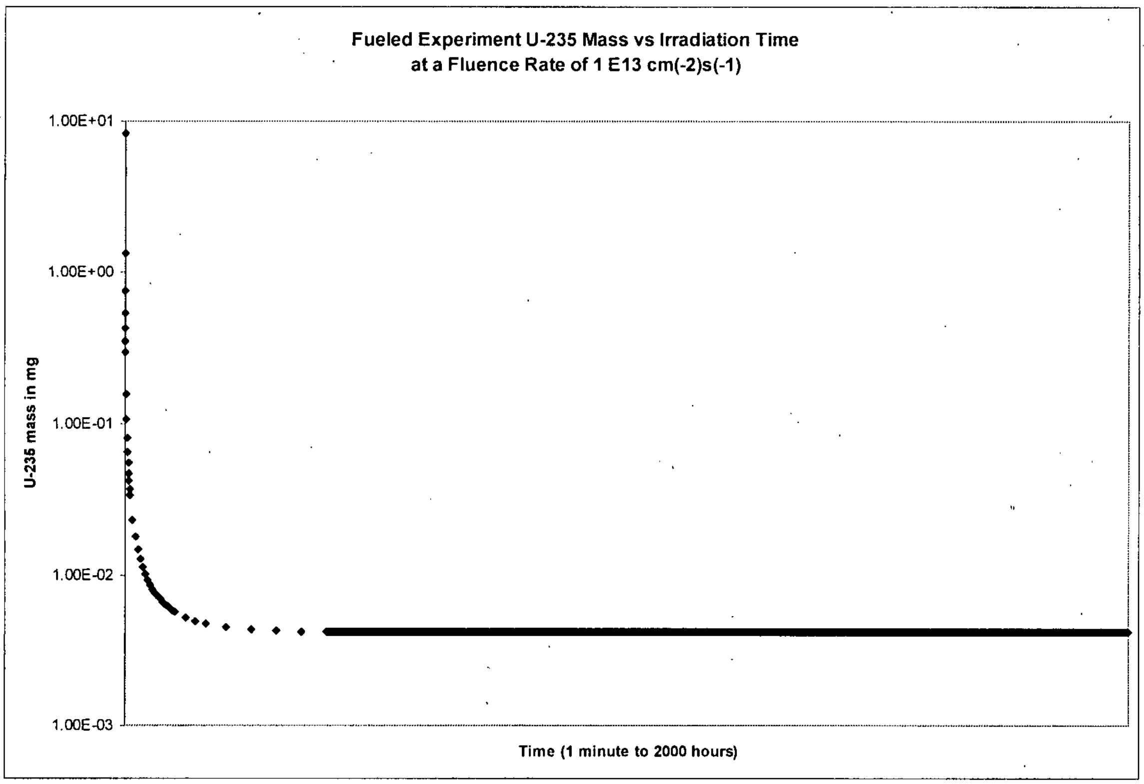
The fission rate and energy release are dependent on the U-235 mass, fluence rate, and irradiation time.

CONCLUSIONS

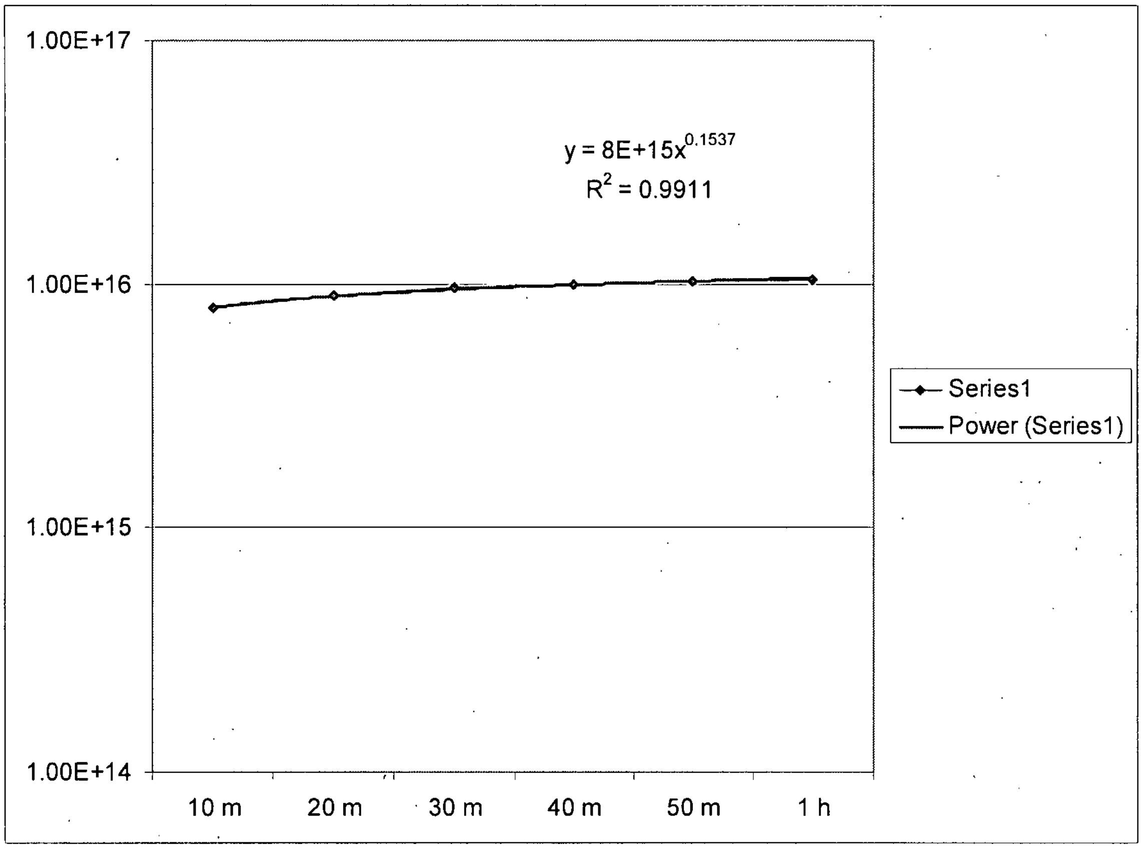
Fueled experiments have the following limitations and conditions:

1. The mass is limited based on the fluence rate and duration of the irradiation to the more limiting of the following:
   - 0.5 rem total effective dose-equivalent to occupational personnel
   - 5 rem committed dose-equivalent to the thyroid
   - rem total effective dose-equivalent to members of the public
2. Fueled experiments require activation of the confinement system.
3. Fueled experiments are limited to the reactor building.
4. Each type of fueled experiment shall be classified as a new (untried) experiment with a documented review. The documented review shall include the following items:
   - i. Meeting license requirements for the receipt, use, and storage of fissionable material.
   - ii. Limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.
   - iii. Radiation monitoring for detection of released fission products.
   - iv. Design criteria related to meeting conditions given in Specifications 3.2 and 3.7.

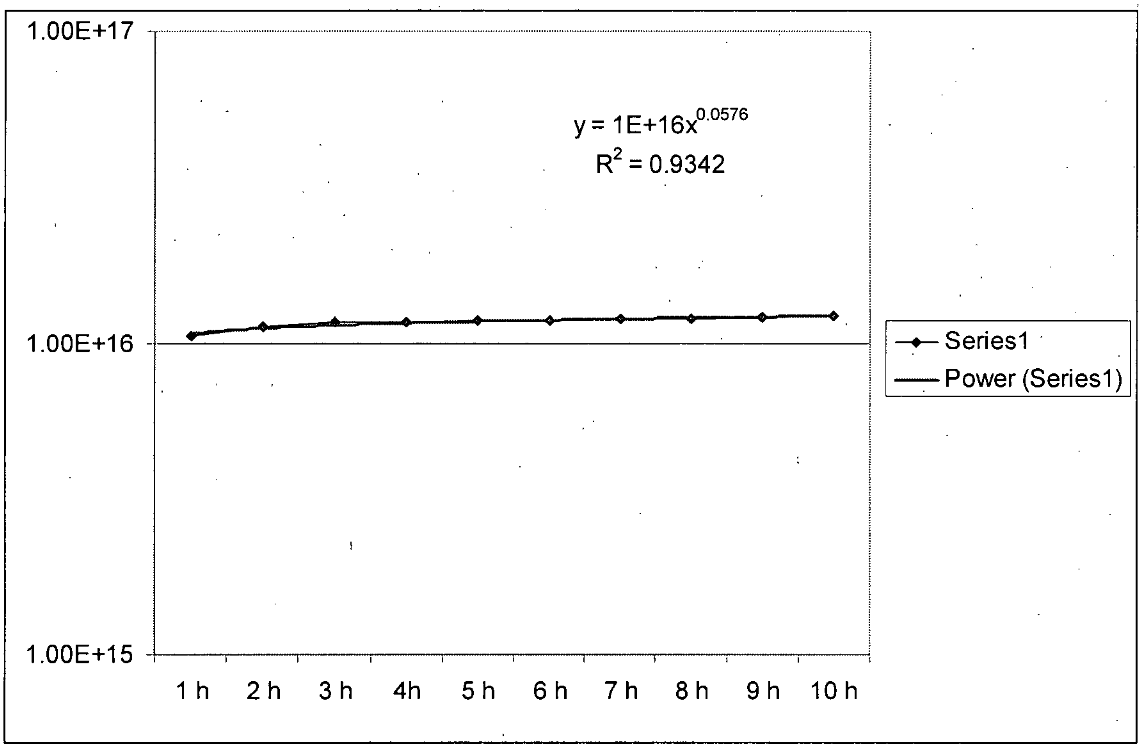
Graph of calculation results: (for times > 500 h the mass limit is the same)



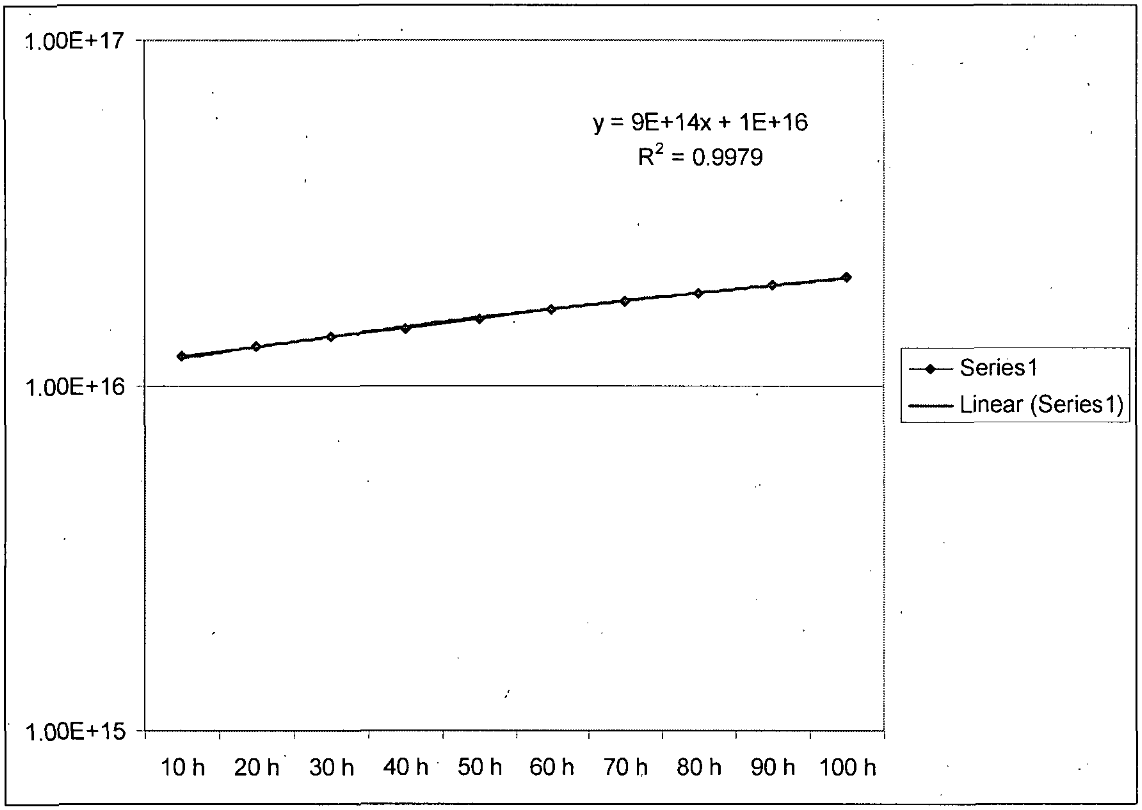
Data for Fueled Experiments at a Fluence Rate of 1 E13 $cm^{-2}s^{-1}$				
Irradiation Time s	U-235 Mass mg	Mass-Fluence mg cm(-2)	Fission Rate f/s	Power milliwatts
6.00E+01	8.34E+00	5.00E+15	1.25E+11	4.01E+03
1.20E+02	4.75E+00	5.70E+15	7.12E+10	2.28E+03
1.80E+02	3.44E+00	6.19E+15	5.16E+10	1.65E+03
3.00E+02	2.30E+00	6.90E+15	3.45E+10	1.10E+03
6.00E+02	1.33E+00	7.98E+15	1.99E+10	6.39E+02
1.20E+03	7.55E-01	9.06E+15	1.13E+10	3.63E+02
1.80E+03	5.36E-01	9.65E+15	8.04E+09	2.57E+02
2.40E+03	4.18E-01	1.00E+16	6.27E+09	2.01E+02
3.00E+03	3.43E-01	1.03E+16	5.14E+09	1.65E+02
3.60E+03	2.92E-01	1.05E+16	4.38E+09	1.40E+02
7.20E+03	1.57E-01	1.13E+16	2.35E+09	7.54E+01
1.08E+04	1.08E-01	1.17E+16	1.62E+09	5.19E+01
1.44E+04	8.13E-02	1.17E+16	1.22E+09	3.90E+01
1.80E+04	6.55E-02	1.18E+16	9.82E+08	3.15E+01
2.16E+04	5.49E-02	1.19E+16	8.23E+08	2.64E+01
2.52E+04	4.74E-02	1.19E+16	7.11E+08	2.28E+01
2.88E+04	4.18E-02	1.20E+16	6.27E+08	2.01E+01
3.24E+04	3.74E-02	1.21E+16	5.61E+08	1.80E+01
3.60E+04	3.39E-02	1.22E+16	5.08E+08	1.63E+01
7.20E+04	1.81E-02	1.30E+16	2.71E+08	8.69E+00
1.08E+05	1.28E-02	1.38E+16	1.92E+08	6.15E+00
1.44E+05	1.02E-02	1.47E+16	1.53E+08	4.90E+00
1.80E+05	8.67E-03	1.56E+16	1.30E+08	4.16E+00
2.16E+05	7.66E-03	1.65E+16	1.15E+08	3.68E+00
2.52E+05	6.95E-03	1.75E+16	1.04E+08	3.34E+00
2.88E+05	6.42E-03	1.85E+16	9.62E+07	3.08E+00
3.24E+05	6.03E-03	1.95E+16	9.04E+07	2.90E+00
3.60E+05	5.72E-03	2.06E+16	8.57E+07	2.75E+00
3.96E+05	5.27E-03	2.09E+16	7.90E+07	2.53E+00
4.32E+05	4.97E-03	2.15E+16	7.45E+07	2.39E+00
4.68E+05	4.77E-03	2.23E+16	7.15E+07	2.29E+00
7.20E+05	4.51E-03	3.25E+16	6.76E+07	2.17E+00
1.08E+06	4.27E-03	4.61E+16	6.40E+07	2.05E+00
1.44E+06	4.21E-03	6.06E+16	6.31E+07	2.02E+00
1.80E+06	4.19E-03	7.54E+16	6.28E+07	2.01E+00
2.16E+06	4.19E-03	9.05E+16	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
1.73E+07	4.19E-03	7.24E+17	6.28E+07	2.01E+00
3.15E+07	4.19E-03	1.32E+18	6.28E+07	2.01E+00



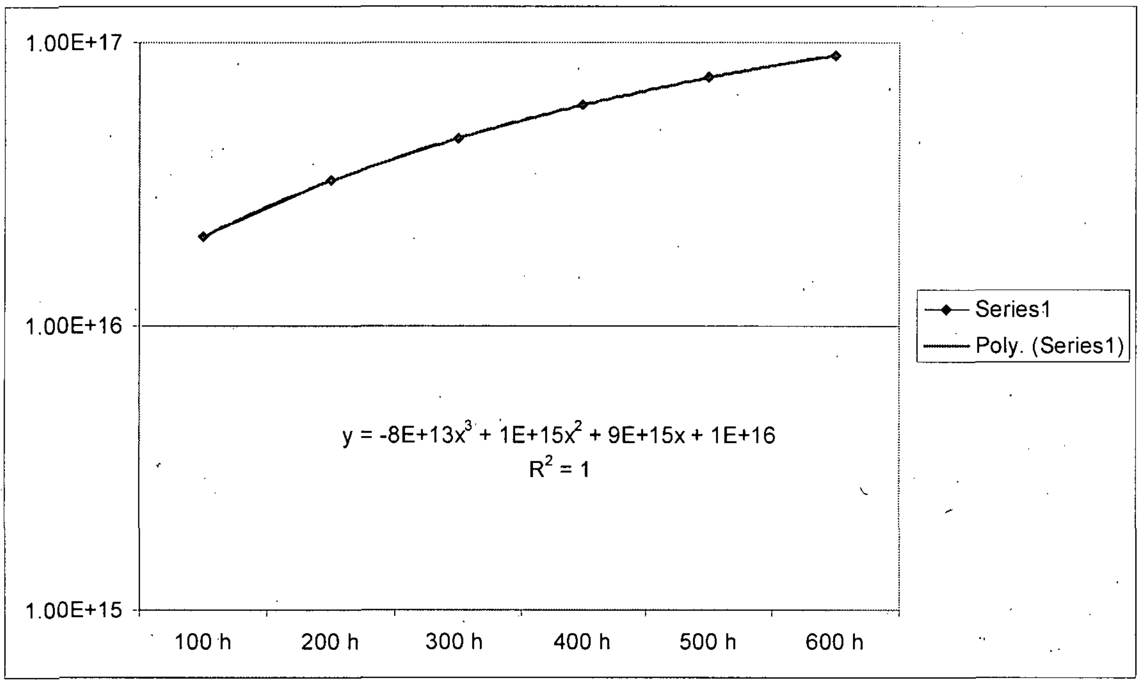
1.00E+17
1.00E+16
1.00E+15
1.00E+14
y = 8E+15x^0.1537
R^2 = 0.9911
10 m
20 m
30 m
40 m
50 m
1 h
Series1
Power (Series1)



1.00E+17
1.00E+16
1.00E+15
y = 1E+16x^0.0576
R^2 = 0.9342
1 h
2 h
3 h
4h
5 h
6 h
7 h
8 h
9 h
10 h
Series1
Power (Series1)



1.00E+17
1.00E+16
1.00E+15
y = 9E+14x + 1E+16
R2 = 0.9979
10 h
20 h
30 h
40 h
50 h
60 h
70 h
80 h
90 h
100 h
Series1
Linear (Series1)



1.00E+17
1.00E+16
1.00E+15
y = -8E+13x3 + 1E+15x2 + 9E+15x + 1E+16
R2 = 1
100 h
200 h
300 h
400 h
500 h
600 h
Series1
Poly. (Series1)

SPREADSHEET EXAMPLE AT TIME OF 60 s and Flux of 1 E13 $cm^{-2}s^{-1}$

Nuclide	Thermal Yield	Fast Yeild	T1/2 seconds	Form (P/V/G)	Partition Factor	Filter Factor	Table 1 uCi/ml	Table 2 uCi/ml
H3	1.08E-02	2.00E-02	3.87E+08	V	1	1	2E-05	1E-07
Kr83m	5.44E-03	5.98E-03	6.70E+03	G	1	1	1E-02	5E-05
Kr85m	1.24E-02	3.98E-03	1.61E+04	G	1	1	2E-05	1E-07
Kr85	1.32E-02	4.08E-03	3.39E+08	G	1	1	1E-04	7E-07
Kr87	2.56E-02	2.59E-02	4.57E+03	G	1	1	5E-06	2E-08
Kr88	3.58E-02	3.41E-02	1.02E+04	G	1	1	2E-06	9E-09
Kr89	4.73E-02	4.48E-02	1.89E+02	G	1	1	1E-07	1E-09
Kr90	5.03E-02	4.18E-02	3.23E+01	G	1	1	1E-07	1E-09
Kr91	3.45E-02	3.50E-02	8.60E+00	G	1	1	1E-07	1E-09
Kr92	1.87E-02	2.72E-02	1.84E+00	G	1	1	1E-07	1E-09
Kr93	4.80E-03	1.75E-02	1.29E+00	G	1	1	1E-07	1E-09
Kr94	1.00E-03	8.47E-03	2.10E-01	G	1	1	1E-07	1E-09
Kr95	2.30E-03	0.00E+00	7.80E-01	G	1	1	1E-07	1E-09
Kr97	5.90E-02	5.94E-02	1.00E-01	G	1	1	1E-07	1E-09
Xe131m	2.93E-02	3.07E-02	1.03E+06	G	1	1	4E-04	2E-06
Xe133m	6.70E-02	6.51E-02	1.89E+05	G	1	1	1E-10	1E-12
Xe133	6.70E-02	6.51E-02	4.53E+05	G	1	1	1E-04	5E-07
Xe135m	6.22E-02	5.66E-02	9.18E+02	G	1	1	9E-06	4E-08
Xe135	6.54E-02	6.15E-02	3.28E+04	G	1	1	1E-05	7E-08
Xe137	6.04E-02	5.26E-02	2.29E+02	G	1	1	1E-07	1E-09
Xe138	6.77E-02	4.67E-02	8.46E+02	G	1	1	4E-06	2E-08
Xe139	6.20E-02	3.21E-02	3.97E+01	G	1	1	1E-07	1E-09
Xe140	3.80E-02	2.04E-02	1.36E+01	G	1	1	1E-07	1E-09
Xe141	1.33E-02	1.17E-02	1.22E+00	G	1	1	1E-07	1E-09
Xe142	3.50E-03	2.22E-02	1.22E+00	G	1	1	1E-07	1E-09
Xe143m	2.29E-03	0.00E+00	9.60E-01	G	1	1	1E-07	1E-09
Xe143	2.80E-03	1.75E-03	3.00E-01	G	1	1	1E-07	1E-09
Xe144	1.20E-03	0.00E+00	1.20E+00	G	1	1	1E-07	1E-09
Xe145	5.00E-04	0.00E+00	9.00E-01	G	1	1	1E-07	1E-09
Br79m	5.60E-04	1.17E-03	4.86E+00	V	1	0.01	1E-07	1E-09
Br82m	3.20E-03	4.18E-03	3.66E+02	V	1	0.01	1E-07	1E-09
Br82	3.20E-03	4.18E-03	1.27E+05	V	1	0.01	1E-10	1E-12
Br83	5.10E-03	5.98E-03	8.64E+03	V	1	0.01	3E-05	9E-08
Br84m	1.00E-02	1.04E-02	3.60E+02	V	1	0.01	1E-07	1E-09
Br84	1.00E-02	1.04E-02	1.91E+03	V	1	0.01	2E-05	8E-08
Br85	1.10E-02	3.98E-03	1.72E+02	V	1	0.01	1E-07	1E-09
Br86	2.02E-02	0.00E+00	5.55E+01	V	1	0.01	1E-07	1E-09
Br87	2.56E-02	0.00E+00	5.59E+01	V	1	0.01	1E-07	1E-09
Br88	3.58E-02	0.00E+00	1.64E+01	V	1	0.01	1E-07	1E-09
Br89	4.73E-02	0.00E+00	4.40E+00	V	1	0.01	1E-07	1E-09
Br90	3.00E-04	0.00E+00	1.90E+00	V	1	0.01	1E-07	1E-09
Br91	3.45E-02	3.50E-02	5.40E-01	V	1	0.01	1E-07	1E-09
Br92	1.87E-02	2.72E-02	3.40E-01	V	1	0.01	1E-07	1E-09
Br93	4.80E-03	1.75E-02	1.00E-01	V	1	0.01	1E-07	1E-09
I129	1.00E-02	1.10E-02	4.95E+14	V	1	0.01	4E-09	4E-11
I131	2.93E-02	3.07E-02	6.93E+05	V	1	0.01	2E-08	2E-10

I132m	4.31E-02	4.77E-02	5.00E+03	V	1	0.01	4E-06	3E-08
I132	4.38E-02	4.77E-02	8.21E+03	V	1	0.01	3E-06	2E-08
I133m	6.20E-02	6.51E-02	9.00E+00	V	1	0.01	1E-07	1E-09
I133	6.70E-02	6.51E-02	7.49E+04	V	1	0.01	1E-07	1E-09
I134m	6.97E-02	4.77E-02	2.22E+02	V	1	0.01	1E-07	1E-09
I134	7.87E-02	6.62E-02	5.27E+01	V	1	0.01	2E-05	6E-08
I135	6.22E-02	5.66E-02	2.37E+04	V	1	0.01	7E-07	6E-09
I136m	1.00E-03	0.00E+00	4.70E+01	V	1	0.01	1E-07	1E-09
I136	3.10E-02	0.00E+00	8.34E+01	V	1	0.01	1E-07	1E-09
I137	6.04E-02	0.00E+00	2.45E+01	V	1	0.01	1E-07	1E-09
I138	6.77E-02	0.00E+00	6.50E+00	V	1	0.01	1E-07	1E-09
I139	7.90E-03	0.00E+00	2.30E+00	V	1	0.01	1E-07	1E-09
I140	1.00E-03	0.00E+00	8.60E-01	V	1	0.01	1E-07	1E-09
I141	1.00E-03	0.00E+00	2.00E-01	V	1	0.01	1E-07	1E-09
I142	3.50E-03	0.00E+00	2.00E-01	V	1	0.01	1E-07	1E-09
Ni72	2.60E-07	0.00E+00	2.10E+00	P	1	0.0003	1E-07	1E-09
Cu72	2.60E-07	0.00E+00	6.60E+00	P	1	0.0003	1E-07	1E-09
Zn72	2.60E-07	0.00E+00	1.67E+05	P	1	0.0003	5E-07	2E-09
Ga72	2.60E-07	0.00E+00	5.08E+04	P	1	0.0003	1E-06	4E-09
Ni73	1.00E-06	0.00E+00	9.00E-01	P	1	0.0003	1E-07	1E-09
Cu73	1.00E-06	0.00E+00	3.90E+00	P	1	0.0003	1E-07	1E-09
Zn73m	1.00E-06	0.00E+00	6.60E+01	P	1	0.0003	1E-07	1E-09
Zn73	1.00E-06	0.00E+00	2.40E+01	P	1	0.0003	1E-07	1E-09
Ga73	1.00E-06	0.00E+00	1.75E+04	P	1	0.0003	6E-06	2E-08
Ni74	3.00E-06	1.40E-05	1.10E+00	P	1	0.0003	1E-07	1E-09
Cu74	3.00E-06	1.40E-05	1.60E+00	P	1	0.0003	1E-07	1E-09
Zn74	3.00E-06	1.40E-05	9.60E+01	P	1	0.0003	1E-07	1E-09
Ga74m	3.00E-06	1.40E-05	1.00E+01	P	1	0.0003	1E-07	1E-09
Ga74	3.00E-06	1.40E-05	4.86E+02	P	1	0.0003	1E-07	1E-09
Cu75	3.00E-06	0.00E+00	1.30E+00	P	1	0.0003	1E-07	1E-09
Zn75	3.00E-06	0.00E+00	1.02E+01	P	1	0.0003	1E-07	1E-09
Ga75	1.10E-05	1.75E-04	1.26E+02	P	1	0.0003	1E-07	1E-09
Ge75m	1.10E-05	1.75E-04	4.80E+01	P	1	0.0003	1E-07	1E-09
Ge75	1.10E-05	1.75E-04	4.97E+03	P	1	0.0003	3E-05	1E-07
Cu76	6.00E-06	0.00E+00	6.40E-01	P	1	0.0003	1E-07	1E-09
Zn76	6.00E-06	0.00E+00	5.70E+00	P	1	0.0003	1E-07	1E-09
Ga76	3.10E-05	2.72E-04	2.90E+01	P	1	0.0003	1E-07	1E-09
Cu77	6.70E-05	4.09E-04	4.70E-01	P	1	0.0003	1E-07	1E-09
Zn77m	6.70E-05	4.09E-04	1.00E+00	P	1	0.0003	1E-07	1E-09
Zn77	6.70E-05	4.09E-04	2.10E+00	P	1	0.0003	1E-07	1E-09
Ga77	6.70E-05	4.09E-04	1.30E+01	P	1	0.0003	1E-07	1E-09
Ge77m	6.70E-05	4.09E-04	5.30E+01	P	1	0.0003	1E-07	1E-09
Ge77	8.30E-05	4.67E-04	4.07E+04	P	1	0.0003	4E-06	1E-08
As77	8.30E-05	4.67E-04	1.40E+05	P	1	0.0003	2E-06	7E-09
Se77m	8.30E-05	4.67E-04	1.74E+01	P	1	0.0003	1E-07	1E-09
Cu78	2.10E-04	0.00E+00	3.40E-01	P	1	0.0003	1E-07	1E-09
Zn78	2.10E-04	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09
Ga78	2.10E-04	0.00E+00	5.09E+00	P	1	0.0003	1E-07	1E-09
Ge78	2.10E-04	7.20E-04	5.22E+03	P	1	0.0003	1E-10	1E-12
As78	2.10E-04	7.39E-04	5.44E+03	P	1	0.0003	9E-06	3E-08
Cu79	5.60E-04	1.17E-03	1.90E-01	P	1	0.0003	1E-07	1E-09
Zn79	5.60E-04	1.17E-03	1.00E+00	P	1	0.0003	1E-07	1E-09

Ga79	5.60E-04	1.17E-03	2.85E+00	P	1	0.0003	1E-07	1E-09
Ge79m	5.60E-04	1.17E-03	3.90E+01	P	1	0.0003	1E-07	1E-09
Ge79	5.60E-04	1.17E-03	1.90E+01	P	1	0.0003	1E-07	1E-09
As79	5.60E-04	1.17E-03	5.40E+02	P	1	0.0003	1E-07	1E-09
Se79m	5.60E-04	1.17E-03	2.35E+02	P	1	0.0003	1E-07	1E-09
Se79	5.60E-04	1.17E-03	1.89E+13	P	1	0.0003	2E-07	8E-10
Zn80	3.00E-04	0.00E+00	5.40E-01	P	1	0.0003	1E-07	1E-09
Ga80	3.00E-04	0.00E+00	1.70E+00	P	1	0.0003	1E-07	1E-09
Ge80	3.00E-04	0.00E+00	2.95E+01	P	1	0.0003	1E-07	1E-09
As80	1.30E-03	1.85E-03	1.60E+01	P	1	0.0003	1E-07	1E-09
Zn81	5.96E-04	0.00E+00	2.90E-01	P	1	0.0003	1E-07	1E-09
Ga81	5.96E-04	0.00E+00	1.22E+00	P	1	0.0003	1E-07	1E-09
Ge81m	5.96E-04	0.00E+00	7.60E+00	P	1	0.0003	1E-07	1E-09
Ge81	5.96E-04	0.00E+00	7.60E+00	P	1	0.0003	1E-07	1E-09
As81	6.80E-04	2.92E-03	3.30E+01	P	1	0.0003	1E-07	1E-09
Se81m	2.00E-03	2.92E-03	3.44E+03	P	1	0.0003	3E-05	1E-07
Se81	2.00E-03	3.02E-03	1.11E+03	P	1	0.0003	9E-05	3E-07
Ga82	3.20E-03	4.18E-03	5.99E-01	P	1	0.0003	1E-07	1E-09
Ge82	3.20E-03	4.18E-03	4.60E+00	P	1	0.0003	1E-07	1E-09
As82m	3.20E-03	4.18E-03	1.37E+01	P	1	0.0003	1E-07	1E-09
As82	3.20E-03	4.18E-03	1.90E+01	P	1	0.0003	1E-07	1E-09
Ga83	2.90E-03	0.00E+00	3.10E-01	P	1	0.0003	1E-07	1E-09
Ge83	2.90E-03	0.00E+00	1.90E+00	P	1	0.0003	1E-07	1E-09
As83	2.90E-03	0.00E+00	1.34E+01	P	1	0.0003	1E-07	1E-09
Se83m	2.90E-03	0.00E+00	7.02E+01	P	1	0.0003	1E-07	1E-09
Se83	5.10E-03	5.64E-03	1.34E+03	P	1	0.0003	5E-05	2E-07
Ga84	1.00E-02	1.01E-02	9.00E-02	P	1	0.0003	1E-07	1E-09
Ge84	1.00E-02	1.01E-02	1.20E+00	P	1	0.0003	1E-07	1E-09
As84m	1.00E-02	1.01E-02	6.00E-01	P	1	0.0003	1E-07	1E-09
As84	1.00E-02	1.01E-02	5.50E+00	P	1	0.0003	1E-07	1E-09
Se84	1.00E-02	1.01E-02	1.92E+02	P	1	0.0003	1E-07	1E-09
Ge85	1.10E-02	1.16E-02	5.40E-01	P	1	0.0003	1E-07	1E-09
As85	1.10E-02	1.16E-02	2.03E+00	P	1	0.0003	1E-07	1E-09
Se85	1.10E-02	1.16E-02	3.20E+01	P	1	0.0003	1E-07	1E-09
As86	2.02E-02	0.00E+00	9.00E-01	P	1	0.0003	1E-07	1E-09
Se86	2.02E-02	0.00E+00	1.50E+01	P	1	0.0003	1E-07	1E-09
Rb86m	2.02E-02	1.88E-02	6.11E+01	P	1	0.0003	1E-07	1E-09
Rb86	2.02E-02	1.88E-02	1.61E+06	P	1	0.0003	3E-07	1E-09
As87	7.00E-03	0.00E+00	8.00E-01	P	1	0.0003	1E-07	1E-09
Se87	7.00E-03	0.00E+00	5.80E+00	P	1	0.0003	1E-07	1E-09
Se88	1.00E-03	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09
Rb88	3.58E-02	3.41E-02	1.06E+03	P	1	0.0003	3E-05	9E-08
Se89	1.40E-03	0.00E+00	4.10E-01	P	1	0.0003	1E-07	1E-09
Rb89	4.73E-02	4.87E-02	9.24E+02	P	1	0.0003	6E-05	2E-07
Sr89	4.73E-02	4.87E-02	4.36E+06	P	1	0.0003	4E-07	1E-09
Rb90m	5.03E-02	0.00E+00	2.58E+02	P	1	0.0003	1E-07	1E-09
Rb90	5.03E-02	5.06E-02	1.56E+02	P	1	0.0003	1E-07	1E-09
Sr90	5.80E-02	5.09E-02	9.07E+08	P	1	0.0003	2E-09	6E-12
Y90m	5.80E-02	5.09E-02	1.15E+04	P	1	0.0003	5E-06	2E-08
Y90	5.80E-02	5.09E-02	2.31E+05	P	1	0.0003	3E-07	9E-10

Se91	3.45E-02	3.50E-02	2.70E-01	P	1	0.0003	1E-07	1E-09
Rb91	5.43E-02	5.15E-02	5.80E+01	P	1	0.0003	1E-07	1E-09
Sr91	5.43E-02	5.35E-02	3.42E+04	P	1	0.0003	1E-06	5E-09
Y91m	5.81E-02	5.35E-02	2.98E+03	P	1	0.0003	7E-05	2E-07
Y91	5.84E-02	5.35E-02	5.05E+06	P	1	0.0003	7E-08	2E-10
Rb92	5.30E-02	5.06E-02	4.48E+00	P	1	0.0003	1E-07	1E-09
Sr92	5.30E-02	5.64E-02	9.76E+03	P	1	0.0003	3E-06	9E-09
Y92	6.03E-02	5.64E-02	1.27E+04	P	1	0.0003	3E-06	1E-08
Rb93	6.10E-02	4.57E-02	5.85E+00	P	1	0.0003	1E-07	1E-09
Sr93	6.10E-02	5.83E-02	4.45E+02	P	1	0.0003	1E-07	1E-09
Y93m	6.10E-02	0.00E+00	8.20E-01	P	1	0.0003	1E-07	1E-09
Y93	6.10E-02	5.93E-02	3.67E+04	P	1	0.0003	1E-06	3E-09
Rb94	2.50E-02	3.51E-02	2.71E+00	P	1	0.0003	1E-07	1E-09
Sr94	5.40E-02	5.65E-02	7.50E+01	P	1	0.0003	1E-07	1E-09
Y94	5.40E-02	6.04E-02	1.12E+03	P	1	0.0003	3E-05	1E-07
Rb95	1.78E-02	2.63E-02	3.77E-01	P	1	0.0003	1E-07	1E-09
Sr95	4.78E-02	5.45E-02	2.51E+01	P	1	0.0003	1E-07	1E-09
Y95	6.43E-02	6.52E-02	6.18E+02	P	1	0.0003	6E-05	2E-07
Zr95	6.43E-02	6.52E-02	5.53E+06	P	1	0.0003	1E-07	4E-10
Nb95m	6.43E-02	6.52E-02	3.12E+05	P	1	0.0003	9E-07	3E-09
Nb95	6.43E-02	6.52E-02	3.02E+06	P	1	0.0003	5E-07	2E-09
Rb96	6.33E-02	5.94E-02	1.99E-01	P	1	0.0003	1E-07	1E-09
Sr96	6.33E-02	5.94E-02	1.07E+00	P	1	0.0003	1E-07	1E-09
Y96m	6.33E-02	5.94E-02	9.60E+00	P	1	0.0003	1E-07	1E-09
Y96	6.33E-02	5.94E-02	5.30E+00	P	1	0.0003	1E-07	1E-09
Nb96	6.33E-02	6.23E-02	8.42E+04	P	1	0.0003	1E-06	3E-09
Rb97	5.90E-02	5.94E-02	1.69E-01	P	1	0.0003	1E-07	1E-09
Sr97	5.90E-02	5.94E-02	4.30E-01	P	1	0.0003	1E-07	1E-09
Y97m	5.90E-02	5.94E-02	1.21E+00	P	1	0.0003	1E-07	1E-09
Y97	5.90E-02	5.94E-02	3.76E+00	P	1	0.0003	1E-07	1E-09
Zr97	5.90E-02	6.59E-02	6.05E+04	P	1	0.0003	1E-10	1E-12
Nb97m	5.90E-02	6.59E-02	5.30E+01	P	1	0.0003	1E-07	1E-09
Nb97	5.90E-02	6.59E-02	4.43E+03	P	1	0.0003	3E-05	1E-07
Rb98	6.00E-05	0.00E+00	1.07E-01	P	1	0.0003	1E-07	1E-09
Sr98	6.00E-05	0.00E+00	6.50E-01	P	1	0.0003	1E-07	1E-09
Y98m	6.00E-05	0.00E+00	2.10E+00	P	1	0.0003	1E-07	1E-09
Y98	6.00E-05	0.00E+00	5.90E-01	P	1	0.0003	1E-07	1E-09
Zr98	5.73E-02	5.55E-02	3.07E+01	P	1	0.0003	1E-07	1E-09
Nb98m	5.79E-02	5.94E-02	3.06E+03	P	1	0.0003	1E-07	1E-09
Nb98	5.79E-02	5.94E-02	2.90E+00	P	1	0.0003	2E-05	7E-08
Rb99	4.00E-04	0.00E+00	5.50E-02	P	1	0.0003	1E-07	1E-09
Sr99	4.00E-04	0.00E+00	2.69E-01	P	1	0.0003	1E-07	1E-09
Y99m	4.00E-04	0.00E+00	9.00E-06	P	1	0.0003	1E-07	1E-09
Y99	4.00E-04	0.00E+00	1.47E+00	P	1	0.0003	1E-07	1E-09
Zr99	4.00E-04	0.00E+00	2.20E+00	P	1	0.0003	1E-07	1E-09
Nb99m	4.00E-04	0.00E+00	1.56E+02	P	1	0.0003	1E-07	1E-09
Nb99	6.10E-02	5.74E-02	1.50E+01	P	1	0.0003	1E-07	1E-09
Mo99	6.10E-02	5.74E-02	2.37E+05	P	1	0.0003	1E-06	4E-09
Tc99m	6.10E-02	5.74E-02	2.16E+04	P	1	0.0003	6E-05	2E-07
Tc99	6.10E-02	5.74E-02	6.71E+12	P	1	0.0003	3E-07	9E-10
Rb100	6.30E-02	5.35E-02	5.30E-02	P	1	0.0003	1E-07	1E-09

Sr100	6.30E-02	5.35E-02	2.01E-01	P	1	0.0003	1E-07	1E-09
Y100m	6.30E-02	5.35E-02	9.40E-01	P	1	0.0003	1E-07	1E-09
Y100	6.30E-02	5.35E-02	7.30E-01	P	1	0.0003	1E-07	1E-09
Zr100	6.30E-02	5.35E-02	7.10E+00	P	1	0.0003	1E-07	1E-09
Nb100m	6.30E-02	5.35E-02	3.00E+00	P	1	0.0003	1E-07	1E-09
Nb100	6.30E-02	5.35E-02	1.50E+00	P	1	0.0003	1E-07	1E-09
Tc100	6.30E-02	5.35E-02	1.58E+01	P	1	0.0003	1E-07	1E-09
Sr101	2.00E-03	0.00E+00	1.16E-01	P	1	0.0003	1E-07	1E-09
Y101	2.00E-03	0.00E+00	4.30E-01	P	1	0.0003	1E-07	1E-09
Zr101	2.00E-03	0.00E+00	2.40E+00	P	1	0.0003	1E-07	1E-09
Nb101	5.20E-02	4.48E-02	7.40E+00	P	1	0.0003	1E-07	1E-09
Mo101	5.20E-02	5.26E-02	8.77E+02	P	1	0.0003	6E-05	2E-07
Tc101	5.20E-02	5.32E-02	8.53E+02	P	1	0.0003	0.0001	5E-07
Rb102	2.00E-03	0.00E+00	4.00E-02	P	1	0.0003	1E-07	1E-09
Sr102	2.00E-03	0.00E+00	6.80E-02	P	1	0.0003	1E-07	1E-09
Y102m	2.00E-03	0.00E+00	3.60E-01	P	1	0.0003	1E-07	1E-09
Y102	2.00E-03	0.00E+00	3.00E-01	P	1	0.0003	1E-07	1E-09
Zr102	2.00E-03	0.00E+00	2.90E+00	P	1	0.0003	1E-07	1E-09
Nb102m	2.00E-03	0.00E+00	4.30E+00	P	1	0.0003	1E-07	1E-09
Nb102	2.00E-03	0.00E+00	1.30E+00	P	1	0.0003	1E-07	1E-09
Mo102	4.30E-02	4.38E-02	6.78E+02	P	1	0.0003	1E-07	1E-09
Tc102m	4.30E-02	4.53E-02	2.64E+02	P	1	0.0003	1E-07	1E-09
Tc102	4.30E-02	4.53E-02	5.30E+00	P	1	0.0003	1E-07	1E-09
Zr103	3.00E-04	0.00E+00	1.30E+00	P	1	0.0003	1E-07	1E-09
Nb103	3.00E-04	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09
Mo103	3.03E-02	3.11E-02	6.78E+01	P	1	0.0003	1E-07	1E-09
Tc103	3.03E-02	3.30E-02	5.40E+01	P	1	0.0003	1E-07	1E-09
Ru103	3.03E-02	3.30E-02	1.41E+05	P	1	0.0003	1E-10	1E-12
Rh103m	3.03E-02	3.30E-02	5.61E+01	P	1	0.0003	1E-07	1E-09
Nb104m	3.00E-04	0.00E+00	1.00E+00	P	1	0.0003	1E-07	1E-09
Nb104m	3.00E-04	0.00E+00	4.80E+00	P	1	0.0003	1E-07	1E-09
Mo104	1.83E-02	1.84E-02	6.00E+01	P	1	0.0003	1E-07	1E-09
Tc104	1.83E-02	2.27E-02	1.09E+03	P	1	0.0003	3E-05	1E-07
Nb105	7.00E-04	0.00E+00	2.00E+00	P	1	0.0003	1E-07	1E-09
Mo105m	7.00E-04	0.00E+00	3.00E+01	P	1	0.0003	1E-07	1E-09
Mo105	9.70E-03	1.26E-02	5.00E+01	P	1	0.0003	1E-07	1E-09
Tc105	9.70E-03	1.46E-02	4.56E+02	P	1	0.0003	1E-07	1E-09
Ru105	9.70E-03	1.46E-02	1.60E+04	P	1	0.0003	1E-10	1E-12
Rh105m	9.70E-03	1.46E-02	4.50E+01	P	1	0.0003	1E-07	1E-09
Rh105	9.70E-03	1.46E-02	1.27E+05	P	1	0.0003	1E-10	1E-12
Nb106	2.20E-04	0.00E+00	1.10E+00	P	1	0.0003	1E-07	1E-09
Mo106	2.20E-04	0.00E+00	8.40E+00	P	1	0.0003	1E-07	1E-09
Tc106	4.02E-03	8.95E-03	3.60E+01	P	1	0.0003	1E-07	1E-09
Ru106	4.02E-03	9.44E-03	3.22E+07	P	1	0.0003	1E-10	1E-12
Rh106m	4.02E-03	9.44E-03	7.85E+03	P	1	0.0003	1E-05	4E-08
Rh106	4.02E-03	9.44E-03	2.98E+01	P	1	0.0003	1E-07	1E-09
Mo107	1.90E-03	3.50E-03	3.50E+00	P	1	0.0003	1E-07	1E-09
Tc107	1.90E-03	3.50E-03	2.12E+01	P	1	0.0003	1E-07	1E-09
Ru107	1.90E-03	3.50E-03	2.28E+02	P	1	0.0003	1E-07	1E-09
Rh107	1.90E-03	3.50E-03	1.30E+03	P	1	0.0003	1E-07	1E-09
Pd107	1.90E-03	3.50E-03	2.05E+14	P	1	0.0003	2E-07	6E-10
Mo108	2.00E-05	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09

Tc108	2.00E-05	0.00E+00	5.00E+00	P	1	0.0003	1E-07	1E-09
Ru108	6.50E-04	2.34E-03	2.76E+02	P	1	0.0003	1E-07	1E-09
Rh108m	6.50E-04	2.34E-03	1.70E+01	P	1	0.0003	1E-07	1E-09
Tc109	4.00E-05	0.00E+00	1.40E+00	P	1	0.0003	1E-07	1E-09
Ru109m	4.00E-05	0.00E+00	1.30E+01	P	1	0.0003	1E-07	1E-09
Ru109	3.00E-04	1.42E-03	3.50E+01	P	1	0.0003	1E-07	1E-09
Rh109	3.00E-04	1.42E-03	8.10E+01	P	1	0.0003	1E-07	1E-09
Pd109	3.00E-04	1.42E-03	4.83E+04	P	1	0.0003	2E-06	6E-09
Tc110	1.00E-04	0.00E+00	8.30E-01	P	1	0.0003	1E-07	1E-09
Ru110	1.00E-04	0.00E+00	1.50E+01	P	1	0.0003	1E-07	1E-09
Rh110m	1.00E-04	0.00E+00	3.10E+00	P	1	0.0003	1E-07	1E-09
Rh110	3.00E-04	1.07E-03	2.90E+01	P	1	0.0003	1E-07	1E-09
Ru111	8.00E-06	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09
Rh111	8.00E-06	0.00E+00	1.10E+01	P	1	0.0003	1E-07	1E-09
Pd111m	1.00E-05	4.48E-04	1.98E+04	P	1	0.0003	1E-07	1E-09
Pd111	2.00E-04	4.48E-04	1.32E+03	P	1	0.0003	1E-07	1E-09
Ag111m	2.00E-04	4.48E-04	6.50E+01	P	1	0.0003	1E-07	1E-09
Ag111	2.00E-04	4.48E-04	6.45E+05	P	1	0.0003	1E-07	1E-09
Ru112	6.00E-05	0.00E+00	4.50E+00	P	1	0.0003	1E-07	1E-09
Rh112	6.00E-05	0.00E+00	8.00E-01	P	1	0.0003	1E-07	1E-09
Pd112	1.60E-04	3.89E-04	7.57E+04	P	1	0.0003	1E-07	1E-09
Ag112	1.60E-04	3.89E-04	1.13E+05	P	1	0.0003	3E-06	1E-08
Rh113	1.40E-05	0.00E+00	9.00E-01	P	1	0.0003	1E-07	1E-09
Pd113m	1.40E-05	0.00E+00	8.90E+01	P	1	0.0003	1E-07	1E-09
Pd113	3.14E-04	3.33E-04	9.80E+01	P	1	0.0003	1E-07	1E-09
Ag113m	3.14E-04	3.33E-04	6.80E+01	P	1	0.0003	1E-07	1E-09
Ag113	3.14E-04	3.33E-04	1.91E+04	P	1	0.0003	1E-10	1E-12
Pd114	3.14E-04	3.29E-04	1.49E+02	P	1	0.0003	1E-07	1E-09
Ag114	3.14E-04	3.33E-04	4.50E+00	P	1	0.0003	1E-07	1E-09
Ag115m	1.10E-04	3.31E-04	1.87E+01	P	1	0.0003	1E-07	1E-09
Ag115	1.10E-04	3.33E-04	1.20E+03	P	1	0.0003	3E-05	1E-07
Cd115m	1.10E-04	3.33E-04	3.85E+06	P	1	0.0003	2E-08	1E-10
Cd115	1.10E-04	3.33E-04	1.92E+05	P	1	0.0003	5E-07	2E-09
In115m	1.10E-04	3.33E-04	1.61E+04	P	1	0.0003	2E-05	6E-08
In115	1.10E-04	3.33E-04	1.39E+22	P	1	0.0003	6E-10	2E-12
Ru113	1.40E-05	0.00E+00	2.70E+00	P	1	0.0003	1E-07	1E-09
Rh114	1.40E-04	0.00E+00	1.70E+00	P	1	0.0003	1E-07	1E-09
Pd115	1.10E-04	3.02E-04	4.70E+01	P	1	0.0003	1E-07	1E-09
Pd116	1.30E-04	0.00E+00	1.27E+01	P	1	0.0003	1E-07	1E-09
Ag116m	1.30E-04	0.00E+00	1.00E+01	P	1	0.0003	1E-07	1E-09
Ag116	1.30E-04	3.49E-04	1.61E+02	P	1	0.0003	1E-07	1E-09
Rh116	2.50E-05	0.00E+00	7.00E-01	P	1	0.0003	1E-07	1E-09
Pd117	2.00E-05	0.00E+00	4.40E+00	P	1	0.0003	1E-07	1E-09
Ag117m	2.00E-05	0.00E+00	5.30E+00	P	1	0.0003	1E-07	1E-09
Ag117	1.30E-04	3.50E-04	7.32E+01	P	1	0.0003	1E-07	1E-09
Cd117m	1.30E-04	3.69E-04	1.22E+04	P	1	0.0003	5E-06	2E-08
Cd117	1.30E-04	3.69E-04	8.96E+03	P	1	0.0003	5E-06	2E-08
In117m	1.30E-04	3.69E-04	6.98E+03	P	1	0.0003	1E-05	5E-08
In117	1.30E-04	3.69E-04	2.64E+03	P	1	0.0003	7E-05	2E-07
Pd118	1.10E-04	3.90E-04	2.10E+00	P	1	0.0003	1E-07	1E-09
Ag118m	1.10E-04	3.90E-04	2.40E+00	P	1	0.0003	1E-07	1E-09
Ag118	1.10E-04	3.90E-04	4.00E+00	P	1	0.0003	1E-07	1E-09

Cd118	1.10E-04	3.90E-04	3.02E+03	P	1	0.0003	1E-07	1E-09
In118	1.10E-04	3.90E-04	5.00E+00	P	1	0.0003	1E-07	1E-09
Ag119	7.00E-05	0.00E+00	2.10E+00	P	1	0.0003	1E-07	1E-09
Cd119m	1.30E-04	1.85E-04	1.32E+02	P	1	0.0003	1E-07	1E-09
Cd119	1.30E-04	1.85E-04	1.61E+02	P	1	0.0003	1E-07	1E-09
In119m	1.30E-04	1.85E-04	1.07E+03	P	1	0.0003	5E-05	2E-07
In119	1.30E-04	1.85E-04	1.38E+02	P	1	0.0003	1E-07	1E-09
Sn119m	1.30E-04	1.85E-04	2.53E+07	P	1	0.0003	1E-06	3E-09
Ag120m	1.30E-04	3.60E-04	3.20E-01	P	1	0.0003	1E-07	1E-09
Ag120	1.30E-04	3.60E-04	1.23E+00	P	1	0.0003	1E-07	1E-09
Cd120	1.30E-04	3.60E-04	5.08E+01	P	1	0.0003	1E-07	1E-09
In120m	1.30E-04	3.60E-04	4.70E+01	P	1	0.0003	1E-07	1E-09
In120	1.30E-04	3.79E-04	3.10E+00	P	1	0.0003	1E-07	1E-09
Ag121	1.50E-04	3.41E-04	7.80E-01	P	1	0.0003	1E-07	1E-09
Cd121m	1.50E-04	3.41E-04	8.00E+00	P	1	0.0003	1E-07	1E-09
Cd121	1.50E-04	3.41E-04	1.35E+01	P	1	0.0003	1E-07	1E-09
In121m	1.50E-04	3.41E-04	2.28E+02	P	1	0.0003	1E-07	1E-09
In121	1.50E-04	3.90E-04	2.30E+01	P	1	0.0003	1E-07	1E-09
Sn121m	1.50E-04	3.90E-04	1.73E+09	P	1	0.0003	2E-07	8E-10
Sn121	1.50E-04	3.90E-04	9.75E+04	P	1	0.0003	5E-06	2E-08
Ag122m	1.60E-04	0.00E+00	1.00E+00	P	1	0.0003	1E-07	1E-09
Ag122	1.60E-04	0.00E+00	5.60E-01	P	1	0.0003	1E-07	1E-09
Cd122	1.60E-04	0.00E+00	5.30E+00	P	1	0.0003	1E-07	1E-09
In122m	1.60E-04	0.00E+00	1.08E+01	P	1	0.0003	1E-07	1E-09
In122	1.60E-04	0.00E+00	1.50E+00	P	1	0.0003	1E-07	1E-09
Ag123	1.60E-04	0.00E+00	3.20E-01	P	1	0.0003	1E-07	1E-09
Cd123m	1.60E-04	0.00E+00	1.84E+00	P	1	0.0003	1E-07	1E-09
Cd123	1.60E-04	0.00E+00	2.10E+00	P	1	0.0003	1E-07	1E-09
In123m	1.60E-04	4.19E-04	4.70E+01	P	1	0.0003	1E-07	1E-09
In123	1.60E-04	4.19E-04	6.00E+00	P	1	0.0003	1E-07	1E-09
Sn123m	1.60E-04	4.97E-04	2.41E+03	P	1	0.0003	5E-05	2E-07
Sn123	1.60E-04	4.97E-04	1.12E+07	P	1	0.0003	7E-08	2E-10
Ag124	5.00E-05	0.00E+00	2.20E-01	P	1	0.0003	1E-07	1E-09
Cd124	5.00E-05	0.00E+00	1.24E+00	P	1	0.0003	1E-07	1E-09
In124m	5.00E-05	0.00E+00	3.40E+00	P	1	0.0003	1E-07	1E-09
In124	2.70E-04	0.00E+00	3.18E+00	P	1	0.0003	1E-07	1E-09
Sb124m	2.70E-04	0.00E+00	7.27E+04	P	1	0.0003	0.0002	8E-07
Sb124	2.70E-04	1.27E-03	5.20E+06	P	1	0.0003	1E-07	3E-10
Cd125m	1.20E-04	0.00E+00	6.00E-01	P	1	0.0003	1E-07	1E-09
Cd125	1.20E-04	0.00E+00	6.80E-01	P	1	0.0003	1E-07	1E-09
In125m	1.20E-04	0.00E+00	1.22E+01	P	1	0.0003	1E-07	1E-09
In125	1.20E-04	0.00E+00	2.36E+00	P	1	0.0003	1E-07	1E-09
Sn125m	2.00E-04	0.00E+00	5.70E+02	P	1	0.0003	1E-07	1E-09
Sn125	3.40E-04	0.00E+00	8.33E+05	P	1	0.0003	1E-07	5E-10
Sb125	3.40E-04	7.59E-03	8.69E+07	P	1	0.0003	2E-07	7E-10
Te125m	3.40E-04	7.59E-03	5.01E+06	P	1	0.0003	2E-07	1E-09
Cd126	1.50E-04	0.00E+00	5.20E-01	P	1	0.0003	1E-07	1E-09
In126m	1.50E-04	0.00E+00	1.63E+00	P	1	0.0003	1E-07	1E-09
In126	1.50E-04	0.00E+00	1.53E+00	P	1	0.0003	1E-07	1E-09
Sn126	5.90E-04	2.92E-03	7.88E+12	P	1	0.0003	2E-08	8E-11
Sb126m	5.90E-04	3.31E-03	1.10E+01	P	1	0.0003	8E-05	3E-07
Sb126	5.90E-04	3.31E-03	1.07E+06	P	1	0.0003	2E-07	7E-10

Cd127	1.44E-03	0.00E+00	4.00E-01	P	1	0.0003	1E-07	1E-09
In127m	1.44E-03	0.00E+00	3.73E+00	P	1	0.0003	1E-07	1E-09
In127	1.44E-03	0.00E+00	1.14E+00	P	1	0.0003	1E-07	1E-09
Sn127m	1.44E-03	1.66E-03	2.49E+02	P	1	0.0003	1E-07	1E-09
Sn127	1.57E-03	1.66E-03	7.63E+03	P	1	0.0003	8E-06	3E-08
Sb127	1.57E-03	1.66E-03	3.32E+05	P	1	0.0003	4E-07	1E-09
Te127m	1.57E-03	1.66E-03	6.80E+06	P	1	0.0003	1E-07	4E-10
Te127	1.57E-03	1.76E-03	3.38E+04	P	1	0.0003	1E-07	3E-10
Cd128	5.90E-04	0.00E+00	2.80E-01	P	1	0.0003	1E-07	1E-09
In128m	5.90E-04	0.00E+00	7.00E-01	P	1	0.0003	1E-07	1E-09
In128	5.90E-04	0.00E+00	8.00E-01	P	1	0.0003	1E-07	1E-09
Sn128	3.70E-03	5.06E-03	3.55E+03	P	1	0.0003	1E-05	4E-08
Sb128m	3.70E-03	5.06E-03	6.06E+02	P	1	0.0003	0.0002	5E-07
Sn128	4.10E-03	6.71E-03	3.28E+04	P	1	0.0003	1E-05	4E-08
Cd129	1.00E-03	0.00E+00	2.70E-01	P	1	0.0003	1E-07	1E-09
In129m	1.00E-03	0.00E+00	7.00E-01	P	1	0.0003	1E-07	1E-09
In129	1.00E-03	0.00E+00	6.30E-01	P	1	0.0003	1E-07	1E-09
Sn129m	1.00E-03	0.00E+00	6.99E+00	P	1	0.0003	1E-07	1E-09
Sn129	1.00E-03	0.00E+00	1.34E+02	P	1	0.0003	1E-07	1E-09
Sb129m	1.00E-02	3.55E-03	1.06E+03	P	1	0.0003	1E-07	1E-09
Sb129	1.00E-02	7.00E-03	1.58E+04	P	1	0.0003	4E-06	1E-08
Te129m	1.00E-02	1.01E-02	2.90E+06	P	1	0.0003	1E-07	3E-10
Te129	1.00E-02	1.10E-02	4.18E+03	P	1	0.0003	3E-05	9E-08
Cd130	1.00E-03	0.00E+00	2.00E-01	P	1	0.0003	1E-07	1E-09
In130m1	1.00E-03	0.00E+00	5.30E-01	P	1	0.0003	1E-07	1E-09
In130m2	1.00E-03	0.00E+00	5.10E-01	P	1	0.0003	1E-07	1E-09
In130	1.00E-03	0.00E+00	2.90E-01	P	1	0.0003	1E-07	1E-09
Sn130m	1.00E-03	0.00E+00	1.02E+02	P	1	0.0003	1E-07	1E-09
Sn130	2.00E-02	8.47E-03	2.23E+02	P	1	0.0003	1E-07	1E-09
Sb130m	2.00E-02	1.75E-02	3.78E+02	P	1	0.0003	1E-07	1E-09
Sb130	2.00E-02	1.75E-02	2.37E+03	P	1	0.0003	3E-05	9E-08
In131m1	1.00E-03	0.00E+00	3.00E-01	P	1	0.0003	1E-07	1E-09
In131m2	1.00E-03	0.00E+00	3.50E-01	P	1	0.0003	1E-07	1E-09
In131	1.00E-03	0.00E+00	2.80E-01	P	1	0.0003	1E-07	1E-09
Sn131m	1.00E-03	0.00E+00	5.84E+01	P	1	0.0003	1E-07	1E-09
Sn131	2.93E-02	1.21E-02	5.60E+01	P	1	0.0003	1E-07	1E-09
Sb131	2.93E-02	2.57E-02	1.38E+03	P	1	0.0003	1E-05	6E-08
Te131m	2.93E-02	2.57E-02	1.17E+05	P	1	0.0003	2E-07	1E-09
Te131	2.93E-02	3.06E-02	1.50E+03	P	1	0.0003	2E-06	2E-08
In132	7.00E-04	0.00E+00	2.00E-01	P	1	0.0003	1E-07	1E-09
Sn132	4.31E-02	9.73E-03	3.97E+01	P	1	0.0003	1E-07	1E-09
Sb132m	4.31E-02	9.73E-03	1.68E+02	P	1	0.0003	1E-07	1E-09
Sb132	4.31E-02	3.21E-02	2.52E+02	P	1	0.0003	1E-07	1E-09
Te132	4.31E-02	4.77E-02	2.76E+05	P	1	0.0003	9E-08	9E-10
Sn133	9.00E-04	0.00E+00	1.44E+00	P	1	0.0003	1E-07	1E-09
Sb133	3.96E-02	3.31E-02	1.50E+02	P	1	0.0003	1E-07	1E-09
Te133m	6.20E-02	3.31E-02	3.32E+03	P	1	0.0003	2E-06	2E-08
Te133	6.20E-02	6.51E-02	7.44E+02	P	1	0.0003	9E-06	8E-08
Sn134	7.00E-04	0.00E+00	1.04E+00	P	1	0.0003	1E-07	1E-09
Sb134m	7.00E-04	0.00E+00	1.04E+01	P	1	0.0003	1E-07	1E-09
Sb134	6.97E-02	1.56E-02	8.00E-01	P	1	0.0003	1E-07	1E-09
Te134	6.97E-02	4.77E-02	2.52E+03	P	1	0.0003	1E-05	7E-08

Sb135	1.30E-03	0.00E+00	1.71E+00	P	1	0.0003	1E-07	1E-09
Te135	6.22E-02	3.31E-02	1.90E+01	P	1	0.0003	1E-07	1E-09
Cs135m	6.54E-02	6.15E-02	3.18E+03	P	1	0.0003	8E-05	3E-07
Cs135	6.54E-02	6.19E-02	7.25E+13	P	1	0.0003	5E-07	2E-09
Sb136	1.00E-03	0.00E+00	8.20E-01	P	1	0.0003	1E-07	1E-09
Te136	1.00E-03	0.00E+00	1.75E+01	P	1	0.0003	1E-07	1E-09
Cs136	6.46E-02	6.13E-02	1.14E+06	P	1	0.0003	3E-07	9E-10
Te137	4.00E-04	0.00E+00	2.50E+00	P	1	0.0003	1E-07	1E-09
Cs137	6.19E-02	6.09E-02	9.47E+08	P	1	0.0003	6E-08	2E-10
Ba137m	6.19E-02	6.09E-02	1.53E+02	P	1	0.0003	1E-07	1E-09
Te138	1.03E-02	0.00E+00	1.40E+00	P	1	0.0003	1E-07	1E-09
Cs138m	6.77E-02	4.67E-02	1.74E+02	P	1	0.0003	1E-07	1E-09
Cs138	6.77E-02	6.28E-02	1.93E+03	P	1	0.0003	2E-05	8E-08
Cs139	6.20E-02	5.45E-02	5.58E+02	P	1	0.0003	1E-07	1E-09
Ba139	6.27E-02	5.94E-02	5.03E+03	P	1	0.0003	1E-05	4E-08
Cs140	6.00E-02	4.57E-02	6.36E+01	P	1	0.0003	1E-07	1E-09
Ba140	6.35E-02	5.54E-02	1.10E+06	P	1	0.0003	6E-07	2E-09
La140	6.35E-02	5.54E-02	1.45E+05	P	1	0.0003	5E-07	2E-09
Cs141	4.60E-02	3.94E-02	2.49E+01	P	1	0.0003	1E-07	1E-09
Ba141	6.30E-02	5.79E-02	1.10E+03	P	1	0.0003	3E-05	1E-07
La141	6.40E-02	6.08E-02	1.40E+04	P	1	0.0003	4E-06	1E-08
Ce141	6.40E-02	6.08E-02	2.81E+06	P	1	0.0003	2E-07	8E-10
Cs142	3.46E-02	4.56E-02	1.80E+00	P	1	0.0003	1E-07	1E-09
Ba142	5.70E-02	6.90E-02	6.42E+02	P	1	0.0003	6E-05	2E-07
La142	6.01E-02	7.50E-02	5.54E+03	P	1	0.0003	9E-06	3E-08
Cs143	2.08E-02	1.56E-02	1.78E+00	P	1	0.0003	1E-07	1E-09
Ba143	5.00E-02	4.19E-02	1.43E+01	P	1	0.0003	1E-07	1E-09
La143	5.96E-02	5.55E-02	1.43E+01	P	1	0.0003	4E-05	1E-07
Ce143	5.96E-02	5.64E-02	1.19E+05	P	1	0.0003	7E-07	2E-09
Pr143	5.96E-02	5.64E-02	1.17E+06	P	1	0.0003	3E-07	9E-10
Cs144	1.20E-03	0.00E+00	1.01E+00	P	1	0.0003	1E-07	1E-09
Ba144	1.20E-03	0.00E+00	1.14E+01	P	1	0.0003	1E-07	1E-09
La144	5.62E-02	4.67E-02	4.07E+01	P	1	0.0003	1E-07	1E-09
Ce144	5.62E-02	5.01E-02	2.46E+07	P	1	0.0003	6E-09	2E-11
Pr144m	5.62E-02	5.01E-02	4.32E+02	P	1	0.0003	1E-07	1E-09
Pr144	5.62E-02	5.01E-02	1.04E+03	P	1	0.0003	5E-05	2E-07
Cs145	3.98E-02	3.74E-02	5.90E-01	P	1	0.0003	1E-07	1E-09
Ba145	3.98E-02	3.75E-02	4.00E+00	P	1	0.0003	1E-07	1E-09
La145	3.98E-02	3.75E-02	2.40E+01	P	1	0.0003	1E-07	1E-09
Ce145	3.98E-02	3.75E-02	1.80E+02	P	1	0.0003	1E-07	1E-09
Pr145	3.98E-02	3.75E-02	2.15E+04	P	1	0.0003	3E-06	1E-08
Cs146	1.00E-03	0.00E+00	3.22E-01	P	1	0.0003	1E-07	1E-09
Ba146	1.00E-03	0.00E+00	2.20E+00	P	1	0.0003	1E-07	1E-09
La146m	1.00E-03	0.00E+00	1.00E+01	P	1	0.0003	1E-07	1E-09
La146	1.00E-03	0.00E+00	6.30E+00	P	1	0.0003	1E-07	1E-09
Ce146	3.07E-02	2.82E-02	8.10E+02	P	1	0.0003	1E-07	1E-09
Pr146	3.07E-02	2.92E-02	1.45E+03	P	1	0.0003	1E-07	1E-09
Cs147	1.00E-03	0.00E+00	2.27E-01	P	1	0.0003	1E-07	1E-09
Ba147	1.00E-03	0.00E+00	8.92E-01	P	1	0.0003	1E-07	1E-09
La147	1.00E-03	0.00E+00	4.02E+00	P	1	0.0003	1E-07	1E-09
Ce147	2.36E-02	2.04E-02	5.60E+01	P	1	0.0003	1E-07	1E-09
Pr147	2.36E-02	2.24E-02	8.04E+02	P	1	0.0003	8E-05	3E-07

Nd147	2.36E-02	2.24E-02	9.49E+05	P	1	0.0003	4E-07	1E-09
Pm147	2.36E-02	2.24E-02	8.26E+07	P	1	0.0003	5E-08	2E-10
Sm147	2.36E-02	2.24E-02	3.34E+18	P	1	0.0003	2E-11	1E-13
Cs148	1.00E-03	0.00E+00	1.50E-01	P	1	0.0003	1E-07	1E-09
Ba148	1.00E-03	0.00E+00	6.40E-01	P	1	0.0003	1E-07	1E-09
La148	1.00E-03	0.00E+00	1.10E+00	P	1	0.0003	1E-07	1E-09
Ce148	1.71E-02	1.36E-02	5.60E+01	P	1	0.0003	1E-07	1E-09
Pr148m	1.71E-02	1.36E-02	1.20E+02	P	1	0.0003	1E-07	4E-10
Pr148	1.71E-02	1.68E-02	1.36E+02	P	1	0.0003	2E-07	7E-10
Ba149	1.00E-03	0.00E+00	3.40E-01	P	1	0.0003	1E-07	1E-09
La149	1.00E-03	0.00E+00	1.05E+00	P	1	0.0003	1E-07	1E-09
Ce149	1.00E-03	0.00E+00	5.20E+00	P	1	0.0003	1E-07	1E-09
Pr149	1.13E-02	0.00E+00	1.38E+02	P	1	0.0003	1E-07	1E-09
Nd149	1.13E-02	1.12E-02	6.23E+03	P	1	0.0003	1E-05	3E-08
Pm149	1.13E-02	1.13E-02	1.91E+05	P	1	0.0003	8E-07	2E-09
Ce150	1.00E-04	0.00E+00	4.40E+00	P	1	0.0003	1E-07	1E-09
Pr150	1.00E-04	0.00E+00	6.20E+00	P	1	0.0003	1E-07	1E-09
Pm150	6.70E-03	8.10E-03	9.65E+03	P	1	0.0003	7E-06	2E-08
Ce151	1.00E-04	0.00E+00	1.00E+00	P	1	0.0003	1E-07	1E-09
Pr151	1.00E-04	0.00E+00	1.89E+01	P	1	0.0003	1E-07	1E-09
Nd151	4.40E-03	3.80E-03	1.25E+01	P	1	0.0003	1E-07	1E-09
Pm151	4.40E-03	5.27E-03	1.02E+05	P	1	0.0003	1E-06	4E-09
Sm151	4.40E-03	5.27E-03	2.84E+09	P	1	0.0003	4E-08	2E-10
Ce152	1.00E-04	0.00E+00	1.40E+00	P	1	0.0003	1E-07	1E-09
Pr152	1.00E-04	0.00E+00	3.20E+00	P	1	0.0003	1E-07	1E-09
Nd152	1.00E-04	0.00E+00	6.84E+02	P	1	0.0003	1E-07	1E-09
Pm152m	1.00E-04	0.00E+00	8.28E+02	P	1	0.0003	1E-07	1E-09
Pm152	2.81E-03	2.98E-03	2.46E+02	P	1	0.0003	1E-07	1E-09
Pr153	1.00E-04	0.00E+00	4.30E+00	P	1	0.0003	1E-07	1E-09
Nd153	1.00E-04	0.00E+00	2.89E+01	P	1	0.0003	1E-07	1E-09
Pm153	1.50E-03	0.00E+00	3.24E+02	P	1	0.0003	1E-07	1E-09
Sm153	1.50E-03	2.04E-03	1.67E+05	P	1	0.0003	1E-06	4E-09
Pr154	1.00E-05	0.00E+00	2.30E+00	P	1	0.0003	1E-07	1E-09
Nd154	1.00E-05	0.00E+00	2.59E+01	P	1	0.0003	1E-07	1E-09
Pm154m	1.00E-05	0.00E+00	1.62E+02	P	1	0.0003	1E-07	1E-09
Pm154	7.70E-04	8.66E-04	1.02E+02	P	1	0.0003	1E-07	1E-09
Eu154m	7.70E-04	9.54E-04	2.80E+03	P	1	0.0003	1E-07	1E-09
Eu154	7.70E-04	9.54E-04	2.71E+08	P	1	0.0003	8E-09	3E-11
Nd155	1.00E-05	0.00E+00	8.90E+00	P	1	0.0003	1E-07	1E-09
Pm155	1.00E-05	0.00E+00	4.20E+01	P	1	0.0003	1E-07	1E-09
Sm155	3.30E-04	6.91E-04	2.22E+01	P	1	0.0003	9E-05	3E-07
Eu155	3.30E-04	6.91E-04	1.50E+08	P	1	0.0003	4E-08	2E-10
Nd156	1.00E-05	0.00E+00	5.50E+00	P	1	0.0003	1E-07	1E-09
Pm156	1.00E-05	0.00E+00	2.67E+01	P	1	0.0003	1E-07	1E-09
Sm156	1.49E-04	2.34E-04	3.38E+04	P	1	0.0003	4E-06	1E-08
Eu156	1.49E-04	2.34E-04	1.31E+06	P	1	0.0003	2E-07	6E-10
Pm157	1.00E-06	0.00E+00	1.09E+01	P	1	0.0003	1E-07	1E-09
Sm157	7.80E-05	1.46E-04	4.80E+02	P	1	0.0003	1E-07	1E-09
Eu157	7.80E-05	1.65E-04	5.46E+04	P	1	0.0003	2E-06	7E-09
Pm158	1.30E-05	0.00E+00	5.00E+00	P	1	0.0003	1E-07	1E-09
Sm158	1.30E-05	0.00E+00	3.18E+02	P	1	0.0003	1E-07	1E-09
Eu158	3.30E-05	8.00E-05	2.75E+03	P	1	0.0003	2E-05	8E-08

Sm159	1.00E-06	0.00E+00	1.14E+01	P	1	0.0003	1E-07	1E-09
Eu159	1.10E-05	3.30E-05	1.09E+03	P	1	0.0003	1E-07	1E-09
Gd159	1.10E-05	3.30E-05	6.66E+04	P	1	0.0003	2E-06	8E-09
Sm160	1.00E-07	0.00E+00	9.60E+00	P	1	0.0003	1E-07	1E-09
Eu160	3.00E-06	0.00E+00	3.80E+01	P	1	0.0003	1E-07	1E-09
Eu161	1.00E-07	0.00E+00	2.60E+01	P	1	0.0003	1E-07	1E-09
Gd161	1.00E-06	4.00E-06	2.20E+02	P	1	0.0003	1E-07	1E-09
Tb161	1.00E-06	4.00E-06	5.97E+05	P	1	0.0003	7E-07	2E-09
Eu162	1.00E-08	0.00E+00	1.10E+01	P	1	0.0003	1E-07	1E-09
Gd162	2.00E-07	0.00E+00	8.49E+00	P	1	0.0003	1E-07	1E-09
Tb162	2.00E-07	0.00E+00	4.56E+02	P	1	0.0003	1E-07	1E-09

Initial Ci	Decay Ci	Bay uCi/ml	Public uCi/ml	Nuclide	Bay Dose	Bay Thy dose	Public dose	Pub Thy dose
				Time =	*0.25*	*0.25*	*24*	*24*
					rem	**rem**	**rem**	**rem**
				TOTAL =	6.00E+01	1.03E+01	5.74E-01	8.46E-04
4.70E-07	4.70E-07	2.09E-10	1.58E-12	H3	2.83E-09		2.13E-10	
1.36E-02	1.36E-02	6.06E-06	4.60E-08	Kr83m	3.14E-07		1.32E-08	
1.30E-02	1.30E-02	5.76E-06	4.37E-08	Kr85m	1.53E-04		8.62E-06	
6.56E-07	6.56E-07	2.91E-10	2.21E-12	Kr85	1.13E-09		8.49E-11	
9.39E-02	9.39E-02	4.17E-05	3.17E-07	Kr87	5.29E-03		1.86E-04	
5.89E-02	5.89E-02	2.62E-05	1.99E-07	Kr88	7.64E-03		3.77E-04	
3.78E+00	3.78E+00	1.68E-03	1.28E-05	Kr89	1.36E+00		1.07E-02	
1.48E+01	1.48E+01	6.56E-03	4.98E-05	Kr90	9.64E-01		7.32E-03	
1.39E+01	1.39E+01	6.16E-03	4.68E-05	Kr91	2.42E-01		1.84E-03	
7.58E+00	7.58E+00	3.37E-03	2.56E-05	Kr92	2.83E-02		2.15E-04	
1.94E+00	1.94E+00	8.64E-04	6.56E-06	Kr93	5.10E-03		3.87E-05	
4.05E-01	4.05E-01	1.80E-04	1.37E-06	Kr94	1.73E-04		1.31E-06	
9.32E-01	9.32E-01	4.14E-04	3.14E-06	Kr95	1.48E-03		1.12E-05	
2.39E+01	2.39E+01	1.06E-02	8.06E-05	Kr97	4.86E-03		3.69E-05	
4.80E-04	4.80E-04	2.13E-07	1.62E-09	Xe131m	2.89E-07		2.16E-08	
5.96E-03	5.96E-03	2.65E-06	2.01E-08	Xe133m	7.17E+00		5.24E-01	
2.49E-03	2.49E-03	1.11E-06	8.40E-09	Xe133	5.99E-06		4.46E-07	
1.12E+00	1.12E+00	4.96E-04	3.76E-06	Xe135m	2.45E-02		3.42E-04	
3.36E-02	3.36E-02	1.49E-05	1.13E-07	Xe135	5.72E-04		3.69E-05	
4.06E+00	4.06E+00	1.80E-03	1.37E-05	Xe137	1.71E+00		1.38E-02	
1.32E+00	1.32E+00	5.85E-04	4.44E-06	Xe138	5.64E-02		7.51E-04	
1.63E+01	1.63E+01	7.25E-03	5.50E-05	Xe139	1.31E+00		9.92E-03	
1.47E+01	1.47E+01	6.52E-03	4.95E-05	Xe140	4.05E-01		3.07E-03	
5.39E+00	5.39E+00	2.39E-03	1.82E-05	Xe141	1.34E-02		1.01E-04	
1.42E+00	1.42E+00	6.30E-04	4.78E-06	Xe142	3.52E-03		2.67E-05	
9.28E-01	9.28E-01	4.12E-04	3.13E-06	Xe143m	1.81E-03		1.37E-05	
1.13E+00	1.13E+00	5.04E-04	3.83E-06	Xe143	6.92E-04		5.25E-06	
4.86E-01	4.86E-01	2.16E-04	1.64E-06	Xe144	1.19E-03		9.00E-06	
2.03E-01	2.03E-01	9.00E-05	6.83E-07	Xe145	3.71E-04		2.81E-06	
2.27E-01	2.27E-01	1.01E-04	7.65E-09	Br79m	1.12E-03		8.50E-08	
1.39E-01	1.39E-01	6.19E-05	4.70E-09	Br82m	4.08E-02		3.70E-06	
4.24E-04	4.24E-04	1.88E-07	1.43E-11	Br82	2.55E-01		1.84E-04	
9.92E-03	9.92E-03	4.41E-06	3.35E-10	Br83	6.40E-05		2.97E-08	
4.42E-01	4.42E-01	1.96E-04	1.49E-08	Br84m	1.28E-01		1.16E-05	
8.73E-02	8.73E-02	3.88E-05	2.95E-09	Br84	5.62E-04		1.21E-07	
9.56E-01	9.56E-01	4.25E-04	3.22E-08	Br85	1.59E-01		1.23E-05	
4.32E+00	4.32E+00	1.92E-03	1.46E-07	Br86	2.41E-01		1.83E-05	
5.44E+00	5.44E+00	2.42E-03	1.84E-07	Br87	3.06E-01		2.33E-05	
1.34E+01	1.34E+01	5.94E-03	4.50E-07	Br88	2.22E-01		1.69E-05	
1.92E+01	1.92E+01	8.52E-03	6.46E-07	Br89	8.57E-02		6.50E-06	
1.22E-01	1.22E-01	5.40E-05	4.10E-09	Br90	2.35E-04		1.78E-08	
1.40E+01	1.40E+01	6.21E-03	4.71E-07	Br91	7.67E-03		5.82E-07	
7.58E+00	7.58E+00	3.37E-03	2.56E-07	Br92	2.62E-03		1.99E-07	
1.94E+00	1.94E+00	8.64E-04	6.56E-08	Br93	1.98E-04		1.50E-08	
3.41E-13	3.41E-13	1.51E-16	1.15E-20	I129	5.12E-12	2.24E-11	3.86E-15	1.69E-14
7.12E-04	7.12E-04	3.17E-07	2.40E-11	I131	2.14E-03	9.38E-03	1.60E-06	7.01E-06

1.45E-01	1.45E-01	6.42E-05	4.87E-09	I132m	2.73E-03	8.96E-03	1.00E-06	3.30E-06
8.97E-02	8.97E-02	3.99E-05	3.02E-09	I132	2.60E-03	7.59E-03	1.18E-06	3.46E-06
2.49E+01	2.49E+01	1.11E-02	8.39E-07	I133m	2.27E-01	9.95E-01	1.72E-05	7.55E-05
1.51E-02	1.51E-02	6.70E-06	5.08E-10	I133	9.03E-03	3.96E-02	6.33E-06	2.77E-05
4.82E+00	4.82E+00	2.14E-03	1.63E-07	I134m	9.92E-01	4.35E+00	7.96E-05	3.49E-04
1.74E+01	1.74E+01	7.73E-03	5.87E-07	I134	1.54E-02	2.02E-02	1.17E-06	1.54E-06
4.43E-02	4.43E-02	1.97E-05	1.49E-09	I135	4.38E-03	1.65E-02	2.68E-06	1.01E-05
2.38E-01	2.38E-01	1.06E-04	8.02E-09	I136m	1.13E-02	4.94E-02	8.56E-07	3.75E-06
4.93E+00	4.93E+00	2.19E-03	1.66E-07	I136	4.12E-01	1.81E+00	3.13E-05	1.37E-04
2.00E+01	2.00E+01	8.88E-03	6.74E-07	I137	4.96E-01	2.17E+00	3.76E-05	1.65E-04
2.74E+01	2.74E+01	1.22E-02	9.24E-07	I138	1.81E-01	7.92E-01	1.37E-05	6.01E-05
3.20E+00	3.20E+00	1.42E-03	1.08E-07	I139	7.48E-03	3.28E-02	5.68E-07	2.49E-06
4.05E-01	4.05E-01	1.80E-04	1.37E-08	I140	3.54E-04	1.55E-03	2.69E-08	1.18E-07
4.05E-01	4.05E-01	1.80E-04	1.37E-08	I141	8.24E-05	3.61E-04	6.25E-09	2.74E-08
1.42E+00	1.42E+00	6.30E-04	4.78E-08	I142	2.88E-04	1.26E-03	2.19E-08	9.59E-08
1.05E-04	1.05E-04	4.68E-08	1.07E-13	Ni72	2.25E-07		5.12E-13	
1.05E-04	1.05E-04	4.67E-08	1.06E-13	Cu72	7.05E-07		1.60E-12	
2.62E-08	2.62E-08	1.16E-11	2.65E-17	Zn72	7.86E-09		1.72E-13	
8.63E-08	8.63E-08	3.83E-11	8.73E-17	Ga72	1.29E-08		2.63E-13	
4.05E-04	4.05E-04	1.80E-07	4.10E-13	Ni73	3.71E-07		8.44E-13	
4.05E-04	4.05E-04	1.80E-07	4.10E-13	Cu73	1.61E-06		3.66E-12	
1.89E-04	1.89E-04	8.42E-08	1.92E-13	Zn73m	1.26E-05		2.86E-11	
3.34E-04	3.34E-04	1.48E-07	3.37E-13	Zn73	8.11E-06		1.85E-11	
9.60E-07	9.60E-07	4.27E-10	9.71E-16	Ga73	2.84E-08		4.89E-13	
1.22E-03	1.22E-03	5.40E-07	1.23E-12	Ni74	1.36E-06		3.09E-12	
1.22E-03	1.22E-03	5.40E-07	1.23E-12	Cu74	1.98E-06		4.50E-12	
4.27E-04	4.27E-04	1.90E-07	4.32E-13	Zn74	4.10E-05		9.34E-11	
1.20E-03	1.20E-03	5.32E-07	1.21E-12	Ga74m	1.21E-05		2.77E-11	
9.97E-05	9.97E-05	4.43E-08	1.01E-13	Ga74	3.42E-05		1.04E-10	
1.22E-03	1.22E-03	5.40E-07	1.23E-12	Cu75	1.61E-06		3.66E-12	
1.19E-03	1.19E-03	5.31E-07	1.21E-12	Zn75	1.24E-05		2.82E-11	
1.25E-03	1.25E-03	5.57E-07	1.27E-12	Ga75	1.56E-04		3.58E-10	
2.58E-03	2.58E-03	1.15E-06	2.61E-12	Ge75m	1.25E-04		2.85E-10	
3.71E-05	3.71E-05	1.65E-08	3.76E-14	Ge75	2.10E-07		2.31E-12	
2.43E-03	2.43E-03	1.08E-06	2.46E-12	Cu76	1.58E-06		3.60E-12	
2.43E-03	2.43E-03	1.08E-06	2.46E-12	Zn76	1.41E-05		3.20E-11	
9.57E-03	9.57E-03	4.25E-06	9.68E-12	Ga76	2.81E-04		6.39E-10	
2.71E-02	2.71E-02	1.21E-05	2.75E-11	Cu77	1.30E-05		2.95E-11	
2.71E-02	2.71E-02	1.21E-05	2.75E-11	Zn77m	2.76E-05		6.28E-11	
2.71E-02	2.71E-02	1.21E-05	2.75E-11	Zn77	5.79E-05		1.32E-10	
2.60E-02	2.60E-02	1.16E-05	2.63E-11	Ga77	3.43E-04		7.82E-10	
1.48E-02	1.48E-02	6.56E-06	1.49E-11	Ge77m	7.88E-04		1.79E-09	
3.44E-05	3.44E-05	1.53E-08	3.48E-14	Ge77	2.05E-06		4.08E-11	
1.00E-05	1.00E-05	4.45E-09	1.01E-14	As77	8.58E-07		1.87E-11	
3.05E-02	3.05E-02	1.36E-05	3.09E-11	Se77m	5.39E-04		1.23E-09	
8.51E-02	8.51E-02	3.78E-05	8.61E-11	Cu78	2.94E-05		6.70E-11	
8.51E-02	8.51E-02	3.78E-05	8.61E-11	Zn78	1.30E-04		2.95E-10	
8.51E-02	8.51E-02	3.78E-05	8.61E-11	Ga78	4.40E-04		1.00E-09	
6.75E-04	6.75E-04	3.00E-07	6.83E-13	Ge78	3.83E-01		4.32E-06	
6.47E-04	6.47E-04	2.88E-07	6.55E-13	As78	1.23E-05		1.41E-10	
2.27E-01	2.27E-01	1.01E-04	2.30E-10	Cu79	4.38E-05		9.98E-11	
2.27E-01	2.27E-01	1.01E-04	2.30E-10	Zn79	2.31E-04		5.25E-10	

2.27E-01	2.27E-01	1.01E-04	2.30E-10	Ga79	6.57E-04	1.50E-09
1.49E-01	1.49E-01	6.61E-05	1.51E-10	Ge79m	5.86E-03	1.33E-08
2.01E-01	2.01E-01	8.95E-05	2.04E-10	Ge79	3.88E-03	8.83E-09
1.68E-02	1.68E-02	7.47E-06	1.70E-11	As79	6.06E-03	1.93E-08
3.68E-02	3.68E-02	1.63E-05	3.72E-11	Se79m	7.92E-03	1.92E-08
4.99E-13	4.99E-13	2.22E-16	5.05E-22	Se79	3.75E-13	8.48E-18
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Zn80	6.67E-05	1.52E-10
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Ga80	2.10E-04	4.78E-10
9.19E-02	9.19E-02	4.08E-05	9.30E-11	Ge80	2.74E-03	6.24E-09
4.88E-01	4.88E-01	2.17E-04	4.93E-10	As80	7.91E-03	1.80E-08
2.41E-01	2.41E-01	1.07E-04	2.44E-10	Zn81	7.12E-05	1.62E-10
2.41E-01	2.41E-01	1.07E-04	2.44E-10	Ga81	2.99E-04	6.82E-10
2.40E-01	2.40E-01	1.07E-04	2.43E-10	Ge81m	1.86E-03	4.23E-09
2.40E-01	2.40E-01	1.07E-04	2.43E-10	Ge81	1.86E-03	4.23E-09
1.97E-01	1.97E-01	8.77E-05	2.00E-10	As81	6.59E-03	1.50E-08
9.74E-03	9.74E-03	4.33E-06	9.86E-12	Se81m	5.37E-05	4.89E-10
2.98E-02	2.98E-02	1.32E-05	3.01E-11	Se81	4.59E-05	2.15E-10
1.30E+00	1.30E+00	5.76E-04	1.31E-09	Ga82	7.90E-04	1.80E-09
1.30E+00	1.30E+00	5.76E-04	1.31E-09	Ge82	6.06E-03	1.38E-08
1.23E+00	1.23E+00	5.49E-04	1.25E-09	As82m	1.72E-02	3.91E-08
1.15E+00	1.15E+00	5.12E-04	1.16E-09	As82	2.22E-02	5.05E-08
1.17E+00	1.17E+00	5.22E-04	1.19E-09	Ga83	3.70E-04	8.43E-10
1.17E+00	1.17E+00	5.22E-04	1.19E-09	Ge83	2.27E-03	5.17E-09
1.12E+00	1.12E+00	4.99E-04	1.14E-09	As83	1.53E-02	3.47E-08
5.25E-01	5.25E-01	2.33E-04	5.31E-10	Se83m	3.70E-02	8.43E-08
6.32E-02	6.32E-02	2.81E-05	6.40E-11	Se83	1.53E-04	7.98E-10
4.05E+00	4.05E+00	1.80E-03	4.10E-09	Ga84	3.71E-04	8.44E-10
4.05E+00	4.05E+00	1.80E-03	4.10E-09	Ge84	4.94E-03	1.13E-08
4.05E+00	4.05E+00	1.80E-03	4.10E-09	As84m	2.47E-03	5.63E-09
4.05E+00	4.05E+00	1.80E-03	4.10E-09	As84	2.26E-02	5.15E-08
7.89E-01	7.89E-01	3.51E-04	7.98E-10	Se84	1.44E-01	3.40E-07
4.46E+00	4.46E+00	1.98E-03	4.51E-09	Ge85	2.45E-03	5.57E-09
4.46E+00	4.46E+00	1.98E-03	4.51E-09	As85	9.20E-03	2.09E-08
3.24E+00	3.24E+00	1.44E-03	3.28E-09	Se85	1.05E-01	2.39E-07
8.18E+00	8.18E+00	3.64E-03	8.28E-09	As86	7.49E-03	1.70E-08
7.67E+00	7.67E+00	3.41E-03	7.76E-09	Se86	1.17E-01	2.66E-07
4.04E+00	4.04E+00	1.80E-03	4.09E-09	Rb86m	2.48E-01	5.66E-07
2.11E-04	2.11E-04	9.39E-08	2.14E-13	Rb86	1.27E-04	2.86E-09
2.84E+00	2.84E+00	1.26E-03	2.87E-09	As87	2.31E-03	5.25E-09
2.83E+00	2.83E+00	1.26E-03	2.87E-09	Se87	1.67E-02	3.80E-08
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Se88	6.18E-04	1.41E-09
5.57E-01	5.57E-01	2.48E-04	5.64E-10	Rb88	2.83E-03	1.29E-08
5.67E-01	5.67E-01	2.52E-04	5.74E-10	Se89	2.36E-04	5.38E-10
8.43E-01	8.43E-01	3.75E-04	8.53E-10	Rb89	1.86E-03	7.79E-09
1.83E-04	1.83E-04	8.11E-08	1.85E-13	Sr89	1.10E-04	2.48E-09
3.03E+00	3.03E+00	1.35E-03	3.07E-09	Rb90m	7.01E-01	1.74E-06
4.77E+00	4.77E+00	2.12E-03	4.83E-09	Rb90	7.25E-01	1.68E-06
1.08E-06	1.08E-06	4.79E-10	1.09E-15	Sr90	1.08E-04	2.44E-09
8.49E-02	8.49E-02	3.77E-05	8.59E-11	Y90m	2.49E-03	3.82E-08
4.24E-03	4.24E-03	1.88E-06	4.29E-12	Y90	2.83E-03	6.24E-08

1.40E+01	1.40E+01	6.21E-03	1.41E-08	Se91	3.84E-03	8.74E-09
1.13E+01	1.13E+01	5.00E-03	1.14E-08	Rb91	6.57E-01	1.50E-06
2.67E-02	2.67E-02	1.19E-05	2.70E-11	Sr91	3.19E-03	6.20E-08
3.26E-01	3.26E-01	1.45E-04	3.30E-10	Y91m	8.86E-04	7.46E-09
1.95E-04	1.95E-04	8.65E-08	1.97E-13	Y91	5.86E-04	1.32E-08
2.15E+01	2.15E+01	9.54E-03	2.17E-08	Rb92	9.77E-02	2.23E-07
9.13E-02	9.13E-02	4.06E-05	9.24E-11	Sr92	5.92E-03	8.61E-08
7.96E-02	7.96E-02	3.54E-05	8.05E-11	Y92	4.68E-03	7.41E-08
2.47E+01	2.47E+01	1.10E-02	2.50E-08	Rb93	1.47E-01	3.34E-07
2.21E+00	2.21E+00	9.81E-04	2.23E-09	Sr93	7.22E-01	2.11E-06
2.47E+01	2.47E+01	1.10E-02	2.50E-08	Y93m	2.06E-02	4.69E-08
2.80E-02	2.80E-02	1.24E-05	2.83E-11	Y93	5.56E-03	1.09E-07
1.01E+01	1.01E+01	4.50E-03	1.02E-08	Rb94	2.79E-02	6.35E-08
9.31E+00	9.31E+00	4.14E-03	9.42E-09	Sr94	7.01E-01	1.60E-06
7.96E-01	7.96E-01	3.54E-04	8.05E-10	Y94	3.69E-03	1.74E-08
7.21E+00	7.21E+00	3.21E-03	7.30E-09	Rb95	2.76E-03	6.29E-09
1.57E+01	1.57E+01	6.97E-03	1.59E-08	Sr95	3.98E-01	9.07E-07
1.70E+00	1.70E+00	7.53E-04	1.72E-09	Y95	3.24E-03	1.10E-08
1.96E-04	1.96E-04	8.70E-08	1.98E-13	Zr95	2.95E-04	6.65E-09
3.47E-03	3.47E-03	1.54E-06	3.51E-12	Nb95m	6.96E-04	1.54E-08
3.59E-04	3.59E-04	1.59E-07	3.63E-13	Nb95	1.08E-04	2.43E-09
2.56E+01	2.56E+01	1.14E-02	2.60E-08	Rb96	5.19E-03	1.18E-08
2.56E+01	2.56E+01	1.14E-02	2.60E-08	Sr96	2.79E-02	6.35E-08
2.53E+01	2.53E+01	1.12E-02	2.56E-08	Y96m	2.47E-01	5.62E-07
2.56E+01	2.56E+01	1.14E-02	2.59E-08	Y96	1.38E-01	3.14E-07
1.27E-02	1.27E-02	5.63E-06	1.28E-11	Nb96	2.53E-03	5.36E-08
2.39E+01	2.39E+01	1.06E-02	2.42E-08	Rb97	4.11E-03	9.35E-09
2.39E+01	2.39E+01	1.06E-02	2.42E-08	Sr97	1.05E-02	2.38E-08
2.39E+01	2.39E+01	1.06E-02	2.42E-08	Y97m	2.94E-02	6.69E-08
2.39E+01	2.39E+01	1.06E-02	2.42E-08	Y97	9.13E-02	2.08E-07
1.64E-02	1.64E-02	7.30E-06	1.66E-11	Zr97	9.83E+00	2.04E-04
1.30E+01	1.30E+01	5.78E-03	1.32E-08	Nb97m	6.94E-01	1.58E-06
2.23E-01	2.23E-01	9.93E-05	2.26E-10	Nb97	1.26E-03	1.31E-08
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Rb98	2.64E-06	6.02E-12
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Sr98	1.61E-05	3.66E-11
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Y98m	5.19E-05	1.18E-10
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Y98	1.46E-05	3.32E-11
1.72E+01	1.72E+01	7.65E-03	1.74E-08	Zr98	5.35E-01	1.22E-06
3.17E-01	3.17E-01	1.41E-04	3.20E-10	Nb98m	1.73E-01	1.47E-06
2.35E+01	2.35E+01	1.04E-02	2.37E-08	Nb98	9.88E-04	2.25E-09
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Rb99	9.06E-06	2.06E-11
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Sr99	4.43E-05	1.01E-10
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Y99m	1.48E-09	3.38E-15
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Y99	2.42E-04	5.51E-10
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Zr99	3.62E-04	8.25E-10
3.79E-02	3.79E-02	1.69E-05	3.84E-11	Nb99m	5.76E-03	1.33E-08
2.32E+01	2.32E+01	1.03E-02	2.34E-08	Nb99	3.52E-01	8.03E-07
4.33E-03	4.33E-03	1.92E-06	4.38E-12	Mo99	6.50E-04	1.44E-08
4.75E-02	4.75E-02	2.11E-05	4.80E-11	Tc99m	1.41E-04	2.54E-09
1.53E-10	1.53E-10	6.81E-14	1.55E-19	Tc99	1.02E-10	2.31E-15
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Rb100	1.38E-03	3.13E-09

2.55E+01	2.55E+01	1.13E-02	2.58E-08	Sr100	5.22E-03	1.19E-08
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Y100m	2.44E-02	5.55E-08
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Y100	1.89E-02	4.31E-08
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Zr100	1.84E-01	4.18E-07
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Nb100m	7.78E-02	1.77E-07
2.55E+01	2.55E+01	1.13E-02	2.58E-08	Nb100	3.89E-02	8.86E-08
2.37E+01	2.37E+01	1.05E-02	2.40E-08	Tc100	3.80E-01	8.64E-07
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Sr101	9.56E-05	2.18E-10
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Y101	3.54E-04	8.07E-10
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Zr101	1.98E-03	4.50E-09
2.10E+01	2.10E+01	9.33E-03	2.12E-08	Nb101	1.58E-01	3.59E-07
9.76E-01	9.76E-01	4.34E-04	9.88E-10	Mo101	2.11E-03	8.62E-09
1.00E+00	1.00E+00	4.45E-04	1.01E-09	Tc101	8.61E-04	3.46E-09
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Rb102	3.30E-05	7.50E-11
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Sr102	5.60E-05	1.28E-10
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Y102m	2.97E-04	6.75E-10
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Y102	2.47E-04	5.63E-10
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Zr102	2.39E-03	5.44E-09
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Nb102m	3.54E-03	8.06E-09
8.10E-01	8.10E-01	3.60E-04	8.20E-10	Nb102	1.07E-03	2.44E-09
1.04E+00	1.04E+00	4.61E-04	1.05E-09	Mo102	4.11E-01	1.46E-06
2.54E+00	2.54E+00	1.13E-03	2.57E-09	Tc102m	5.97E-01	1.48E-06
1.74E+01	1.74E+01	7.74E-03	1.76E-08	Tc102	9.38E-02	2.13E-07
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Zr103	1.61E-04	3.66E-10
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Nb103	1.85E-04	4.22E-10
5.63E+00	5.63E+00	2.50E-03	5.69E-09	Mo103	3.84E-01	8.73E-07
6.59E+00	6.59E+00	2.93E-03	6.67E-09	Tc103	3.59E-01	8.17E-07
3.61E-03	3.61E-03	1.60E-06	3.65E-12	Ru103	2.17E+00	4.71E-05
6.42E+00	6.42E+00	2.86E-03	6.50E-09	Rh103m	3.63E-01	8.27E-07
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Nb104m	1.24E-04	2.81E-10
1.22E-01	1.22E-01	5.40E-05	1.23E-10	Nb104m	5.93E-04	1.35E-09
3.71E+00	3.71E+00	1.65E-03	3.75E-09	Mo104	2.24E-01	5.10E-07
2.77E-01	2.77E-01	1.23E-04	2.80E-10	Tc104	1.28E-03	5.91E-09
2.84E-01	2.84E-01	1.26E-04	2.87E-10	Nb105	5.77E-04	1.31E-09
2.13E-01	2.13E-01	9.45E-05	2.15E-10	Mo105m	6.46E-03	1.47E-08
2.22E+00	2.22E+00	9.86E-04	2.25E-09	Mo105	1.12E-01	2.55E-07
3.43E-01	3.43E-01	1.52E-04	3.47E-10	Tc105	1.14E-01	3.36E-07
1.02E-02	1.02E-02	4.54E-06	1.03E-11	Ru105	6.03E+00	1.02E-04
2.37E+00	2.37E+00	1.05E-03	2.40E-09	Rh105m	1.08E-01	2.45E-07
1.28E-03	1.28E-03	5.70E-07	1.30E-12	Rh105	7.70E-01	1.67E-05
8.91E-02	8.91E-02	3.96E-05	9.02E-11	Nb106	9.97E-05	2.27E-10
8.85E-02	8.85E-02	3.93E-05	8.96E-11	Mo106	7.55E-04	1.72E-09
1.12E+00	1.12E+00	4.96E-04	1.13E-09	Tc106	4.06E-02	9.24E-08
2.10E-06	2.10E-06	9.35E-10	2.13E-15	Ru106	1.27E-03	2.86E-08
8.61E-03	8.61E-03	3.83E-06	8.71E-12	Rh106m	1.25E-04	1.67E-09
1.23E+00	1.23E+00	5.45E-04	1.24E-09	Rh106	3.69E-02	8.41E-08
7.70E-01	7.70E-01	3.42E-04	7.79E-10	Mo107	2.74E-03	6.23E-09
6.62E-01	6.62E-01	2.94E-04	6.69E-10	Tc107	1.42E-02	3.24E-08
1.28E-01	1.28E-01	5.70E-05	1.30E-10	Ru107	2.70E-02	6.52E-08
2.42E-02	2.42E-02	1.08E-05	2.45E-11	Rh107	1.16E-02	5.97E-08
1.56E-13	1.56E-13	6.95E-17	1.58E-22	Pd107	1.57E-13	3.54E-18
8.10E-03	8.10E-03	3.60E-06	8.20E-12	Mo108	1.24E-05	2.81E-11

8.10E-03	8.10E-03	3.60E-06	8.20E-12	Tc108	4.12E-05	9.37E-11
3.68E-02	3.68E-02	1.64E-05	3.73E-11	Ru108	8.95E-03	2.25E-08
2.41E-01	2.41E-01	1.07E-04	2.43E-10	Rh108m	4.15E-03	9.44E-09
1.62E-02	1.62E-02	7.20E-06	1.64E-11	Tc109	2.31E-05	5.25E-11
1.55E-02	1.55E-02	6.91E-06	1.57E-11	Ru109m	2.05E-04	4.67E-10
8.45E-02	8.45E-02	3.76E-05	8.55E-11	Ru109	2.99E-03	6.81E-09
4.88E-02	4.88E-02	2.17E-05	4.94E-11	Rh109	3.96E-03	9.03E-09
1.04E-04	1.04E-04	4.64E-08	1.06E-13	Pd109	1.04E-05	2.11E-10
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Tc110	3.42E-05	7.78E-11
3.80E-02	3.80E-02	1.69E-05	3.84E-11	Ru110	5.78E-04	1.32E-09
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Rh110m	1.28E-04	2.91E-10
9.26E-02	9.26E-02	4.11E-05	9.37E-11	Rh110	2.72E-03	6.18E-09
3.24E-03	3.24E-03	1.44E-06	3.28E-12	Ru111	4.94E-06	1.13E-11
3.17E-03	3.17E-03	1.41E-06	3.21E-12	Rh111	3.54E-05	8.05E-11
8.50E-06	8.50E-06	3.78E-09	8.60E-15	Pd111m	5.04E-06	8.91E-11
2.51E-03	2.51E-03	1.12E-06	2.54E-12	Pd111	1.21E-03	6.27E-09
3.83E-02	3.83E-02	1.70E-05	3.87E-11	Ag111m	2.50E-03	5.70E-09
5.22E-06	5.22E-06	2.32E-09	5.28E-15	Ag111	3.14E-06	7.03E-11
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Ru112	1.11E-04	2.53E-10
2.43E-02	2.43E-02	1.08E-05	2.46E-11	Rh112	1.98E-05	4.50E-11
3.56E-05	3.56E-05	1.58E-08	3.60E-14	Pd112	2.13E-05	4.49E-10
2.38E-05	2.38E-05	1.06E-08	2.41E-14	Ag112	1.43E-06	3.08E-11
5.67E-03	5.67E-03	2.52E-06	5.74E-12	Rh113	5.19E-06	1.18E-11
2.12E-03	2.12E-03	9.41E-07	2.14E-12	Pd113m	1.89E-04	4.30E-10
4.40E-02	4.40E-02	1.96E-05	4.45E-11	Pd113	4.30E-03	9.82E-09
5.82E-02	5.82E-02	2.59E-05	5.89E-11	Ag113m	3.98E-03	9.06E-09
2.77E-04	2.77E-04	1.23E-07	2.80E-13	Ag113	1.64E-01	2.88E-06
3.10E-02	3.10E-02	1.38E-05	3.14E-11	Pd114	4.51E-03	1.04E-08
1.27E-01	1.27E-01	5.65E-05	1.29E-10	Ag114	5.82E-04	1.32E-09
3.97E-02	3.97E-02	1.77E-05	4.02E-11	Ag115m	7.53E-04	1.72E-09
1.52E-03	1.52E-03	6.75E-07	1.54E-12	Ag115	7.15E-06	3.50E-11
4.81E-07	4.81E-07	2.14E-10	4.87E-16	Cd115m	2.89E-06	6.53E-11
9.63E-06	9.63E-06	4.28E-09	9.74E-15	Cd115	2.89E-06	6.35E-11
1.15E-04	1.15E-04	5.09E-08	1.16E-13	In115m	1.13E-06	1.90E-11
0.00E+00	0.00E+00	0.00E+00	0.00E+00	In115	0.00E+00	0.00E+00
5.67E-03	5.67E-03	2.52E-06	5.74E-12	Ru113	1.56E-05	3.54E-11
5.67E-02	5.67E-02	2.52E-05	5.74E-11	Rh114	9.80E-05	2.23E-10
2.62E-02	2.62E-02	1.16E-05	2.65E-11	Pd115	1.24E-03	2.82E-09
5.07E-02	5.07E-02	2.25E-05	5.13E-11	Pd116	6.53E-04	1.49E-09
5.18E-02	5.18E-02	2.30E-05	5.25E-11	Ag116m	5.26E-04	1.20E-09
1.20E-02	1.20E-02	5.33E-06	1.21E-11	Ag116	1.87E-03	4.35E-09
1.01E-02	1.01E-02	4.50E-06	1.02E-11	Rh116	7.21E-06	1.64E-11
8.10E-03	8.10E-03	3.60E-06	8.20E-12	Pd117	3.62E-05	8.25E-11
8.10E-03	8.10E-03	3.60E-06	8.20E-12	Ag117m	4.36E-05	9.93E-11
2.28E-02	2.28E-02	1.01E-05	2.31E-11	Ag117	1.68E-03	3.82E-09
1.79E-04	1.79E-04	7.94E-08	1.81E-13	Cd117m	5.24E-06	8.21E-11
2.44E-04	2.44E-04	1.08E-07	2.47E-13	Cd117	7.09E-06	1.00E-10
3.13E-04	3.13E-04	1.39E-07	3.16E-13	In117m	3.60E-06	4.62E-11
8.23E-04	8.23E-04	3.66E-07	8.33E-13	In117	2.21E-06	1.74E-11
4.46E-02	4.46E-02	1.98E-05	4.51E-11	Pd118	9.51E-05	2.17E-10
4.46E-02	4.46E-02	1.98E-05	4.51E-11	Ag118m	1.09E-04	2.48E-10
4.46E-02	4.46E-02	1.98E-05	4.51E-11	Ag118	1.81E-04	4.12E-10

6.10E-04	6.10E-04	2.71E-07	6.17E-13	Cd118	3.32E-04	2.81E-09
4.46E-02	4.46E-02	1.98E-05	4.51E-11	In118	2.26E-04	5.15E-10
2.84E-02	2.84E-02	1.26E-05	2.87E-11	Ag119	6.05E-05	1.38E-10
1.42E-02	1.42E-02	6.33E-06	1.44E-11	Cd119m	1.85E-03	4.25E-09
1.20E-02	1.20E-02	5.32E-06	1.21E-11	Cd119	1.87E-03	4.35E-09
2.00E-03	2.00E-03	8.89E-07	2.02E-12	In119m	4.59E-06	2.10E-11
1.37E-02	1.37E-02	6.09E-06	1.39E-11	In119	1.86E-03	4.28E-09
8.65E-08	8.65E-08	3.84E-11	8.75E-17	Sn119m	1.74E-08	3.92E-13
5.27E-02	5.27E-02	2.34E-05	5.33E-11	Ag120m	1.71E-05	3.90E-11
5.27E-02	5.27E-02	2.34E-05	5.33E-11	Ag120	6.59E-05	1.50E-10
2.94E-02	2.94E-02	1.31E-05	2.98E-11	Cd120	1.51E-03	3.43E-09
3.09E-02	3.09E-02	1.37E-05	3.13E-11	In120m	1.47E-03	3.34E-09
5.27E-02	5.27E-02	2.34E-05	5.33E-11	In120	1.66E-04	3.78E-10
6.08E-02	6.08E-02	2.70E-05	6.15E-11	Ag121	4.82E-05	1.10E-10
6.04E-02	6.04E-02	2.69E-05	6.12E-11	Cd121m	4.91E-04	1.12E-09
5.80E-02	5.80E-02	2.58E-05	5.87E-11	Cd121	7.94E-04	1.81E-09
1.01E-02	1.01E-02	4.50E-06	1.03E-11	In121m	2.13E-03	5.15E-09
5.08E-02	5.08E-02	2.26E-05	5.14E-11	In121	1.18E-03	2.69E-09
1.46E-09	1.46E-09	6.48E-13	1.48E-18	Sn121m	1.10E-09	2.48E-14
2.59E-05	2.59E-05	1.15E-08	2.62E-14	Sn121	7.77E-07	1.66E-11
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Ag122m	6.59E-05	1.50E-10
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Ag122	3.69E-05	8.40E-11
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Cd122	3.49E-04	7.94E-10
6.34E-02	6.34E-02	2.82E-05	6.42E-11	In122m	6.95E-04	1.58E-09
6.48E-02	6.48E-02	2.88E-05	6.56E-11	In122	9.88E-05	2.25E-10
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Ag123	2.11E-05	4.80E-11
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Cd123m	1.21E-04	2.76E-10
6.48E-02	6.48E-02	2.88E-05	6.56E-11	Cd123	1.38E-04	3.15E-10
3.81E-02	3.81E-02	1.69E-05	3.85E-11	In123m	1.80E-03	4.11E-09
6.48E-02	6.48E-02	2.88E-05	6.55E-11	In123	3.95E-04	8.99E-10
1.11E-03	1.11E-03	4.94E-07	1.12E-12	Sn123m	2.95E-06	2.19E-11
2.41E-07	2.41E-07	1.07E-10	2.44E-16	Sn123	7.26E-07	1.64E-11
2.03E-02	2.03E-02	9.00E-06	2.05E-11	Ag124	4.53E-06	1.03E-11
2.03E-02	2.03E-02	9.00E-06	2.05E-11	Cd124	2.55E-05	5.81E-11
2.03E-02	2.03E-02	9.00E-06	2.05E-11	In124m	7.00E-05	1.59E-10
1.09E-01	1.09E-01	4.86E-05	1.11E-10	In124	3.54E-04	8.05E-10
6.25E-05	6.25E-05	2.78E-08	6.33E-14	Sb124m	4.68E-08	9.83E-13
8.75E-07	8.75E-07	3.89E-10	8.85E-16	Sb124	1.75E-06	3.96E-11
4.86E-02	4.86E-02	2.16E-05	4.92E-11	Cd125m	2.97E-05	6.75E-11
4.86E-02	4.86E-02	2.16E-05	4.92E-11	Cd125	3.36E-05	7.65E-11
4.70E-02	4.70E-02	2.09E-05	4.76E-11	In125m	5.82E-04	1.33E-09
4.86E-02	4.86E-02	2.16E-05	4.92E-11	In125	1.17E-04	2.66E-10
5.70E-03	5.70E-03	2.53E-06	5.77E-12	Sn125m	2.10E-03	6.86E-09
6.88E-06	6.88E-06	3.06E-09	6.96E-15	Sn125	8.27E-06	1.86E-10
6.59E-08	6.59E-08	2.93E-11	6.67E-17	Sb125	5.67E-08	1.28E-12
1.14E-06	1.14E-06	5.08E-10	1.16E-15	Te125m	6.88E-07	1.55E-11
6.08E-02	6.08E-02	2.70E-05	6.15E-11	Cd126	3.21E-05	7.32E-11
6.08E-02	6.08E-02	2.70E-05	6.15E-11	In126m	1.01E-04	2.29E-10
6.08E-02	6.08E-02	2.70E-05	6.15E-11	In126	9.45E-05	2.15E-10
1.26E-12	1.26E-12	5.61E-16	1.28E-21	Sn126	9.49E-12	2.15E-16
2.34E-01	2.34E-01	1.04E-04	2.36E-10	Sb126m	8.69E-06	1.98E-11
9.28E-06	9.28E-06	4.12E-09	9.39E-15	Sb126	7.97E-06	1.79E-10

5.83E-01	5.83E-01	2.59E-04	5.90E-10	Cd127	2.37E-04	5.40E-10
5.83E-01	5.83E-01	2.59E-04	5.90E-10	In127m	2.21E-03	5.03E-09
5.83E-01	5.83E-01	2.59E-04	5.90E-10	In127	6.76E-04	1.54E-09
8.97E-02	8.97E-02	3.99E-05	9.08E-11	Sn127m	2.02E-02	4.96E-08
3.46E-03	3.46E-03	1.54E-06	3.50E-12	Sn127	6.66E-05	8.85E-10
7.97E-05	7.97E-05	3.54E-08	8.07E-14	Sb127	4.79E-05	1.06E-09
3.89E-06	3.89E-06	1.73E-09	3.93E-15	Te127m	5.85E-06	1.32E-10
7.81E-04	7.81E-04	3.47E-07	7.90E-13	Te127	1.55E-03	3.02E-08
2.39E-01	2.39E-01	1.06E-04	2.42E-10	Cd128	6.81E-05	1.55E-10
2.39E-01	2.39E-01	1.06E-04	2.42E-10	In128m	1.70E-04	3.87E-10
2.39E-01	2.39E-01	1.06E-04	2.42E-10	In128	1.94E-04	4.43E-10
1.75E-02	1.75E-02	7.77E-06	1.77E-11	Sn128	2.41E-04	2.24E-09
9.94E-02	9.94E-02	4.42E-05	1.01E-10	Sb128m	7.54E-05	2.53E-10
2.11E-03	2.11E-03	9.36E-07	2.13E-12	Sn128	3.14E-05	6.07E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Cd129	1.11E-04	2.53E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In129m	2.88E-04	6.57E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In129	2.60E-04	5.91E-10
4.04E-01	4.04E-01	1.80E-04	4.09E-10	Sn129m	2.87E-03	6.53E-09
1.08E-01	1.08E-01	4.81E-05	1.10E-10	Sn129	1.43E-02	3.28E-08
1.56E-01	1.56E-01	6.91E-05	1.57E-10	Sb129m	7.12E-02	3.24E-07
1.06E-02	1.06E-02	4.72E-06	1.07E-11	Sb129	6.27E-04	1.05E-08
5.80E-05	5.80E-05	2.58E-08	5.87E-14	Te129m	1.16E-04	2.62E-09
4.01E-02	4.01E-02	1.78E-05	4.06E-11	Te129	2.50E-04	2.52E-09
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Cd130	8.24E-05	1.88E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In130m1	2.18E-04	4.97E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In130m2	2.10E-04	4.78E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In130	1.19E-04	2.72E-10
1.36E-01	1.36E-01	6.03E-05	1.37E-10	Sn130m	1.38E-02	3.15E-08
1.38E+00	1.38E+00	6.12E-04	1.39E-09	Sn130	2.85E-01	6.86E-07
8.44E-01	8.44E-01	3.75E-04	8.54E-10	Sb130m	2.52E-01	6.94E-07
1.41E-01	1.41E-01	6.26E-05	1.43E-10	Sb130	8.30E-04	6.12E-09
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In131m1	1.24E-04	2.81E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In131m2	1.44E-04	3.28E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	In131	1.15E-04	2.63E-10
2.06E-01	2.06E-01	9.17E-05	2.09E-10	Sn131m	1.21E-02	2.76E-08
6.22E+00	6.22E+00	2.77E-03	6.30E-09	Sn131	3.51E-01	7.99E-07
3.52E-01	3.52E-01	1.57E-04	3.57E-10	Sb131	2.85E-03	1.52E-08
4.23E-03	4.23E-03	1.88E-06	4.28E-12	Te131m	2.54E-03	5.48E-08
3.25E-01	3.25E-01	1.44E-04	3.28E-10	Te131	8.02E-03	4.49E-08
2.84E-01	2.84E-01	1.26E-04	2.87E-10	In132	5.77E-05	1.31E-10
1.13E+01	1.13E+01	5.04E-03	1.15E-08	Sn132	4.55E-01	1.03E-06
3.83E+00	3.83E+00	1.70E-03	3.87E-09	Sb132m	6.22E-01	1.45E-06
2.66E+00	2.66E+00	1.18E-03	2.69E-09	Sb132	6.03E-01	1.49E-06
2.63E-03	2.63E-03	1.17E-06	2.66E-12	Te132	1.75E-03	3.88E-08
3.65E-01	3.65E-01	1.62E-04	3.69E-10	Sn133	5.34E-04	1.22E-09
3.88E+00	3.88E+00	1.73E-03	3.93E-09	Sb133	5.70E-01	1.32E-06
3.12E-01	3.12E-01	1.39E-04	3.16E-10	Te133m	8.58E-03	7.67E-08
1.37E+00	1.37E+00	6.07E-04	1.38E-09	Te133	7.00E-03	2.61E-08
2.84E-01	2.84E-01	1.26E-04	2.87E-10	Sn134	3.00E-04	6.83E-10
2.78E-01	2.78E-01	1.24E-04	2.82E-10	Sb134m	2.94E-03	6.69E-09
2.82E+01	2.82E+01	1.26E-02	2.86E-08	Sb134	2.30E-02	5.23E-08
4.62E-01	4.62E-01	2.05E-04	4.68E-10	Te134	3.53E-03	2.70E-08

5.27E-01	5.27E-01	2.34E-04	5.33E-10	Sb135	9.16E-04	2.08E-09
2.24E+01	2.24E+01	9.94E-03	2.26E-08	Te135	4.31E-01	9.81E-07
3.44E-01	3.44E-01	1.53E-04	3.48E-10	Cs135m	6.28E-04	5.48E-09
1.52E-11	1.52E-11	6.76E-15	1.54E-20	Cs135	4.57E-12	1.03E-16
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Sb136	3.38E-04	7.69E-10
3.68E-01	3.68E-01	1.63E-04	3.72E-10	Te136	6.52E-03	1.48E-08
9.57E-04	9.57E-04	4.25E-07	9.69E-13	Cs136	6.40E-04	1.44E-08
1.62E-01	1.62E-01	7.20E-05	1.64E-10	Te137	4.12E-04	9.38E-10
1.10E-06	1.10E-06	4.89E-10	1.11E-15	Cs137	3.31E-06	7.48E-11
5.96E+00	5.96E+00	2.65E-03	6.03E-09	Ba137m	8.91E-01	2.06E-06
4.17E+00	4.17E+00	1.85E-03	4.22E-09	Te138	5.94E-03	1.35E-08
5.83E+00	5.83E+00	2.59E-03	5.90E-09	Cs138m	9.77E-01	2.28E-06
5.84E-01	5.84E-01	2.60E-04	5.91E-10	Cs138	3.76E-03	2.46E-08
1.80E+00	1.80E+00	8.02E-04	1.83E-09	Cs139	6.60E-01	2.13E-06
2.09E-01	2.09E-01	9.30E-05	2.12E-10	Ba139	2.96E-03	3.28E-08
1.17E+01	1.17E+01	5.19E-03	1.18E-08	Cs140	7.46E-01	1.70E-06
9.71E-04	9.71E-04	4.32E-07	9.83E-13	Ba140	2.92E-04	6.57E-09
7.38E-03	7.38E-03	3.28E-06	7.47E-12	La140	2.21E-03	4.82E-08
1.51E+01	1.51E+01	6.72E-03	1.53E-08	Cs141	3.81E-01	8.68E-07
9.49E-01	9.49E-01	4.22E-04	9.60E-10	Ba141	4.38E-03	2.03E-08
7.67E-02	7.67E-02	3.41E-05	7.76E-11	La141	4.51E-03	7.35E-08
3.84E-04	3.84E-04	1.71E-07	3.89E-13	Ce141	2.89E-04	6.51E-09
1.40E+01	1.40E+01	6.23E-03	1.42E-08	Cs142	2.57E-02	5.84E-08
1.45E+00	1.45E+00	6.44E-04	1.47E-09	Ba142	2.81E-03	9.70E-09
1.82E-01	1.82E-01	8.09E-05	1.84E-10	La142	3.45E-03	4.00E-08
8.43E+00	8.43E+00	3.75E-03	8.53E-09	Cs143	1.52E-02	3.47E-08
1.92E+01	1.92E+01	8.51E-03	1.94E-08	Ba143	2.78E-01	6.33E-07
2.28E+01	2.28E+01	1.01E-02	2.31E-08	La143	3.31E-03	7.53E-09
8.44E-03	8.44E-03	3.75E-06	8.54E-12	Ce143	2.53E-03	5.47E-08
8.56E-04	8.56E-04	3.81E-07	8.67E-13	Pr143	5.72E-04	1.29E-08
4.86E-01	4.86E-01	2.16E-04	4.92E-10	Cs144	4.99E-04	1.14E-09
4.74E-01	4.74E-01	2.10E-04	4.79E-10	Ba144	5.48E-03	1.25E-08
1.46E+01	1.46E+01	6.48E-03	1.47E-08	La144	5.99E-01	1.36E-06
3.85E-05	3.85E-05	1.71E-08	3.90E-14	Ce144	1.16E-03	2.62E-08
2.09E+00	2.09E+00	9.29E-04	2.11E-09	Pr144m	6.73E-01	1.95E-06
8.95E-01	8.95E-01	3.98E-04	9.06E-10	Pr144	2.03E-03	9.13E-09
1.61E+01	1.61E+01	7.17E-03	1.63E-08	Cs145	9.67E-03	2.20E-08
1.61E+01	1.61E+01	7.17E-03	1.63E-08	Ba145	6.55E-02	1.49E-07
1.33E+01	1.33E+01	5.90E-03	1.34E-08	La145	3.23E-01	7.35E-07
3.33E+00	3.33E+00	1.48E-03	3.37E-09	Ce145	5.74E-01	1.34E-06
3.11E-02	3.11E-02	1.38E-05	3.15E-11	Pr145	1.85E-03	3.32E-08
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Cs146	1.33E-04	3.02E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Ba146	9.06E-04	2.06E-09
3.99E-01	3.99E-01	1.77E-04	4.04E-10	La146m	4.05E-03	9.22E-09
4.05E-01	4.05E-01	1.80E-04	4.09E-10	La146	2.59E-03	5.89E-09
6.22E-01	6.22E-01	2.77E-04	6.30E-10	Ce146	2.63E-01	1.03E-06
3.51E-01	3.51E-01	1.56E-04	3.55E-10	Pr146	1.72E-01	9.46E-07
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Cs147	9.35E-05	2.13E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Ba147	3.67E-04	8.37E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	La147	1.65E-03	3.77E-09
5.01E+00	5.01E+00	2.23E-03	5.07E-09	Ce147	2.83E-01	6.44E-07
4.82E-01	4.82E-01	2.14E-04	4.88E-10	Pr147	6.77E-04	2.63E-09

4.19E-04	4.19E-04	1.86E-07	4.24E-13	Nd147	2.52E-04	5.66E-09
4.81E-06	4.81E-06	2.14E-09	4.87E-15	Pm147	1.45E-05	3.27E-10
0.00E+00	0.00E+00	0.00E+00	0.00E+00	Sm147	0.00E+00	0.00E+00
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Cs148	6.18E-05	1.41E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Ba148	2.64E-04	6.00E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	La148	4.53E-04	1.03E-09
3.63E+00	3.63E+00	1.61E-03	3.67E-09	Ce148	2.05E-01	4.66E-07
2.03E+00	2.03E+00	9.02E-04	2.05E-09	Pr148m	6.03E-01	1.38E-06
1.82E+00	1.82E+00	8.10E-04	1.84E-09	Pr148	3.49E-01	8.02E-07
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Ba149	1.40E-04	3.19E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	La149	4.33E-04	9.85E-10
4.05E-01	4.05E-01	1.80E-04	4.10E-10	Ce149	2.14E-03	4.87E-09
1.19E+00	1.19E+00	5.29E-04	1.21E-09	Pr149	1.62E-01	3.72E-07
3.05E-02	3.05E-02	1.35E-05	3.08E-11	Nd149	5.82E-04	7.11E-09
9.96E-04	9.96E-04	4.43E-07	1.01E-12	Pm149	2.99E-04	6.57E-09
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Ce150	1.81E-04	4.12E-10
4.05E-02	4.05E-02	1.80E-05	4.09E-11	Pr150	2.55E-04	5.80E-10
1.17E-02	1.17E-02	5.19E-06	1.18E-11	Pm150	3.40E-04	4.93E-09
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Ce151	4.12E-05	9.38E-11
3.60E-02	3.60E-02	1.60E-05	3.65E-11	Pr151	6.90E-04	1.57E-09
1.72E+00	1.72E+00	7.64E-04	1.74E-09	Nd151	2.18E-02	4.96E-08
7.25E-04	7.25E-04	3.22E-07	7.34E-13	Pm151	1.09E-04	2.33E-09
2.61E-08	2.61E-08	1.16E-11	2.65E-17	Sm151	7.87E-08	1.78E-12
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Ce152	5.77E-05	1.31E-10
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Pr152	1.32E-04	3.00E-10
2.39E-03	2.39E-03	1.06E-06	2.42E-12	Nd152	9.51E-04	3.39E-09
1.98E-03	1.98E-03	8.82E-07	2.01E-12	Pm152m	8.44E-04	3.33E-09
1.77E-01	1.77E-01	7.87E-05	1.79E-10	Pm152	3.95E-02	9.68E-08
4.05E-02	4.05E-02	1.80E-05	4.10E-11	Pr153	1.77E-04	4.03E-10
3.09E-02	3.09E-02	1.37E-05	3.13E-11	Nd153	9.04E-04	2.06E-09
7.32E-02	7.32E-02	3.25E-05	7.41E-11	Pm153	1.99E-02	5.20E-08
1.52E-04	1.52E-04	6.74E-08	1.53E-13	Sm153	2.28E-05	4.98E-10
4.05E-03	4.05E-03	1.80E-06	4.10E-12	Pr154	9.47E-06	2.16E-11
3.24E-03	3.24E-03	1.44E-06	3.28E-12	Nd154	8.49E-05	1.93E-10
9.17E-04	9.17E-04	4.08E-07	9.28E-13	Pm154m	1.44E-04	3.35E-10
1.04E-01	1.04E-01	4.64E-05	1.06E-10	Pm154	1.06E-02	2.42E-08
4.61E-03	4.61E-03	2.05E-06	4.66E-12	Eu154m	2.49E-03	2.02E-08
4.79E-08	4.79E-08	2.13E-11	4.85E-17	Eu154	9.61E-07	2.17E-11
4.01E-03	4.01E-03	1.78E-06	4.06E-12	Nd155	3.63E-05	8.26E-11
2.55E-03	2.55E-03	1.13E-06	2.58E-12	Pm155	1.08E-04	2.46E-10
1.13E-01	1.13E-01	5.03E-05	1.15E-10	Sm155	8.48E-06	1.93E-11
3.72E-08	3.72E-08	1.65E-11	3.76E-17	Eu155	1.12E-07	2.53E-12
4.05E-03	4.05E-03	1.80E-06	4.10E-12	Nd156	2.26E-05	5.15E-11
3.20E-03	3.20E-03	1.42E-06	3.24E-12	Pm156	8.64E-05	1.97E-10
7.41E-05	7.41E-05	3.29E-08	7.50E-14	Sm156	4.42E-06	8.59E-11
1.91E-06	1.91E-06	8.49E-10	1.93E-15	Eu156	1.92E-06	4.31E-11
3.96E-04	3.96E-04	1.76E-07	4.01E-13	Pm157	4.38E-06	9.98E-12
2.62E-03	2.62E-03	1.17E-06	2.65E-12	Sm157	8.93E-04	2.69E-09
2.40E-05	2.40E-05	1.07E-08	2.43E-14	Eu157	2.05E-06	4.22E-11
5.27E-03	5.27E-03	2.34E-06	5.33E-12	Pm158	2.67E-05	6.09E-11
6.46E-04	6.46E-04	2.87E-07	6.53E-13	Sm158	1.73E-04	4.51E-10
2.00E-04	2.00E-04	8.90E-08	2.03E-13	Eu158	1.35E-06	1.09E-11

3.95E-04	3.95E-04	1.75E-07	3.99E-13	Sm159	4.57E-06		1.04E-11
1.67E-04	1.67E-04	7.44E-08	1.69E-13	Eu159	7.70E-05		3.55E-10
2.78E-06	2.78E-06	1.24E-09	2.81E-15	Gd159	2.08E-07		4.34E-12
4.00E-05	4.00E-05	1.78E-08	4.05E-14	Sm160	3.90E-07		8.87E-13
8.09E-04	8.09E-04	3.59E-07	8.18E-13	Eu160	3.10E-05		7.07E-11
3.23E-05	3.23E-05	1.44E-08	3.27E-14	Eu161	8.51E-07		1.94E-12
6.99E-05	6.99E-05	3.11E-08	7.07E-14	Gd161	1.43E-05		3.43E-11
2.82E-08	2.82E-08	1.25E-11	2.86E-17	Tb161	8.48E-09		1.90E-13
3.96E-06	3.96E-06	1.76E-09	4.01E-15	Eu162	4.42E-08		1.01E-13
8.04E-05	8.04E-05	3.57E-08	8.14E-14	Gd162	6.93E-07		1.58E-12
7.06E-06	7.06E-06	3.14E-09	7.15E-15	Tb162	2.34E-06		6.92E-12

Appendix A

Technical Specifications for the North Carolina State University PULSTAR Reactor

Facility License No. R-120

Docket No. 50-297

Amendment 17

TABLE OF CONTENTS

FIGURES

TABLES

1.0. INTRODUCTION

1.1. Purpose

These Technical Specifications provide limits within which operation of the reactor will assure the health and safety of the public, the environment and on-site personnel. Areas addressed are Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features, and Administrative Controls.

Included in this document are the "Bases" for the Technical Specifications. The bases provide the technical support for the individual technical specification and are included for information purposes only. The bases are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2. Definitions

1.2.1. **Channel**: A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.2.2. **Channel Calibration**: A channel calibration is an adjustment of the channel, such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include a Channel Test.

1.2.3. **Channel Check**: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

1.2.4. **Channel Test**: A channel test is the introduction of a signal into the channel for verification that it is operable.

1.2.5. **Cold Critical**: The condition of the reactor when it is critical, with negligible xenon, and the fuel and bulk water are both at an isothermal temperature of 70°F.

1.2.6. **Confinement**: Confinement means a closure on the overall facility that controls the movement of air into and out of the facility through a controlled path.

1.2.7. **Control Rod**: A control rod is a neutron absorbing blade having an in-line drive which is magnetically coupled and has SCRAM capability.

1.2.8. **Excess Reactivity**: Excess reactivity is that amount of reactivity that would exist if all control rods (and Shim Rod) were fully withdrawn from the point where the reactor is exactly critical ($k_{eff}=1$).

1.2.9. **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 tube or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design. Specific categories of experiments include:

a. **Tried Experiment**: Tried experiments are those experiments that have been previously performed in this reactor. Specifically, a tried experiment has similar size, shape, composition and location of an experiment previously approved and performed in the reactor.

b. **Secured Experiment**: A secured experiment is any experiment, experimental facility, or component of an experiment that 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.

c. **Non-Secured Experiment**: A non-secured experiment is an experiment that does not meet the criteria for being a "secured" experiment.

d. **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.

e. **Fueled Experiment**: A fueled experiment is an experiment which contains fissionable material.

1.2.10. **Experimental Facilities**: Experimental facilities are facilities used to perform experiments. They include beam tubes, thermal columns, void tanks, pneumatic transfer systems, in-core facilities at single-assembly positions, out-of-core irradiation facilities, and the bulk irradiation facility.

1.2.11. **Limiting Condition for Operation**: Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10CFR50.36).

1.2.12. **Limiting Safety System Setting**: Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded (10CFR50.36).

1.2.13. **Measured Value**: The measured value is the value of a parameter as it appears on the output of a channel.

1.2.14. **Operable**: Operable means a component or system is capable of performing its intended function.

1.2.15. **Operating**: Operating means a component or system is performing its intended function.

1.2.16. **pcm**: A unit of reactivity that is the abbreviation for "percent millirho" and is equal to 10^{-5} $\Delta k/k$ reactivity. For example, 1000 pcm is equal to 1.0% $\Delta k/k$.

1.2.17. **Reactor Building**: The Reactor Building includes the Reactor Bay, Control Room and Ventilation Room, the Mechanical Equipment Room (MER), and the Primary Piping Vault (PPV). The Nuclear Regulatory Commission R-120 license applies to the areas in the Reactor Building and the Waste Tank Vault.

1.2.18. **Reactor Operation**: Reactor operation is any condition when the reactor is not secured or shutdown.

1.2.19. **Reactor Operator**: A reactor operator (RO) is an individual who is licensed under 10 CFR 55 to manipulate the controls of the facility.

1.2.20. **Reactor Operator Assistant (ROA)**: An individual who has been certified by successful completion of an in-house training program to assist the licensed reactor operator during reactor operation.

1.2.21. **Reactor Safety System**: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.2.22. **Reactor Secured**: The 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, **or**

b. The following conditions exist:

i. All scrammable neutron absorbing control rods are fully inserted, **and**

ii. The reactor key switch is in the OFF position and the key is removed from the lock, **and**

iii. 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, **and**

iv. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding one dollar (730 pcm).

1.2.23. **Reactor Shutdown**: That subcritical condition of the reactor where the absolute value of the negative reactivity of the core is equal to or greater than the shutdown margin.

1.2.24. **Reportable Event**: A Reportable Event is any of the following:

a. Violation of a Safety Limit.

b. Release of radioactivity from the site above allowed limits.

c. Operation with actual Safety System Settings (SSS) for required systems less conservative than the Limiting Safety System Settings (LSSS) specified in these specifications.

d. Operation in violation of Limiting Conditions for Operation (LCO) established in these Technical Specifications.

e. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdown. (For components or systems other than those required by these Technical Specifications, the failure of the extra component or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function).

f. An unanticipated or uncontrolled change in reactivity greater than one dollar (730 pcm). Reactor trips resulting from a known cause are excluded.

g. Abnormal or significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks), which could result in exceeding radiological limits for personnel or environment, or both, as prescribed in the facility Emergency Plan.

h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence of an unsafe condition with regard to reactor operations.

1.2.25. **Safety Limit**: Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity (10CFR50.36).

1.2.26. **Shim Rod**: A shim rod is a neutron absorbing rod having an in-line drive which is mechanically, rather than magnetically, coupled and does not have a SCRAM capability.

1.2.27. **Senior Reactor Operator**: A senior reactor operator (SRO) is an individual who is licensed under 10 CFR 55 to manipulate the controls of the facility and to direct the activities of licensed reactor operators.

1.2.28. **Shutdown Margin**: Shutdown margin means the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition with the most reactive scrammable rod fully withdrawn, the non-scrammable rod (Shim rod) fully withdrawn, and experiments considered at their most reactive condition, and finally, that the reactor will remain subcritical without further operator action.

1.2.29. **Total Nuclear Peaking Factor**: The factor obtained by multiplying the measured local radial and axial neutron fluence peaking factors.

1.2.30. **True Value**: The true value is the actual value of a parameter.

1.2.31. **University Management:** University Management is the Chancellor or Office of the Chancellor other University Administrator(s) having authority designated by the Chancellor or as specified in University policies.

1.2.32 **Unscheduled Shutdown**: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused 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 check-out operations.

2.0. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1. Safety Limits (SL)

2.1.1. Safety Limits for Forced Convection Flow

Applicability

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance with forced convection flow. These interrelated variables are:

P Reactor Thermal Power

W Reactor Coolant Flow Rate

H Height of Water Above the Top of the Core

T_{inlet} Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of forced convection flow, the Safety Limit shall be as follows:

a. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 2.1-1 under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time outside the operating envelope shown in Figure 2.1-1.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

Above 80 percent of the full core flow of 500 gpm in the region of full power operation, the criterion used to establish the Safety Limit was no bulk boiling at the outlet of any coolant channel. This was found to be far more limiting than the criterion of a minimum allowable burnout heat flux ratio of 2.0. The analysis is given in the SAR Appendix 3B.

In the region below 80 percent of full core flow, where, under a loss of flow transient at power the flow coasts down to zero, reverses, and then establishes natural convection, the criterion for selecting a Safety Limit is taken as a fuel cladding temperature. The analysis of a loss of flow transient is presented in Appendix 3B of the SAR. For initial conditions of full flow and an operating power of 1.4 MWt, the maximum clad temperature reached under the conservative assumptions of the analysis was 273°F which is well below the temperature at which fuel clad damage could possibly occur. The Safety Limit shown in Figure 2.1-1 for flow less than 80 percent of full flow is the steady state power corresponding to the maximum fuel clad temperature of 273°F with natural convection flow, namely, 1.4 MWt.

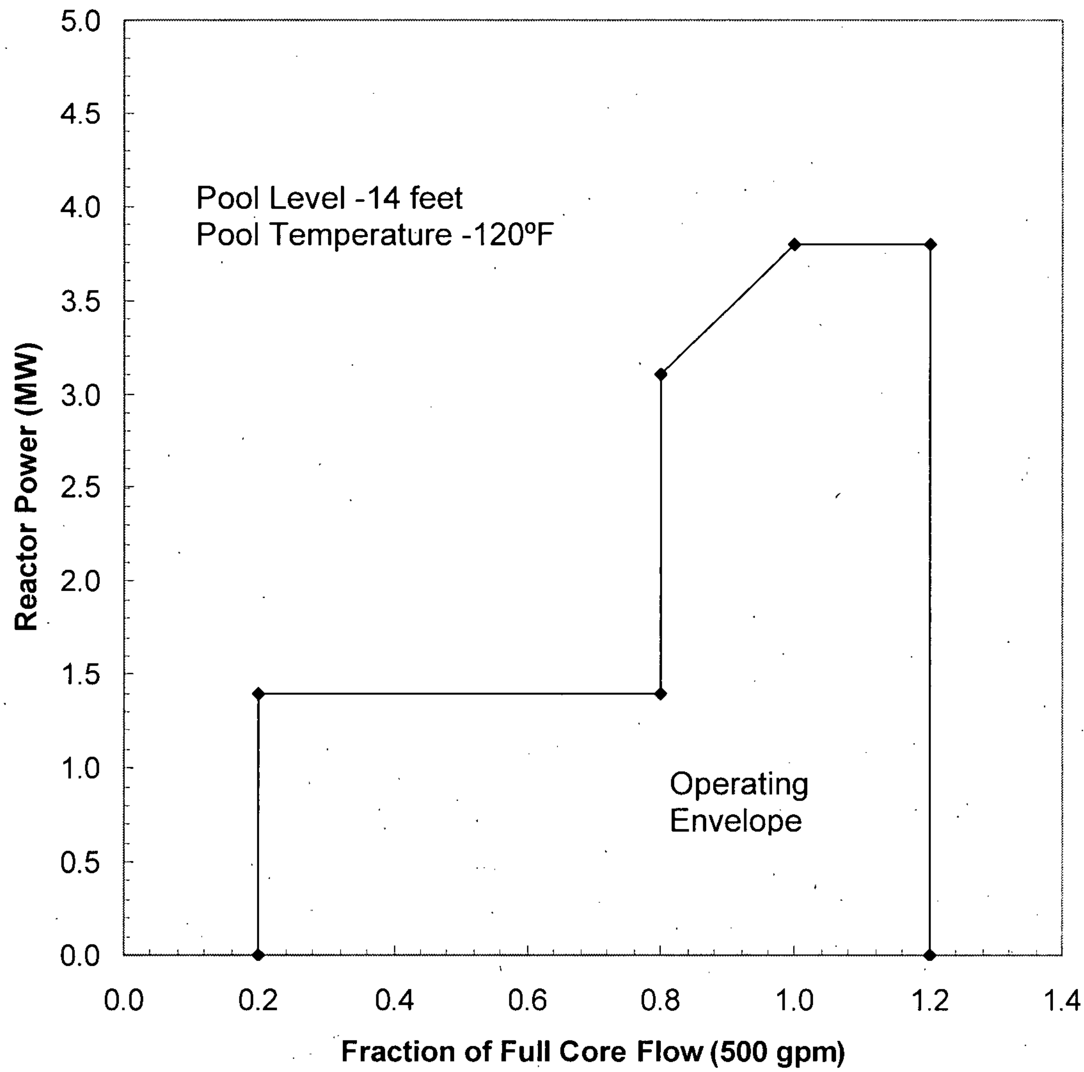


Figure 2.1-1: Power-Flow Safety Limit Curve

2.1.2. Safety Limits for Natural Convection Flow.

Applicability

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance with natural convection flow. These interrelated variables are:

P	Reactor Thermal Power
H	Height of Water Above the Top of the Core
$\mathbf{T}_{inlet}$	Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of natural convection flow, the Safety Limit shall be as follows:

a. The true value of reactor thermal power (P) shall not exceed 1.4 MWt.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

The criterion for establishing a Safety Limit with natural convection flow is established as the fuel clad temperature. This is consistent with Figure 2.1-1 for forced convection flow during a transient. The analysis of natural convection flow given in Appendix 3B and 3C of the SAR shows that at 1.4 MWt the maximum fuel clad temperature is 273°F which is well below the temperature at which fuel clad damage could occur. The flow with natural convection at this power is 98 gpm. This flow is based on data from natural convection tests with fuel assemblies of the same design performed in the prototype PULSTAR Reactor, as referenced in Section 3 of the SAR.

2.2. Limiting Safety System Settings

2.2.1. Limiting Safety System Settings (LSSS) for Forced Convection Flow

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), height of water above the top of the core (H), and pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under the condition of forced convection flow, the Limiting Safety System Settings shall be as follows:

P	1.3 MWt (max.)
W	450 gpm (min.)
H	14 feet, 2 inches (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in the Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded during the most limiting anticipated transient (loss of flow). The safety margin that is provided between the Limiting Safety System Settings and the Safety Limits also allows for the most adverse combination of instrument uncertainties associated with measuring the observable parameters. These instrument uncertainties include a flow variation of ten percent, a pool level variation of two inches and a power level variation of seven percent.

The analysis presented in Section 3 of the SAR of a loss of flow transient indicates that if the interrelated variables were at their LSSS, as specified in 2.2.1 above, at the initiation of the transient, the Safety Limits specified in 2.1.1 would not be exceeded.

2.2.2. Limiting Safety System Settings (LSSS) for Natural Convection Flow

Applicability

This specification applies to the setpoints for the safety channel monitoring reactor thermal power (P), the height of water above the core (H), and the pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specifications

Under the condition of natural convection flow, the Limiting Safety System Settings shall be as follows:

P	250 kWt (max.)
H	14 feet, 2 inches (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage (for these relatively low power levels) which was the criterion used in establishing the Safety Limits (see Specification 2.1.2). It is desirable to minimize to the greatest extent practical, N-16 dose at the pool surface which might be aided by steam bubble rise during up-flow in natural convection. Analysis of coolant bulk boiling given in SAR, Section 3, indicates that the large safety margin on reactor thermal power assumed in Specification 2.2.2 above will satisfy this additional criterion of no bulk boiling in any channel.

3.0. LIMITING CONDITIONS FOR OPERATION

3.1. Reactor Core Configuration

Applicability

This specification applies to the reactor core configuration during forced convection or natural convection flow operations.

Objective

The objective is to assure that the reactor will be operated within the bounds of established Safety Limits.

Specification

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

a. A maximum of twenty-five fuel assemblies.

b. A maximum of ten reflector assemblies of either graphite or beryllium or a combination of these located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rod guides are in place.

e. The maximum worth of a single fuel assembly shall not exceed 1590 pcm.

f. The total nuclear peaking factor in any fuel assembly shall not exceed 2.92.

Bases

Specifications 3.1.a through 3.1.d require that the core be configured such that there is no bypass cooling flow around the fuel through the grid plate.

Specification 3.1.e provides assurances that a fuel loading accident will not result in a Safety Limit to be exceeded as discussed in SAR Section 13.2.2.1.

Specification 3.1.f provides assurances that core hot channel power are bounded by the SAR assumptions in Appendix 3-B.

3.2. Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods, shim rod and experiments.

Objective

The objective is to assure that the reactor can be shutdown at all times and that the Safety Limits will not be exceeded.

Specifications

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

a. The shutdown margin, with the highest worth scrammable control rod fully withdrawn, with the shim rod fully withdrawn, and with experiments at their most reactive condition, relative to the cold critical condition, is greater than 400 pcm.

b. The excess reactivity is not greater than 3970 pcm.

c. The drop time of each control rod is not greater than 1.0 second.

d. The rate of reactivity insertion of the control rods is not greater than 100 pcm per second (critical region only).

e. The absolute reactivity worth of experiments or their rate of reactivity change shall not exceed the values indicated in Table 3.2-1.

f. The sum of the absolute values of the reactivity worths of all experiments shall not be greater than 2890 pcm.

Table 3.2-1: Reactivity Limits for Experiments

Experiment	Limit
Movable	300 pcm or 100 pcm/sec, whichever is more limiting
Non-secured	1000 pcm
Secured	1590 pcm

Bases

The shutdown margin required by Specification 3.2.a assures that the reactor can be shut down from any operating condition and will remain shutdown after cool down and xenon decay, even if the highest worth scrammable rod should be in the fully withdrawn position. Refer to Section 3.1.2.1.

The upper limit on excess reactivity ensures that an adequate shutdown margin is maintained.

The rod drop time required by Specification 3.2.c assures that the Safety Limit will not be exceeded during the flow reversal which occurs upon loss of forced convection coolant flow. The rise in fuel temperature due to heat storage is partially controlled by the reactivity insertion associated with the SCRAM. The analysis of this transient is based upon this SCRAM reactivity insertion taking the form of a ramp function of two second duration. This analysis is found in SAR Section 3.2.4 and Appendix 3B. The rod drop time is the time interval measured between the instant of a test signal input to the SCRAM Logic Unit and the instant of the rod seated signal.

The maximum rate of reactivity insertion by the control rods which is allowed by Specification 3.2.d assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Refer to SAR Section 13.

Experiments affecting the reactivity condition of the reactor are commonly categorized by the sign of the reactivity effect produced by insertion of the experiment. An experiment having a large reactivity effect of either sign can also produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculations and the calibration of Safety Channels.

The Specification 3.2.e is intended to prevent inadvertent reactivity changes during reactor operation caused by the insertion or removal of an experiment. It further provides assurance that the failure of a single experiment will not result in a reactivity insertion which could cause the Safety Limit to be exceeded. Analyses indicate that the inadvertent reactivity insertion of these magnitudes will not result in consequences greater than those analyzed in the SAR Sections 3 and 13.

The total limit on reactivity associated with experiments ensures that an adequate shutdown margin is maintained.

3.3. Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels which must be operable in order to assure that the Safety Limits are not exceeded.

Specification

The reactor shall not be operated unless the reactor safety system channels described in Table 3.3-1 are operable.

Table 3.3-1: Required Safety and Safety Related Channels

	Measuring Channel	Function
a.	Startup Power Level(1)	Inhibits Control Rod withdrawal when neutron count is ≤2 cps
b.	Safety Power Level	SCRAM at ≤1.3 MW (LSSS) Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
c.	Linear Power Level	SCRAM at ≤1.3 MW (LSSS)
d.	Log N Power Level	Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
e.	Flow Monitoring(2)	SCRAM when flapper not closed and Flow/Flapper SCRAMs are enabled
f.	Primary Coolant Flow(2)	SCRAM at ≥450 gpm (LSSS) when Flow/Flapper SCRAMs are enabled
g.	Pool Water Temperature Monitoring Switch	ALARM at ≤117°F
h.	Pool Water Temperature Measuring Channel	SCRAM at ≤117°F (LSSS)
i.	Pool Water Level	SCRAM at ≥14 feet 2 inches
j.	Manual SCRAM Button	SCRAM
k.	Reactor Key Switch	SCRAM
l.	Over-the-Pool Radiation Monitor(3)	Alarm (100 mR/hr)

[(1)] Required only for reactor startup when power level is less than 4 watts.

[(2)] Either the Flapper SCRAM or the Flow SCRAM may be bypassed during maintenance testing and/or performance of a startup checklist in order to verify each SCRAM is independently operable. The reactor must be shutdown in order to use these bypasses.

[(3)] May be bypassed for less than two minutes during the return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

Bases

The Startup Channel inhibit function assures the required startup neutron source is sufficient and in its proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to assure adequate information is available to the operator.

The reactor power level SCRAMs provide the redundant protection channels to assure that, if a condition should develop which would tend to cause the reactor to operate at an abnormally high power level, an immediate automatic protective action will occur to prevent exceeding the Safety Limit.

The primary coolant flow SCRAMs provide redundant channels to assure when the reactor is at power levels which require forced flow cooling that, if sufficient flow is not present, an immediate automatic shutdown of the reactor will occur to prevent exceeding a Safety Limit. The Log N Power Channel is included in this section since it is one of the two channels which enables the two flow SCRAMs when the reactor is above 250 kW (LSSS).

The pool water temperature channel provides for shutdown of the reactor and prevents exceeding the Safety Limit due to high pool water temperature.

The pool water level channel together with the Over-the-Pool (Bridge) radiation monitor, provides two diverse channels for shutdown of the reactor and prevents exceeding the Safety Limit due to insufficient pool height.

To prevent unnecessary initiation of the evacuation and confinement systems during the return of the pneumatic capsule from the core to the unloading station or during the removal of experiments from the reactor pool, the Over-the-Pool monitor may be bypassed for the specified time interval.

The manual SCRAM button and the Reactor Key switch provide two manual SCRAM methods to the reactor operator if unsafe or abnormal conditions should occur.

3.4. Reactor Instrumentation

Applicability

This specification applies to the instrumentation that shall be available to the reactor operator to support the safe operation of the reactor, but are not considered reactor safety systems.

Objective

The objective is to require that sufficient information be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the following are operable:

a. N-16 Power Measuring Channel when reactor power is greater than 500 kW

b. Control Rod Position Indications for each control rod and the Shim Rod

c. Differential pressure gauge for "Bay with Respect to Atmosphere"

Bases

The N-16 Channel provides the necessary power level information to allow adjustment of Safety and Linear Power Channels.

Control rod position indications give the operator information on rod height necessary to verify shutdown margin..

The differential pressure gauge provides the pressure difference between the Reactor Bay and the outside ambient and confirms air flow in the ventilation stream for both normal and confinement modes.

3.5. Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated unless the radiation monitoring equipment listed in Table 3.5-1 is operable.(1)(2)(3)

a. Three fixed area monitors operating in the Reactor Building with their setpoints as listed in Table 3.5-1.(1)(3)(4)

b. Particulate and gas building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as listed in Table 3.5-1.(1)(3)(4)

c. The Radiation Rack Recorder.(5)

Table 3.5-1: Required Radiation Area Monitors

Monitor	Alert Setpoint	Alarm Setpoint
Control Room	≤ 2 mR/hr	≤ 5 mR/hr
Over-the-Pool	≤ 5 mR/hr	≤ 100 mR/hr
West Wall	≤ 5 mR/hr	≤ 100 mR/hr
Stack Gas	≤ 1000 Ar-41 AEC(6)	≤ 5,000 Ar-41AEC(6)
Stack Particulate	≤ 1000 Co-60 AEC(6)	≤ 5,000 Co-60 AEC(6)

(1) For periods of time, not to exceed ninety days, for maintenance to the radiation monitoring channel, the intent of this specification will be satisfied if one of the installed channels is replaced with a gamma-sensitive instrument which has its own alarm audible or observable in the control room. Refer to SAR Section 5.

(2) The Over-the-Pool Monitor may be bypassed for less than two minutes during return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

(3) Stack Gas and Particulate are based on the AEC quantities present in the ventilation flow stream as it exits the stack. Refer to SAR Section 10 for setpoint bases for the radiation monitoring equipment.

(4) May be bypassed for less than one minute immediately after starting the pneumatic blower system.

(5) During repair and/or maintenance of the recorder not to exceed 90 days, the specified area and effluent monitor readings shall be recorded manually at a nominal interval of 30 minutes when the reactor is not shutdown. Refer to SAR Section 5.

(6) Airborne Effluent Concentrations (AEC) values from 10CFR20 Appendix B, Table 2

Bases

A continued evaluation of the radiation levels within the Reactor Building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Section 5 of the SAR.

Evaluation of the continued discharge air to the environment will be made using the information recorded from the particulate and gas monitors.

When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, the building will be automatically placed in confinement as described in SAR Section 5.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule from the core to the unloading station or during removal of experiments from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval. Refer to SAR Section 5.

3.6. Confinement and Main HVAC Systems

Applicability

This specification applies to the operation of the Reactor Building confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated, nor shall irradiated fuel be moved within the pool area, unless the following equipment is operable, and conditions met:

Table 3.6-1: Required Main HVAC and Confinement Conditions

	Equipment/Condition	Function
a.	All doors, except the Control Room, and basement corridor entrance: self-latching, self-closing, closed and locked.	To maintain reactor building negative differential pressure (dp).(1)
b.	Control room and basement corridor entrance door: self-latching, self-closing and closed.	To maintain reactor building negative differential pressure.(2)
c.	Reactor Building under a negative differential pressure of not less than 0.2" H_2O with the normal ventilation system or 0.1" H_2O with one confinement fan operating.	To maintain reactor building negative differential pressure with reference to outside ambient.(3)
d.	Confinement system	Operable(4)(5)(7)
e.	Evacuation system	Operable(6)

(1) Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indications are available for the reactor operator to verify door status. Refer to SAR Section 5.

(2) Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and Reactor Building. Refer to SAR Section 5.

(3) During an interval not to exceed 30 minutes after a loss of dp is identified with Main HVAC operating, reactor operation may continue while the loss of dp is investigated and corrected. Refer to SAR Section 5.

(4) Operability also demonstrated with an auxiliary power source.

(5) One filter train may be out of service for the purpose of maintenance, repair, and/or surveillance for a period of time not to exceed 45 days. During the period of time in which one filter train is out of service, the standby filter train shall be verified to be operable every 24 hours if the reactor is operating with the Reactor Building in normal ventilation.

(6) The public address system can serve temporarily for the Reactor Building evacuation system during short periods of maintenance.

(7) When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, listed in Table 3.5-1, the building will be automatically placed in confinement as described in SAR Section 5.

Bases

In the event of a fission product release, the confinement initiation system will secure the normal ventilation fans and close the normal inlet and exhaust dampers. In confinement mode, a confinement system fan will: maintain a negative pressure in the Reactor Building and insure in-leakage only; purge the air from the building at a greatly reduced and controlled flow through charcoal and absolute filters; and control the discharge of all air through a 100 foot stack on site. Section 5 of the SAR describes the confinement system sequence of operation.

The allowance for operation under a temporary loss of dp when in normal ventilation is based on the requirement of having the confinement system operable and therefore ready to respond in the unlikely event of an airborne release.

3.7. Limitations of Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities. Fueled experiments must also meet the requirements of Specification 3.8.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

a. All materials to be irradiated shall be either corrosion resistant or encapsulated within a corrosion resistant container to prevent interaction with reactor components or pool water. Corrosive materials, liquids, and gases shall be doubly encapsulated.

b. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2. Pressure buildup inside any container shall be limited to 200 psi.

c. Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the saturation temperature of the reactor pool water.

d. Experimental apparatus, material or equipment to be inserted in the reactor shall be positioned so as to not cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.

e. Concerning the material content of experiments, the following will apply:

   i. No experiment will be performed unless the major constituent of the material to be irradiated is known and a reasonable effort has been made to identify trace elements and impurities whose activation may pose the dominant radiological hazard. When a reasonable effort does not give conclusive information, one or more short irradiations of small quantities of material may be performed in order to identify the activated products.

   ii. Attempts will be made to identify and limit the quantities of elements having very large thermal neutron absorption cross sections, in order to quantify reactivity effects.

iii. Explosive material[1] shall not be allowed in the reactor. Experiments in which the material is considered to be potentially explosive, either while contained, or if it leaks from the container, shall be designed to maintain seal integrity even if detonated, to prevent damage to the reactor core or to the control rods or instrumentation and to prevent any change in reactivity.

iv. Each experiment will be evaluated with respect to radiation induced physical and/or chemical changes in the irradiated material, such as decomposition effects in polymers.

v. Experiments involving cryogenic liquids[1] within the biological shield, flammable[1], or highly toxic materials[1] require specific procedures for handling and shall be limited in quantity and approved as specified in Specification 6.2.3.

f. Credible failure of any experiment shall not result in releases or exposures in excess of the annual limits established in 10CFR20.

[1] Defined as follows (reference - *Handbook of Laboratory Safety* - Chemical Rubber Company, 4th Ed., 1995, unless otherwise noted):

Toxic: A substance that has the ability to cause damage to living tissue when inhaled, ingested, injected, or absorbed through the skin (*Safety in Academic Chemistry Laboratories* - The American Chemical Society, 1994).

Flammable: Having a flash point below 73°F and a boiling point below 100°F. The flash point is defined as the minimum temperature at which a liquid forms a vapor above its surface in sufficient concentrations that it may be ignited as determined by appropriate test procedures and apparatus as specified.

Explosive: Any chemical compound, mixture, or device, where the primary or common purpose of which is to function by explosion with substantially simultaneous release of gas and heat, the resultant pressure being capable of destructive effects. The term includes, but is not limited to, dynamite, black powder, pellet powder, initiating explosives, detonators, safety fuses, squibs, detonating cord, igniter cord, and igniters.

Cryogenic: A cryogenic liquid is considered to be a liquid with a normal boiling point below -238°F (reference - *National Bureau of Standards Handbook 44*).

Bases

Specifications 3.7.a, 3.7.b, 3.7.c, and 3.7.d 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 and untried experiments.

Specification 3.7.e ensures that no physical or nuclear interferences compromise the safe operation of the reactor, specifically, an experiment having a large reactivity effect of either sign could produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculation and/or safety channels calibrations. Review of experiments using the specifications of Section 3 and Section 6 will ensure the insertion of experiments will not negate the considerations implicit in the Safety Limits and thereby violate license conditions.

3.8. Operations with Fueled Experiments

Applicability

This specification applies to the operation of the reactor with any fueled experiment.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

Fueled experiments may be performed in experimental facilities of the reactor with the following conditions and limitations:

a. The mass, fission rate and power are limited as indicated in Figure 3.8-1 and Table 3.8-1.

b. The reactor shall not be operated with a fueled experiment unless the ventilation system is operated in the confinement mode.

c. Specification 3.2 pertaining to reactivity shall be met.

d. Specification 3.7 pertaining to reactor experiments shall be met.

e. Specification 6.5 pertaining to the review of experiments shall be met.

   Each type of fueled experiment shall be classified as a new (untried) experiment with a documented review. The documented review shall include the following items:

   i. Meeting license requirements for the receipt, use, and storage of fissionable material.

   ii. Limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.

   iii. Radiation monitoring for detection of released fission products.

   iv. Design criteria related to meeting conditions given in Specifications 3.2 and 3.7.

f. Credible failure of any fueled experiment shall not result in releases or exposures in excess of 10% of the annual limits established in 10CFR20.

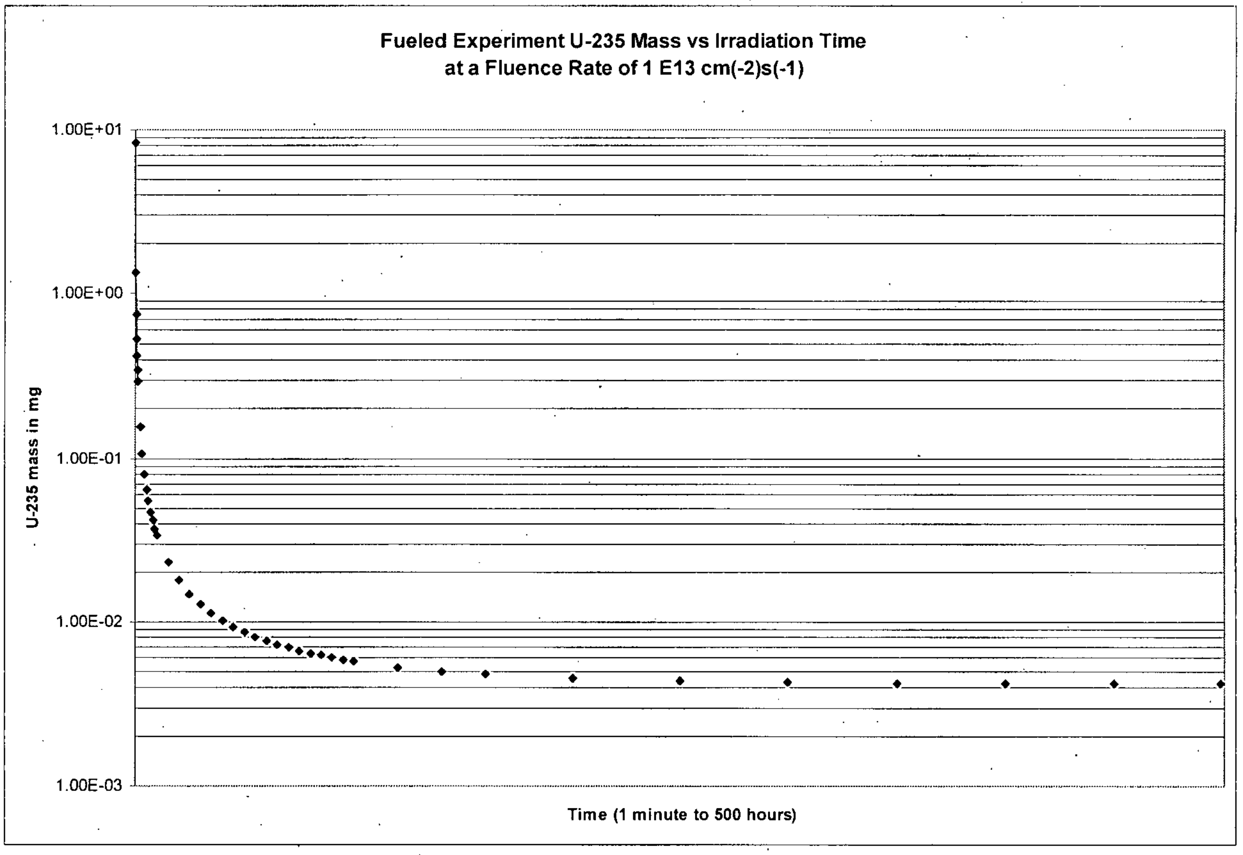


Figure 3.8-1

NOTE: The mass at 500 hours may be used for periods up to 1 y (8760 hours).

Table 3.8-1: Data for Fueled Experiments at a Fluence Rate of 1 E13 $cm^{-2}s^{-1}$

Irradiation Time, s	U-235 Mass mg	Mass-Fluence mg cm(-2)	Fission Rate f/s	Power milliwatts
6.00E+01	8.34E+00	5.00E+15	1.25E+11	4.01E+03
1.20E+02	4.75E+00	5.70E+15	7.12E+10	2.28E+03
1.80E+02	3.44E+00	6.19E+15	5.16E+10	1.65E+03
3.00E+02	2.30E+00	6.90E+15	3.45E+10	1.10E+03
6.00E+02	1.33E+00	7.98E+15	1.99E+10	6.39E+02
1.20E+03	7.55E-01	9.06E+15	1.13E+10	3.63E+02
1.80E+03	5.36E-01	9.65E+15	8.04E+09	2.57E+02
2.40E+03	4.18E-01	1.00E+16	6.27E+09	2.01E+02
3.00E+03	3.43E-01	1.03E+16	5.14E+09	1.65E+02
3.60E+03	2.92E-01	1.05E+16	4.38E+09	1.40E+02
7.20E+03	1.57E-01	1.13E+16	2.35E+09	7.54E+01
1.08E+04	1.08E-01	1.17E+16	1.62E+09	5.19E+01
1.44E+04	8.13E-02	1.17E+16	1.22E+09	3.90E+01
1.80E+04	6.55E-02	1.18E+16	9.82E+08	3.15E+01
2.16E+04	5.49E-02	1.19E+16	8.23E+08	2.64E+01
2.52E+04	4.74E-02	1.19E+16	7.11E+08	2.28E+01
2.88E+04	4.18E-02	1.20E+16	6.27E+08	2.01E+01
3.24E+04	3.74E-02	1.21E+16	5.61E+08	1.80E+01
3.60E+04	3.39E-02	1.22E+16	5.08E+08	1.63E+01
7.20E+04	1.81E-02	1.30E+16	2.71E+08	8.69E+00
1.08E+05	1.28E-02	1.38E+16	1.92E+08	6.15E+00
1.44E+05	1.02E-02	1.47E+16	1.53E+08	4.90E+00
1.80E+05	8.67E-03	1.56E+16	1.30E+08	4.16E+00
2.16E+05	7.66E-03	1.65E+16	1.15E+08	3.68E+00
2.52E+05	6.95E-03	1.75E+16	1.04E+08	3.34E+00
2.88E+05	6.42E-03	1.85E+16	9.62E+07	3.08E+00
3.24E+05	6.03E-03	1.95E+16	9.04E+07	2.90E+00
3.60E+05	5.72E-03	2.06E+16	8.57E+07	2.75E+00
3.96E+05	5.27E-03	2.09E+16	7.90E+07	2.53E+00
4.32E+05	4.97E-03	2.15E+16	7.45E+07	2.39E+00
4.68E+05	4.77E-03	2.23E+16	7.15E+07	2.29E+00
7.20E+05	4.51E-03	3.25E+16	6.76E+07	2.17E+00
1.08E+06	4.27E-03	4.61E+16	6.40E+07	2.05E+00
1.44E+06	4.21E-03	6.06E+16	6.31E+07	2.02E+00
1.80E+06	4.19E-03	7.54E+16	6.28E+07	2.01E+00
2.16E+06	4.19E-03	9.05E+16	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
1.73E+07	4.19E-03	7.24E+17	6.28E+07	2.01E+00
3.15E+07	4.19E-03	1.32E+18	6.28E+07	2.01E+00

Bases

NUREG 1537 provides guidelines for the format and content of non-power reactor licensing. Guidelines on operating conditions and accident analysis for fueled experiments are given in NUREG 1537. These guidelines include (1) actuation of engineered safety features (ESF) to prevent or mitigate the consequences of damage to fission product barriers caused by overpower or loss of cooling events, (2) use of ESF to control of radioactive material released by accidents, (3) radiation monitoring of fission product effluent and accident releases, (4) accidental analysis for loss of cooling or other experimental malfunction resulting in liquefaction or volatilization of fissile materials, (5) accident analysis for catastrophic failure of the experiment in the reactor pool or air, (6) accident analysis for insertion of excess reactivity leading to fuel melting, and (7) emergency plan activation and classification.

The limitations given in Specification 3.8 ensure that (1) fueled experiments performed in experimental facilities at the reactor prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure, (2) radiation doses to occupational personnel and the public and radioactive material releases are ALARA, (3) adequate radiation monitoring is in place, and (4) in the event of failure of a fueled experiment with the subsequent release of radioactive material, the resulting dose to personnel and the public at any location are well within limits set in 10 CFR 20.

Specification 3.8 e ensures that each type of fueled experiment is reviewed, approved, and documented as required by Specification 6.5. This includes (1) meeting applicable limitations on experiments given in Specifications 3.2 and 3.7, (2) limiting the amount of fissile material to ensure that experimental reactivity conditions are met and that radiation doses are well within 10 CFR 20 limits following maximum fission product release from a failed experiment, and (3) limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.

3.9. Primary Coolant

Applicability

This specification applies to the water quality and flow path of the primary coolant.

Objective

The objective is to ensure that primary coolant quality be maintained to acceptable values in order to reduce the potential for corrosion and limit the buildup of activated contaminants in the primary piping and pool.

Specification

The reactor shall not be operated unless the pool water meets the following limits:

a. The resistivity shall be ≥ 500 k$\Omega\cdot$cm.

b. The pH shall be within the range of 5.5 to 7.5.

Bases

The limits on resistivity are based on reducing the potential for corrosion in the primary piping or pool liner and to reduce the potential for activated contaminants in these systems.

4.0. SURVEILLANCE REQUIREMENTS

All surveillance tests required by these specifications are scheduled as described; however, some system tests may be postponed at the required intervals if that system or a closely associated system is undergoing maintenance. Any pending surveillance tests will be completed prior to reactor startup. Any surveillance item(s) which require reactor operation will be completed immediately after reactor startup. Surveillance requirements scheduled to occur during extended operation which cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown.

The intent of the surveillance interval (e.g., annually, but not to exceed fifteen months) is to maintain an average cycle, with occasional extensions as allowed by the interval tolerance. If it is desired to permanently change the scheduled date of surveillance, the particular surveillance item will be performed at an earlier date and the associated interval normalized to this revised earlier date. In no cases will permanent scheduling changes, which yield slippage of the surveillance interval routine scheduled date, be made by using the allowed interval tolerance.

4.1. Fuel

Applicability

This specification applies to the surveillance requirement for the reactor fuel.

Objective

The objective is to monitor the physical condition of the PULSTAR fuel.

Specification

a. All fuel assemblies shall be visually inspected for physical damage biennially but at intervals not to exceed thirty (30) months.

b. The reactor will be operated at such power levels necessary to determine if an assembly has had fuel pin cladding failure.

Bases

Each fuel assembly is visually inspected for physical damage that would include corrosion of the end fitting, end box, zircaloy box, missing fasteners, dents, severe surface scratches, and blocked coolant channels.

Based on a long history of prototype PULSTAR operation in conjunction with primary coolant analysis, biennial inspections of PULSTAR fuel to ensure fuel assembly integrity have been shown to be adequate for Zircoloy-2 (Zr-2) clad fuel. Any assembly that appears to have leaking fuel pin(s) will be disassembled to confirm and isolate damaged fuel pins. Damaged fuel pins will be logged as such and permanently removed from service.

4.2. Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods, shim rod, and control rod drive mechanisms (CRDM).

Objective

The objective is to assure the operability of the control rods and shim rod, and to provide current reactivity data for use in verifying adequate shutdown margin.

Specification

a. The reactivity worth of the shim rod and each control rod shall be determined annually but at intervals not to exceed fifteen (15) months for the steady state core in current use. The reactivity worth of all rods shall be determined for any new core or rod configuration, prior to routine operation.

b. Control rod drop times[1] and control rod drive times shall be determined:

   i. Annually but at intervals not to exceed fifteen (15) months.

   ii. After a control assembly is moved to a new position in the core or after maintenance or modification is performed on the control rod drive mechanism.

c. The control rods shall be visually inspected biennially but at intervals not to exceed thirty (30) months.

d. The values of excess reactivity and shutdown margin shall be determined monthly, but at intervals not to exceed six (6) weeks, and for new core configurations.

(1) Applies only to magnetically coupled rods.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The measurement of reactivity worths on an annual basis provides a correction for the slight variations expected due to burnup. This frequency of measurement has been found acceptable at similar research reactor facilities, particularly the prototype PULSTAR which has a similar slow change of rod value with burn-up.

Control rod drive and drop time measurements are made to determine whether the rods are functionally operable. These time measurements may also be utilized in reactor transient analysis.

Visual inspections include: detection of wear or corrosion in the rod drive mechanism; identification of deterioration, corrosion, flaking or bowing of the neutron absorber material; and verification of rod travel setpoints.

Control rod surveillance procedures will document proper control rod system reassembly after maintenance and recorded post-maintenance data will identify significant trends in rod performance.

4.3. Reactor Instrumentation and Safety Systems

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System and other required reactor instruments.

Objective

The objective is to assure that the required instrumentation and Safety Systems will remain operable and will prevent the Safety Limits from being exceeded.

Specification

a. A channel check of each measuring channel in the RSS shall be performed daily when the reactor is in operation.

b. A channel test of each channel in the RSS shall be performed prior to operation each day, or prior to each operation extending more than one day.

c. A channel calibration of the N-16 Channel shall be made semi-annually, but at intervals not to exceed seven and one-half (7½) months. A calorimetric measurement shall be performed to determine the N-16 detector current associated with full power operation.

d. A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half (7½) months.(1)

   i. Pool Water Temperature
   ii. Primary Cooling and Flow Monitoring (Flapper)
   iii. Pool Water Level
   iv. Primary Heat Exchanger Inlet and Outlet Temperature
   v. Safety and Linear Power Channels

(1) A channel calibration shall also be required after repair of a channel component that has the potential of affecting the calibration of the channel.

Bases

The daily channel tests and checks will assure the Reactor Safety Systems are operable and will assure operations within the limits of the operating license. The semi-annual calibrations will assure that long term drift of the channels is corrected. The calorimetric calibration of the reactor power level, in conjunction with the N-16 Channel, provides a continual reference for adjustment of the Linear, Log N and Safety Channel detector positions.

4.4. Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

a. The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen (15) months.

b. The setpoints shall be verified weekly, but at intervals not to exceed ten (10) days.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations. Therefore, the weekly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

4.5. Confinement and Main HVAC System

Applicability

This specification applies to the surveillance requirements for the confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is operable.

Specification

a. The confinement and evacuation system shall be verified to be operable within seven (7) days prior to reactor operation.

b. Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six (6) weeks.(1)

c. A visual inspection of the door seals and closures, dampers and gaskets of the confinement and ventilation systems shall be performed semi-annually but at intervals not to exceed seven and one-half (7½) months to verify they are operable.

d. The control room differential pressure (dp) gauges shall be calibrated annually but at intervals not to exceed fifteen (15) months.

e. The confinement filter train shall be tested biennially but at intervals not to exceed thirty (30) months and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate removal efficiency and leak testing of the filter housing.(2)

f. The air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not to exceed fifteen (15) months. The air flow shall be not less than 600 CFM.

(1) Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

(2) Testing shall also be required following major maintenance of the filters or housing.

Bases

Surveillance of this equipment will verify that the confinement of the Reactor Building is maintained as described in Section 5 of the SAR.

4.6. Primary and Secondary Coolant

Applicability

This specification applies to the surveillance requirement for monitoring the radioactivity in the primary and secondary coolant.

Objective

The objective is to monitor the radioactivity in the pool water to verify the integrity of the fuel cladding and other reactor structural components. The secondary water analysis is used to confirm the boundary integrity of the primary heat exchanger.

Specification

a. The primary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. The analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample, neutron activation analysis (NAA) of an aliquot, and pH and resistivity measurements.

b. The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. This analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample.

Bases

Radionuclide analysis of the pool water samples will allow detection of fuel clad failure, while neutron activation analysis will give corrosion data associated with primary system components in contact with the coolant. Refer to SAR Section 10. The detection of activation or fission products in the secondary coolant provides evidence of a primary heat exchanger leak. Refer to SAR Section 10.

5.0. DESIGN FEATURES

5.1. Reactor Fuel

a. The reactor fuel shall be UO_2 with a nominal enrichment of 4% in U-235, zircaloy clad, with fabrication details as described in Section 3 of the Safety Analysis Report.

b. Total burn-up on the reactor fuel is limited to 20,000 MWD/MTU.

5.2. Reactor Building

a. The reactor shall be housed in the Reactor Building, designed for confinement. The minimum free volume in the Reactor Building shall be 2.25×10^9 cm^3 (refer to SAR Section 13 analysis).

b. The Reactor Building ventilation and confinement systems shall be separate from the Burlington Engineering Laboratories building systems and shall be designed to exhaust air or other gases from the building through a stack with discharge at a minimum of 100 feet above ground level.

c. The openings into the Reactor Building are the truck entrance door, personnel entrance doors, and air supply and exhaust ducts.

d. The Reactor Building is located within the Burlington Engineering Laboratory complex on the north campus of North Carolina State University at Raleigh, North Carolina. Restricted Areas as defined in 10CFR20 include the Reactor Bay, Ventilation Room, Mechanical Equipment Room, Primary Piping Vault, and Waste Tank Vault. The PULSTAR Control Room is part of the Reactor Building, however it is also a controlled access area and a Controlled Area as defined in 10CFR20. The facility license applies to the Reactor Building and Waste Tank Vault. Figure 5.2-1 depicts the licensed area as being within the operations boundary.

5.3. Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water except in cases where a fuel shipping container is used, then the licensed limit for the k_{eff} limit of the container shall apply.

5.4. Reactivity Control

Reactivity control is provided by four neutron absorbing blades. Each control blade is nominally comprised of 80% silver, 15% indium, and 5% cadmium with nickel cladding. Three of these neutron absorbing blades are magnetically coupled and have scramming capability. The remaining neutron absorbing blade is non-scrammable. One of the scrammable rods may be used for automatic servo-control of reactor power. When in use, the servo-control maintains a constant power level as indicated by the Linear Power Channel.

5.5. Primary Coolant System

The primary coolant system consists of the aluminum lined reactor tank, a N-16 delay tank, a pump, and heat exchanger, and associated stainless steel piping. The nominal capacity of the primary system is 15,600 gallons. Valves are located adjacent to the biological shield to allow isolation of the pool, and at major components in the primary system to permit isolation.

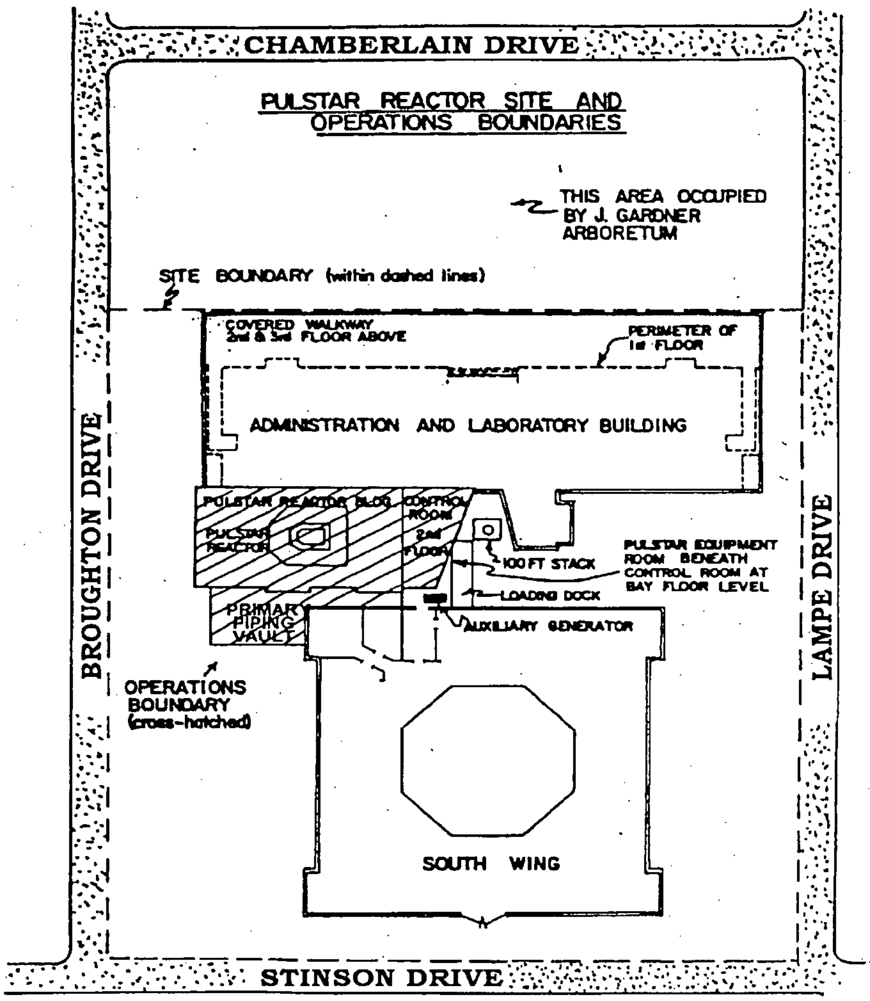


Figure 5.2-1: NCSU PULSTAR Reactor Site Map

6.0. ADMINISTRATIVE CONTROLS

6.1. Organization

The reactor facility shall be an integral part of the Department of Nuclear Engineering of the College of Engineering of North Carolina State University. The reactor shall be related to the University structure as shown in Figure 6.1-1.

6.1.1. Organizational Structure:

The reporting chain is given in Figure 6.1-1. The following specific organizational levels (as defined by ANSI/ANS-15.1-1990) and positions shall exist at the PULSTAR Facility:

Level 1 – Administration

This level shall include the Chancellor, the Dean of the College of Engineering, and the Nuclear Engineering Department Head. Within three months of appointment, the Nuclear Engineering Department Head shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility.

Level 2 – Facility Management

This level shall include the Nuclear Reactor Program (NRP) Director. The NRP Director is responsible for the safe and efficient operation of the facility as specified in the facility license and Technical Specifications, general conduct of reactor performance and NRP operations, long range development of the NRP, and NRP personnel matters. The NRP Director evaluates new service and research applications, develops new facilities and support for needed capital investments, and controls NRP budgets. The NRP Director works through the Manager of Engineering and Operations to monitor daily operations and with the Reactor Health Physicist to monitor radiation safety practices and regulatory compliance. The minimum qualifications for the NRP Director are a Master of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. Within three months of appointment, the NRP Director shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility. The NRP Director is a faculty member and reports to the Nuclear Engineering Department Head.

Level 3 – Manager of Engineering and Operations

The Manager of Engineering and Operations (MEO) performs duties as assigned by the NRP Director associated with the safe and efficient operation of the facility as specified in the facility license and Technical Specifications. The MEO is responsible for coordination of operations, experiments, and maintenance at the facility, including reviews and approvals of experiments as defined in Technical Specification 1.2.9 and 6.5, and making minor changes to procedures as stated in Technical Specification 6.4. The MEO shall receive appropriate facility specific training within three months of appointment and be certified as a Senior Reactor Operator within one year of appointment. The minimum qualifications for the MEO are a Bachelor of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. The MEO reports to the NRP Director.

Level 4 – Operating and Support Staff

This level includes licensed Senior Reactor Operators (SRO),licensed Reactor Operators (RO), and other personnel assigned to perform maintenance and technical support of the facility. Senior Reactor Operators and Reactor Operators are responsible for assuring that operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications, applicable Nuclear Regulatory Commission regulations, and the provisions of the Radiation Safety Committee and Reactor Safety and Audit Committee. All Senior Reactor Operators shall have three years of nuclear experience and shall have a high school diploma or successfully completed a General Education Development test. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of nuclear experience as applicable to research reactors for Senior Reactor Operators. Other Level 4 personnel shall have a high school diploma or shall have successfully completed a General Education Development test. All Level 4 personnel report to the Manager of Engineering and Operations.

Reactor Health Physicist

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP shall have a high school diploma or shall have successfully completed a General Education Development test and have three years of relevant experience in applied radiation safety. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of experience in radiation safety as applicable to research reactors. The RHP reports directly to the Nuclear

Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6.1-1.

6.1.2. Responsibility:

Responsibility for the safe operation of the PULSTAR Reactor shall be with the chain of command established in Figure 6.1-1.

Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, the Technical Specifications, and federal regulations.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon the appropriate qualifications.

6.1.3. Minimum Staffing:

The minimum staffing when the reactor is not secured shall be:

a. A licensed reactor operator or senior reactor operator shall be present in the Control Room.

b. A Reactor Operator Assistant (ROA), capable of being at the reactor facility within five (5) minutes upon request of the reactor operator on duty.

c. A Designed Senior Reactor Operator (DSRO). This individual shall be readily available on call, meaning:

   i. Has been specifically designated and the designation known to the reactor operator on duty.

   ii. Keeps the reactor operator on duty informed of where he may be rapidly contacted and the telephone number.

   iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

d. A Reactor Health Physicist or his designated alternate. This individual shall also be on call, under the same limitations as prescribed for the Designed Senior Reactor Operator under Specification 6.1.3.c.

6.1.4. Senior Reactor Operator Duties:

The following events shall require the presence of a licensed Senior Reactor Operator at the facility or its administrative offices:

a. Initial startup and approach to power.

b. All fuel or control rod relocations within the reactor core or pool.

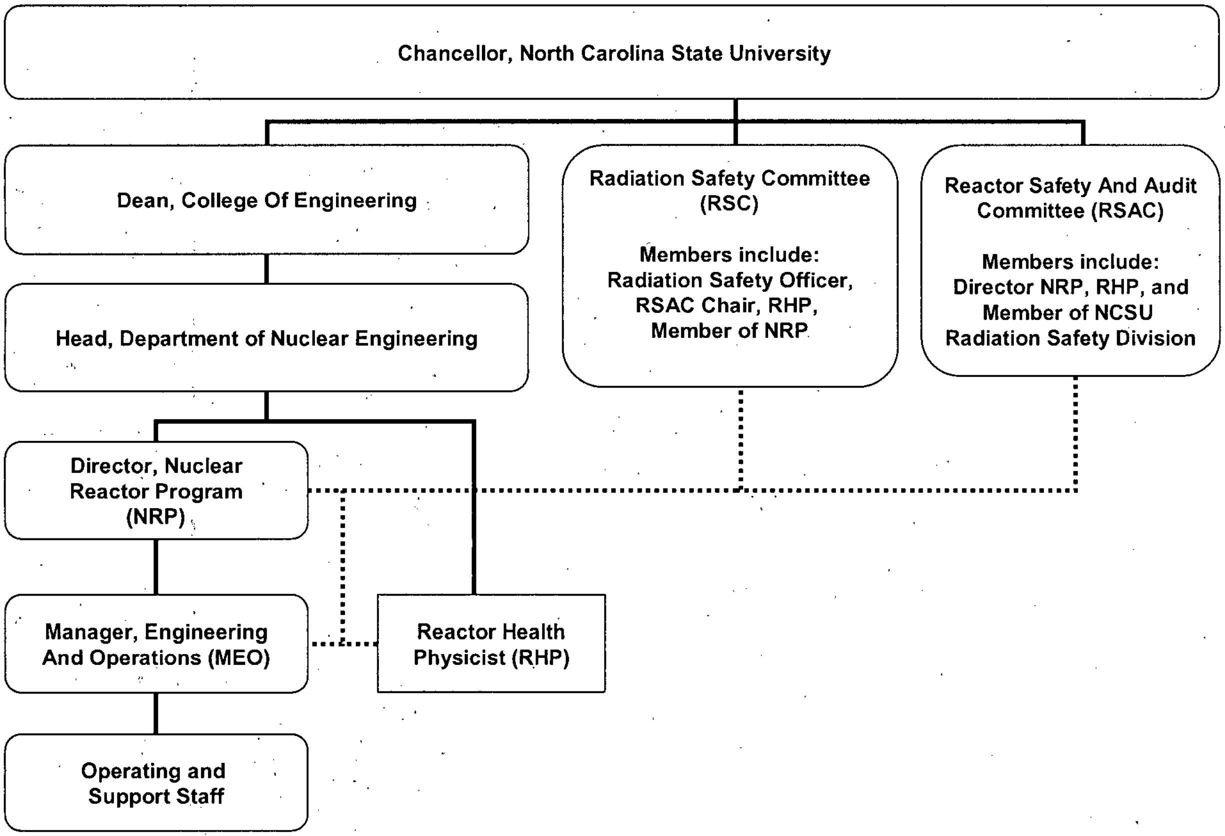
c. Relocation of any in-core experiment with a reactivity worth greater than one dollar (730 pcm).

d. Recovery from unplanned or unscheduled shutdown or significant power reduction (documented verbal concurrence from a licensed Senior Reactor Operator is required).

6.1.5. Selection and Training:

All operators will undergo a selection, training and licensing program prior to unsupervised operation of the PULSTAR reactor. All licensed operators will participate in a requalification program, which will be conducted over a period not to exceed two (2) years. The requalification program will be followed by successive two (2) year programs.

Figure 6.1-1: NCSU PULSTAR Reactor Organizational Chart



NOTES: Line of direct communication ———

Line of advice and liaison – – –

Nuclear Reactor Program (NRP) includes:

- Director, NRP
- Manager, Engineering and Operations
- Operating and Support Staff

Reactor Health Physicist (RHP) reports to the Head, Department of Nuclear Engineering and serves both the NRP and Department of Nuclear Engineering.

Communication on reactor operations, experiments, radiation safety, and regulatory compliance occurs between the NRP, RHP, Reactor Safety and Audit Committee, Radiation Safety Committee, and campus Radiation Safety Division as described in these Technical Specifications and facility procedures.

6.2. Review and Audit

The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at the University are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all experiments involving the potential release of radioactive material conducted at the University and provides oversight of the University Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

RSAC has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, Technical Specifications, and all applicable regulations. RSAC performs an annual audit of the operations and performance of the NRP.

6.2.1. RSC and RSAC Composition and Qualifications:

a. RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety and individuals from the line organization shown in Figure 6.1-1. RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

b. RSAC shall consist of at least five individuals who have expertise in one or more of the component areas of nuclear reactor safety. These include Nuclear Engineering, Nuclear Physics, Health Physics, Electrical Engineering, Chemical Engineering, Material Engineering, Mechanical Engineering, Radiochemistry, and Nuclear Regulatory Affairs.

   At least three of the RSAC members are appointed from the general faculty. The faculty members shall be as follows:

   i. NRP Director

   ii. One member from an appropriate discipline within the College of Engineering

   iii. One member from the general faculty

The remaining RSAC members are as follows:

iv. Reactor Health Physicist (RHP)

v. Member from the campus Radiation Safety Division of the Environmental Health and Safety Center

vi. One additional member from an outside nuclear related establishment may be appointed

At the discretion of RSAC, specialist(s) from other universities and outside establishments may be invited to assist in its appraisals.

The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center are permanent members of RSAC.

6.2.2. RSC and RSAC Rules

a. RSC and RSAC committee member appointments are made by University Management for terms of three (3) years.

b. RSC shall meet as required by the broad scope radioactive materials license issued to the University by the State of North Carolina. RSC may also meet upon call of the committee Chair.

c. RSAC shall each meet at least four (4) times per year, with intervals between meetings not to exceed six months. RSAC may also meet upon call of the committee Chair.

d. A quorum of RSC or RSAC shall consist of a majority of the full committee membership and shall include the committee Chair or a designated alternate for the committee Chair. Members from the line organization shown in Figure 6.1-1 shall not constitute a majority of the RSC or RSAC quorum.

6.2.3. RSC and RSAC Review and Approval Function

a. The following items shall be reviewed and approved by the RSC:

   i. All new experiments or classes of experiments that could result in the release of radioactivity.

   ii. Proposed changes to the facility license or Technical Specifications, excluding safeguards information.

b. The following items shall be reviewed and approved by the RSAC:

   i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and Technical Specification requirements.

   ii. All new procedures and major revisions having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

   iii. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

   iv. Proposed changes to the facility license or Technical Specifications, including safeguards information.

c. The following items shall be reviewed by the RSC and RSAC:

   i. Violations of the facility license or Technical Specifications

   ii. Violations of internal procedures or instructions having safety significance.

   iii. Operating abnormalities having safety significance.

   iv. Reportable Events as defined in Specification 1.2.24.

Distribution of RSC summaries and meeting minutes shall include the RSAC Chair and Director of the Nuclear Reactor Program.

A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to the Dean of the College of Engineering, the Nuclear Engineering Department Head, the Director of the Nuclear Reactor Program, the RSC Chair, Director of Environmental Health and Safety, RSAC Chair, and the Manager of Engineering and Operations prior to the next scheduled RSAC meeting.

6.2.4. RSAC Audit Function

The audit function shall consist of selective, but comprehensive, examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall also be used as appropriate. The RSAC shall be responsible for this audit function. In no case shall an individual immediately responsible for the area perform an audit in that area. This audit shall include:

a. Facility operations for conformance to the facility license and Technical Specifications, annually, but at intervals not to exceed fifteen (15) months.

b. The retraining and requalification program for the operating staff, biennially, but at intervals not to exceed thirty (30) months.

c. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed fifteen (15) months.

d. The Emergency Plan and Emergency Procedures, biennially, but at intervals not to exceed thirty (30) months.

e. Radiation Protection annually, but at intervals not to exceed fifteen (15) months.

Deficiencies uncovered that affect reactor safety shall be immediately reported to the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC.

The annual audit report made by the RSAC, including any recommendations, is provided to the RSC.

6.3. Radiation Safety

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6.1-1.

6.4. Operating Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the following:

a. Startup, operation and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Maintenance of major components of systems that could have an affect on reactor safety.

d. Surveillance checks, calibrations and inspections required by the facility license or Technical Specifications or those that may have an affect on the reactor safety.

e. Personnel radiation protection, consistent with applicable regulations and that include commitment and/or programs to maintain exposures and releases as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of the Emergency Plan and Security Plan.

Substantive changes to the above procedures shall be made effective only after documented review and approval by the RSAC and by the Manager of Engineering and Operations.

Minor modifications to the original procedures which do not change their original intent may be made by the Manager of Engineering and Operations, but the modifications shall be approved by the Director of the Nuclear Reactor Program within fourteen (14) days.

Temporary deviations from procedures may be made by Designed Senior Reactor Operator as defined by Specification 6.1.3.c or the Manager of Engineering and Operations, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Director of the Nuclear Reactor Program.

6.5. Review of Experiments

6.5.1. New (untried) Experiments

All new experiments or class of experiments, referred to as "untried" experiments, shall be reviewed and approved by the RSC, the RSAC, the Director of the Nuclear Reactor Program, Manager of Engineering and Operations, and the Reactor Health Physicist, prior to initiation of the experiment.

The review of new experiments shall be based on the limitations prescribed by the facility license and Technical Specifications and other Nuclear Regulatory Commission regulations, as applicable.

6.5.2. Tried Experiments

All proposed experiments are reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation and/or analysis associated with an existing approved experiment and therefore constitutes an untried experiment that will require the approval process detailed under Specification 6.5.1.

If the Manager of Engineering and Operations and the Reactor Health Physicist concur that the experiment is a tried experiment, then the request may be approved.

Substantive changes to previously approved experiments will require the approval process detailed under Specification 6.5.1.

6.6. Required Actions

6.6.1. Action to be Taken in Case of Safety Limit Violation

In the event a Safety Limit is violated:

a. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The Safety Limit violation shall be promptly reported to the Director of the Nuclear Reactor Program, or his designated alternate.

c. The Safety Limit violation shall be reported to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

d. A Safety Limit violation report shall be prepared that describes the following:

   i. Circumstanced leading to the violation including, when known, the cause and contributing factors.

   ii. Effect of violation upon reactor facility components, systems, or structures and on the health and safety of facility personnel and the public.

   iii. Corrective action(s) to be taken to prevent recurrence.

The report shall be reviewed by the RSC and RSAC and any follow-up report shall be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

6.6.2 Action to be Taken for Reportable Events (other than SL Violation)

In case of a Reportable Event (other than violation of a Safety Limit), as defined by Specification 1.2.24, the following actions shall be taken:

a. Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operation shall not be resumed unless authorized by the Director of the Nuclear Reactor Program, or his designated alternate.

b. The occurrence shall be reported to the Director of the Nuclear Reactor Program, and to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

c. The occurrence shall be reviewed by the RSC and RSAC at their next scheduled meeting.

6.7. Reporting Requirements

6.7.1. Reportable Event

For Reportable Events as defined by Specification 1.2.24, there shall be a report not later than the following work day by telephone to the Nuclear Regulatory Commission Operations Center followed by a written report within fourteen (14) days that describes the circumstances of the event.

6.7.2. Permanent Changes in Facility Organization

Permanent changes in the facility organization involving either Level 1 or 2 personnel (refer to Specification 6.1.1) shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.3. Changes Associated with the Safety Analysis Report

Significant changes in the transient or accident analysis as described in the Safety Analysis Report shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.4. Annual Operating Report

An annual operating report for the previous calendar year is required to be submitted no later than March 31st of the present year to the Nuclear Regulatory Commission Document Control Desk. The annual report shall contain as a minimum, the following information:

a. A brief narrative summary:

   i. Operating experience including a summary of experiments performed.

   ii. Changes in performance characteristics related to reactor safety that occurred during the reporting period.

   iii. Results of surveillance, tests, and inspections.

b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality.

c. The number of emergency shutdowns and unscheduled SCRAMs, including reasons and corrective actions.

d. Discussion of the corrective and preventative maintenance performed during the period, including the effect, if any, on the safety of operation of the reactor.

e. A brief description, including a summary of the analyses and conclusions of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10CFR50.59.

f. A summary of the nature and amount of radioactive effluent 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, including:

Liquid Waste (summarized by quarter)

i. Radioactivity released during the reporting period:

1. Number of batch releases.
2. Total radioactivity released (in microcuries).
3. Total liquid volume required (in liters).
4. Diluent volume required (in liters).
5. Tritium activity released (in microcuries)
6. Total (yearly) tritium released.
7. Total (yearly) activity released.

ii. Identification of fission and activation products:

Whenever the undiluted concentration of radioactivity in the waste tank at the time of release exceeds 2×10^{-5} μCi/ml, as determined by gross beta/gamma count of the dried residue of a one liter sample, a subsequent analysis shall also be performed prior to release for principle gamma emitting radionuclides. An estimate of the quantities present shall be reported for each of the identified nuclides.

iii. Disposition of liquid effluent not releasable to the sanitary sewer system:

Any waste tank containing liquid effluent failing to meet the requirements of 10CFR20, Appendix B, to include the following data:

1. Method of disposal.
2. Total radioactivity in the tank (in microcuries) prior to disposal.
3. Total volume of liquid in tank (in liters).
4. The dried residue of one liter sample shall be analyzed for the principle gamma-emitting radionuclides. The identified isotopic composition with estimated concentrations shall be reported. The tritium content shall be included.

Gaseous Waste

i. Radioactivity discharged during the reporting period (in curies) for:

1. Gases

2. Particulates, with half lives greater than eight days.

ii. The Airborne Effluent Concentration (AEC) used and the estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis. (AEC values are given in 10CFR20, Appendix B, Table 2.)

Solid Waste

i. The total amount of solid waste packaged (in cubic feet).

ii. The total activity involved (in curies).

iii. The dates of shipment and disposition (if shipped off-site).

g. A summary of radiation exposures received by facility personnel and visitors, including pertinent details of significant exposures.

h. A summary of the radiation and contamination surveys performed within the facility and significant results.

i. A description of environmental surveys performed outside the facility.

6.8. Retention of Records

Records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as detailed below. In addition, any additional federal requirement in regards to record retention shall be met.

6.8.1 Records to be retained for a period of at least five (5) years:

a. Normal plant operation and maintenance.
b. Principal maintenance activities.
c. Reportable Events.
d. Equipment and components surveillance activities as detailed in Specification 4.
e. Experiments performed with the reactor.
f. Changes to Operating Procedures.
g. Facility radiation and contamination surveys other than those used in support of personnel radiation monitoring.
h. Audit summaries.
i. RSC and RSAC meeting minutes.

6.8.2 Records to be retained for the life of the facility:

a. Gaseous and liquid radioactive waste released to the environs.
b. Results of off-site environmental monitoring surveys.
c. Radiation exposures for monitored personnel and associated radiation and contamination surveys used in support of personnel radiation monitoring.
d. Fuel inventories and transfers.
e. Drawings of the reactor facility.

6.8.3 Records to be retained for at least one (1) license period of six (6) years:

Records of retraining and requalification of certified operating personnel shall be maintained at all times the individual is employed, or until the certification is renewed.