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

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6 February 2007

US Nuclear Regulatory Commission
Washington, DC

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

Enclosed please find proposed Amendment 17 to our facility license. Changes are being requested to the Technical Specifications for consistency with ANSI/ANS-15.1-1990. The entire document was rewritten and all pages are being reset to Amendment 17.

Major changes are proposed to Specification 3.5 regarding limiting conditions for operation for radiation monitor set points, Specification 3.8 regarding limiting conditions for operation for fueled experiments, and Specifications in Section 6 regarding the line organization and review committees. Other changes are also proposed to Specifications in Sections 1 through 6. A summary and description of and reason(s) for the proposed changes are given on the following pages.

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 6 February 2007.

Ayman Hawari

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

Enclosures: Summary of Changes with Attachment 1
Copy of Emergency Plan Revision 8 Appendix B
Technical Specification Amendment 17

A001

Summary of Changes

Proposed changes to Technical Specifications (TS) in Amendment 17 are made for consistency with ANSI/ANS-15.1-1990.

In addition, changes to Specifications 3.5 and 3.8 regarding radiation monitoring set points and fueled experiments, respectively, are proposed based on 10 CFR 20, the facility radiation protection program, and the facility Emergency Plan.

Also, changes to Specifications in Section 6 regarding the line organization, Radiation Safety Committee (RSC), and Reactor Safety and Audit Committee (RSAC) are proposed for administrative efficiency and consistency with their areas of expertise and responsibility. 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. 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. All items currently listed in TS continue to be reviewed by either RSAC or both RSAC and RSC with this TS amendment.

Other changes are made involving grammar, format, consistency with the Safety Analysis Report, and wording clarification. Because of the number of changes proposed and the number of revisions currently in place, the entire TS document was rewritten with all pages reset to Amendment 17 to avoid confusion.

Description of Changes

Table of Contents

Table of contents was updated with new titles and specification numbers. Titles for the third level of specifications (1.2.3) were deleted.

Figures

Title was changed from "List of Figures" to "Figures". Page numbers on which the figures appear were revised.

Tables

Added table numbers, table titles, and page numbers on which the tables appear.

Section 1

Specification 1.1

The following text was added regarding the purpose statement:

"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."

With the addition of the above text, the remaining text in Specification 1.1 was changed as follows:

"Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. ..." was changed to "Included in this document are the "Bases" for the Technical Specifications. ..." The remaining text in this paragraph is not changed.

Specification 1.2

The following definitions were modified or added:

Channel Calibration: Reworded to be consistent with ANSI/ANS-15.1-1990.

Channel Check: Reworded to be consistent with ANSI/ANS-15.1-1990.

Channel Test: Reworded to be consistent with ANSI/ANS-15.1-1990.

Confinement: Reworded to be consistent with ANSI/ANS-15.1-1990.

Excess Reactivity: Reworded to be consistent with ANSI/ANS-15.1-1990.

Experiment: Reworded to be consistent with ANSI/ANS-15.1-1990.

Tried Experiment: Reworded to be consistent with ANSI/ANS-15.1-1990.

Limiting Conditions for Operations: This definition was added.

Limiting Safety System Setting: This definition was added.

Operating: Reworded to be consistent with ANSI/ANS-15.1-1990.

Reactor Building: Added Ventilation Room to description. The space above the Control Room was modified by an approved design change to house ventilation equipment. Previously, access between the two spaces was achieved by a ladder and an access panel. Now, a secured door on the 3rd floor of the administrative building (Burlington) is used for access and access from the Control Room has been removed.

Reactor Operator: Reworded to be consistent with 10 CFR 50.55.

Reactor Secured: The use of the unit Δk/k was deleted. For consistency and to limit confusion, "pcm" will be the only unit used for reactivity.

Reportable Event: The use of the unit Δk/k was deleted. For consistency and to limit confusion, "pcm" will be the only unit used for reactivity.

Safety Limit: This definition was added.

Senior Reactor Operator: Reworded to be consistent with 10 CFR 50.55.

Unscheduled Shutdown: Reworded to be consistent with ANSI/ANS-15.1-1990.

Section 2

Minor formatting changes were made. Mainly, the ordering of the interrelated variables associated with core thermal and hydraulic performance in Specification 2.1.1 and 2.1.2 was changed. These variables are now ordered consistently throughout the specifications.

Section 3

Specification 3.1.e

The use of the unit Δk/k was deleted. For consistency and to limit confusion, "pcm" will be the only unit used for reactivity.

Specification 3.2.a,b,d,e and f

The use of the unit Δk/k was deleted. For consistency and to limit confusion, "pcm" will be the only unit used for reactivity.

Specification 3.2.e

Requirements are now given in Table 3.2-1. Secured experiment reactivity was reduced from 1600 pcm to 1590 pcm to be consistent with the Section 13.2.2.1 of the Safety Analysis Report.

Specification 3.2.f

The sum of the reactivity of all experiments was reduced from 2900 pcm to 2890 pcm to be consistent with the Section 13.2.2.1 of the Safety Analysis Report.

Specifications 3.3

Requirements are now given in Table 3.3-1.

Specification 3.3.g Pool Water Temperature Monitoring Switch

The requirement for the operator to manually SCRAM was removed. The requirement for the Alarm will remain, but the appropriate operator response for this alarm will be controlled through procedures relating to responses to alarms and abnormal conditions.

Specification 3.3.1 Over-the-Pool Radiation Monitor

The requirement for the operator to manually SCRAM was removed. The requirement for the Alarm will remain, but the appropriate operator response for this alarm will be controlled through procedures relating to responses to alarms and abnormal conditions.

Specification 3.4.a

Footnote was moved into the wording of the specification.

Specification 3.5

Requirements are now given in Table 3.5-1. Area radiation monitors set points are based on 10 CFR 20 dose limits. 2 mR/h is used as the instantaneous exposure rate for members of the public and non-occupational personnel. 5 mR/h is the exposure rate based on the definition of radiation area. 100 mR/h is the exposure rate based on the definition for high radiation area. Area radiation monitors provide exposure rate readings which are taken to be equivalent to dose-equivalent rates, i.e. for gamma radiation mR/h is equivalent to mrem/h.

Changes to effluent monitor set points are based on 10 CFR 20, the facility radiation protection program, and the facility Emergency Plan as detailed in Attachment 1.

Specifications 3.6.a

Requirements are now given in Table 3.6-1.

Specification 3.7.e.(iii)

Text was changed from "Experiments reviewed by the Radiation Safety Committee in which material is considered to be potentially..." to "Experiments in which the material is considered to be potentially..."

Specification 3.8

Changes to this specification were based on 10 CFR 20, the facility radiation protection program, and the facility Emergency Plan. Similar statements to those being requested are given in Section 3.8 of ANSI/ANS-15.1-1990. The changes repeat and are consistent with Specifications 3.2 regarding reactivity, 3.7 regarding limitations on experiments, and 6.5 regarding review of experiments.

The previous Specification 3.8 limited fueled experiments to 400 mg of U-235 and placed restrictions on fluence and fission rate within the sample. The previous TS 3.8 bases indicated that the exposure to releases of radioactive material from fueled experiments was bounded by the FSAR analysis for a fuel handling accident, which assumes a failure of three PULSTAR reactor fuel pins.

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. Limiting the off-site exposure or dose from the release of radioactive materials that may potentially occur from failure of a fueled experiment to that evaluated in the FSAR for clad defects in three fuel pins assumed in the fuel handling accident is very restrictive. The maximum off-site dose given in the FSAR for the fuel handling accident is well below 1 mrem. The purpose of the FSAR analysis for the fuel handling accident was to ensure that potential dose was well within 10 CFR 20 limits and therefore the possibility of exceeding 10 CFR 20 limits is unlikely.

The changes requested simply state that credible failures of fueled experiments are not allowed to exceed 10 CFR 20 limits. This statement is consistent with the intent of the FSAR analysis, ANSI/ANS-15.1-1990 Section 3.8, and Specification 3.7. 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).

Based on these variables, specific restrictions on time, mass, and sample fission rate and fluence (or exposure) are likely to vary dramatically and therefore those specifications have been revised and Figure 3.8-1 has been deleted. The proposed specification states that limitations on several conditions, including time, mass, and fission rate, will be documented, reviewed, and approved for each type of fueled experiment.

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 is limited as defined in 10 CFR 20.

Furthermore, the limitations given in Specification 3.8 ensure that each type of fueled experiment is reviewed, approved, and documented as required in Specification 6.5. This review addresses applicable limitations given in Specifications 3.2 and 3.7. Limiting the amount of fissile material ensures that experimental reactivity conditions are met and that radiation doses are within 10 CFR 20 limits following maximum fission product release from a failed experiment. 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. This review process is similar to that used to meet Specification 3.7 for non-fueled experiments.

The bases for Specification 3.8 were rewritten.

Figure 3.8-1

Figure 3.8-1 was deleted.

Section 4

Specification 4.3.b

Text was reworded to from "A channel test of each channel in the RSS shall be performed prior to each day's operation, or prior to …" to "A channel test of each channel in the RSS shall be performed prior to operation each day, or prior to ..."

Specification 4.3.d.(i) and (iv) (previously 4.3.d.1 and 4)

The references to T_2, T_5 and T_6 were deleted. Deleting the number associated with the temperature sensor (i.e. T_2, T_5, T_6) allows renumbering of the sensors at a later date without seeking a license amendment.

Specification 4.6.b

Text was changed to from "The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed 18 days. This analysis shall include gamma spectroscopy of a liquid sample." to "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."

Gross beta analysis of dried residue of a 1 liter sample or gamma spectroscopy for analyses of secondary coolant was added. Analyses of primary and secondary coolant are now consistent. Sensitivity to environmental radioactivity limits is capable using common gross beta radiation counting equipment.

Section 5

Specification 5.2.d, deleted PULSTAR, added "Ventilation Room".

Section 6

Specification 6.1

Some text was moved to specification 6.1.2 to be consistent with ANSI/ANS-15.1-1990.

Figure 6.1-1

Figure 6.1-1 was changed to reflect the new organizational structure.

Specification 6.1.1

This specification was significantly modified to meet the needs of the NRP and to be consistent with ANSI/ANS-15.1-1990.

Level One personnel now include the Chancellor, The Dean of the College of Engineering, and the Head of the Department of Nuclear Engineering.

Level Two personnel now include the Director of the Nuclear Reactor Program.

Level Three personnel now include the Manager of Engineering and Operations (MEO), the position created from the combination of the Associate Director and Reactor Operations Manager positions. The MEO position qualification requirements meet or exceed those listed in ANSI/ANS-15.4-1988.

Level Four personnel now include all operating and support staff. Qualification for licensed operators was added.

Reactor Health Physicist position description and qualifications were changed to meet those given in ANSI/ANS-15.1-1990. Regulatory compliance monitoring function was added.

Specification 6.1.2

This specification was created to be consistent with ANSI/ANS-15.1-1990 section 6.1.2 and contains text that was previously given in Specification 6.1.

Specification 6.1.3.a (previously 6.1.2.a)

The terminology of "certified reactor operator" was deleted to be consistent with 10 CFR 50.55.

Specification 6.1.3.c (previously 6.1.2.c)

Specification was reworded to remove ambiguity and confusion.

Specification 6.2 (previously 6.2.1)

The wording was changed based on the primary responsibilities of the review committees. With this change, RSC reviews only 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.

Specification 6.2.1 a (previously 6.2.2 a)

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 appointment information is now given in Specification 6.2.2.

Specification 6.2.1 b (previously 6.2.2 b)

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. Membership appointments to the RSAC have been listed separately for clarification. RSAC appointment information is now given in Specification 6.2.2.

Specification 6.2.2 a

RSC and RSAC member appointments were separated in this specification and contains text previously given in specifications 6.2.2 a and 6.2.2 b to be consistent with ANSI/ANS-15.1-1990. "University management" was listed as making committee term appointments rather than a specific office within the University.

Specification 6.2.2 b (previously 6.2.2 d)

The proposed TS change states that the RSC meeting frequency is specified in the broad scope radioactive materials license issued by the State of North Carolina for the University and that additional meetings may be called by the RSC Chair.

Specification 6.2.2 c (previously 6.2.2 d)

No changes to the RSAC meeting frequency are proposed.

Specification 6.2.2 d (previously 6.2.2 c)

Committee rules regarding RSC and RSAC continue to ensure that members from the NRP line organization do not constitute a majority of the quorum.

Specification 6.2.3

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 with the exception of 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. RSAC members will undergo fingerprinting and criminal history checks as required for having access to SGI. Furthermore, RSC has the primary mission of radiation safety for the campus rather than physical security at the reactor and therefore does not have a need-to-know SGI. RSAC on the other hand, has the mission of verifying compliance with license conditions at the reactor and therefore has a need-to-know SGI.

Distribution of RSC summaries and meeting minutes was changed to include the RSAC Chair and NRP Director rather than listing all recipients since RSC has approval authority for reactor experiments and licensing documents. The RSAC Chair and NRP Director are the appropriate individuals who need to be informed of RSC actions affecting the reactor facility.

The Associate Director was changed to the Manager of Engineering and Operations for receiving a summary of RSAC meeting minutes.

Wording referencing the summary of the annual audit was moved to Specification 6.2.4.

<u>Specification 6.2.3.a</u>

The specification was rewritten to require the RSC to review and approve experiments that could result in the release of radioactivity and changes to the license and technical specifications, excluding SGI. This change is consistent with the expertise and purpose of the RSC.

<u>Specification 6.2.3.b</u>

The specification was rewritten to require the RSAC to review and approve changes made to equipment, systems, tests and experiments (all experiments including those that could result in the release of radioactivity or that could affect reactivity), and procedures to verify that license requirements are met. Review of SGI is performed only by RSAC. This change is consistent with the expertise of the RSAC members and agrees with the 10CFR50.59 changes that became effective in March 2000.

<u>Specification 6.2.3.c</u>

The specification was rewritten to require a review of items associated with the facility license by both RSC and RSAC.

<u>Specification 6.2.4</u>

The wording "under the authority of the RSC" was deleted in reference to the RSAC Audit. The RSAC is solely responsible for performing the audit. RSC continues to receive the audit report.

The Associate Director was deleted from receiving reports of deficiencies uncovered by the audit. The NRP Director may notify members within the NRP.

The following text was added: "In no case shall an individual immediately responsible for an area perform an audit in that area." This basic audit practice has always been observed, but is being specifically listed to be consistent with ANSI/ANS-15.1-1990.

Wording referencing the summary of the annual audit was moved from Specification 6.2.3 to Specification 6.2.4.

<u>Specification 6.2.4.e</u>

A frequency of annually, not to exceed fifteen (15) months was added to the radiation protection section of the audit. This is consistent with ANSI/ANS-15.11-1993 and 10 CFR 20.1101. The RSAC audit in this area may be used to partially fulfill the 10 CFR 20 requirement for an annual review of the content and implementation of the radiation protection program. The annual review required by 10 CFR 20 is performed by the Reactor Health Physicist using internal procedures and includes applicable audits and inspections performed by other organizations.

Specification 6.3

This section was created to be consistent with ANSI/ANS-15.1-1990.

Specification 6.4 (previously 6.3)

The following changes were made to be consistent with the restructuring of the NRP:

Approval for substantive changes was changed from RSC and the former Associate Director position to RSAC and Manager of Engineering and Operations.

Making minor changes to procedures was changed from Reactor Operations Manager to Manager of Engineering and Operations and approval within 14 days was changed from the Associate Director to the Director.

For temporary deviations, Reactor Operations Manager was replaced with Manager of Engineering and Operations and such deviations are reported now to the Director.

Specification 6.5.1 (previously 6.4.1)

Untried Experiments are now approved by the RSC, RSAC, the Manager of Engineering and Operations, and the Reactor Health Physicist.

Wording previously given in Specification 6.4.1 regarding issuance of an authorized experiment by the RSC was deleted since the requirement for review and approval of experiments is provided in Specification 6.2.3.

Specification 6.5.2 (previously 6.4.2)

All proposed experiments are now reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist.

Wording previously given in Specification 6.4.2 regarding scheduling of an experiment request was deleted since scheduling is a time management issue rather than a licensing or safety issue.

Specification 6.6.1 (previously 6.5)

Specification was reworded to require Safety Limit violations to be reported to the Director of the Nuclear Reactor Program instead of the former Associate Director position.

The report of the event is reviewed by both the RSC and RSAC because licensing issues are involved.

Minor grammar change was made to 6.6.1 d. iii.

Specification 6.6.2 (previously 6.6)

Specification was reworded to require the Director, rather than the former Associate Director position, take action regarding reportable events other than SL violations. The occurrence is reviewed by both the RSC and RSAC because compliance issues are involved.

Specification 6.7

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.

Specification 6.8.

Specification 6.8 subsections were renumbered for consistent format with other TS.

Specification 6.8.1.d (formerly 6.8.a.iv.) was changed: The phrase "as detailed in Specification 4" was added to be consistent with ANSI/ANS-15.1-1990.

Specification 6.8.1.g was added: "Facility radiation and contamination surveys" was moved from lifetime records to 5 year records to be consistent with ANSI/ANS-15.1-1990. Surveys used to document personnel exposures are retained for the life of the facility under the newly worded Specification 6.8.2.c.

Specifications 6.8.1.h and i were renumbered.

Specification 6.8 2.c (formerly 6.8.b.iii.), Records to be retained for the life of the facility was changed: The phrases "for monitored personnel" and "and associated radiation and contamination surveys" were added to agree with ANSI/ANS-15.1-1990 and 10 CFR 20.2103.

Specification 6.8.c was renumbered to 6.8.3.

Conclusion

The proposed changes are justified based on the following reasons:

1. Conformity with ANSI/ANS-15.1-1990, which is the applicable standard for TS at research reactors.
2. Radiation monitoring set points changes are based on requirements given in 10 CFR 20, Emergency Plan Emergency Action Levels, and monitoring of fueled experiments.
3. Specifications for fueled experiments were changed based on meeting 10 CFR 20 requirements and license conditions. Limitations on fueled experiments are determined, similarly to those determined for non-fueled experiments, by the reactor staff, Reactor Health Physicist, and review committees for each type of fueled experiment.

4. Regarding the RSC and the RSAC, RSAC serves as the review and audit group assigned to the reactor facility while RSC serves in an oversight capacity for the entire university. Therefore, RSAC is the primary review and audit group that fulfills the requirements of ANSI/ANS-15.1-1990 for the reactor facility. This is evident by the phrases given in the previous specifications 6.2.3, 6.3, and 6.4 "by referral to the RSAC" and "or RSAC as applicable". The RSC reviews and approves experiments using radiation and radioactive materials conducted anywhere at the university, including the reactor, for consistency and compliance with applicable regulations and license conditions. The RSC has limited expertise in nuclear research reactor operations and relies on the RSAC. Therefore, the RSAC and the RSC have a relationship that must and continues to be described in the Technical Specifications. The specifications regarding the RSC were reworded to ensure an adequate review of reactor related items within their area of expertise and responsibility. RSAC membership and function has not been changed, but has been reorganized and clarified, by this TS amendment.

5. Other changes are associated with grammar, format, consistency with statements made in the Safety Analysis Report, and clarification of wording.

In conclusion, the proposed changes to the Technical Specifications improve conformity with the applicable standard, provide organizational and administrative efficiency, and make necessary editorial corrections.

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:

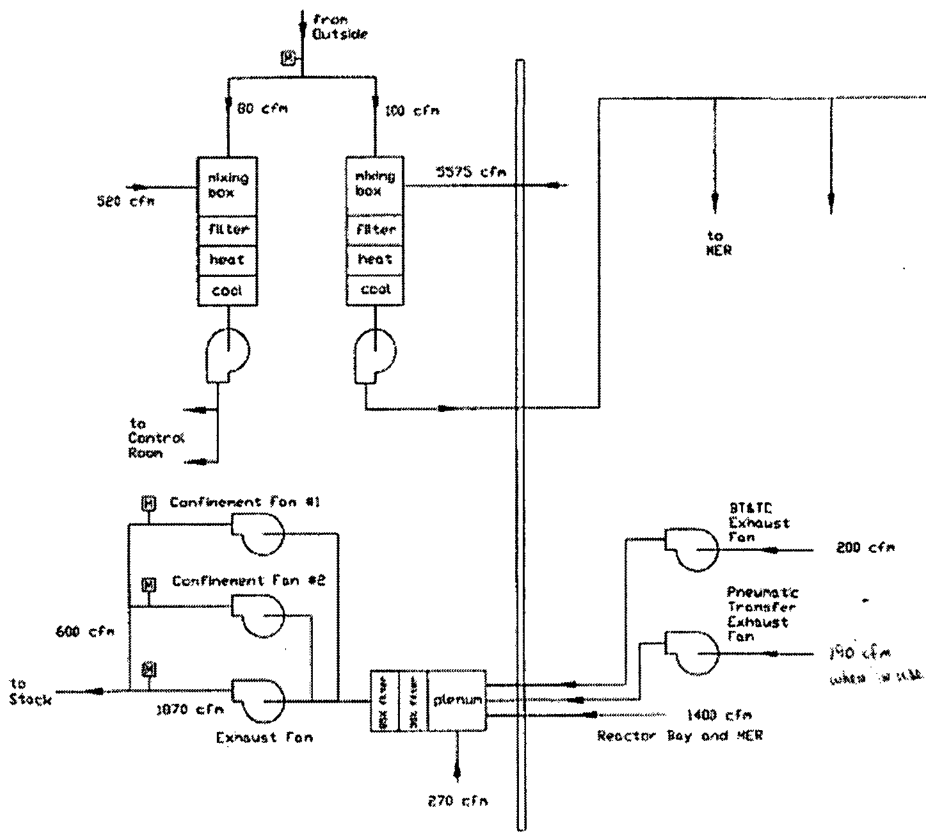
Total normal exhaust =	10,050 cfm
PN fan (intermittent use) =	190 cfm
Total normal exhaust with PN use =	10,240 cfm

New normal ventilation system data:

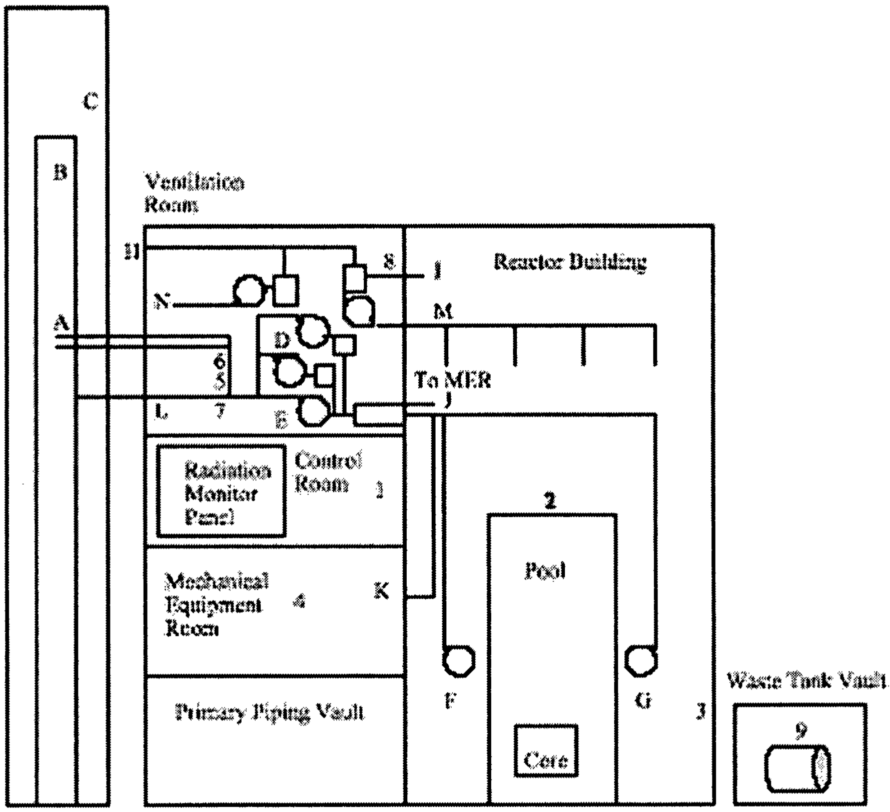
Total normal exhaust =	1870 cfm
PN fan (intermittent use) =	190 cfm
Total normal exhaust with PN use =	2060 cfm

10,240 cfm / 2060 cfm = 5

Ventilation System Diagram:



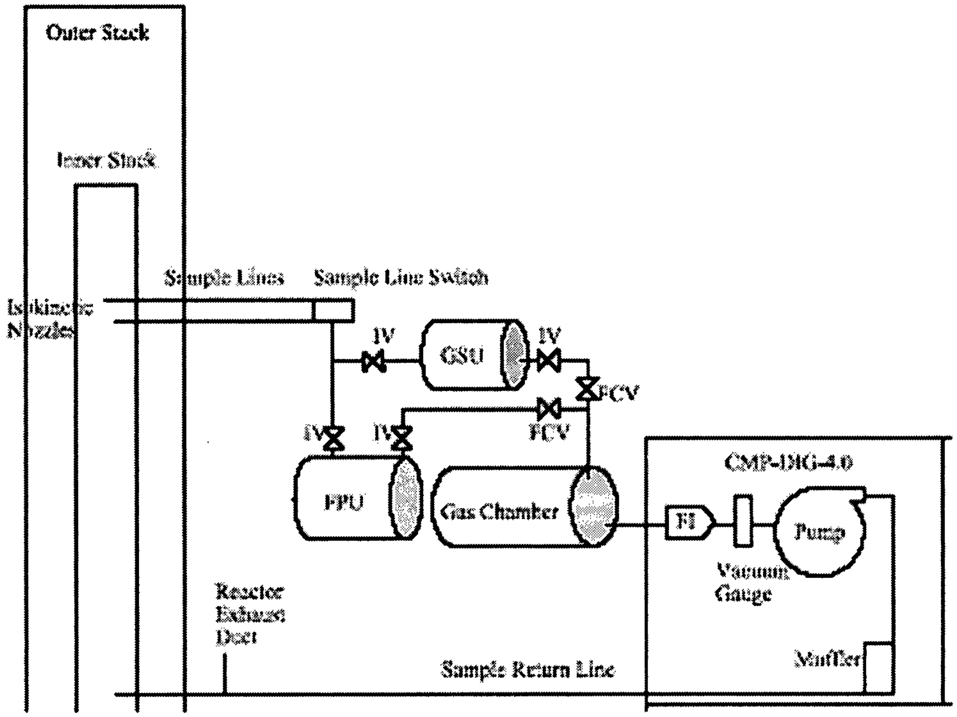
VENTILATION AND RADIATION MONITORING SYSTEM DIAGRAM



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	PN 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 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 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.

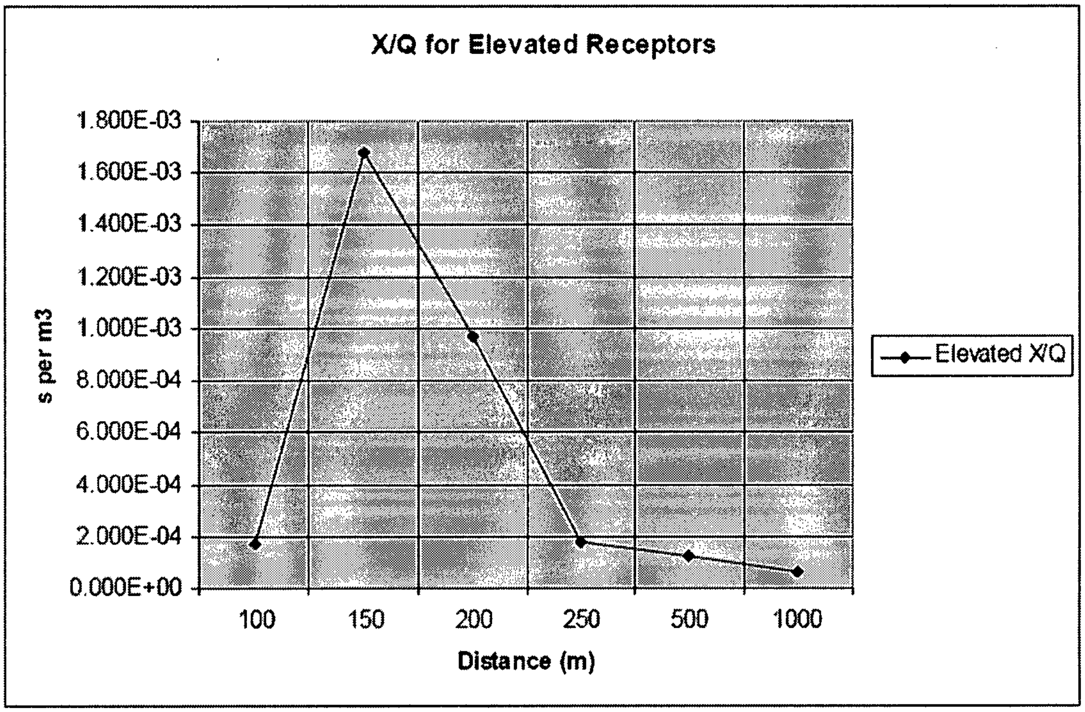
3. The maximum average X/Q of occurs at distance of 150 m and a height of 30 m, giving a minimum ADF in excess of 5 E2.

NOTES:

Elevated receptors within a distance of 150 m are at a height of ≤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 ≤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



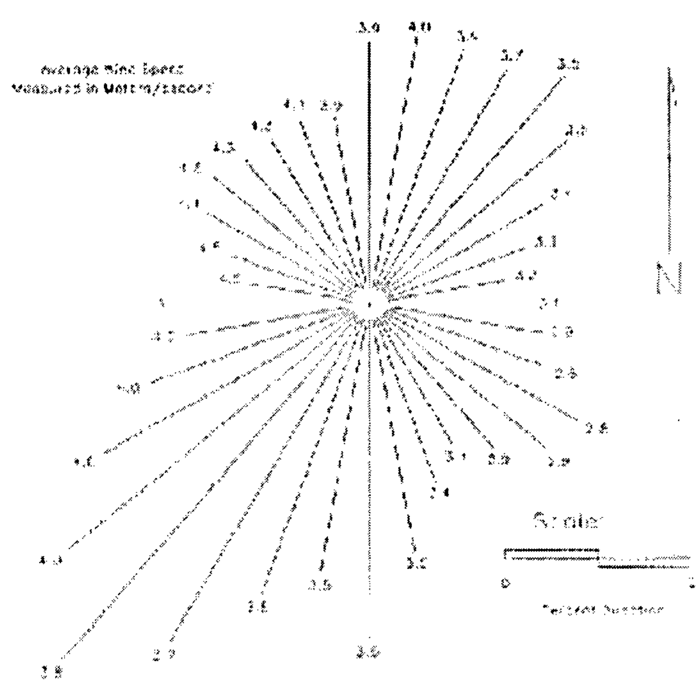
d. Weather data, such as wind speed and direction, at the RDU airport is applicable to NCSU and is available in Section 2 of the FSAR and data from the COMPLY code. Both references give a maximum wind frequency for 16 sectors at 0.1.

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 E3 (i.e. 1/(0.1*1.7E-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 rem per year.

Based on item (2) in the previous paragraph, the AEC for all airborne effluent is limited to 5 E3 AEC for the PULSTAR nuclear reactor for distances at or beyond 100 m from the reactor stack. 5000 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 mph**	**H_{shine} Confinement Ventilation rem/h/ per µCi/ml at 1 mph**	**H_{shine} Confinement Ventilation with Dilution rem/h per µCi/ml at 1 mph**
Ar-41	0 m	8.4 E-2	2.8 E-2	2.2 E-3
Ar-41	20 m	2.2 E-1	7.1 E-2	4.5 E-3
Fuel Failure	0 m	1.6 E-2	5.1 E-3	4.1 E-4
Fuel Failure	20 m	4.2 E-2	1.4 E-2	8.4 E-4

The maximum value is 2.2 E-1 rem/h per uCi/ml at a wind speed of 1 mph for Ar-41.

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

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

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

$$5\text{ E3 AEC} = (100\text{ mrem per y}) / \left(\frac{2\text{ E-2 mrem per y}}{\text{AEC}}\right)$$

EAL Based Set Points

Emergency classification for airborne effluent was analyzed as a result of the ventilation system changes in Appendix B of Revision 8 to the facility Emergency Plan. From that analysis it is concluded that:

a. Emergency classification would be necessary at 2500 AEC for noble gases and 5000 AEC for radionuclides other than noble gases

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

For distances at or beyond 100 m from the reactor stack, 2500 AEC and 5000 AEC may therefore be used as the basis for an alarm set point for noble gases and radionuclides other than noble gases, respectively. Again, 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. Thus, exceeding the EAL for a continuous 24 hour period is not likely if the alarm is set at a level associated with the airborne effluent concentration based EAL for the location with the lowest ADF.

Conclusions

Based on the calculations and discussion given above and the Emergency Plan and FSAR, the following conclusions are made regarding effluent monitor set points:

- The stack gas radiation monitor alarm is based on the ADF value of ≤ 2500 AEC associated with EAL criteria for noble gases

    - External public dose considerations for locations within 100 m of the reactor stack limit noble gas effluent to ≤ 5000 AEC

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

- The stack particulate radiation monitor alarm is based on the ADF value of ≤ 5000 AEC associated with either:

    - EAL criteria for radionuclides other than noble gases, OR

    - Limiting the annual average concentration of particulate airborne effluent to that given in Table 2 of 10 CFR 20 Appendix B

- 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.

NOTE: 2500 AEC for noble gases and 5000 AEC for radionuclides other than noble gases result in the same dose to members of the public. Dose from noble gas exposure is external, so all age groups are equally exposed. Dose from radionuclides other than noble gases is based on inhalation and metabolism, which has an associated factor of 2 for other age groups (i.e. children) vs. adults. Annual dose from AEC for radionuclides other than noble gases is based on 50 mrem for adults, or 100 mrem for other age groups.

APPENDIX B

EFFLUENT RADIONUCLIDE MEASUREMENTS

This appendix 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 Attachments B-1 and B-2, respectively. 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.

$$\textbf{SDF} = [12{,}500 + 600]\ \textbf{cfm} / 600\ \textbf{cfm} \approx 20$$

$$\textbf{SDF} \equiv 1\ \textbf{for all other ventilation modes}$$

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: $A(\infty) = \sigma\phi N \times 1$ uCi / 3.7E4 dps = 4.2 E4 uCi of Ar-41

where, σ is 0.5 E-24 cm^2, ϕ is 1 E13 $cm^{-2}s^{-1}$, N is number of Ar-40 atoms
PN tube volume = $\pi(2.54\ cm)^2(61 cm)$ = 1236 ml
N = (1236 ml)(1.2 E-3 g/ml)(0.01g Ar/g of air)(0.996 Ar-40)(6.022 E23 at/mol)/28.964 g/mol
N = 3.1 E20 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 hour average [C] = {(33uCi/ml)(0.015s)+(8E-6 uCi/ml)(24 h)(3600 s/h)}/{(24h)(3600 s/h)
= **1.4 E-5 uCi/ml**

where, 8 E-6 uCi/ml is based on 830 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
720 cpm = 130 cpm (historical cpm value at 0.9 MW for current ventilation system)
x {(10,050 cfm + 190 cfm) / (1870 cfm + 190 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 600 cpm or ~ 7 E-6 uCi/ml is present at the detector and 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;

4.2 E4 uCi / 2.25 E9 ml = 1.9 E-5 uCi/ml in reactor bay from PN blower rupture
8.0 E-6 uCi/ml from PN use

24 hour average [C] = 1.9 E-5 uCi/ml + 8 E-6 uCi/ml = **2.7 E-5 uCi/ml**

(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. (If in confinement during BT opening, the 24 hour average is the same as for normal ventilation.);

Saturation activity of Ar-41 in BT: $A(\infty) = \sigma\phi N$ x 1 uCi / 3.7E4 dps = 5.8 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
BT tube volume = 1.7 E5 ml (6 cubic feet)
N = (1.7E5ml)(1.2 E-3g/ml)(0.01gAr/g of air)(0.996 Ar-40)(6.022 E23at/mol)
÷ 28.964 g/mol
N = 4.2 E21 atoms of Ar-40
Reactor bay [C] = 5.8 E5 uCi/ 2.25E9 ml = 2.6 E-4 uCi/ml
BT exhaust time = (6 cubic feet / 7445 cfm) (60 s/m) = 0.05 s

24 hour average [C] ={(2.6E-4 uCi/ml)(0.05 s)+(8E-6 uCi/ml)(24h)(3600s/h)}/{(24h)(3600 s/h)}
= **8.0 E-6 uCi/ml**

The second 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.

<u>Determination of AEC Fraction</u>

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:

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

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	9.3 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 9.3 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).

Alternately 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	2.7 E-5	2.7 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

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:

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

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:

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

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-2 s/m^3 at a wind speed of 1 mph at a distance of ~ 140 m (refer to Attachment B-3).

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

$$AECFraction = 7.1E-4 \bullet \Sigma_j \Sigma_i [\frac{C_i}{AEC_i}]_j \bullet [\frac{T}{SDF}]_j$$

where, (1/24)(1.7E-2) = 7.1 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	46	46	2.3
Fuel Failure	13	4.8 E-1	2.4 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 with clean air.

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.

<u>Release Rate Determination</u>

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	2.4 E-05	7.6 E-06	3.8 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

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

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

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

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

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

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: (See Attachment B-3)

- 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 ith radionuclide in μCi/ml

DCF is dose rate conversion constant in rem/h per μCi/ml for the ith radionuclide

F is stack exhaust flow rate in cfm

U is wind speed in mph

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 ith radionuclide

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

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
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. These locations are illustrated in Attachment B-4. 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. Reference source terms (stack [C], Q, and line activities) and H_{stack} and H_{line} results are given in Attachment B-4. 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 1 mph: (Refer to Attachment B-4 or B-5 for data)

RADIO-NUCLIDE	ELEVATION	H_{shine} Normal Ventilation rem/h per μCi/ml at 1 mph	H_{shine} Confinement Ventilation rem/h/ per μCi/ml at 1 mph	H_{shine} Confinement Ventilation with Dilution rem/h per μCi/ml at 1 mph
Ar-41	0 m	8.4 E-2	2.8 E-2	2.2 E-3
Ar-41	20 m	2.2 E-1	7.1 E-2	4.5 E-3
Fuel Failure	0 m	1.6 E-2	5.1 E-3	4.1 E-4
Fuel Failure	20 m	4.2 E-2	1.4 E-2	8.4 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. Upon detection of this concentration, the confinement system is automatically initiated. 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 2.7 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.

$$\mathbf{H_{ede} = H_{xyz} \bullet T}$$ for distances ≥ 100 m

$$\mathbf{H_{ode} = H_{thy} \bullet T}$$ for distances ≥ 100 m

$$\mathbf{H_{ede} = H_{ode} = H_{shine} \bullet T}$$ for distances < 100 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. Worse case χ/Q values for locations of interest were determined from Attachment B-3. Reference calculation results are given in Attachment B-5.

Dose projections for Ar-41 releases and the fuel handling accident described in the FSAR were calculated using Attachment B-5. H_{xyz} calculation results for typical annual Ar-41 releases and the FSAR fuel handling accident are given in Attachment B-6. All projected doses are well below PAG values and 10 CFR 20 limits. Doses for multiple locations are given in Attachment B-6. 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)

Comparison of calculation results for the fuel failure accident given in Attachment B-6 and Section 13 of the FSAR was made. Most results agree within a factor of 10 and the maximum projected effective dose is on the order of a few mrem.

Waste Liquid Effluent

Three 904 gallon liquid waste holding tanks are used at the PULSTAR Nuclear Reactor and associated BEL laboratories. All waste water from these facilities is collected in sumps located in the Reactor Building. The Mechanical Equipment Room and Primary Piping Vault sump pumps pump water to the waste tanks. A common surge line is used to prevent overfilling of individual waste tanks. Sump pumps may be disabled to prevent overfilling of the waste tanks as necessary. The waste tank vault is monitored by a gamma sensitive detector and digital ratemeter which provides indication and alarms in the Control Room. The waste tank radiation monitor may be used as an area radiation monitor using a conversion factor determined from the most recent calibration to convert cpm to mR/h. Tank level indication may be measured. Waste water in the each tank is mixed, sampled, and analyzed for tritium, gross beta-gamma activity, and gamma emitting radioisotopes using calibrated laboratory counting systems. Liquid discharges are made to the sanitary sewer in accordance with 10 CFR 20, State of North Carolina regulations, and City of Raleigh ordinances using approved PULSTAR Health Physics procedures.

Conclusion

The calculation method used in this appendix is considered to be suitable for emergency dose projects since it is in fair agreement with the FSAR, includes the dose quantities required by the State of NC and 10 CFR 20, and accounts for the various ventilation modes possible at the PULSTAR nuclear reactor.

Sample Calculation for Demonstration Purposes Only

Conditions and data:

Fuel failure is detected
Normal Ventilation, F = 1870 cfm
Stack Sample Flow = 2 cfm, Stack Sample Time = 24 hours or 1440 minutes
Duration is 4 hours
Windspeed is 5 mph
Radiation Monitor Readings Channel 5 = 100 net cpm
Channel 6 = 1000 net cpm

Concentration at Stack Monitor, $[C]_i$:

Noble Gas:	100 / 5.0 E 6	=	2.0 E-5 μCi/ml
Particulate:	1000/(8.2E9*2*1440)	=	4.3 E-11 μCi/ml
Iodines:	(3E-2)(2.0 E-5)	=	6.0 E-7 μCi/ml
Bromines:	(3E-2)(2.0 E-5)	=	6.0 E-7 μCi/ml
TOTAL [C]		=	2.1 E-5 μCi/ml

Release Rate, Q^{total} = (2.1E-5)(1870)(4.72E-4) = 1.9 E-5 Ci/s

AEC fraction at the exhaust stack:

= [2.0E-5/5E-7 + 4.3 E-11/6E-12 + 6E-7/6E-10+ 6E-7/9E-8]•(4)/(24)
= [40 + 7.2 + 1000 + 7]•(4)/24
= 1054 * 4/24 = 176

NOTE: AEC fraction at receptor location is at least 50 times lower, or 3.5

Dose at location of interest at elevation of 0 m and distance of < 100 m:

Using Attachment B-5 provides data at 1 mph, which is conservative;

(0.016 rem/h per uCi/ml at 1 mph)(2.1 E-5 uCi/ml)(4 h) = 1.3 E-6 rem or 1.3 E-3 mrem

Using Attachment B-4 to adjust for wind speed of 5 mph provides a better dose estimate;

H_{stack} = (4.0E-5)(4.05)(2.0E-5 uCi/ml) = 3.2 E-9 rem/h,
where 4.0 E-5 is taken from Attachment B-4
H_{line} =(7.8E-5)(224)(1.9E-5 Ci/s) = 3.3E-7 rem/h,
where 7.8 E-5 is taken from Attachment B-4
H_{shine} = H_{ext} = 3.2 E-9 + (3.3 E-7 / 5) = 6.9 E-8 rem/h
H_{ede} = H_{ode} = (6.9 E-8 rem/h)(4h) = 2.8 E-7 rem

Dose at location of interest at elevation of 30 m and distance of 200 m:

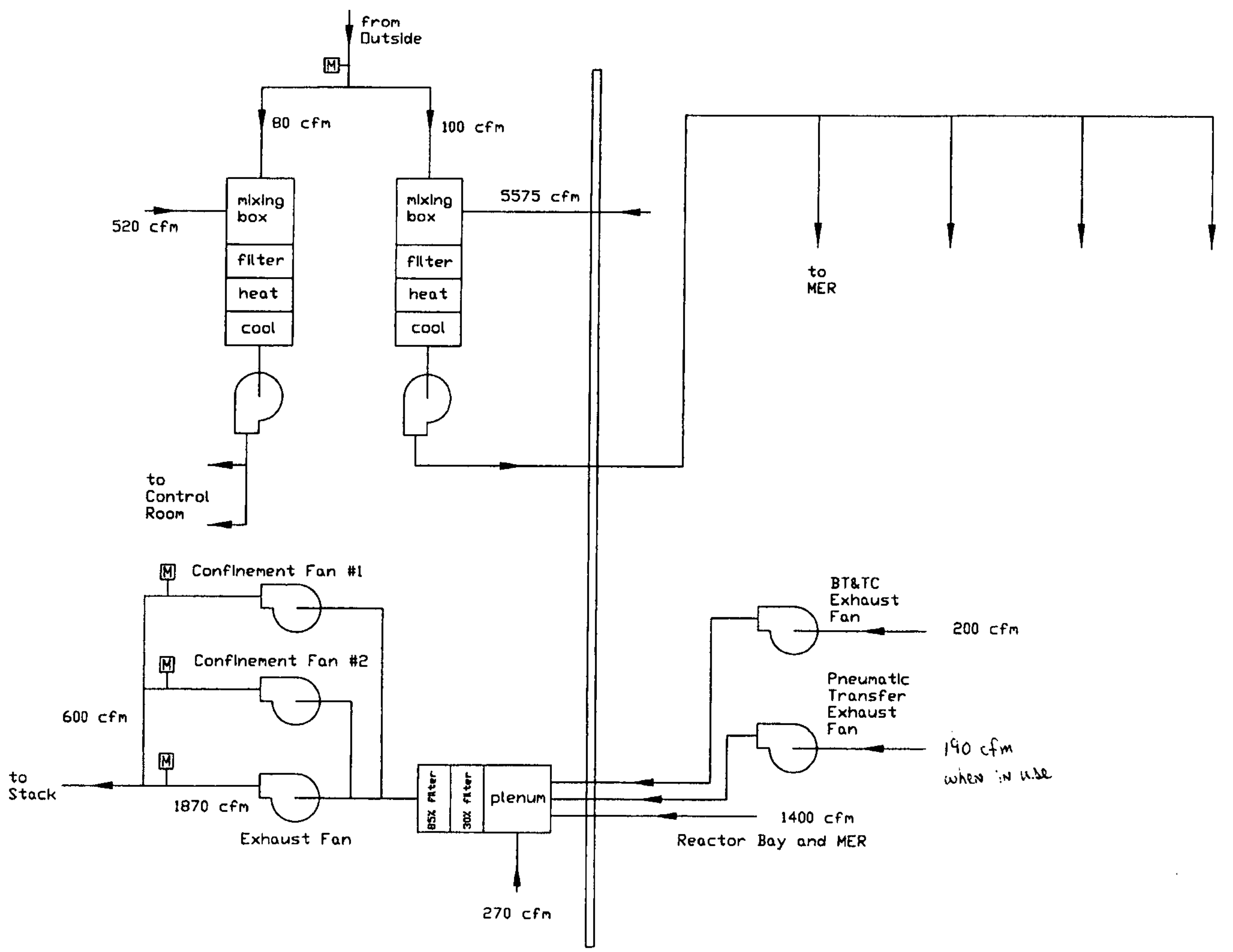
H_{xyz} ={(2.0E-5)(1.1)+(4.3E-11)(1.4 E4)+(6.0E-7)(2.8E2)+(6.0E-7)(6.3E-1)}/5
where, reference dose data are taken from Attachment B-5
H_{xyz} = 1.9 E-4/5 = 3.8 E-5 rem/h
H_{ede} = 3.8 E-5 • 4 = 1.5 E-4 rem
H_{thy} = [H_{ext}+(6.0E-7)(6.3E3)/5] = H_{ext}+7.6E-4 rem/h,
where 6.3 E3 is taken from Attachment B-6
H_{ode} = (H_{ext} + 7.6 E-4) (4) = (4)(6.9 E-8 + 7.6 E-4 rem) = 3.0 E-3 rem, or 30 mrem,
where H_{ext} is estimated from that at a distance of < 100m

References

- PULSTAR Reactor FSAR Sections 5, 10, 13
- 10 CFR 20 "Standards for Protection Against Radiation"
- EPA 400 R-92-001 "Manual for Protective Action Guides and Protective Actions for Nuclear Incidents"
- NUREG 0849 "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors"
- ANSI/ANS 15.16-1982 "Emergency Planning for Research Reactors"
- ANSI/ANS 15.7-1977 "Research Reactor Site Evaluation"
- Meteorology and Atomic Energy 1968, US Atomic Energy Commission
- Letter from Victoreen Instrument Corp., "Technical Data to be used in Conjunction with the Victoreen Radiation Monitoring Equipment Installed in NCSU PULSTAR Reactor", June 1, 1972
- Victoreen Beta Point Source Calibration Data Sheet for Detector Model 943-25T Ser No 216, March 25, 1992 (Stack Particulate Monitor)
- Calculation NRP 91-001 "PULSTAR Reactor Facility Offsite Dose Calculation"
- Calculation NRP 94-001 "Radiation Monitor Set Point Bases and Calculations"

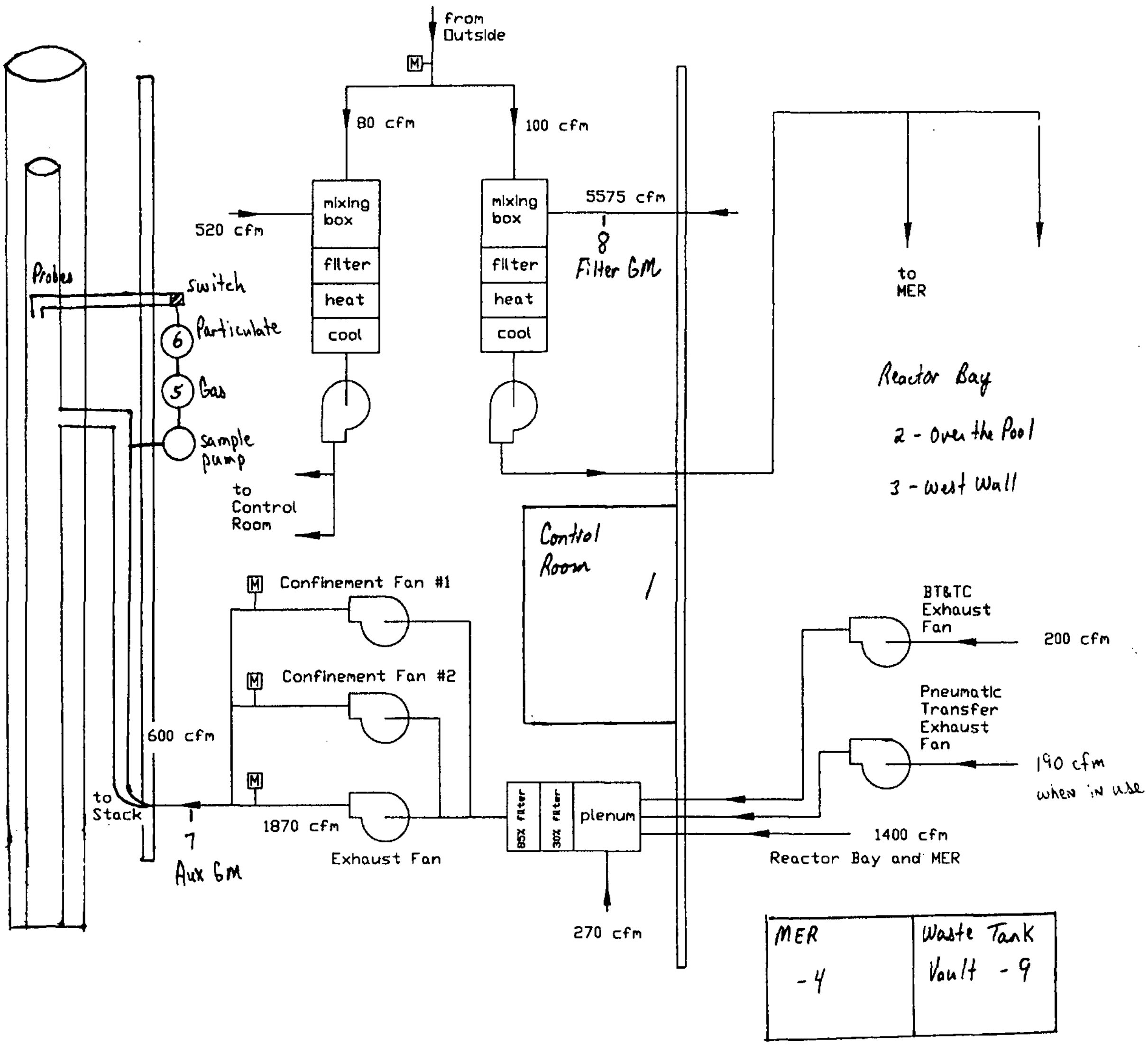
ATTACHMENT B-1
PULSTAR REACTOR VENTILATION SYSTEM

Third floor - above Control Room || Reactor Bay and Mechanical Equipment Room



ATTACHMENT B-2
PULSTAR REACTOR RADIATION MONITORING SYSTEM

ATTACHMENT B-2



ATTACHMENT B-3
χ/Q VALUE CALCULATION

Pulstar Reactor
X/Q Factors

Stability Class	Ave Wind Speed m/s	Stack Height m	Sigma Y m	Sigma Z m	Downwind Distance m	Elevation z m	X/Q s per m3 at 1 mph	Dilution Factor 1/(X/Q)
A	1	30	24.6	15.5	100	0	2.871E-04	3.483E+03
B	1	30	20.6	11.21	100	0	8.592E-05	1.164E+04
C	1	30	14.7	7.8	100	0	3.812E-06	2.624E+05
D	1	30	9	5.5	100	0	4.984E-09	2.006E+08
E	1	30	7.4	3.7	100	0	1.380E-16	7.248E+15
F	1	30	4.9	2.5	100	0	3.129E-33	3.196E+32
A	1	30	24.6	15.5	100	20	7.639E-04	1.309E+03
B	1	30	20.6	11.21	100	20	1.036E-03	9.650E+02
C	1	30	14.7	7.8	100	20	1.366E-03	7.321E+02
D	1	30	9	5.5	100	20	1.378E-03	7.256E+02
E	1	30	7.4	3.7	100	20	3.374E-04	2.964E+03
F	1	30	4.9	2.5	100	20	9.755E-06	1.025E+05
A	1	30	24.6	15.5	100	30	9.348E-04	1.070E+03
B	1	30	20.6	11.21	100	30	1.543E-03	6.482E+02
C	1	30	14.7	7.8	100	30	3.107E-03	3.219E+02
D	1	30	9	5.5	100	30	7.197E-03	1.390E+02
E	1	30	7.4	3.7	100	30	1.301E-02	7.686E+01
F	1	30	4.9	2.5	100	30	2.908E-02	3.439E+01
A	1	30	35	25	150	0	3.963E-04	2.523E+03
B	1	30	25	15	150	0	2.571E-04	3.889E+03
C	1	30	19	11	150	0	8.269E-05	1.209E+04
D	1	30	12	7	150	0	8.711E-07	1.148E+06
E	1	30	9	5	150	0	2.411E-10	4.147E+09
F	1	30	6.5	3.2	150	0	2.815E-21	3.552E+20
A	1	30	35	25	150	20	4.309E-04	2.321E+03
B	1	30	25	15	150	20	7.643E-04	1.308E+03
C	1	30	19	11	150	20	1.128E-03	8.869E+02
D	1	30	12	7	150	20	1.529E-03	6.542E+02
E	1	30	9	5	150	20	1.071E-03	9.334E+02
F	1	30	6.5	3.2	150	20	1.297E-04	7.707E+03
A	1	30	35	25	150	30	4.300E-04	2.326E+03
B	1	30	25	15	150	30	9.503E-04	1.052E+03
C	1	30	19	11	150	30	1.704E-03	5.867E+02
D	1	30	12	7	150	30	4.241E-03	2.358E+02
E	1	30	9	5	150	30	7.916E-03	1.263E+02
F	1	30	6.5	3.2	150	30	1.713E-02	5.839E+01

ATTACHMENT B-3
χ/Q VALUE CALCULATION

Stability Class	Ave Wind Speed m/s	Stack Height m	Sigma Y m	Sigma Z m	Downwind Distance m	Elevation z m	X/Q s per m3 at 1 mph	Dilution Factor 1/(X/Q)
A	1	30	49.9	33.3	200	0	2.857E-04	3.500E+03
B	1	30	34	20.93	200	0	3.584E-04	2.790E+03
C	1	30	25	15.2	200	0	2.674E-04	3.740E+03
D	1	30	17.5	9.4	200	0	2.660E-05	3.760E+04
E	1	30	12.5	6.6	200	0	2.817E-07	3.550E+06
F	1	30	8.3	4.2	200	0	1.704E-13	5.868E+12
A	1	30	49.9	33.3	200	20	2.744E-04	3.645E+03
B	1	30	34	20.93	200	20	4.755E-04	2.103E+03
C	1	30	25	15.2	200	20	7.592E-04	1.317E+03
D	1	30	17.5	9.4	200	20	1.230E-03	8.132E+02
E	1	30	12.5	6.6	200	20	1.370E-03	7.298E+02
F	1	30	8.3	4.2	200	20	6.004E-04	1.666E+03
A	1	30	49.9	33.3	200	30	2.567E-04	3.896E+03
B	1	30	34	20.93	200	30	5.088E-04	1.965E+03
C	1	30	25	15.2	200	30	9.378E-04	1.066E+03
D	1	30	17.5	9.4	200	30	2.166E-03	4.618E+02
E	1	30	12.5	6.6	200	30	4.318E-03	2.316E+02
F	1	30	8.3	4.2	200	30	1.022E-02	9.786E+01
A	1	30	60	40	250	0	2.241E-04	4.463E+03
B	1	30	40	25	250	0	3.468E-04	2.884E+03
C	1	30	30	18	250	0	3.290E-04	3.040E+03
D	1	30	20	11	250	0	7.856E-05	1.273E+04
E	1	30	15	7.9	250	0	4.442E-06	2.251E+05
F	1	30	10	5	250	0	2.170E-10	4.608E+09
A	1	30	60	40	250	20	2.118E-04	4.721E+03
B	1	30	40	25	250	20	3.771E-04	2.652E+03
C	1	30	30	18	250	20	5.793E-04	1.726E+03
D	1	30	20	11	250	20	1.071E-03	9.335E+02
E	1	30	15	7.9	250	20	1.349E-03	7.412E+02
F	1	30	10	5	250	20	9.642E-04	1.037E+03
A	1	30	60	40	250	30	1.966E-04	5.086E+03
B	1	30	40	25	250	30	3.762E-04	2.658E+03
C	1	30	30	18	250	30	6.622E-04	1.510E+03
D	1	30	20	11	250	30	1.619E-03	6.176E+02
E	1	30	15	7.9	250	30	3.006E-03	3.326E+02
F	1	30	10	5	250	30	7.125E-03	1.404E+02

ATTACHMENT B-3
χ/Q VALUE CALCULATION

Pulstar Reactor
X/Q Factors

Stability Class	Ave Wind Speed m/s	Stack Height m	Sigma Y m	Sigma Z m	Downwind Distance m	Elevation z m	X/Q s per m3 at 1 mph	Dilution Factor 1/(X/Q)
A	1	30	100	124.9	500	0	5.542E-05	1.804E+04
B	1	30	84	52.5	500	0	1.372E-04	7.288E+03
C	1	30	60	35.6	500	0	2.339E-04	4.276E+03
D	1	30	40	18.6	500	0	2.608E-04	3.835E+03
E	1	30	30	13.6	500	0	1.533E-04	6.524E+03
F	1	30	20	8.3	500	0	6.249E-06	1.600E+05
A	1	30	100	124.9	500	20	5.476E-05	1.826E+04
B	1	30	84	52.5	500	20	1.307E-04	7.654E+03
C	1	30	60	35.6	500	20	2.225E-04	4.494E+03
D	1	30	40	18.6	500	20	4.273E-04	2.340E+03
E	1	30	30	13.6	500	20	6.673E-04	1.499E+03
F	1	30	20	8.3	500	20	1.039E-03	9.629E+02
A	1	30	100	124.9	500	30	5.393E-05	1.854E+04
B	1	30	84	52.5	500	30	1.228E-04	8.142E+03
C	1	30	60	35.6	500	30	2.071E-04	4.829E+03
D	1	30	40	18.6	500	30	4.814E-04	2.077E+03
E	1	30	30	13.6	500	30	8.732E-04	1.145E+03
F	1	30	20	8.3	500	30	2.146E-03	4.660E+02
A	1	30	183	550	1000	0	7.068E-06	1.415E+05
B	1	30	154	122.1	1000	0	3.676E-05	2.720E+04
C	1	30	110	65.9	1000	0	8.861E-05	1.129E+04
D	1	30	71.4	33	1000	0	2.000E-04	4.999E+03
E	1	30	55	22.3	1000	0	2.350E-04	4.255E+03
F	1	30	36.6	13.5	1000	0	1.221E-04	8.192E+03
A	1	30	183	550	1000	20	7.063E-06	1.416E+05
B	1	30	154	122.1	1000	20	3.630E-05	2.755E+04
C	1	30	110	65.9	1000	20	8.543E-05	1.171E+04
D	1	30	71.4	33	1000	20	1.924E-04	5.198E+03
E	1	30	55	22.3	1000	20	2.862E-04	3.494E+03
F	1	30	36.6	13.5	1000	20	5.487E-04	1.822E+03
A	1	30	183	550	1000	30	7.058E-06	1.417E+05
B	1	30	154	122.1	1000	30	3.574E-05	2.798E+04
C	1	30	110	65.9	1000	30	8.161E-05	1.225E+04
D	1	30	71.4	33	1000	30	1.801E-04	5.551E+03
E	1	30	55	22.3	1000	30	2.982E-04	3.353E+03
F	1	30	36.6	13.5	1000	30	7.210E-04	1.387E+03

ATTACHMENT B-4
EXTERNAL DOSE RATE CALCULATIONS FOR REFERENCE LINE SOURCES

H_{stack} and H_{line}

Overhead Plume - Line length of 100 m

Stack - line length of 20 m

- Elevated point at 20 m height and 50 m away
- Ground point at 0 m height and 50 m away

RADIO-NUCLIDE	ELEV-ATION	H_{line} Normal Ventilation rem/h/Ci at 1 mph	H_{line} Confinement Ventilation rem/h/Ci at 1 mph	H_{stack} Normal Ventilation rem/h/Ci at 1 mph	H_{stack} Confinement Ventilation rem/h/Ci at 1 mph
Ar-41	Ground	4.2 E-4	4.2 E-4	2.1 E-4	2.1 E-4
Ar-41	20 m	1.1 E-3	1.1 E-3	2.4 E-4	2.4 E-4
Fuel Failure	Ground	7.8 E-5	7.7 E-5	4.0 E-5	4.1 E-5
Fuel Failure	20 m	2.1 E-4	2.1 E-4	4.2 E-5	4.2 E-5

RADIO-NUCLIDE	ELEV-ATION	H_{shine} Normal Ventilation rem/h/μCi/ml at 1 mph	H_{shine} Confinement Ventilation rem/h/μCi/ml at 1 mph	H_{shine} Confinement Ventilation with Dilution rem/h/μCi/ml at 1 mph
Ar-41	Ground	8.4 E-2	2.8 E-2	2.2 E-3
Ar-41	20 m	2.2 E-1	7.1 E-2	4.5 E-3
Fuel Failure	Ground	1.6 E-2	5.1 E-3	4.1 E-4
Fuel Failure	20 m	4.2 E-2	1.4 E-2	8.4 E-4

ATTACHMENT B-5
DOSE RATE CALCULATION FOR REFERENCE RELEASE

	Distance x (m)	Elevation h (m)	X/Q (s/m3 at 1 mph)	Hxyz REFERENCE RELEASE DOSE RATES Normal (rem/h per uCi/ml at 1 mph)	Confine	Confine with Dilution
Ar-41 DCF = 1100	50	0		8.4E-02	2.8E-02	2.2E-03Hext
rem/h	50	20		2.2E-01	7.1E-02	4.5E-03Hext
per uCi/ml	100	0	2.90E-04	2.8E-01	9.0E-02	4.5E-03
	100	20	1.40E-03	1.4E+00	4.4E-01	2.2E-02
	150	0	4.00E-04	3.9E-01	1.2E-01	6.2E-03
	150	20	1.50E-03	1.5E+00	4.7E-01	2.3E-02
	150	30	1.70E-02	1.7E+01	5.3E+00	2.6E-01
	200	0	3.60E-04	3.5E-01	1.1E-01	5.6E-03
	200	20	1.40E-03	1.4E+00	4.4E-01	2.2E-02
	200	30	1.00E-02	9.7E+00	3.1E+00	1.6E-01
	250	0	3.50E-04	3.4E-01	1.1E-01	5.5E-03
	250	20	1.40E-03	1.4E+00	4.4E-01	2.2E-02
	250	30	7.10E-03	6.9E+00	2.2E+00	1.1E-01
	500	0	2.60E-04	2.5E-01	8.1E-02	4.0E-03
	500	20	1.00E-03	9.7E-01	3.1E-01	1.6E-02
	500	30	2.20E-03	2.1E+00	6.9E-01	3.4E-02
Fuel Fail Gas DCF 123	50	0		1.6E-02	5.1E-03	4.1E-04Hext
rem/h	50	20		4.2E-02	1.4E-02	8.4E-04Hext
per uCi/ml	100	0	2.90E-04	3.1E-02	1.0E-02	5.1E-04
	100	20	1.40E-03	1.5E-01	4.9E-02	2.4E-03
	150	0	4.00E-04	4.3E-02	1.4E-02	7.0E-04
	150	20	1.50E-03	1.6E-01	5.2E-02	2.6E-03
	150	30	1.70E-02	1.8E+00	5.9E-01	3.0E-02
	200	0	3.60E-04	3.9E-02	1.3E-02	6.3E-04
	200	20	1.40E-03	1.5E-01	4.9E-02	2.4E-03
	200	30	1.00E-02	1.1E+00	3.5E-01	1.7E-02
	250	0	3.50E-04	3.8E-02	1.2E-02	6.1E-04
	250	20	1.40E-03	1.5E-01	4.9E-02	2.4E-03
	250	30	7.10E-03	7.7E-01	2.5E-01	1.2E-02
	500	0	2.60E-04	2.8E-02	9.1E-03	4.5E-04
	500	20	1.00E-03	1.1E-01	3.5E-02	1.7E-03
	500	30	2.20E-03	2.4E-01	7.7E-02	3.8E-03

ATTACHMENT B-5
DOSE RATE CALCULATION FOR REFERENCE RELEASE

		Distance x (m)	Elevation h (m)	X/Q (s/m3 at 1 mph)	Hxyz REFERENCE RELEASE DOSE RATE Normal (rem/h per uCi/ml at 1 mph)	Confine	Confine with Dilution
Particulate DCF rem/h per uCi/ml	270000	100	0	2.90E-04	6.9E+01	2.2E+01	1.1E+00
		100	20	1.40E-03	3.3E+02	1.1E+02	5.4E+00
		150	0	4.00E-04	9.5E+01	3.1E+01	1.5E+00
		150	20	1.50E-03	3.6E+02	1.1E+02	5.7E+00
		150	30	1.70E-02	4.1E+03	1.3E+03	6.5E+01
		200	0	3.60E-04	8.6E+01	2.8E+01	1.4E+00
		200	20	1.40E-03	3.3E+02	1.1E+02	5.4E+00
		200	30	1.00E-02	2.4E+03	7.6E+02	3.8E+01
		250	0	3.50E-04	8.3E+01	2.7E+01	1.3E+00
		250	20	1.40E-03	3.3E+02	1.1E+02	5.4E+00
		250	30	7.10E-03	1.7E+03	5.4E+02	2.7E+01
		500	0	2.60E-04	6.2E+01	2.0E+01	9.9E-01
		500	20	1.00E-03	2.4E+02	7.6E+01	3.8E+00
		500	30	2.20E-03	5.2E+02	1.7E+02	8.4E+00
Fuel Fail Part DCF rem/h per uCi/ml	1600000	100	0	2.90E-04	4.1E+02	1.3E+02	6.6E+00
		100	20	1.40E-03	2.0E+03	6.3E+02	3.2E+01
		150	0	4.00E-04	5.6E+02	1.8E+02	9.1E+00
		150	20	1.50E-03	2.1E+03	6.8E+02	3.4E+01
		150	30	1.70E-02	2.4E+04	7.7E+03	3.9E+02
		200	0	3.60E-04	5.1E+02	1.6E+02	8.2E+00
		200	20	1.40E-03	2.0E+03	6.3E+02	3.2E+01
		200	30	1.00E-02	1.4E+04	4.5E+03	2.3E+02
		250	0	3.50E-04	4.9E+02	1.6E+02	7.9E+00
		250	20	1.40E-03	2.0E+03	6.3E+02	3.2E+01
		250	30	7.10E-03	1.0E+04	3.2E+03	1.6E+02
		500	0	2.60E-04	3.7E+02	1.2E+02	5.9E+00
		500	20	1.00E-03	1.4E+03	4.5E+02	2.3E+01
		500	30	2.20E-03	3.1E+03	1.0E+03	5.0E+01

ATTACHMENT B-5
DOSE RATE CALCULATION FOR REFERENCE RELEASE

	Distance x (m)	Elevation h (m)	X/Q (s/m3 at 1 mph)	Hxyz REFERENCE RELEASE DOSE RATE Normal (rem/h per uCi/ml at 1 mph)	Confine	Confine with Dilution
Fuel Fail						
Iodines Effective DCF 32000	100	0	2.90E-04	8.2E+00	2.6E+00	1.3E-01
rem/h	100	20	1.40E-03	4.0E+01	1.3E+01	6.3E-01
per uCi/ml	150	0	4.00E-04	1.1E+01	3.6E+00	1.8E-01
	150	20	1.50E-03	4.2E+01	1.4E+01	6.8E-01
	150	30	1.70E-02	4.8E+02	1.5E+02	7.7E+00
	200	0	3.60E-04	1.0E+01	3.3E+00	1.6E-01
	200	20	1.40E-03	4.0E+01	1.3E+01	6.3E-01
	200	30	1.00E-02	2.8E+02	9.1E+01	4.5E+00
	250	0	3.50E-04	9.9E+00	3.2E+00	1.6E-01
	250	20	1.40E-03	4.0E+01	1.3E+01	6.3E-01
	250	30	7.10E-03	2.0E+02	6.4E+01	3.2E+00
	500	0	2.60E-04	7.3E+00	2.4E+00	1.2E-01
	500	20	1.00E-03	2.8E+01	9.1E+00	4.5E-01
	500	30	2.20E-03	6.2E+01	2.0E+01	1.0E+00
Fuel Fail						
Bromines DCF 71	100	0	2.90E-04	1.8E-02	5.8E-03	2.9E-04
rem/h	100	20	1.40E-03	8.8E-02	2.8E-02	1.4E-03
per uCi/ml	150	0	4.00E-04	2.5E-02	8.0E-03	4.0E-04
	150	20	1.50E-03	9.4E-02	3.0E-02	1.5E-03
	150	30	1.70E-02	1.1E+00	3.4E-01	1.7E-02
	200	0	3.60E-04	2.3E-02	7.2E-03	3.6E-04
	200	20	1.40E-03	8.8E-02	2.8E-02	1.4E-03
	200	30	1.00E-02	6.3E-01	2.0E-01	1.0E-02
	250	0	3.50E-04	2.2E-02	7.0E-03	3.5E-04
	250	20	1.40E-03	8.8E-02	2.8E-02	1.4E-03
	250	30	7.10E-03	4.4E-01	1.4E-01	7.1E-03
	500	0	2.60E-04	1.6E-02	5.2E-03	2.6E-04
	500	20	1.00E-03	6.3E-02	2.0E-02	1.0E-03
	500	30	2.20E-03	1.4E-01	4.4E-02	2.2E-03

ATTACHMENT B-5
DOSE RATE CALCULATION FOR REFERENCE RELEASE

Fuel Fail	**Distance**	**Elevation**	**X/Q**	**Hxyz REFERENCE RELEASE DOSE RATE**		
Thy DCF	**x**	**h**	**(s/m3**	**Normal**	**Confine**	**Confine with Dilution**
710000	**(m)**	**(m)**	**at 1 mph)**	**(rem/h per uCi/ml at 1 mph)**		
rem/h	100	0	2.90E-04	1.8E+02	5.8E+01	2.9E+00
per uCi/ml	100	20	1.40E-03	8.8E+02	2.8E+02	1.4E+01
	150	0	4.00E-04	2.5E+02	8.0E+01	4.0E+00
	150	20	1.50E-03	9.4E+02	3.0E+02	1.5E+01
	150	30	1.70E-02	1.1E+04	3.4E+03	1.7E+02
	200	0	3.60E-04	2.3E+02	7.2E+01	3.6E+00
	200	20	1.40E-03	8.8E+02	2.8E+02	1.4E+01
	200	30	1.00E-02	6.3E+03	2.0E+03	1.0E+02
	250	0	3.50E-04	2.2E+02	7.0E+01	3.5E+00
	250	20	1.40E-03	8.8E+02	2.8E+02	1.4E+01
	250	30	7.10E-03	4.4E+03	1.4E+03	7.1E+01
	500	0	2.60E-04	1.6E+02	5.2E+01	2.6E+00
	500	20	1.00E-03	6.3E+02	2.0E+02	1.0E+01
	500	30	2.20E-03	1.4E+03	4.4E+02	2.2E+01

ATTACHMENT B-6
DOSE CALCULATION FOR Ar-41 ACCIDENT RELEASE and FSAR FUEL HANDLING ACCIDENT

	Distance x(m)	Elevation h(m)	PREDICTED Hxyz DOSE Normal (rem at 1 mph)	Confine	Confine with Dilution
Ar41	50	0	5.44E-05	1.78E-05	1.41E-06
Accident	50	20	1.42E-04	4.58E-05	2.89E-06
Release	100	0	1.82E-04	5.73E-06	2.87E-07
	100	20	8.81E-04	2.77E-05	1.38E-06
	150	0	2.52E-04	7.91E-06	3.95E-07
	150	20	9.44E-04	2.96E-05	1.48E-06
	150	30	**1.07E-02**	**3.36E-04**	**1.68E-05**
	200	0	2.26E-04	7.12E-06	3.56E-07
	200	20	8.81E-04	2.77E-05	1.38E-06
	200	30	6.29E-03	1.98E-04	9.88E-06
	250	0	2.20E-04	6.92E-06	3.46E-07
	250	20	8.81E-04	2.77E-05	1.38E-06
	250	30	4.47E-03	1.40E-04	7.02E-06
	500	0	1.64E-04	5.14E-06	2.57E-07
	500	20	6.29E-04	1.98E-05	9.88E-07
	500	30	1.38E-03	4.35E-05	2.17E-06

ATTACHMENT B-6
DOSE CALCULATION FOR Ar-41 ACCIDENT RELEASE and FSAR FUEL HANDLING ACCIDENT

	Distance x(m)	Elevation h(m)	PREDICTED Hxyz DOSE Normal (rem at 1 mph)	Confine	Confine with Dilution
Fuel	50	0	3.16E-07	2.97E-07	2.41E-08
Handle	50	20	8.44E-07	7.93E-07	4.91E-08
Accident	100	0	5.21E-06	6.40E-07	3.20E-08
Effective	100	20	2.51E-05	3.09E-06	1.54E-07
Dose Rates	150	0	7.18E-06	8.83E-07	4.41E-08
	150	20	2.69E-05	3.31E-06	1.65E-07
	150	30	**3.05E-04**	**3.75E-05**	**1.88E-06**
	200	0	6.46E-06	7.94E-07	3.97E-08
	200	20	2.51E-05	3.09E-06	1.54E-07
	200	30	1.80E-04	2.21E-05	1.10E-06
	250	0	6.29E-06	7.72E-07	3.86E-08
	250	20	2.51E-05	3.09E-06	1.54E-07
	250	30	1.27E-04	1.57E-05	7.83E-07
	500	0	4.67E-06	5.74E-07	2.87E-08
	500	20	1.80E-05	2.21E-06	1.10E-07
	500	30	3.95E-05	4.85E-06	2.43E-07
Fuel	50	0	3.16E-07	2.97E-07	2.41E-08
Handle	50	20	8.44E-07	7.93E-07	4.91E-08
Accident	100	0	1.02E-04	1.03E-06	5.14E-08
Organ	100	20	4.94E-04	4.96E-06	2.48E-07
Dose Rate	150	0	1.41E-04	1.42E-06	7.09E-08
	150	20	5.29E-04	5.32E-06	2.66E-07
	150	30	**5.99E-03**	**6.02E-05**	**3.01E-06**
	200	0	1.27E-04	1.28E-06	6.38E-08
	200	20	4.94E-04	4.96E-06	2.48E-07
	200	30	3.53E-03	3.54E-05	1.77E-06
	250	0	1.23E-04	1.24E-06	6.20E-08
	250	20	4.94E-04	4.96E-06	2.48E-07
	250	30	2.50E-03	2.52E-05	1.26E-06
	500	0	9.17E-05	9.21E-07	4.61E-08
	500	20	3.53E-04	3.54E-06	1.77E-07
	500	30	7.76E-04	7.80E-06	3.90E-07

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 10CFR50.55 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 10CFR50.55 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. **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.

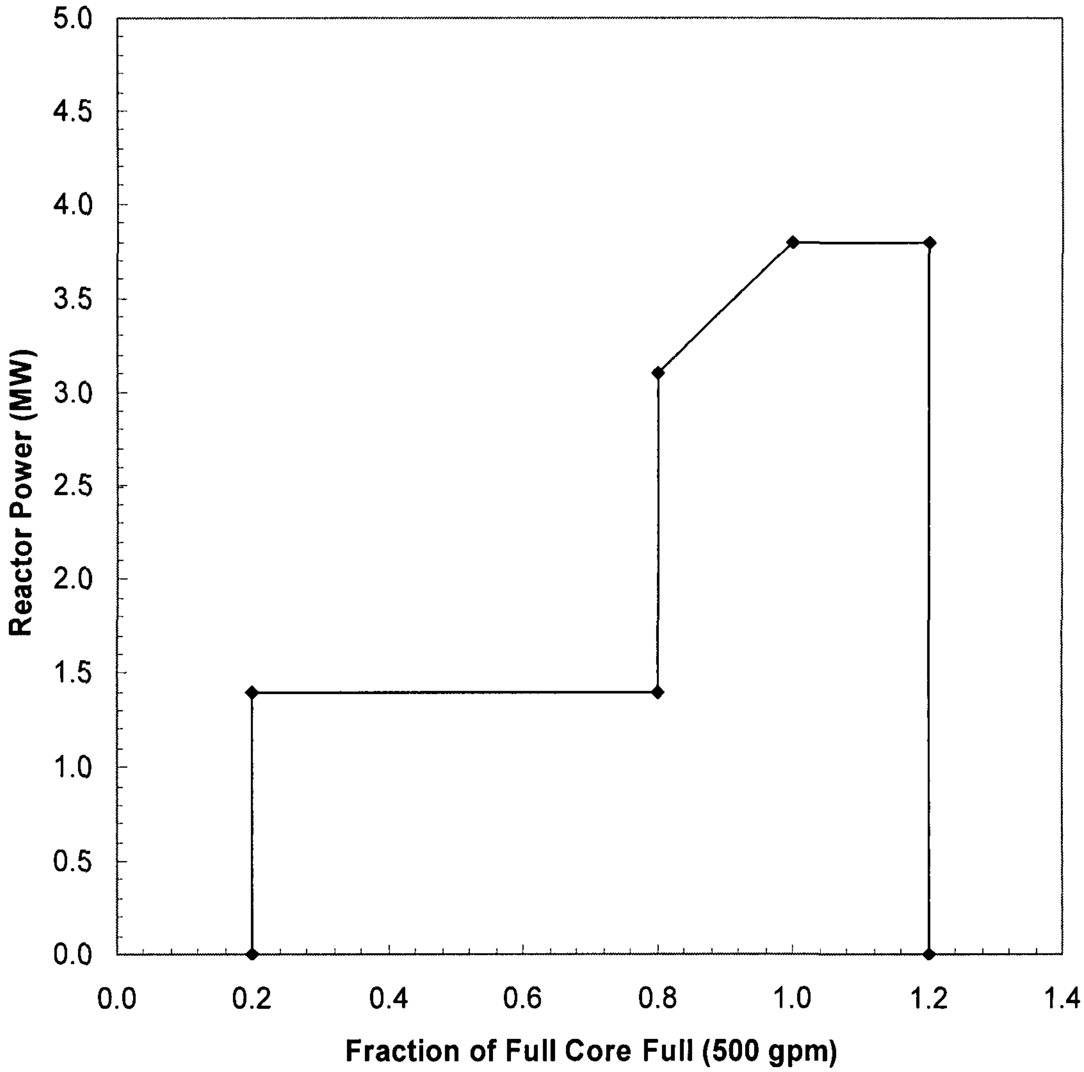


Figure 2.1-1: Power-Flow Safety Limit Curve

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)	≤ 2,500 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 as defined in Specification 1.2.9.e.

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

b. Specification 3.2 pertaining to reactivity shall be met.

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

d. 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. Limitation on sample mass.

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

   iii. Limitation on power or fission rate produced in the sample.

   iv. Limitation on irradiation time.

   v. Radiation monitoring for detection of released fission products.

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

   vii. Design criteria related to keeping personnel and public radiation doses and radioactive material releases As Low As Reasonably Achievable (ALARA).

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

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 is limited as defined in 10 CFR 20.

Specification 3.8 d 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 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 $\geq$500 kΩ·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.

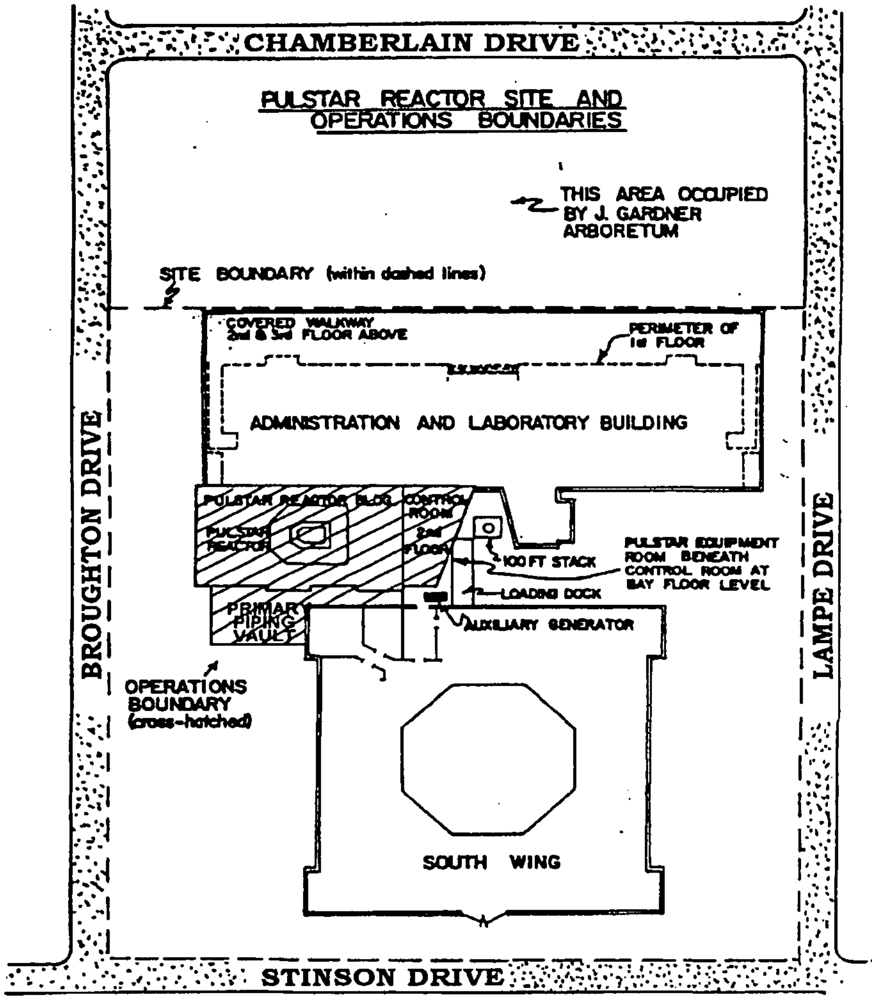


Figure 5.2-1: NCSU PULSTAR Reactor Site Map

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 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.

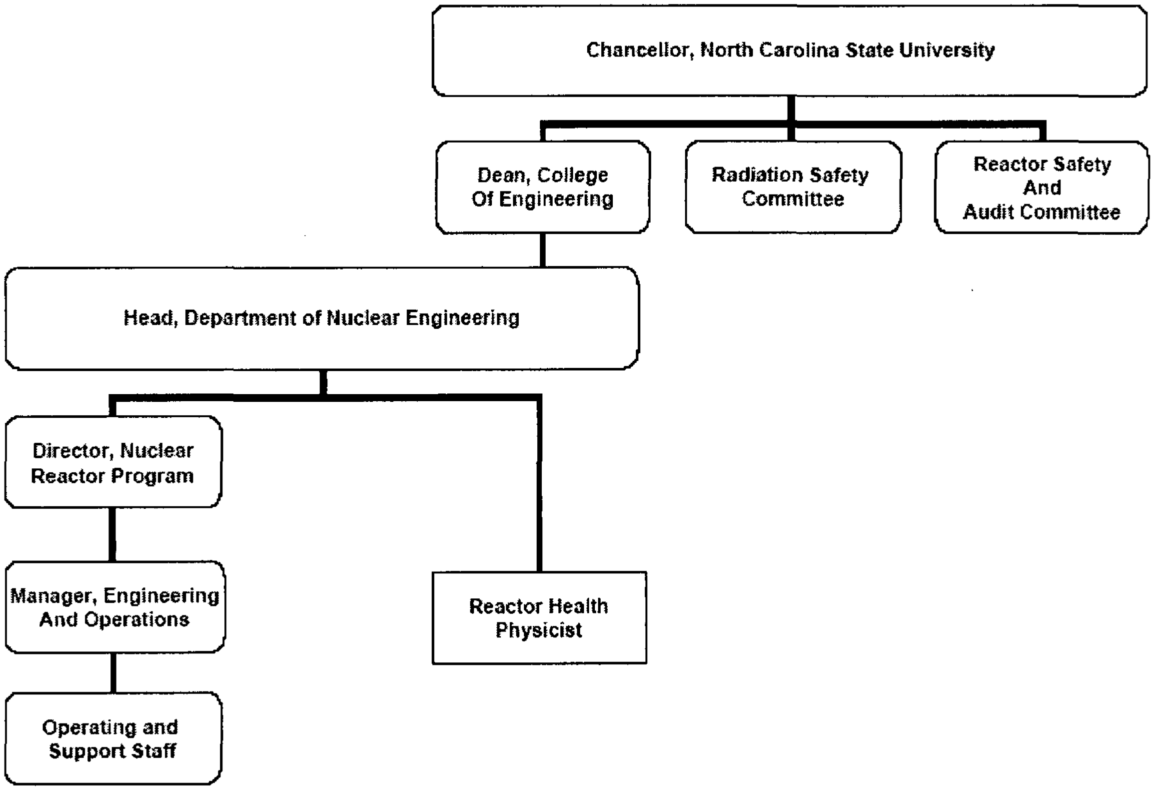
c. Relocation of any in-core experiment with a reactivity worth greater than one dollar (730 pcm).

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:

Lines of supervision are shown.

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.

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 positions of NRP Director, RHP, and 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 Charter and 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 or 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, and the RSAC Chair 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 action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods 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.

A summary of the annual audit made by the RSAC is forwarded 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 February 28th 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) training cycle:

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