Niagara Mohawk Power Corporation
Nine Mile Point Unit 2
Application for Amendment to Special Nuclear Material License

1.1 Reactor and Fuel

Pursuant to 10 CFR 70.21 and 70.22, Niagara Mohawk Power Corporation, et al*, hereby applies for an amendment to special nuclear material license SNM 1895 for Nine Mile Point Unit 2, Docket No. 70-2948. Niagara Mohawk is the holder of Construction Permit No. CPPR-112 issued by the Atomic Energy Commission on June 15, 1972. CPPR-112 permits construction of Nine Mile Point Unit 2 on the southeast shore of Lake Ontario in Oswego County, New York. Niagara Mohawk is an investor owned utility incorporated in the state of New York with its corporate headquarters located in Syracuse, New York. The company address is as follows: Niagara Mohawk Power Corporation, 300 Erie Boulevard West, Syracuse, New York 13202. Further information regarding Niagara Mohawk Power Corporation and Nine Mile Point Unit 2 is contained in the application for operating license for Nine Mile Point Unit 2 filed with the NRC (Docket No. 50-410). Nine Mile Point Unit 2 has been assigned the Reporting Identification Symbol XZU.

A. The receipt, possession, inspection and storage of: uranium enriched in the U-235 isotope contained in fuel assemblies.

B. The packaging of fuel assemblies for delivering to a carrier in accordance with 10 CFR Part 71.

It is hereby requested that the issued special nuclear material license be amended for Nine Mile Point Nuclear Station - Unit 2 to September 1987 or upon conversion of Construction Permit No. CPPR-112 to an operating license, whichever is earlier. It is currently anticipated that new fuel will be received onsite beginning in September 1985.

A fuel assembly is described in the Final Safety Analysis Report (FSAR) for the Nine Mile Point Unit 2, Section 4.3 (Docket Number 50-410) and on Tables 1, 2, and Figure 1 and 2.

The Nine Mile Point Unit 2 initial core is composed of the following:

92 fuel assemblies containing 0.711 wt.% U-235 which is 1.31 kilograms of U-235 per fuel assembly.

240 fuel assemblies containing 1.761 wt.% U-235 which is 3.23 kilograms of U-235 per fuel assembly.

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* Niagara Mohawk Power Corporation is submitting this application for itself, it's co-owners, Central Hudson Gas and Electric Corporation, Long Island Lighting Company, New York State Electric and Gas Corporation and Rochester Gas and Electric. Attachment 3 provides the principal officers of Niagara Mohawk.

432 fuel assemblies containing 2.191 wt.% U-235 which is 4.01 kilograms of U-235 per fuel assembly.

The average weight of uranium is 183.6 kilograms (U) per fuel assembly. The fuel assemblies contain no U-233, Pu, depleted uranium or thorium.

The inner dimension of the fuel assembly channel is 5.258 in. along the active fuel and the thickness is 0.100 in. The length of fuel assembly channels is 166.906 in. and the material is Zircaloy - 4.

Special nuclear material will be contained in a number of fuel assemblies which shall not exceed 800, although only 764 fuel assemblies are required and are planned to be delivered to load the reactor core of Nine Mile Point Unit 2. A license authorizing 800 assemblies is requested to allow for contingencies. The highest pin enrichment of the 36 spare fuel highest average assemblies will not exceed 3.00 wt. % U-235. The total quantity of U-235 (including 36 spares at 4.01 kilograms U-235 per assembly) is 2772.4 kilograms. Additional fuel data is shown on Table 1.

1.2 Storage Conditions

Construction of the Nine Mile Point 2 Fuel Storage Facilities has been completed, and preoperational testing of the handling equipment shall be complete prior to its use for handling new fuel. Detailed information regarding the fuel storage facility structures, systems, components and design basis is provided in the Nine Mile Point 2 FSAR Sections 9.1 (provided as Attachment 1). A general arrangement drawing of the Nine Mile Point 2 refueling floor is shown on FSAR Figures 9.1-25, 9.1-1a and 9.1-2 (Attachment 1).

New fuel is received in the fuel receiving area (Figure 3) and stored temporarily prior to being removed from the shipping crate. Each crate contains two fuel bundles supported by an inner metal shipping container. The 25 ton auxiliary hoist of the Reactor Building Polar Crane will lift three inner containers up through the equipment hatch to the refueling floor (Figure 4). Upon removal from the inner metal shipping container, the fuel assembly is transported to the new fuel inspection stand for channelling and inspection. The assemblies are then transported to either the new fuel storage vault or the spent fuel pool. The fuel may be stored either with or without its channel. For analysis purposes, the NRC should conservatively assume the fuel is not channeled.

The new fuel will be temporarily stored in shipping containers on the refueling floor before being inspected and placed in either the new fuel storage vault or the spent fuel storage pool (refer to Section 2.1 for new fuel inspection surveillance requirements and acceptable limits). The shipping container is a reuseable metal container designed for shock and vibration isolation, humidity control and leak tightness, to protect fuel assemblies from damage during normal handling and shipping at temperatures from -40°F to +150°F. Each container may contain one or two fuel assemblies.

The container lateral clearance dimensions are 31" high by 31" wide. A 216" long shipping container is used to ship the fuel assemblies and weighs approximately 3000 pounds fully loaded.

Transient combustibles are not stored in the storage vault and combustible materials are controlled in the area. A manual fire fighting system of appropriate capacity and capability is present should combustible material be inadvertently introduced.

Adequate ventilation (from permanent or temporary equipment) will be provided to ensure proper cleanliness of the fuel inspection and permanent storage areas. Any high levels of radioactivity released are detected by the installed (or temporary) area radiation monitors.

New Fuel Storage Facility

The design of the new fuel facility is based on the following criteria:

- General Design Criterion 2 - Design for protection against natural phenomena;
- General Design Criterion 3 - Fire Protection;
- General Design Criterion 4 - Environmental and missile design bases;
- General Design Criterion 5 - Sharing of structures, systems and components;
- General Design Criterion 61 - Fuel storage, handling and radioactive control;
  a. Capability of periodic inspection;
  b. Shielding for radiation protection;
  c. Provisions for containment and confinement;
- General Design Criterion 62 - Prevention of criticality in fuel storage and handling;
- General Design Criterion 63 - Monitoring fuel and waste storage.
- Regulatory Guide 1.29

The Nine Mile Point 2 Nuclear Station will have new fuel storage racks located within a New Fuel Storage Vault. Both the New Fuel Storage Vault and the new fuel storage racks are designed as Category 1 structures. The new fuel storage racks are arranged to provide dry storage for 270 new fuel assemblies. The new fuel storage vault contains 27 sets of racks, each of which may contain up to ten fuel assemblies. These racks provide support for the fuel assemblies. The minimum center-to-center spacing for the fuel assembly between rows is 11 inches (and the nominal is 12.25 inches). The minimum center-to-center spacing within the rows is 7.00 in. Fuel assembly placement between rows is not possible. The new fuel storage racks are constructed of aluminum with guides which provide for easy entry of the assemblies into the racks. The racks, along with additional restraints, restrict lateral movement of the fuel assemblies

during storage. New fuel racks have 32 covers. There are two covers for each row. See Figure EV-160A. Each aluminum cover plate covers 10 fuel assemblies in the new fuel vault. The maximum number of fuel assemblies that will be uncovered at any one time in the new fuel vault is 10.

The following accidents (see Attachment 1) are considered in the criticality design of the new fuel storage area:

- Flooding: Complete immersion of the entire array in pure, unborated, room temperature water.
- Envelopment of the entire array in a uniform density aqueous foam of optimum density (that density which maximizes the reactivity of the array), for example as a result of fire fighting.

Accidents resulting in an increase in K_{eff} because of geometrical changes of the racks or fuel handling accidents are not considered credible due to the following design bases:

- The facility is designed in accordance with GDC 2, 3, 4, 5, 61, 62 and 63.
- The racks are designed to Seismic Category 1 requirements.
- The racks and anchorages can withstand the maximum uplift force available without a significant change in geometry.
- The design of the Fuel Handling System and administrative procedures will ensure subcritical spacing of fuel assemblies.

The introduction and retention of moderators into the facility is prevented by the following:

- The storage of the new fuel within the New Fuel Storage Vault precludes flooding of the new fuel assemblies by the probable maximum flood (PMF) and the probable maximum precipitation (PMP).
- There is no piping routed through this area, the rupture of which could introduce moderator into the storage vault.
- The floor of the new fuel storage vault is sloped to a nonclosable drain located at the low point. This drain removes any water that may be accidentally and unknowingly introduced into the vault. The drain is part of the reactor building equipment drains.
- Administrative policy has been established to preclude the use of hydrogeneous foam fire fighting material into the storage vault. Extinguishers provided for this use are of the dry chemical or CO_2 type and halon (see page 5).

Two gamma area radiation monitors, which alarm both locally and in the control room area upon the detection of preset radiation levels, will be used to monitor the New Fuel Storage Vault and the area of the spent fuel pool. One radiation monitor is located in the New Fuel Storage Vault; the other is located immediately adjacent to the top opening vault cover plates. These gamma radiation monitors are self testing in that each has an installed checksource which may be remotely operated at any time and may also be programmed to automatically actuate at prescribed intervals. Should either unit become inoperable due to an inadequate checksource response, detector failure (loss of counts), high voltage out of range or detector saturation, indication of the failed state is provided in the Control Room. These monitors will be source checked monthly, channel functional test performed semi-annually and calibrated once per operating cycle (once per 731 days). In the event that the above radiation monitors are not operable, portable instruments having both local read out and alarm capabilities will be used.

The use of hydrogenous fire fighting foam in this fuel storage area is excluded. However, hose stations are provided for fighting fires which could occur in the fuel receiving area. Dry chemical or CO_2 and halon fire extinguishers are also provided for this purpose, and their use is encouraged while the hose stations are used as a backup. Training and administrative controls will preclude the fire fighting crews from using these hose stations to spray water into the new fuel vault or from spraying the fuel receiving areas if new fuel is being transferred from the shipping container to its storage location. Transient combustibles will be stored at a distance of 20 feet from the new fuel fault. In addition, the new fuel racks are designed so that water disperses through all the racks and cannot accumulate in any given single rack.

Procedural controls are established to eliminate sources of moderation around the new fuel vault in order to preclude the possibility of achieving a uniform moderator distribution in the peak range of approximately 0.1 to 0.2 g/cc water. Noncombustible covers provide a means to preclude exceeding k-eff = 0.98 with optimum moderation by limiting the individual array sizes. The maximum number of fuel assemblies that will be uncovered at any one time in the fuel vault is 10. The basis for these controls are described below:

The controls were based on a series of generic analyses which demonstrated that an infinite array of high reactivity fuel stored in a new fuel vault will exceed k-eff = 0.98 for water densities ranging from approximately 0.05 to 0.60 g/cc, with the peak occurring between 0.1 and 0.2 g/cc. This is precisely the moderator range over which radial and axial neutron leakage effects become important.

These generic calculations, performed in the early 1970s, applied a leakage correction to the infinite array for several different new fuel vault array sizes. In addition, actual new fuel reactivities were assumed. The results could not justify that all new fuel designs could be stored in all the new fuel vault configurations. There are dual unit plants which share a common new fuel vault resulting in an array which is double that of a single unit vault. The associated k-eff was at or exceeding 0.98 without accounting for uncertainties. As a result, GE recommended the use of plant procedural controls to preclude the optimum

moderator density condition rather than show compliance to k-eff = 0.98 under all conditions. These recommendations were first presented to the utilities via SIL-152 issued in March 1976.

NMPC has followed SIL-152 to eliminate sources of moderation from the immediate area of the new fuel vault. In addition, NMP2 has metallic noncombustible covers over the new fuel vault. These covers allow access to a group of fuel bundles which, if sprayed with an optimum density moderator, could not exceed k-eff = 0.98 due to the small critical array size. These covers are not required to be water tight at the connections since water "sheeting" does not constitute an optimum moderator condition.

In summary, the intent of the procedural controls are to eliminate sources of moderation around the new fuel vault in order to preclude the possibility of achieving a uniform moderator distribution in the peak range of approximately 0.1 to 0.2 g/cc water. Non-combustible covers provide a means to preclude exceeding k-eff = 0.98 with optimum moderation by limiting the individual array sizes. Realistic three dimensional Monte Carlo analysis accounting for neutron leakage have shown that k-eff remains below 0.98.

The following firefighting systems shall be functional.

1. The construction or permanent Gaitronics fire alarm system.
2. A fire pump capable of a minimum of 1000 gpm at 125 psig or the permanent fire pump.
3. The wet pipe sprinkler systems in the railroad passageway.
4. Hose reels in the reactor building.
5. Fire detector zones 242NW and 281NZ, or a fire watch.

Spent Fuel Storage Facility

The design bases of the spent fuel storage facility are the following:

- The prevention of criticality during storage
- The prevention of damage of the fuel
- Adequate radiation shielding
- Protection against radioactivity release
- Adequate monitoring of the fuel storage
- Ability to withstand design seismic loads

The fuel assemblies are held in a vertical position by the spent fuel pool storage racks (FSAR Figure 9.1-4 in Attachment 1). The fuel assemblies are supported within the fuel storage racks by a stainless steel plate located approximately six inches above the fuel pool floor. The spent fuel racks absorber material is Boraflex.

The composition of Boraflex is as follows:

Boron Carbide (B_4C) Powder in Silicon Rubber (Slygard 170A and 170B)

The finished product B10 areal density is 0.028 gm/sq. cm.

Percent by weight - Elastomer - 51.08%
(typical) Boron carbide - 48.91%

Minimum B-10 Content - .1850 weight % (laboratory evaluation)

Total boron content in B_4C - 0.788 weight %

Boraflex density - 1.7259 g/cc

Density of B_4C - 2.48 g/cc

The spent fuel rack supplier, U.S. Tool and Die (UST&D), procured the Boraflex and manufactured the racks to SWEC Specification No. NMP2-P232G. The UST&D Quality Assurance Program requires documented receipt inspection and verification that the Boraflex received by UST&D is in agreement with the purchase order. In-process inspection throughout fabrication of the rack requires that all manufacture and assembly be in accordance with the approved drawings. The Quality Control Department cannot accept components that are not in accordance with the approved drawings. Capture of the Boraflex is ensured by the design, which provides local raised box wall areas, where the boxes are welded together, between which the Boraflex sheet is captured. In addition, capture at the bottom is ensured by a stainless steel strip welded to one of the two adjacent boxes.

There will be provisions for approximately 4000 fuel assembly storage spaces for spent fuel.

Approved fuel handling procedures will be used for movement of new and irradiated fuel. Reactor Analyst instructions will provide spent fuel storage rack and reactor core locations for all fuel bundles. These instructions, when completed, are reviewed by the Reactor Analyst and provide a record of bundle storage locations.

In accordance with Administrative Procedure 3.6 "Special Nuclear Material Control Procedure", the Station Superintendent on the unit affected is responsible for preparation and approval of detailed procedures required to ensure safe handling of SNM at the station.

In accordance with Table 7.1 of Administrative Procedure 2.0 "Production And Control of Procedures", fuel handling procedures are approved by the General Superintendent, the Station Superintendent, and the Reactor Analyst Supervisor. In addition the Chemistry and Radiation Management and Operations (S.R.O.) departments will review fuel handling procedures prior to approval. Other safety related procedures will be reviewed and approved in accordance with Table 7.1 of AP-2.0. (Attachment A)

The Reactor Building Polar Crane or jib crane with a general purpose grapple is used to transport new fuel assemblies from the New Fuel Vault to the Spent Fuel Pool and lowers the fuel assembly to the pool bottom so that it may be picked up by the telescoping refueling grapple. Limit switches are provided on the refueling grapple to maintain approximately 8'-6" of water above active fuel. However, the initial core new fuel may be loaded into the reactor vessel without water in the spent fuel pool.

The heaviest objects which could possibly be moved over the spent fuel pool racks are the two spent fuel pool weir gates and cask storage pit gates. Overhead loads greater than 1,000 lbs. will be administratively controlled when being moved in the vicinity of fuel storage areas. The spent fuel pool racks are designed to protect stored fuel assemblies from damage resulting from a dropped fuel assembly. When subjected to this impact, those members that maintain spacing to ensure k_{eff} 0.95 remain intact.

1.3 Security and Physical Protection

The Security Plan and Contingency Plan associated with storage of new fuel will be submitted to the NRC for review under separate cover.

1.4 Transfer of Special Nuclear Material

The General Electric Corporation is responsible for the shipment of new fuel to the Nine Mile Point Unit #2 Nuclear Station.

Should the need arise for the NMP-2 Nuclear Station to package and transport new fuel, the station would ship the new fuel in General Electric Corporation owned new fuel shipping containers. This would be in accordance with the provisions of 10CFR Part 71 and DOT regulations.

1.5 Financial Protection

Nine Mile Point Unit 2 is covered under the existing policy for Nine Mile Point Unit 1.

Nine Mile Point Unit 2 has a Nuclear Energy Liability Policy with American Nuclear Insurance under policy #NF161 and Mutual Atomic Energy Liability Underwriters policy #MF46. These policies have been enforced since 1969 and are reviewed and submitted to the NRC yearly.

2.1 Radiation Control [1]

The qualifications of those personnel responsible for the control of Special Nuclear Material at Nine Mile Point #2 are provided in Attachment 2.

Training and qualification of personnel in Radiation Protection in accordance with 10CFR19.12 are the responsibility of the Supervisor of Chemistry and Radiation Protection and are performed under his direction.

All administrative aspects of training, such as scheduling and documentation are handled by the Nuclear Training Department. The Nuclear Training Department also administers the general standardized Radiation Protection Training, as approved by the Superintendent of Chemistry and Radiation Management.

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1 Should any of this equipment not be totally functional at Nine Mile Unit 2 upon receipt of material covered under this license, provisions shall be made to use similar equipment and facilities capabilities onsite at the Nine Mile Unit 1.

The C&RM Department also conducts the Offsite Radiological Monitoring Program and Emergency Planning for the station.

The three basic objectives of the Health Physics Program at Nine Mile Point are to:

- Protect station personnel
- Protect the public
- Protect the station

Protection of personnel means surveillance and control over the internal and external radiation exposure of personnel and maintaining the exposure of all personnel within permissible limits (10 CFR 20.101 and 10 CFR 20.103), and as low as reasonably achievable in compliance with applicable regulations and license conditions.

Protection of the public means surveillance and control over all station conditions and operations that may affect the health and safety of the public. It includes such activities as radioactive gaseous, liquid and solid waste disposal and the shipment of radioactive materials. It also involves conducting an environmental radioactivity monitoring program and maintaining an effective emergency plan.

Protection of the station means the continuous determination and evaluation of the radiological status of the station for operational safety and radiation exposure control purposes. This work is done in order to warn of possible detrimental changes and exposure hazards, to determine changes or improvements needed and to note trends for planning future maintenance work.

Duties concerning radioactive liquid, gaseous and solid waste disposal are performed under Chemistry and Radiation Management (C&RM) Department direction. The detailed analyses and records required to characterize the nature of these releases, both qualitatively and quantitatively, are under the control of the C&RM Department. In addition, solid waste disposal and shipments of radioactive materials are under the control of the C&RM Department.

The program organization is as follows:

The General Superintendent - Nuclear Generation is responsible for the protection of all persons against radiation and for compliance with NRC regulations and license conditions. This responsibility is in turn shared by all supervisors. Furthermore, all personnel are required to work safely and to follow the regulations, rules and procedures that have been established for their protection.

The Superintendent of Chemistry and Radiation Management (Radiation Protection Manager - RPM) establishes the Health Physics Program and develops the health physics procedures for NMP-2 that are designed to assure compliance with applicable regulations, licenses and regulatory guides. He also provides technical guidance for conducting this program, audits the effectiveness and the result of the program and modifies it as required. He also provides technical assistance to the General Superintendent - Nuclear Generation, who has management authority to

implement the "as low as reasonably achievable" (ALARA), occupational exposure policy, to which Niagara Mohawk Power Corporation is committed.

The Superintendent of Chemistry and Radiation Management (RPM) has the qualifications equivalent to those required by Regulatory Guide 1.8.

The Supervisor of Chemistry and Radiation Protection is responsible for conducting the Health Physics Program that has been established for the station. This supervisor has the duty and the authority to measure and control the radiation exposure of personnel to a level that is as low as reasonably achievable and within regulatory exposure limits; to continuously evaluate and review the radiological status of the station; to make recommendations for control or elimination of radiation hazards; to train personnel in radiation safety; to assist all personnel in carrying out their radiation protection responsibilities; and to protect the health and safety of the public both on-site and in the surrounding area.

In order to achieve the goals of the Health Physics Program and fulfill these responsibilities for radiation protection, radiation monitoring, survey and personnel exposure control work are performed for all station operations and maintenance. The extent of this surveillance is outlined below.

The Health Physics section performs the major portion of the Health Physics work for the station. Personnel in the Health Physics section normally work on the day shift, five days a week, during periods of routine operation; and deploy onto the other shifts for major maintenance, shutdown and the refueling work. The Health Physics Section is organized into two units, each headed by a supervisor. These units are: (1) Operations and (2) Support Functions.

For the purpose of defining and assigning work to be performed by the operating shifts and the Health Physics Sections, the routine station radiation surveillance work can be described as consisting of radiation monitoring, radiation survey, radiation exposure control and radioactive waste disposal activities.

The Radiation Protection Technicians perform radiation monitoring and exposure control work for the routine and special shift operations. This work is performed under the direction of the appropriate Radiation Protection Supervisor. Radiation Protection Instructions (RPIs) and/or assignments prepared by appropriate Radiation Protection Supervision, will designate routine work to be performed. Radiation Protection Technicians receive training in the use of equipment and procedures for dealing with radiological concerns and job related accidents.

The C&RM Department also performs essentially all of the work necessary to calibrate and maintain (other than repair) the Counting Room instruments and the portable radiation monitoring instruments.

The chemistry and radiochemistry laboratory where radioactive samples are chemically analyzed and/or prepared for radiochemical analysis is located in the Unit 1 Turbine Building at El. 261 ft. The laboratory is equipped with fume hoods with filters, an emergency shower, an eyewash and

miscellaneous chemistry laboratory equipment for chemical analysis. The laboratory supports routine chemistry and radiochemistry analysis. In addition, satellite laboratory facilities are located at Unit 2 in the

Turbine Building sample room El. 250' and in the Radwaste Building sample room El. 261'. Radiation-measuring instrumentation is calibrated and repaired at the Unit 1 instrument calibration laboratory located in the Unit 1 turbine building at El. 261'. These facilities are equipped for conducting the health physics and chemistry programs for the station, for detecting, analyzing and measuring all types of radiation and for evaluating any radiological problem that may reasonably be expected. Equipment for preparing environmental radioactivity samples and for radiobioassay is also included. Measurements for internal personnel dosimetry purposes are performed on site.

Should the Unit 2 Health Physics facilities not be totally functional upon receipt of the materials covered under this license, temporary provisions shall be made to ensure that adequate capabilities are provided. Facilities would be organized within 1) the Unit 1 RP office area and/or 2) in those Unit 2 station areas that have been completed prior to receipt of materials.

Additional Facilities and Access Provisions

Storage areas and/or change room facilities are provided where personnel obtain clean protective clothing and equipment for station work. A personnel decontamination facility is located in the turbine building at both el. 306 ft. and 250 ft. These two facilities are strategically located at main access points and each is equipped with a dressing area, sink and shower. A personnel decontamination facility is also located in the auxiliary service building south at el. 261 ft. Locker rooms and toilet facilities for men and women are located in the auxiliary service building south at el. 261 ft. Laundry facilities for decontamination of protective clothing and respiratory equipment are located at the Nine Mile Point Unit 1 turbine building at el. 261 ft. Equipment decontamination facilities are also provided at the station for large and small items of station equipment, components and tools. This facility also contains equipment for repair of contaminated plant components.

Drains from all these facilities go to appropriate radioactive liquid waste drain tanks. Written procedures govern the proper use of protective clothing, the change rooms and the decontamination facilities.

Contamination control checkpoints that are equipped with appropriate monitoring instrumentation are located at the main access points. Additional exits with appropriate monitoring instrumentation are provided to support work as deemed necessary. All other personnel access points into the controlled access areas in the Restricted Area are protected by restricted-in/free-out doors and are for emergency exits only. Stairs and elevators are provided for personnel access from one elevation to another. Contamination control checkpoints will be appropriately located and strategically placed throughout the Restricted Area to prevent the spread of contamination within the area.

Before leaving the Restricted Area, personnel are required to monitor themselves with personnel friskers (thin window G-M detectors (count rate meters) positioned near each exit door) to make sure that they are free of significant contamination.

Portable and Laboratory Equipment

Different types of instruments are selected to cover the entire spectrum of radiation measurement requirements expected. This includes instruments for detecting and measuring alpha, beta, gamma and neutron radiation. These consist of Counting Room and portable radiation survey/monitoring instruments. These instruments are required to provide protection against radiation for station personnel (for surveys required by 10CFR20.201); to control the release of effluents for the protection of the health and safety of the public; and to provide for all other radiological measurements necessary for personnel and public safety and for the protection of property. Sufficient quantities are available to allow for use, calibration, maintenance and repair.

Instruments available on site for radioactivity measurements include the following:

- Computer-based multi-channel gamma analyzer with multiple Ge(Li) type detectors, used for identification and measurement of gamma emitting radionuclides in samples of reactor primary coolant, liquid and gaseous waste, airborne contaminants and similar samples. This instrument will be calibrated quarterly.

- Beta sample counter used primarily for gross beta measurements of surface contamination on swipes. This instrument will be calibrated quarterly.

- Alpha counter-scaler used for gross alpha measurements such as uranium or plutonium in reactor water samples or alpha contamination from surface or air samples. This instrument will be calibrated quarterly.

- Shielded body burden and thyroid burden analyzer used for measurement of possible internally deposited radioisotopes for determination of internal dose of personnel. This instrument will calibrated quarterly.

Portable radiation survey and monitoring instruments for routine use are selected to cover the entire range from background to high levels for the radiation types of concern. These include (with nominal range characteristics and indicated):

- Beta-gamma meters (Geiger counters 0-50,000 CPM) used for detection of radioactive contamination on surfaces. These instruments will be calibrated quarterly.

- GM survey meters (0-1000 R/hr) with telescoping probe used to cover very high radiation ranges with the lowest possible radiation field to the operator. These instruments will be calibrated quarterly.

- Low and high range beta-gamma ionization chamber survey meters (0 mR/hr - 50 R/hr) used to cover the general range of dose rate measurements necessary for radiation protection purposes. These instruments will be calibrated quarterly.

- Neutron survey meter (BF_3 tube within cadmium-loaded polyethylene sphere, 0-5000 mrem/hr) used to give reasonably correct dose equivalent response to fast neutrons. This instrument will be calibrated once per year.

- Alpha scintillation counters (0-500,000 CPM) used for measurement of alpha contamination on various surfaces that may result from any uranium or plutonium in the reactor water, for example. These instruments will be calibrated semi-annually.

The Process Monitoring System is relied upon for continuous monitoring of airborne radioactivity. This is supplemented by grab air samples collected and analyzed by Radiation Protection during maintenance and routine and abnormal operations where airborne radioactivity may be involved.

Airborne gaseous, particulate and iodine samplers are also available for routine use as well as an assortment of special purpose and emergency type radiation survey instruments including bubblers for tritium, gas sample containers, low volume air samplers, particulate filters and activated charcoal and silver zeolite cartridges. All of this equipment is kept in areas under control. Necessary emergency instruments are also located onsite and at a remote assembly point.

In addition to the portable radiation monitoring instruments, beta-gamma count rate meters are located at exits from the Restricted Area. These instruments are intended to prevent any contamination on personnel, materials or equipment from being spread to the unrestricted areas of the station. Appropriate monitoring instruments are also available at various locations within the Restricted Area for contamination control purposes. Portal monitors (scintillation detectors) are also utilized, as appropriate, to monitor personnel leaving the station.

All of the above instruments are subjected to initial operational checks and calibration prior to use and to a continuing quality control program to assure the accuracy of all measurements of radioactivity and radiation levels. These instruments are recalibrated with standards whenever their operation appears statistically to be out of the accepted limits. In addition, routine calibrations are performed periodically on all of this equipment and after all repairs. A shielded calibrator with a range capable of exposure rates from essentially background to hundreds of R/Hr is used for calibration of radiation monitoring instruments. Also available are small (mCi level) sources for certain low level calibrations and a nominal 0.5 gr. Pu-Be neutron source for neutron instrument calibration checks. The gamma sources are calibrated with an exposure rate meter traceable to the National Bureau of Standards.

The whole body counter is calibrated using uniformly distributed sources of radionuclides of concern (CO 60, Cs 137 and Ba 133 (for I 131)) within a tissue equivalent phantom. This detector is used in conjunction with a computer-based multi-channel gamma analyzer and associated readout to obtain a permanent record.

A record of all calibrations is kept and personnel dosimetry, survey and monitoring records, etc. are maintained as required by NRC regulations.

The instrument storage room used for storage of portable survey instrumentation, respiratory equipment and radiation protection supplies is located in the turbine building at el. 306 ft.

The station is divided into controlled access areas for the purpose of protecting individuals from radiation exposure. Access control can be accomplished by barricaded areas, locked barriers such as doors or gates or a posted guard. Access control within the fenced area of the site for radiological purposes is determined by the radiation level, contamination level or the presence of radioactive materials. Controlled areas are posted as required by 10CFR20. Posted areas include the following: 1) Restricted Area - Radioactive Materials, 2) Radiation Area, 3) High Radiation Area, and 4) Airborne Radioactivity Area. Administrative controls are also used in conjunction with the above.

A Radiation Work Permit system, governed by written procedures outlining proper use, is also utilized to control access within the restricted area. Radiation Work Permits are approved and signed by a Station Shift Supervisor prior to their use.

Personnel who are required to utilize protective clothing obtain these items in the Change Rooms and/or from designated storage areas. They first consult the Radiation Work Permit, remove street clothes, don the required protective clothing and then proceed to the job location. After completing work, they remove outer contaminated protective clothing at the exit of the Radiation Control Zone previously established for the work area. They then proceed to the Change Room, where they monitor themselves; and then proceed to put on their personal clothing and leave.

All persons entering the Restricted Area of the station shall receive training pursuant to 10CFR19.12, and wear the personnel monitoring equipment (film badge, TLD, pocket dosimeters, etc.) prescribed by Chemistry and Radiation Management Department personnel in accordance with NRC regulations and shall comply with applicable Radiation Work Permits.

All work on systems or in locations where certain radiological conditions exist, as determined by established Radiation Protection procedures, require a specific Radiation Work Permit (RWP). RWPs are issued by qualified radiation protection personnel under the supervision of the Supervisor, Chemistry and Radiation Protection, and must be approved by the Station Shift Supervisor on duty. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit. The Radiation Work Permit lists the precautions to be taken including radiation levels (for external and internal exposure), protective clothing to be worn and any radiation monitoring that may be required during the

performance of the work. The permit is issued to the people who perform the work; a copy is available for the Shift Supervisor and a working copy is maintained by personnel in the Radiation Protection Section.

All personnel working under a permit are required to read the instructions on the permit and to fill out the information necessary before and after entering the work area. The information from the permit is entered into computer programs and serves, in part, as a personnel monitoring record for the individuals involved.

In accordance with 10CFR20.202, personnel monitoring equipment consisting of film badges, thermoluminescent dosimeters (TLDs) and self-reading pocket dosimeters, as appropriate, are assigned by the Chemistry and Radiation Management Department and worn by all personnel (employees and visitors) whose jobs involve radiation exposure as defined in 10CFR20. The film badge readings are normally used as the official station record. Additional personnel dosimetry equipment such as high range self-reading pocket dosimeters and extremity TLDs may be assigned as needed depending on the radiological conditions encountered.

Administrative limits of 1000 mrem per quarter and the use of a computer dose tracking program are used to ensure that the limits of 10 CFR 20.101 are not exceeded.

Personnel whose jobs require them to frequently enter the Restricted Area of the station may ordinarily be assigned a permanent personnel monitoring film badge and a dosimeter, whereas personnel working under a specific Radiation Work Permit in a job situation where a sizable fraction of the quarterly allowable dose may be received in a relatively short period of time may additionally be assigned a high range self-reading dosimeter and/or extremity monitoring equipment, depending on job conditions. Extremity monitoring equipment is used for jobs or situations where extremity dose is expected to be limiting or controlling in accordance with station procedures. The use of additional personnel monitoring equipment beyond that routinely used, depends on the job and on existing radiological conditions as evaluated and determined by Chemistry and Radiation Management Department personnel.

Neutron exposure is monitored in accordance with Regulatory Guide 8.14 through the use of dose rate/stay time calculations or neutron/gamma dose rate ratios.

Records of radiation exposure history and current occupational exposure are maintained by the C&RM Department for each individual for whom personnel monitoring is required. The external radiation dose to personnel is determined on a daily basis by means of self-reading pocket dosimeters. Personnel monitoring badges (film badges & TLDs) are processed at least twice per month. If necessary, they may be procesed more frequently.

A whole body counting system for routine screening of personnel to determine internal exposure is available on site. Outside services for radio-bioassay and whole body counting may be used as required for backup and support of the program. The station equipment is sufficiently

sensitive to detect in the thyroid, lungs or whole body a small fraction of the permissible body burden for those gamma emitting radionuclides expected.

Whole body counts are performed as soon as practicable on individuals entering a bioassay area, who have been newly assigned a visitor or permanent film badge or who are terminating employment or assignment at the Nine Mile Point site. In addition, all permanently badged individuals are required to receive at least one whole body count every two years.

Anyone on site, whether badged or not, who was involved in a radiological accident where internal exposure was likely, would be given a whole body count as soon as practicable. If radioactive material uptake had occurred, appropriate action would be taken.

Film badges are supplied and processed by an outside vendor. Quality assurance is provided by the use of badges which are "spiked" in-house and then processed by the vendor along with the regularly issued badges.

Thermoluminescent dosimeters (TLDs) are supplied by a central in-house service which is responsible for the calibration and maintenance of all TLD and TLD readout equipment. Self-reading pocket dosimeters are calibrated and leak tested at the station in accordance with station procedures.

Should the permanent facilities not be totally functional upon receipt of materials covered under this license, temporary provisions shall be made which adequately provide for protective clothing, personnel decontamination and equipment decontamination.

The site counting room facilities are shielded on all sides to facilitate low level counting work. The instrument calibration room also has shield walls. In addition, extensive shielding of components has been utilized throughout the station for the protection of personnel, both for routine operation and for maintenance.

Special "protective" and "anti-contamination" clothing is furnished and worn as necessary to protect personnel against contact with radioactive contamination. This consists of coveralls, lab coats, surgeon caps, hoods, gloves and shoe covers. Change areas will be conveniently located within the station for proper utilization of this protective clothing. Approved respiratory protective equipment is also available to supplement process containment and ventilation controls, for the protection of personnel against airborne radioative contamination and the possibility of internal radiation exposure. This equipment consists of full face air-purifying respirators and self-contained breathing apparatus providing protection factors allowed per 10 CFR 20.103. Also, a breathing air system is being installed in the station, and respiratory protective equipment consisting of air-line full-face respirators, hoods and plastic suits are available, should their use become necessary or desirable.

Maintenance of the respiratory equipment is in general accordance with the manufacturer's recommendations and rules of good practice, such as those in NUREG 0041 entitled "Manual of Respiratory Protection Against Airborne Radioactive Materials." The use and maintenance of protective clothing and respiratory protective equipment is under the direct control of the Chemistry and Radiation Management Department. The Chemistry and Radiation Management Department and personnel are trained in the use of this equipment before using it in the performance of their work.

New Fuel Inspection

Upon receipt of the new fuel shipment(s), a general area beta and gamma radiation exposure level survey and a general area loose contamination survey (smear survey) analyzed for alpha and beta-gamma activity will be performed on the transport vehicle and the new fuel shipping crates. Smear surveys of the fuel containers and fuel bundles will be performed as each layer of wrapping is removed. The last smear survey will be of the fuel pins and other structural parts of the fuel bundle. These smear surveys will also be counted for alpha and beta-gamma activity. Radiation Protection will provide continuous surveillance during handling and inspection of the new fuel.

In the unlikely event loose contamination greater than 10 dpm/100 cm^2 alpha and/or 100 dpm/100 cm^2 beta-gamma above background is detected, the person(s) monitoring the shipments will contact the Supervisor of Chemistry and Radiation Protection or his designee. The Supervisor of Chemistry and Radiation Protection or his designee will take the appropriate action to assure that the contaminated item is controlled.

2.2 Nuclear Criticality Safety

Applicable qualifications of personnel are shown on Table 1. The program organization is as follows:

General Superintendent Nuclear Generation

The General Superintendent Nuclear Generation is directly responsible for the safe and efficient operation of all generating units on the site. He is responsible for safeguarding the general public and all onsite personnel from radiation exposure and for adherence to the operating licenses and technical specifications. His duties include responsibility for the fire protection program and for implementation of the emergency plan and procedures.

Site Superintendent Maintenance Nuclear

The management of station electrical, mechanical and structural maintenance is under the direction of the Site Superintendent, Maintenance Nuclear. He is assisted in his duties by the Superintendent Electrical Maintenance Nuclear and the Superintendent Mechanical Maintenance Nuclear.

Technical Superintendent Nuclear

Management of the technical and clerical staff is under the direction of the Technical Superintendent Nuclear who is responsible to the General

Superintendent Nuclear Generation. He is responsible for the proper execution of technical and clerical work in the station. He has under his direction the Superintendent Technical Services, the Fire Protection Supervisor, the Planning Coordinator, the Office Supervisor, and the Superintendent Inservice Inspection, Nuclear.

Superintendent Technical Services Nuclear

The Superintendent Technical Services Nuclear provides technical and administrative guidance for instrumentation and control, the Reactor Analyst, Computer Operations and Maintenance and the Site Technical Support Staff.

Supervisor Reactor Analysis

The Supervisor Reactor Analysis and the Unit Supervisor Reactor Analysis are on the Site Technical Staff and assist in the operation of station on a functional basis. They provide guidance for fuel management. They maintain performance and fuel accountability records.

Station Superintendent

The Station Superintendent is responsible to the General Superintendent for the safe, orderly and efficient functional operation of his station and the plant modifications. The Station Superintendent also is responsible for radiation protection and fire protection at his station.

The operating forces report both functionally and administratively to the Superintendent. General direction of operations is provided through the Supervisor Operations and Assistant Supervisor Operations who in turn direct the activities of the Shift Supervisors.

Supervisor Operations

This Supervisor directs the functional conduct of shift operations and, when required, performs the duties of the Station Shift Supervisor. He shall hold an NRC Senior Reactor Operator license for his station. In the absence of the Station Superintendent, he is designated to act as Station Superintendent.

Assistant Supervisor Operations

This Supervisor assists the Supervisor Operations in the functional conduct of shift operations. He shall hold an NRC Senior Reactor Operator license. In the absence of the Supervisor Operations Nuclear, he is designated to act in his behalf.

Station Shift Supervisor

The Shift Supervisor is in charge of all operations on his assigned shift. Under the general direction of the Supervisor Operations, his function includes direction of shift activities, authorization of equipment releases for maintenance, ensuring that the plant is operated safely and within the license and technical specifications and ensuring

that plant operations are conducted in accordance with approved procedures. As overall supervisor of operations for his shift, the Station Shift Supervisor should avoid becoming personally involved in the manipulative tasks or details of operation of any one portion of the plant, so that he may retain a comprehensive perspective of general station conditions at all times. In an emergency situation, however, should the Shift Supervisor choose to perform manipulative functions to ensure that the plant is in a safe condition, he shall coordinate his actions with the Chief Shift Operator. Whenever he determines that the safety of the reactor is in immediate jeopardy or when circuit set points and automatic shutdown should but does not occur, he has the responsibility and the authority to order shutdown of the reactor, or to personally effect the shutdown.

The Shift Supervisor shall hold a NRC Senior Reactor Operator license. He shall be continuously present at the plant for the duration of his assigned shift until properly relieved by the oncoming Shift Supervisor. It is his responsibility to provide direction for returning the reactor to power following a trip or an unscheduled power reduction.

During emergencies, accidents, or incidents requiring special procedures, the Shift Supervisor shall remain continuously in the control room until relieved by the oncoming shift supervisor or a senior licensed operator designated by the Operations Supervisor or higher authorty. From the control room, he shall continuously assess the condition of the station and provide general direction for all operating actions.

FUEL HANDLING

Fuel assembly movements, while being removed from or replaced in approved storage, shall be controlled by approved fuel handling procedures.

Handling of new fuel from initial receipt through storage in the new fuel vault or spent fuel pool (for the initial core) is performed by maintenance personnel. Maintenance personnel (performing fuel inspection) will receive specialized training resulting in their certification as fuel inspectors. In addition, all personnel involved in fuel receipt receive basic training in radiation protection and the site emergency plan.

All fuel handling from storage in the storage racks (ie. either the new fuel vault or the spent fuel pool) through loading into the core is performed by a licensed operator and supervised by a senior licensed operator. As part of license training, these personnel receive instruction in safe fuel handling, emergency plan procedures, and postulated accidents as analyzed in the FSAR.

A Senior Licensed Reactor Operator shall be responsible for all movement of new fuel within the site boundary. Senior Reactor Operators will be licensed on Unit 1 (if not yet licensed on Unit 2) or will hold a Unit 2 license.

The fuel will be stored in the shipping containers only as a temporary condition until they can be loaded into the spent fuel or new fuel racks. Fuel bundles may be stored temporarily in shipping containers. If they are stored in this way, the shipping containers will be stored in an array which is no more active than the array used during shipping. Containers

will be stacked no more than three containers high when fuel bundles are contained within. Shipping containers will be temporarily located in fuel receiving area on elevation 261 feet.

The fuel bundles are shipped in a steel container encased in a wooden shipping crate. One (1) steel container is contained in each wooden shipping crate. Two (2) fuel bundles are contained in each steel container. The container and crate are described in General Eletric Company drawing numbers 769E321, 769E232, and 769E229. The new and spent fuel racks are described in Attachment 1. The accident analysis is described in Attachment 1.

New fuel movement from the receipt area, fuel inspection, fuel channelling and movement of new fuel to the predetermined storage location is performed by the Maintenance Department or Operations Department. In addition, all activities associated with fuel receipt, movement, inspection and channelling will be performed in accordance with approved fuel handling procedures with formal training provided to the personnel involved (Attachment 4).

Preoperational testing of applicable fuel handling equipment will be done in accordance with approved procedures. This testing will be completed prior to their use for handling the new fuel shipment.

Fuel assemblies may be removed from storage (for other than core loading) for the following reasons:

- Inspection - This involves a visual or otherwise nondestructive examination of the fuel assembly to determine its acceptability before exposure in the reactor core.
- Precharacterization - This involves measurements and examination to determine the pre-exposure characteristics of a given fuel assembly.
- Repair - This includes moving the fuel bundles to the inspection stand or laydown area for evaluation and repair.

All receipt, shipment and internal transfer of special nuclear material in the form of fuel assemblies will be performed in accordance with approved fuel handling procedures, under the control of the Station Superintendent. The General Superintendent approves the procedures and changes to the procedures. This approval is documented by the General Superintendent's signature.

Criticality Control/Spent Fuel Pool

The design of the spent fuel storage racks provides for a subcritical multipliction factor (k_{eff}) for both normal and abnormal storage conditions. For normal and abnormal conditions, k_{eff} is equal to or less than 0.95. Normal conditions exist when the fuel storage racks are located in the pool and are covered with a depth of water approximately 25 feet above the stored fuel for radiation shielding and with the maximum number of fuel assemblies or bundles in their design storage position. An abnormal condition

may result from accidental dropping of a fuel assembly or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

The spent fuel storage array is such that k_{eff} is less than 0.95 due to the presence of the neutron absorber material.

Neutron poison is used in the spent fuel racks. No credit is taken for burnable poisons which may be contained in any fuel bundles.

For additional information on Spent Fuel Pool, refer to FSAR Section 9.1. The safety evaluation of the Spent Fuel pool is provided in Section 9.1 of the FSAR. Criticality analysis is presented in Section 9.1.

Criticality Safety Based on Other Than Maximum Enrichment of Fuel

This section of Regulatory Guide 3.15 is not applicable. Criticality safety is based on new fuel with a nominal flat U-235 enrichment of 3.6 w/o. For additional information, refer to above section of this document and Section 9.1 of the FSAR.

Criticality Safety Based on the Reactivity Effects of Neutron Absorber Materials

Refer to Criticality Control Section (above) of this document and Section 9.1 of the FSAR.

Criticality Safety Based on Moderation Control

Refer to Criticality Control Section (above) of this document and Section 9.1 of the FSAR.

Validation of Calculational Method for Criticality Safety

Description of the computer codes and methodology utilized in the verification of the criticality analysis is presented in FSAR Section 9.1.

Maximum Number of Fuel Assemblies Out of Authorized Locations

The maximum number of fuel assemblies that will be allowed outside a normal, approved storage location or normal shipping container is three (3) above the refuel floor and one (1) below the floor in the spent fuel pool confines. Fuel assemblies outside approved storage locations or shipping containers must maintain an edge-to-edge spacing of 12 inches or more from all other fuel. A fuel array of more than four assemblies outside approved fuel storage locations or shipping containers is prohibited.

No more than two metal shipping container containing fuel may be opened at any one time, and this container must be closed if all fuel is not immediately removed.

Removal of wooden crates is done in the fuel receiving area at elevation 261 feet.

ACCIDENT ANALYSIS

Detailed accident analyses of fuel handling equipment and storage areas are provided in FSAR (Attachment 1). The accidents considered that could affect the safety of new fuel in the fuel handling and storage area are as follows:

Railroad Access Area

Dropping of three containers containing two fuel assemblies each in the receiving area lifting bay while being lifted by the reactor building polar crane.

The consequences of this accident would be limited to impact damage to the dropped container and any container impacted in the Railroad Access area. Fuel damage from this accident would be limited to the possible rupture of fuel rods in the dropped and impacted containers. Since this accident affects only new fuel, the consequences would be limited to the potential release of unirradiated uranium dioxide fuel. No potential for a criticality condition exists in this accident since the maximum number of containers is enveloped by the 10CFR71 analysis for the shipping containers.

Other Accidents

All other handling accidents involve only one fuel assembly and are discussed in the FSAR (Attachment 1). Overhead loads greater than 1,000 lbs. will be administratively controlled when being moved in the vicinity of fuel storage areas.

The seismic design of the Reactor Building and of cranes, racks and pools precludes the credibility of more severe accidents. In the unlikely event of a dropped new fuel assembly in the storage areas, the consequences would be minimal. Due to the spacing of storage arrays, a criticality condition would not be possible under these accident conditions. The consequences of these accidents would be limited to the possible rupture of new fuel rods and subsequent release of unirradiated uranium dioxide fuel.

To preclude damage from falling objects, no construction loads will be allowed over the fuel in the fuel receiving area in the railroad access area.

QUALIFICATIONS OF SITE PERSONNEL

Title	Section of ANSI/ANS 3.1-1978 Containing Qualifications
General Superintendent	4.2.1
Station Superintendent	4.2.1
Site Maintenance Superintendent Nuclear	4.2.3
Technical Superintendent Nuclear	4.2.4
Superintendent Chemistry and Radiation Management	4.4.3* or 4.4.4
Supervisor Operations	4.2.2
Assistant Operations Supervisor	4.2.2
Supervisor Chemistry and Radiation Protection	4.4.3* or 4.4.4
Unit Radiation Protection Supervisor	4.4.4
Unit Chemistry Supervisor	4.4.3
Supervisor Instrument Support	4.7.2
Station Shift Supervisor	4.3.1
Superintendent Technical Services Nuclear	4.2.4

* Fulfills the requirements of Regulatory Guide 1.8 when performing "Radiation Protection Manager" duties

TABLE 1

Fuel Assembly Data:

Overall Length 176.16" (Attachment 2)
Nominal Active Fuel Length 150.0"
Fuel Rod Pitch 0.640"
Rod Array 8x8
Rods/Assembly 64 (includes 2 water rods)
Location of Water Rods (See Attachment)

Fuel Rod Data:

Fuel Pellet Material	$UO_2 + UO_2/Gd_2O_3$
Clad Outside Diameter	0.483"
Clad Thickness	0.032"
Clad Inside Diameter	0.419"
Fuel Pellet Immersion Density	95.0% theoretical
Fuel Pellet Diameter	0.410"
Maximum Pin Enrichment	3.00 w/o U235
Maximum Quantity U235	4.01 kg U235/183.2 kg U @ 2.19 w/o

Table 2

CURRENT FUEL DESIGN DATA

1.a.	Core Description	
	Core Arrangement (Drawing)	Attached
	Lattice (Drawing)	Attached
	Number of Assemblies	764
	Number of Control Rods	185
	Equivalent Core Diameter (in.)	187.1
	Core Power Density (kw/liter)	49.15
b.	Control Description	
	Type of Drive	Locking Piston
	Notch Length (in.)	6
	Shape	Cruciform
	Control Rod Pitch (in.)	12.0
	Blade Thickness (in.)	0.260
	Blade Span (in.)	9.810
	Sheath Thickness (in.)	0.030
	Material	304 Stainless
	Tube Diameter (in.)	0.188
	Tube Wall Thickness (in.)	0.025
	Poison Tubes	B_4C
	Poison Density (%, Theoretical)	70
c.	Fuel Description	
	Fuel Material	UO_2
	Cladding Material	ZR2
	Spacer Materials	ZR4, Inconel X750
	End Fittings Material	ZR2
	Channel Material	ZR4
	Fuel Bundle Weight (Less Channel)	
	Total Fuel Bundle Weight (UO_2 Kgs)	207.6
	Cladding (Including End Plugs) (Kgs)	53.42
	Spacers (Total) (Kgs)	2.31
	Number and Position of Spacers	Attached
	End Fittings and Hardware (Kgs)	8.86
	Channel Weight (Kgs)	36.7
	Number of Fuel Bundles	Attached
	Average Initial Enrichment	Attached
	Fuel Stack Density, % Theoretical	94.215
	Burnable Poison	
	Number and Location of Gd Rods	Attached
	Distribution and Concentration	Attached
	Dimensional Data on Fuel Bundle	Attached

Table 2
(Continued)

CURRENT FUEL DESIGN DATA
(Continued)

d. Thermal Hydraulic Data

Core Pressure Drop, psid	24.7*	
Channel Flow Rate/Bundle Power	Attached	
Total Coolant Flow Rate (External)	11.2%**	
Orifice Diameter (in.)	Central	Periph
	2.430	1.488
Number of Fuel Bundles per Orifice Region	Attached	
Fuel Bundle Outline Drawing	Attached	

2. Fuel Bundle Outline Drawing — Attached

3. Channel Outline Drawing — Attached

* Rated Power
** Rated Power Including Water Rods

Table, Section 7.1 AP-2.0

	Procedure Group	Approval	Review
a.	All procedures for which Technical Specification 6.8 Unit 1 or 2 requires the approval of the General Superintendant, Nuclear Generation.	General Superintendent Nuclear	Per Groups d through m
b.	All Administrative Procedures and other procedures of Technical Specification 6.8 which pertain to the responsibilities of a Station Superintendent.	Station Superintendent, Unit 1 and Station Superintendent, Unit 2	Per Groups d through m
c.	All Administrative Procedures	Manager Quality Assurance, Nuclear, Concurrence	
d.	Operating and Special Operating Procedures Unit 1 or Unit 2	Supervisor Operations, Unit 1 or Unit 2	Chemistry and Radiation Management Technical Department | 1
e.	Instrument and Control Procedures Unit 1 and Unit 2	Supervisor, Instrument and Control	Operations (SRO) Chemistry and Radiation Management
f.	Chemistry and Radiation Protection, Unit 1 and Unit 2	Superintendent Chemistry and Radiation Management	Operations (SRO) Maintenance Technical Department | 1
g.	Reactor and Plant Performance Procedures, Fuel Handling Procedures, Unit 1 and Unit 2	Reactor Analyst Supervisor	Operations (SRO) Chemistry and Radiation Management | 1
h.	Computer and Communications Procedures Unit 1 and Unit 2	Supervisor, Computer Operations and Maintenance	Operations (SRO) Technical Department including Administrative Services if involved Chemistry and Radiation Management | 1
i.	All Electrical, Structural and Mechanical Maintenance Procedures Unit 1 and Unit 2	Site Maintenance Superintendent	Operations (SRO) Chemistry and Radiation Management | 1 Quality Assurance

Table, Section 7.1 AP-2.0(cont'd)

	Procedure Group	Approval	Review	
j.	Modification Installation Procedures	Site Maintenance Superintendent or Supervisor Instrument and and Control or other site department head involved;	Operations Chemistry and Radiation Management Quality Assurance	1
k.	Training Procedures	Superintendent Training, Nuclear Supervisor or Superintendent of Department involved	Operations (SRO) Chemistry and Radiation Management	1
l.	Inservice Inspection Procedures	Superintendent, Inservice Inspection	Chemistry and Radiation Management Quality Assurance Operations (SRO) Maintenance and/or I&C involved	1
m.	Fire Protection Procedures	Supervisor Fire Protection, Nuclear	Chemistry and Radiation Management Operation (SRO)	
n.	Preoperational Tests	Technical Superintendent, Nuclear Generation	Nuclear Operations (SRO) Chemistry and Radiation Management Technical Department Quality Assurance	1
o.	Start-Up Procedures	Technical Superintendent, Nuclear Generation	Nuclear Operations (SRO) Technical Department Chemistry and Radiation Management Quality Assurance	1

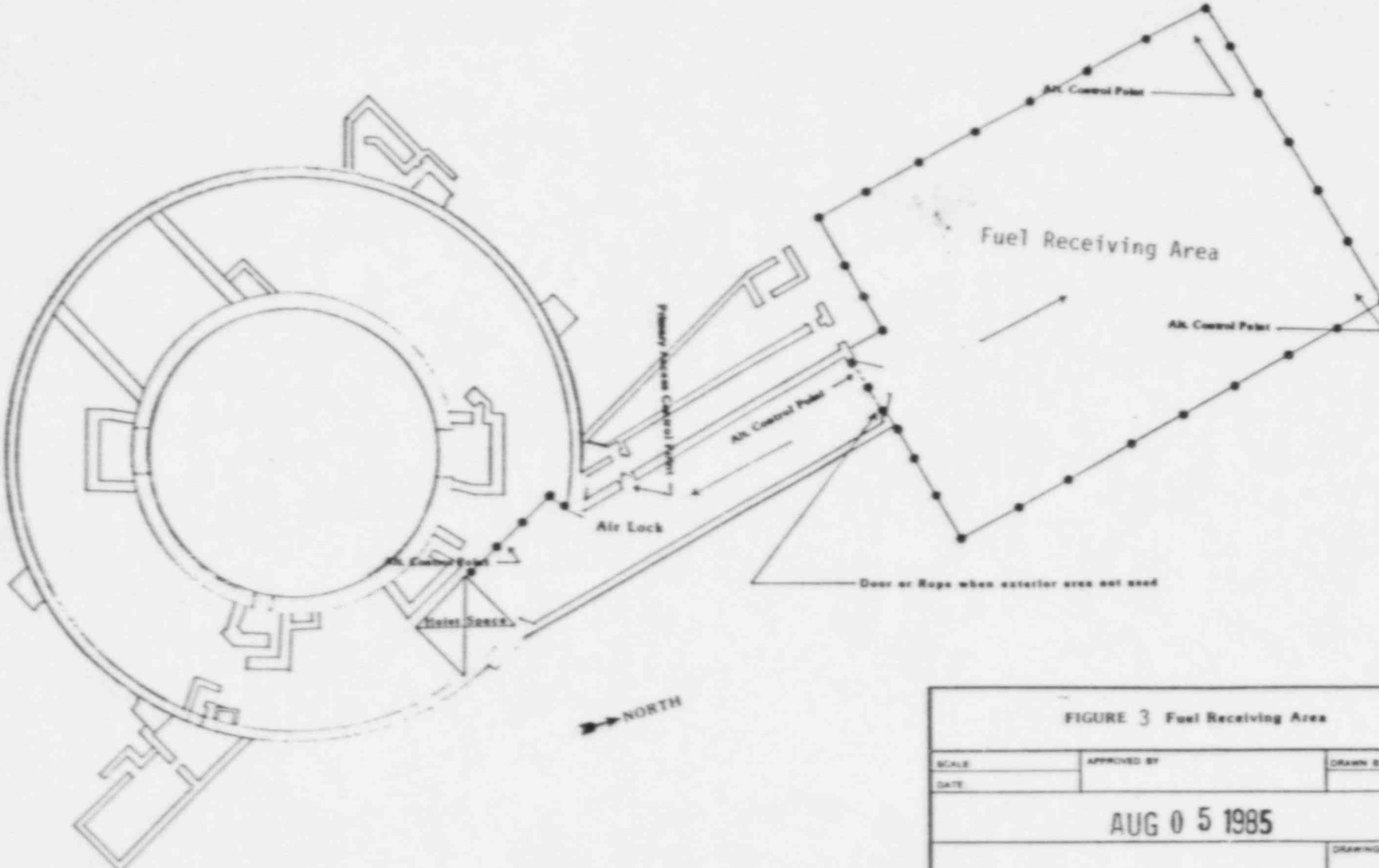

FIGURE 3 Fuel Receiving Area

SCALE / DATE	APPROVED BY	DRAWN BY
AUG 0 5 1985		
		DRAWING NUMBER

REQUEST FOR ADDITIONAL INFORMATION
NINE MILE POINT, UNIT 2

1. State the inner dimension, thickness, and height of the fuel assembly channel and the construction of material. Page 2 Section 1.1.

   Will the fuel assemblies be channeled before storage? Page 2 Section 1.2

2. Page 2 of your application gives the maximum pin enrichment of the new fuel as 2.191 w/o U-235, however, Table 1 states the maximum pin enrichment to be 3.00 w/o U-235, please clarify. Page 24 Revised Table 1

3. State the total quantity of U-235 requested in your license application. Page 2 Section 1.1

4. State the maximum number of fuel assemblies that will be covered by each steel cover plate in the New Fuel Vault. Page 4

   State the maximum number of fuel assemblies that will be uncovered at any one time in the new fuel vault. Page 4

5. Demonstrate the maximum number of assemblies in the New Fuel Vault that will remain uncovered at one time cannot be made critical independent of the degree of water (or mist) moderation. Page 5

6. Provide the composition of boraflex, include percent of weight of the various constituents (Boron, Carbon, Impurities), minimum B-10 content, and the Boraflex density (g/cm^3). Page 6-7

7. Describe the quality assurance program to assure that the boraflex composition, plate fabrication, and installation meet design specifications. Page 7

   Describe the quality assurance program to assure that the boraflex is securely captured into the stainless steel wall of the specified storage cell in accordance with storage rack design. Page 7

8. Identify the position responsible for developing and implementing procedures involving the control and handling of fuel assemblies. Page 7

9. Identify the position responsible for developing the health physics procedures. Page 7

10. Identify the position/committee responsible for the review and approval of the fuel handling and safety-related procedures. Page 7

11. Describe the fire alarm and control system. Page 6

12. Provide the minimum qualifications for the positions responsible for nuclear criticality safety and fuel handling. Page 19 and 23

13. Explain the statement "The spent fuel rack density (storage density) is 100 percent." Page 6

14. Confirm that the Station Shift Supervisor (p16) and the Station Shift Supervisor Nuclear (p21) are the same position or provide the minimum qualifications and job descriptions for both positions. Page 19

15. Confirm that all equipment used will be calibrated prior to use. Page 13-14

16. Confirm that al, personnel involved in fuel handling activities will receive prior training related to the safety of their operations, which will include emergency procedures and accident conditions. Page 19

17. Document that all personnel involved with radiation control activities will receive training in use of equipment and procedures for dealing with radiological problems or job-related accidents. Page 19

18. Confirm that all personnel will receive training pursuant to 10 CFR 19.12. Page 14

Please note that Figure 3 in the June 12, 1985 submittal was incorrect. The attached Figure 3 is correct. This figure was also part of the August 8, 1985 Special Nuclear Material Security Plan which was sent to Mr. G. McCorkle of the NRC.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
Niagara Mohawk Power Corporation) Docket No. 50-410
(Nine Mile Point Unit 2))

AFFIDAVIT

T. E. Lempges, being duly sworn, states that he is Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

T E Lempges

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 25 day of September, 1985.

Janis M. Macro
Notary Public in and for
Onondaga County, New York

My Commission expires:

JANIS M. MACRO
Notary Public in the State of New York
Qualified in Onondaga County No. 4704??8
My Commission Expires March 30, 19..

70-2948

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PDR

RETURN TO 396-SS

September 27, 1985

RECEIVED
SEP 30 1985
U. S. NUCLEAR REGULATORY COMMISSION
NMSS
Mail Section

Mr. John G. Davis
Director, Office of
Nuclear Material Safety
and Safeguards
U. S. Nuclear Regulatory
Commission
Washington, D.C. 20555

In the Matter of
Niagara Mohawk Power Corporation
(Nine Mile Point 2)
Docket No. 70-2948

Dear Mr. Davis:

On behalf of Niagara Mohawk Power Corporation, I am enclosing for filing the original and ten copies of an amendment to the Application to Amend Special Nuclear Material License SNM 1895, originally submitted to you on June 12, 1985. The amendment consists of replacement pages for the body of the application, Tables 1 and 2 and a new table entitled Table Section 7.1 AP-2.0. A corrected Figure 3 is also a part of the amendment.

Also enclosed is a restatement of your staff's July 29, 1985 Request for Additional Information which indicates the pages on which the requested information has been supplied in the amended application. A related affidavit of T. E. Lempges, Vice President of Niagara Mohawk Power Corporation is also enclosed.

Sincerely,

Mark J. Wetterhahn

Mark J. Wetterhahn
Counsel for Niagara Mohawk
Power Corporation

DOCKETED
USNRC
SEP 30 1985
NMSS
MAIL SECTION
DOCKET CLERK

MJW:sdd
Enclosures

FEE EXEMPT

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