PURDUE
UNIVERSITY

RADIOLOGICAL & ENVIRONMENTAL MANAGEMENT

February 8, 2011

Mr. Merritt N. Baker
Fuel Manufacturing Branch
Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
Mail Stop E-2-C40M
11555 Rockville Pike
Rockville, MD 20852

RE: REQUEST FOR ADDITIONAL INFORMATION REGARDING RENEWAL OF SNM-142 (TAC L32882)

Dear Mr. Baker:

This letter is sent to you in response to your letter dated September 24, 2010 for request for additional information regarding renewal of SNM-142.

Attached are the answers to the Request for Additional Information. In order to maintain clarity we have submitted revision in its' entirety (minus original attachments) of our SNM license renewal request. The tables below indicate where the information can be found in the revised SNM-142 license renewal request, relating to your questions. In addition, there is an attached SNM-142 license renewal request with indicating marks (markup), to allow a clearer comparison to the previous submission.

If you should have any questions regarding this information please contact me at 765-494-2350. Thanks for your attention to this matter.

Sincerely,

James F. Schweitzer
Radiation Safety Officer

Attachments:

(1) Table indicating location of responses to RAI
(2) US NRC Request for additional information answers regarding renewal of SNM-142 (TAC L32882) dated February 3, 2011
(3) SNM-142 License Renewal Request, Revision 1 (dated February 3, 2011)
(4) SNM-142 License Renewal Request, Revision 1 markup (dated February 3, 2011)

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REQUEST FOR ADDITIONAL INFORMATION REGARDING RENEWAL OF SNM-142
(TAC L32882)

Radiation Protection	***Purdue University Response Changes to Application***
RP-1)	Sections 7.3 (2nd paragraph) and 10.11 (last paragraph)
RP-2)	Addressed in Section 10.9 (2nd paragraph)
RP-3)	Addressed in Section 8.2 (1st paragraph)
RP-4)	Corrected
RP-5)	Addressed in Section 10.7 (last paragraph)
RP-6)	Addressed in Sections 7.3 (3rd paragraph) and 7.4 (last paragraph)
RP-7)	Addressed in Section 10.4 (last paragraph)
RP-8)	Addressed in Section 7.5 (1st paragraph)

Criticality Safety	***Purdue University Response Changes to Application***
CS-1)	Addressed in Section 6.0 (a)
CS-2)	Addressed in Section 6.0 (c)
CS-3)	Addressed in answers
CS-4)	Addressed in answers
CS-5)	Addressed in Section 9.2
CS-6)	Addressed in Section 9.2
CS-7)	Addressed in answers
CS-8)	Addressed in answers
CS-9)	Not Addressed see Section 9.2
CS-10)	Addressed in Section 9.2
CS-11)	Addressed in answers
CS-12)	Not Addressed see Section 9.2
CS-13)	Addressed in answers

Fire Protection	*Purdue University Response Changes to Application*
FP-1)	Addressed in Section 9.1 (Duncan Annex-General; Building Construction Information)
FP-2)	Addressed in Section 9.1 (Duncan Annex-General; Fire Protection Features)
FP-3)	Addressed in Section 9.1 (Duncan Annex-General; Fire Protection Equipment Inspection, Testing, and Maintenance)
FP-4)	Addressed in Section 9.1 (Duncan Annex-General; Combustible Loading)
FP-5)	Addressed in Section 9.1 (Duncan Annex-General; Housed Radioactive Material)
FP-6)	Addressed in Section 9.1 (Duncan Annex-General; Hazardous Material and Processes)
FP-7)	Addressed in Section 9.1 (Duncan Annex-General; NFPA Code Compliance)
FP-8)	Addressed in Section 9.1 (Duncan Annex-General; Fire Safety and Emergency Training for Facility Workers)
FP-9)	Addressed in Section 9.1 (Duncan Annex-General; Fire Department Support)

RP-1) Guidance available to the U.S. Nuclear Regulatory Commission (NRC) staff (specifically NUREG-1520, Section 4.4.2.3) considers an application for a fuel cycle facility to be acceptable if it includes commitments for an as low as is reasonably achievable (ALARA) Committee with clear responsibilities to ensure that the occupational radiation exposure limits are not exceeded during normal conditions. The ALARA Committee will review the ALARA program and should include evaluations of audit results, radiation levels in the facility, contamination levels, employee exposures, effluent releases, etc. The review should look for upward trends and determine if exposures, releases, and contamination levels are in accordance with the ALARA concept. The ALARA program's goals and objectives should be reviewed regularly and revised, as appropriate. Recommendations of the committee should be documented and tracked to completion. The staff considers this portion of the guidance to be applicable to Purdue. Please revise the application, Sections 7.3 and 10.10, as appropriate, to establish the Radiation Safety Committee as an acceptable substitute for the ALARA Committee with commitments to perform ALARA reviews as described in NUREG-1520. This is needed to ensure compliance with10 CFR 20.1101(b).

The following will be added to Section 7.3:

The RSC also acts as the ALARA (as low as reasonably achievable) Committee. This is achieved by its consideration of radiation safety program audit results, laboratory survey results (which includes contamination evaluations), radiation exposure reports for personnel, etc. The Committee has the authority to mandate action, should it determine a need for further restrictions or procedural changes to achieve and maintain ALARA.

The following will be added to Section 10.11:

The Radiation Safety Committee (RSC) also acts as the ALARA Committee (see Section 7.3 of this document). All aspects of the ALARA program are reviewed on a continual basis by the RSC.

RP-2) 10 CFR 70.22(i)(1)(i) requires an evaluation showing that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 1 roentgen equivalent man (rem) effective dose equivalent or an intake of 2 milligrams of soluble uranium or else an emergency plan is required. You have stated that an emergency plan is not required but have not submitted an evaluation showing that the maximum dose to a member of the public does not exceed 1 rem effective dose equivalent. Please provide this evaluation to comply with the referenced regulation.

A separate "Emergency Plan" is not required for this license. This finding is based on the type of material, the geometry of material, and the conditions under which the material is stored. The most credible accident scenario with release to the environment would be a facility fire. The following material and facility characteristics make the intake of 2 grams of soluble uranium or 1 rem effective dose equivalent highly unlikely. Uranium SNM is predominately sealed in aluminum or stainless steel cladding. This first level of containment would be resistant to fire and provides a barrier for release of material to the environment. Additionally other licensed material is in the form of sealed sources and would similarly prevent release of licensed material to the environment. All material is stored within a room in a building which would act to further contain any material that could be released. Some material is further contained within cabinets or in stainless steel converters.

Finally the building would further act to contain material. The building structures are composed of brick, block, concrete and other fire resistant materials. The buildings with the exception of the SNM storage areas are sprinklered and would prevent a fire from spreading easily. The ventilation system would be shut down as a precaution to minimize the air entering and exiting the building and as a result minimize the release of combustion products. Fire department and police personnel would also ensure evacuation of the building(s) and this would prevent exposure to members of the public (non-radiation workers).

RP-3) Inclusion of the Radiation Safety Manual in the License Application potentially creates a situation where any revision to the manual may require a license amendment. It may clarify things to state in the application (and possibly stamped on the attachment) that this attachment is for information only and include in the application that a Radiation Safety Manual is maintained online that addresses topic areas ... (as are present in the current revision of the manual). Please clarify the intention of including the Radiation Safety Manual in the application.

The application has been changed in Section 8.2 and refers to an online version of the Radiation Safety Manual.

RP-4) There are two sections numbered 10.7 in the application. Please revise to avoid confusion when referring to a specific section.

The sections have been renumbered.

RP-5) In Section labeled 10.7, page 14, last paragraph, there are references to MPC (assumed to be maximum permissible concentration as was referenced in 10 CFR 20 prior to 1992). This reference is outdated. Please revise as appropriate.

The section has been corrected to reference 10 CFR 20 Appendix B.

RP-6) There is no mention of Radiation Work Permits in the application. How does Purdue address work involving licensed material that is outside the scope of developed procedures?

The RSO is authorized to review and approve minor deviations in work procedures (equivalent to "Radiation Work Permits"). These actions are subsequently reviewed and approved by the Radiation Safety Committee at the next scheduled meeting.

RP-7) Please clarify the use of oxidants, solvents, and isotope dilution solutions for personnel decontamination on page 12, Section 10.4, last paragraph.

The reference to oxidants, solvents, and isotope dilution solutions has been removed.

RP-8) Section 7.5 of the application references a 1982 Draft Regulatory Guide for RSO Qualifications. This is an outdated and difficult to locate reference. Is there no better guidance available that could be referenced—if such is desired?

The RSO should meet the requirements as described in Section 8.7.1 of NUREG 1556, Volume 7.

CS-1) Confirm that your statement in Section 6.0, "Other various samples of both enriched and natural uranium may be obtained for non-destructive testing, chemical and thermal analysis, and other analytical and other analytical and developmental techniques" does not imply that you may exceed your possession limits listed in Section 5.0. This is an open-ended statement that could be construed as allowing materials beyond those in the possession limits to be acquired.

The text of §6.0.a has been modified to read as follows (new text in italics):

Other various samples of both enriched and natural uranium may be obtained for non-destructive testing, chemical and thermal analysis, and other analytical and developmental techniques; *however, material acquisition above license limits will not be allowed. A license amendment will be requested to allow for higher possession limits if necessary.*

CS-2) Confirm that only storage activities will be conducted with licensed special nuclear material; and that an amendment request will be provided for any handling, movement, or rearrangement of the material.

The material discussed in §6.0.c are for storage only. The material addressed in the other sections (§6.0.a, §6.0.b and §6.0.d) are still in use, and will be used according to previous practices and procedures.

CS-3) Explain what type of evaluation you do to demonstrate that the storage configurations described in the License Application are adequately subcritical under normal and credible abnormal conditions. Although calculations appear to have been performed (because the application refers to specific effective multiplication factor values), there is no reference to criticality safety evaluations. Provide the following information regarding your conduct of such evaluations:

- ***a. Explain your criteria for when a criticality safety evaluation is needed (e.g., when fissile mass exceeds a certain threshold).***
- ***b. Describe the contents of your criticality safety evaluations.***
- ***c. Describe the qualifications of staff authorized to perform criticality safety evaluations, and the process for reviewing and approving them.***
- ***d. List and briefly describe all criticality safety evaluations that establish the safety basis for your facility.***

The SPERT fuel rods and 1.3% enriched fuel rods kept in Duncan Annex are licensed for storage only (see the SNM-142 licenses granted in 1999 and 1994, or earlier), and all storage configurations meet or exceed the guidance of ANSI/ANS-8.1-1998. Therefore, no additional criticality calculations have been performed.

The SPERT material in the Physics Building Fast Breeder Blanket Facility (FBBF) in Physics B28 is contained within two converter assemblies, one within the FBBF itself, and another in a secure cage. Both of these converter assemblies are made of stainless steel and are welded shut, with no rearrangement of the fuel contained within possible. Two independent sets of criticality calculations were performed on the converter assemblies (one by Purdue staff, and another by staff at the University of California at

Berkeley), both of which were submitted to and evaluated by NRC staff, and subsequently approved (see NRC Safety Evaluation Report issued 17 December 1976 for details). Since the fuel is presently licensed for storage only, and no credible normal or abnormal conditions exist which could lead to an unplanned criticality with this material, no further calculations have been performed.

CS-4) Describe your process for validating calculations used to demonstrate subcriticality, including
- **(a) the specific computer codes or methods used;**
- **(b) the validation methodology;**
- **(c) the critical experiments used to perform the validation; and**
- **(d) the resulting upper subcritical limit, including the minimum margin of subcriticality and the method's area of applicability.**

As described in the answer to CS-3, no criticality evaluations have been required since the fuel is licensed for storage only, and is stored in a way that meets or exceeds the requirements for slab geometries as described in ANSI/ANS-8.1-1998. Therefore there is no process in place for validation of calculations. Any movement of fuel would be subject to an NRC amendment and a criticality calculation with subsequent NRC approval.

CS-5) Summarize the administrative controls and limits for ensuring criticality safety in your facility (e.g., geometry, enrichment, moderator limits, and the controls to maintain these limits).

The administrative controls and limits for ensuring criticality safety for the material in the facility are based on ANSI/ANS-8.1-1998 and ANSI/ANS-8.7-1998. Criticality is controlled by maintaining safe geometry based upon enrichment for the material in question. All of the fuel is arranged in a rack securely mounted to the concrete walls in the storage areas. Only three staff members have access to the material, and each of those individuals are aware of the requirements for maintaining the established criticality controls.

CS-6) Address the susceptibility of storage racks, cabinets, and other equipment credited with ensuring criticality safety to external events such as earthquakes, floods, and sprinkler activation or firefighting activities following a fire (e.g., from a sodium fire). The main concern would be introduction of water or rearrangement of licensed material following such an event. This information is needed to provide reasonable assurance that processes will be subcritical under normal and credible abnormal conditions.

SNM stored in the Duncan Annex are in windowless rooms in a steel frame, concrete and block building with no sprinkler systems. There are no flammable materials stored or used in close proximity to the material, and no ignition sources, so the possibility of fire is very low. The building characteristics are described elsewhere in this response. The material is stored in accordance with ANSI/ANS-8.1-1998 guidelines for slab thickness. The SPERT fuel has a maximum thickness of 8.7 cm, which is less than the 10 cm limit for 4.8% oxide fuel, and the 1.3% is in a slab less than 14.0 thick, which is less than the 28.0 cm limit. The ANSI/ANS-8.1-1998 limits provided in Figure 3 provide an adequate factor of safety for prevention of criticality in all credible normal and abnormal conditions, and the Purdue limits are more conservative.

Sturdy steel storage racks used in the Duncan Annex will keep the fuel geometries secure and prevent rearrangement of the fuel in all credible normal and abnormal scenarios, including fire, flood or earthquake. Even in the event of detachment from the concrete walls, the fuel will remain in the aforementioned slab geometry of controlled thickness.

CS-7) Describe the positions having responsibility for criticality safety, and their responsibilities and qualifications.

The following positions have responsibility for criticality safety under SNM-142.

Position	Education	Years Experience	Additional information
RSO	PhD Health Phys.	23	Certified Health Physicist
Dir. Rad. Labs	MS Nucl. Eng.	9	Facility Director for PUR-1 Senior Reactor Op., SOP 70468

CS-8) Explain what a "converter" is (geometry, material composition, fissile mass contents). The License Application makes reference to this equipment but does not describe it.

The converter has an active (fueled) height of 92 cm, and was used to modify the source spectrum of FBBF so that the neutron spectrum measured at the converter-blanket interface closely matched that at the core-blanket interface of a large Liquid Metal Fast Breeder Reactor (LMFBR) (See attached pictures). The converter has both an inner and outer region. The inner converter has an inner radius of 3.5 cm and an outer radius of 16 cm. The inner converter consists of 4.8% enriched UO_2 fuel rods with stainless steel cladding arranged in a close-packed, hexagonal pattern with the spaces between the fuel rods filled with boron carbide powder and housed in a welded-shut stainless steel container . The total number of rods in the inner converter is 528 rods. The outer converter has an outer radius of 22 cm and is loaded with 196 rods of the same type of fuel. The outer converter not only has 4.8% enriched UO_2 fuel rods but also sodium-filled, welded-shut, stainless steel cans. All free space is filled with boron carbide powder and is also housed in a welded- shut stainless steel container. There is one additional converter that contains 288 rods 4.8% enriched UO_2 fuel. All free space is filled with boron carbide powder and welded-shtu n a stainless steel can.

CS-9) Explain your statement in Section 9.2 of the License Application that "A minimum of one of these monitors [Remote Area Monitors] will be in operation at all times." Describe any compensatory measures that will be taken during in the event of any short periods of alarm outages for maintenance (including the allowed duration of such outages). 10 CFR 70.24(a) requires that each area will be covered by two detectors.

The current criticality monitors were put in place since there was active movement of fuel in the FBBF (Physics B28). Those operations have ceased and since the material in question is presently stored in a secure and fixed location, with geometry, shielding, and inventory unalterable, there is no credible scenario where a criticality accident could occur. As required by our license conditions, there is no movement of fuel so it is not credible that a criticality event would be possible. The area is not occupied and only trained individuals have access to this area. Therefore we request that this area be exempted from criticality monitoring under 10 CFR 70.24(a).

CS-10) Provide a detailed justification for your requested exemption to 10 CFR 70.24(a) for the Duncan Annex. The stated justification that the facility is only for fuel storage is not sufficient. 10 CFR 70.17 states that exemptions must be authorized by law, that they must not endanger life and property or the common defense and security, and must otherwise be in the public interest. Demonstrate that these criteria are met for the Duncan Annex. (You may refer to previous NRC approvals of this exemption, provided the bases for the exemption remain valid.)

As determined in the Safety Evaluation Report issued by the NRC on 30 July 1999, "The SPERT fuel rods and fuel rods are only authorized for storage, are being stored in safe slab geometry [according to ANSI/ANS-8.1-1998], and an inadvertent nuclear criticality is not possible", and based on the fact that the bases for this exemption remain valid, we are requesting a continuation of the exemption to 10 CFR 70.24(a).

CS-11) Explain your procedures for protecting health and safety in the event of a criticality accident.

Based on the determinations detailed in the Safety Evaluation Report for SNM-142 issued by the NRC on 30 July 1999, neither the material stored in Duncan Annex, nor the material stored in FBBF can become critical. Specifically, for the material in Duncan, "The SPERT fuel rods and fuel rods are only authorized for storage, are being stored in safe slab geometry, and an inadvertent nuclear criticality is not possible," and for the material in Physcis B28, the maximum calculated k-eff for FBBF was 0.74 under credible abnormal conditions as reviewed and approved by NRC staff (see question CS-3). Therefore, inadvertent criticality is not possible, and as such, there are no specific procedures for a criticality accident.

CS-12) Describe how you will ensure adequate emergency response, including testing of the criticality alarm system and conduct of evacuation drills. State whether you comply with ANSI/ANS-8.3-1997 and ANSI/ANS-8.23-1997, and explain any exceptions to these standards.

As stated in the answer to CS-11, a criticality accident is not possible under normal or abnormal conditions. Therefore we request an exemption to 10 CFR 70.24(a). If so exempted we would not be required to meet the requirements of ANSI/ANS-8.3-1997.

CS-13) Explain whether the liquid waste referred to in Section 11 of the License Application contains fissile material. If so, describe the measures for ensuring criticality safety. If not, describe the physical characteristics (form, composition, fissile mass) of this material.

The discussion of the liquid waste in section 11 of the application was the description of the whole Waste Management program at the university as set forth under the licenses granted under 10 CFR 20, 10 CFR 50 and the SNM license. Since the SNM covered by this license is in a sealed or sequestered form, there is no liquid waste containing fissile material generated by the storage of the material onsite.

FP-1) Describe the facility's building construction, fire area determination, electrical installation, emergency lighting, life safety/egress, ventilation, and lightning protection.

The facility is located in the Duncan Annex of the Electrical Engineering Building. The Duncan Annex was originally constructed in 1940 for use as a high voltage laboratory. In 1957, extensive modifications to the facility created space for nuclear engineering laboratories. The 1957 work created a basement, first and second floor in addition to a "ground floor" enclosed equipment mezzanine and an attic.

The building is type IIB construction. The floor area of the basement, first and second floors is approximately 7,400 ft^2; the area of the ground floor mezzanine is approximately 2,600 ft^2. The construction drawings for the 1957 renovation show the beams supporting the first floor were to be fireproofed. A shaft extends from the mezzanine through to the attic; ductwork is not protected with fire dampers, and a pair of doors opening from the mezzanine to room B70 are not rated.

The Duncan Annex is served at 12.47 kV by a three phase, 500 kVA, 139/240 V transformer and a single phase, 333 kVA, 120/240 V transformer located in a vault outside room B70; there is no access from the vault to room B70. The transformers were installed in 2007. Distribution switchgear is located in room 71 on the ground floor; this equipment was installed in the 1957 renovation. The facility does not have an emergency generator, and emergency egress lighting is provided by wall mounted, battery back-up emergency lights.

Occupants on the first and second floor of the Duncan Annex have access to two (2) stair enclosures on the northeast and southeast ends of the structure. Note that the northern stair enclosure (which also serves the basement) opens into a corridor on the ground floor (mezzanine) leading to an exit on the east side of the building. Openings from the mechanical and electrical spaces on the mezzanine into the corridor do not have a fire resistance rating.

Occupants in the basement have access to the northern stair enclosure and can reach a stair leading into the Electrical Engineering (EE) Building by moving through room B87. The southern stair enclosure can be accessed through room B70A; however access to B70A is restricted. There is a second route from the Duncan Annex to the EE building through room B70 and then through a storage tunnel and a machine shop to a stair enclosure and corridor before finally exiting on the south side of EE.

HVAC equipment for the Duncan Annex is located in a mechanical room on the ground floor mezzanine. The mechanical plans from the 1957 renovation do not show fire dampers in any of the ductwork serving the Annex. There is not a dedicated lightning protection system for the Duncan Annex.

FP-2) Describe the facility's fire protection features (suppression, alarm, detection, fire-rated walls/opening protection).

The Duncan Annex (and the Electrical Engineering Building generally) is protected by an automatic sprinkler system designed in accordance with NFPA 13. The Duncan Annex system was installed in 1995 with modifications made to the system in 1999. The space is not fully sprinklered; sprinklers were not installed in rooms B70A, B77A and B84 because of concerns about criticality potential. The design density for the basement floor is 0.15 gpm / 1500 ft^2 and includes a 100 gpm standpipe allowance. There are no standpipes in the Duncan Annex.

The existing fire alarm system in the EE Building was installed in 1987. Fire alarm initiating devices in the Duncan Annex are manual pull stations and waterflow switches; occupant notification is by audible and visual devices installed in accordance with NFPA 72. None of the walls in the annex appear to have been explicitly designed to provide a fire-resistance rating.

FP-3) Describe any inspection, testing, and maintenance of fire protection systems.

The building sprinkler system is tested semi-annually; tests include activation of flow switches at each floor zone as well as the building main flow switch. The dry pipe system is also tested on a semi-annual schedule. Fire alarm supervisory, initiating and notification devices are tested annually.

FP-4) Describe, by fire area, any potential combustible loading, possible fire scenarios, the potential consequences, and any mitigative controls.

The contents of room B77A are limited to the radioactive material stored in the space; this space does not have sprinkler protection. Room B77 is a sprinklered laboratory with two fume hoods and several lab benches. During our visit to the facility, approximately 10 liters of a flammable liquid was in use in the laboratory. Mitigative controls in the space are the sprinkler system in room B77 and the passive fire protection afforded by the non-combustible construction of the laboratory itself. Training staff and students to work safely with the materials present in the lab space also provides a degree of protection from the risk of fire. Rooms B77 and B77A do not have systems to limit the spread smoke or other products of combustion from a fire occurring in the space.

FP-5) Is the radioactive material situated at separate locations on the site? What amount(s) is/are located where? Describe the physical barriers separating the radioactive material from a single fire incident. Do these barriers have a fire rating?

Yes the radioactive material is located in a number of different areas. The SPERT fuel (4.8%) is stored in room B77A. The walls are constructed from 6" concrete block; the door to B77A is hollow metal and has an air transfer grill. Additional SPERT fuel is stored in converters (sealed stainless steel in Physics B-28). The construction of the Physics building is similar in nature to the Duncan Annex. Both natural and enriched uranium is stored in Physics B28 which is awaiting transfer to DOE and ultimate disposal. Sealed plutonium sources are housed in Duncan Annex B84.

FP-6) Are there any hazardous chemicals or processes which may contribute to the fire hazards in radiological areas?

There are no hazardous chemical processes or hazardous chemicals in excess of the maximum allowable quantity per control area which may contribute to the fire hazards in radiological areas.

FP-7) Is the facility compliant with NFPA 45, Standard for Fire Protection in Laboratory Facilities, and/or NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials?

The facility does not fully comply with either NFPA 45 or NFPA 801. The facility was in compliance with Indiana Building and Fire Codes at the time of construction.

FP-8) Describe the frequency and scope of any training for facility workers in response to a fire (fire extinguisher, safe shutdown, evacuation, etc.).

Facility personnel complete fire extinguisher training upon initial employment. All personnel are provided the Building Emergency Plan for the facility, which covers procedures on fire or other hazards and their evacuation plan, as well as shelter-in-place information. Additionally all staff responsible for oversight/storage of SNM and those using sources participate annually in the Purdue University reactor drill. These drills cover a wide variety of scenarios such as fires, explosions, injuries, etc. that would be useful in a response to an incident in the facility.

FP-9) Is the responding fire department located onsite? Describe the responding fire department's qualifications and training. Describe any pre-fire plan coordination with the responding fire department(s) (fire drills, preparation for hazardous materials response, etc.).

The Purdue University Fire Department is located on campus, and has an average response time of 2.3 minutes to the Duncan Annex.

The department is extensively trained in regards to fire suppression, as well as hazardous material emergency response. All of the firefighters (currently staffed with 29 personnel) are trained to a minimum of Firefighter I/II - this certification is in compliance with the National Fire Protection Agency standard 1001- Standard for Firefighter Professional Qualifications. Additionally, all firefighters are trained to Hazardous Material Technician, in compliance with National Fire Protection Agency standard 472- Competence of Responders to Hazardous Materials/Weapons of Mass Destruction Incidents. All firefighters complete annual radiation safety training, as well as monthly Haz-Mat training regarding response, detection, monitoring and mitigation procedures.

Pre-fire plan coordination involves routine building walkthroughs, during which special hazards or concerns are identified. Roundtable discussions are conducted on response duties and responsibilities regarding these issues. Additionally the firefighters participate on a routine basis in the reactor emergency drill.