Niagara Mohawk

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October 26, 2001
NMP1L 1620

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63
TAC No. MB2441

Subject: ***Proposed Technical Specification Changes - Section 6.0, Administrative Controls***

Gentlemen:

Niagara Mohawk Power Corporation (NMPC) hereby transmits an Application for Amendment to Nine Mile Point Unit 1 (NMP1) Operating License DPR-63. Enclosed are proposed changes to the Technical Specifications (TS) set forth in Appendix A to the above mentioned license. These changes are included as Attachment A to this letter.

Section 6.0 of the NMP1 TS delineates the Administrative Controls required at NMP1. Section 6.0 includes a discussion of plant management responsibilities, station organization, staff qualifications and training, review and audit activities, procedures, reporting requirements, record retention, high radiation areas, and various plant programs. Recently, Nine Mile Point Unit 2 (NMP2) converted to the Improved Standard Technical Specifications (ITS) in License Amendment No. 91. Section 5.0 of the NMP2 ITS delineates the Administrative Controls required at NMP2. NMPC proposes to revise the format and content of Section 6.0 of the NMP1 TS in a manner similar to NMP2 ITS Section 5.0; however, changes to incorporate the recommendations of Generic Letter 89-01 regarding radiological effluent technical specifications (RETS) are not included. Changes associated with RETS and updates to 10 CFR Part 20 references are the subject of a separate submittal. Consistency between the NMP1 and NMP2 Administrative Controls TS is necessary to avoid confusion and improve efficiency, since many of the processes and programs described are common to both units.

A "marked-up" copy of the TS pages and the associated supporting information discussing and justifying each change are included in Attachment B to this letter. The presentation format is similar to that employed in the original NMP2 ITS submittal dated October 16, 1998. Analyses demonstrating that the proposed changes to TS Section 6.0 involve no significant hazards consideration pursuant to 10 CFR 50.92 are included in Attachment C. NMPC's determination that the proposed changes meet the criteria for categorical exclusion from performing an environmental assessment is included as Attachment D.

A001

Upon NRC approval of this application, NMPC requests that the license amendment be issued with at least 90 days allowed for implementation.

Pursuant to 10 CFR 50.91(b)(1), NMPC has provided a copy of this License Amendment request and the associated analyses regarding no significant hazards consideration to the appropriate state representative.

Very truly yours,

John H. Mueller
Senior Vice President and
Chief Nuclear Officer

JHM/DEV/cld
Attachments

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
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Records Management

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	**Docket No. 50-220**
	)	
Nine Mile Point Unit 1	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission, Niagara Mohawk Power Corporation (NMPC), holder of Facility Operating License No. DPR-63, hereby requests that Section 6.0 of the Technical Specifications set forth in Appendix A to that license be amended. The proposed changes have been reviewed in accordance with Section 6.5 of the Technical Specifications (TS).

Section 6.0 of the Nine Mile Point Unit 1 (NMP1) TS delineates the Administrative Controls required at NMP1. Section 6.0 includes a discussion of plant management responsibilities, station organization, staff qualifications and training, review and audit activities, procedures, reporting requirements, record retention, high radiation areas, and various plant programs. Recently, Nine Mile Point Unit 2 (NMP2) converted to the Improved Standard Technical Specifications (ITS) in License Amendment No. 91. Section 5.0 of the NMP2 ITS delineates the Administrative Controls required at NMP2. NMPC proposes to revise the format and content of Section 6.0 of the NMP1 TS in a manner similar to NMP2 ITS Section 5.0; however, changes to incorporate the recommendations of Generic Letter 89-01 regarding radiological effluent technical specifications (RETS) are not included. Changes associated with RETS and updates to 10 CFR Part 20 references are the subject of a separate submittal. Consistency between the NMP1 and NMP2 Administrative Controls TS is necessary to avoid confusion and improve efficiency, since many of the processes and programs described are common to both units.

The proposed changes will not authorize any change in the type of effluents or in the authorized power level of the facility. Supporting information and analyses which demonstrate that the proposed changes involve no significant hazards consideration pursuant to 10 CFR 50.92 are included as Attachment C.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License DPR-63 be amended in the form attached hereto as Attachment A.

NIAGARA MOHAWK POWER CORPORATION

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10/25/05

By [signature]
John H. Mueller
Senior Vice President and
Chief Nuclear Officer

Subscribed and sworn to before
me on this 26th day of October 2001

[signature]
NOTARY PUBLIC

ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Proposed Changes to Technical Specifications

Replace the existing Technical Specification (TS) pages listed below with the attached revised pages. The revised pages have been retyped in their entirety, with marginal markings (revision bars) to indicate changes to the text.

Remove	Insert
v	v
vi	vi
8	8
11	11
131	131
296	296
301	301
302	302
304	304
306	306
315	315
324	324
331	331
332	332
337	337
347 through 374	347 through 374

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1.28 Ventilation Exhaust Treatment System

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

1.29 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

SAFETY LIMIT

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point (5 feet below minimum normal water level). The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operating during the major maintenance with the water level below the low-low level set point.

In addition to the requirement that at least one licensed Operator be in the control room when fuel is in the reactor, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

LIMITING SAFETY SYSTEM SETTING

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux for IRM range 9 or lower.

   The IRM scram trip setting shall not exceed 38.4% of rated neutron flux for IRM range 10.

c. The reactor high pressure scram trip setting shall be ≤ 1080 psig.

d. The reactor water low level scram trip setting shall be no lower than –12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than –5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be ≥ 850 psig when the reactor mode switch is in the run position or the IRMs are on range 10.

g. The main-steam-line isolation valve closure scram setting shall be ≤ 10 percent of valve closure (stem position) from full open.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10 CFR 100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

To assure that periodic surveillances of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Specification:

Whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required, the primary containment leakage rate shall be limited to:

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. The primary containment leakage rates shall be demonstrated at test schedules and in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan as described in Specification 6.5.4.

b. The provisions of Specification 4.0.1 are not applicable, and the surveillance interval extensions are in accordance with the 10 CFR 50 Appendix J Testing Program Plan.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

(2) Dose

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas (see Figures 5.1-1) shall be limited:

(a) During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

(b) During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

(2) Dose

Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual, prior to each release of a batch of liquid waste.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(2) Air Dose The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following: (a) During any calendar quarter: Less than or equal to 5 milliroentgen for gamma radiation and less than or equal to 10 mrads for beta radiation and, (b) During any calendar year: Less than or equal to 10 milliroentgen for gamma radiation and less than or equal to 20 mrads for beta radiation. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6, a Special Report that identifies the cause(s) for Exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	(2) Air Dose Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

LIMITING CONDITION FOR OPERATION

(3) Tritium, Iodines and Particulates

The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

(b) During any calendar year: Less than or equal to 15 mrems to any organ.

With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENT

(3) Tritium, Iodines and Particulates

Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined monthly in accordance with the method-ology and parameters in the Offsite Dose Calculation Manual.

(a) The LLD is defined in notation (a) of Table 4.6.15-1.

(b) Purge is defined in Section 1.23.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report pursuant to Specification 6.6.3.

(d) Sampling and analysis shall also be performed following shutdown, startup or an increase on the recombiner discharge monitor of greater than 50 percent, factoring out increases due to changes in thermal power level or dilution flow; or when the stack release rate is in excess of 1000 μCi/second and steady-state gaseous release rate increases by 50 percent.

(e) The sample flow rate and the stack flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.6.15.b.(1).(b) and 3.6.15.b.(3).

(f) When the release rate is in excess of 1000 μCi/sec and steady state gaseous release rate increases by 50 percent. The iodine and particulate collection device shall be removed and analyzed to determine the changes in iodine-131 and particulate release rate. The analysis shall be done daily following each change until it is shown that a pattern exists which can be used to predict the release rate; after which it may revert to weekly sampling frequency. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(g) When RAGEMS is inoperable the LLD for noble gas gross gamma analysis shall be 1×10^{-4}.

(h) Tritium grab samples shall be taken weekly from the station ventilation exhaust (stack) when fuel is offloaded until stable tritium release levels can be demonstrated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.6.15.a.(2)(b), 3.6.15.b.(2)(b) and 3.6.15.b.(3)(b), calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above listed 40CFR190 limits have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report.	Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual. This requirement is applicable only under conditions set forth in Specification 3.6.15.d.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6, Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.

c. Solid

The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

c. Solid

The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms).

(1) If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program.

(2) If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting levels of Table 6.6.6-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification 6.6.6. The Special Report shall identify the cause(s) for exceeding the limit(s) and define the corrective action(s) to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3). When more than one of the radionuclides in Table 6.6.6-1 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \text{......} \geq 1.0$$

When radionuclides other than those in Table 6.6.6-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specification 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3).

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.2.

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N.545 (1975), Section 4.3. Allowable exceptions to ANSI N.545 (1975), Section 4.3 are contained in the Nine Mile Point Unit 1 Offsite Dose Calculation Manual (ODCM).

(c) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \bullet V \bullet 2.22 \bullet Y \bullet \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, where applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for the particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.6.2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If the D/Q value at a new milk sampling location is significantly greater (50%) than the D/Q value at an existing milk sampling location, add the new location to the radiological environmental monitoring program within 30 days. The sampling location(s) excluding the control station location, having the lowest calculated D/Q may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.6.3 identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).	

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or a designee shall approve, prior to implementation, each proposed test and experiment not addressed in the UFSAR or Technical Specifications, and each modification to systems or equipment that affect nuclear safety.

6.1.2 The Station Shift Supervisor – Nuclear (SSS) shall be responsible for the control room command function. During any absence of the SSS from the control room while the unit is in the power operating or hot shutdown conditions, an individual with an active Senior Reactor Operator license shall be designated to assume the control room command function. During any absence of the SSS from the control room while the unit is in the cold shutdown or refueling conditions, an individual with an active Senior Reactor Operator license or Reactor Operator license shall be designated to assume the control room command function.

6.2 Organization

6.2.1 Onsite and Offsite Organization

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions or in equivalent forms of documentation. The organization charts and the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR. The functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions shall be documented in procedures.

b. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. The individuals who train the operating staff, carry out radiation protection, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Unit Staff

The unit organization shall be subject to the following:

a. At least two non-licensed operators shall be assigned when the unit is in the power operating condition; and at least one non-licensed operator shall be assigned when the unit is in the hot shutdown, cold shutdown, or refueling conditions. In addition, if the process computer is out of service for greater than 8 hours, at least three non-licensed operators shall be assigned when the unit is in the power operating, hot shutdown, cold shutdown, or refueling conditions.

b. The Shift Crew Composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specification 6.2.2.a for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements.

c. An individual qualified to implement radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence of on-duty personnel, provided immediate action is taken to fill the required position.

d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, key radiation protection personnel, auxiliary operators and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

e. As a minimum, either the Manager Operations or the General Supervisor Operations shall hold a senior reactor operator license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift supervision in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

6.3 Unit Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for; the Manager Operations who, in lieu of meeting the senior reactor operator license requirements of ANSI N18.1-1971, shall 1) hold a senior reactor operator license at the time of appointment, or 2) have held a senior reactor operator license at Nine Mile Point Nuclear Station Unit 1 or at a similar unit, or 3) have been certified for equivalent senior reactor operator knowledge; and the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed Reactor Operator are those individuals who, in addition to meeting the requirements of Specification 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 Procedures

6.4.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and cover the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for radioactive effluent and radiological environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 6.5.

6.5 Programs and Manuals

6.5.1 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, Containment Spray, Emergency Cooling, Shutdown Cooling, Reactor Cleanup, Vacuum Relief, Reactor Water Sampling, Containment Atmosphere Dilution (CAD) H_2O_2 Monitor, Drywell Containment Atmosphere Monitoring (CAM), Post Accident Sampling, Radioactive Gaseous Effluent Monitoring (RAGEMS), Offgas Effluent Stack Monitoring (OGESMS), and Post Accident Vent to Reactor Building Emergency Ventilation. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. System leak test requirements for each system at 24 month intervals.

The provisions of Specification 4.0.1 are applicable to the 24 month frequency for performing system leak test activities.

6.5.3 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to the Bases without prior NRC approval provided the changes do not involve either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of 6.5.3.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.4 10 CFR 50 Appendix J Testing Program Plan

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

   1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topical Report BN-TOP-1, and

   2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

6.5.4 10 CFR 50 Appendix J Testing Program Plan (cont'd)

b. The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

c. The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

d. Leakage Rate Surveillance Test acceptance criteria are:

1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.

2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.

3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

4. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a minimum pathway basis, at all times when containment integrity is required.

e. The provisions of Specification 4.0.1 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

6.5.5 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.6.1 Occupational Radiation Exposure Report

A tabulation shall be submitted on an annual basis which includes the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according

6.6.1 Occupational Radiation Exposure Report (cont'd)

to work and job functions; e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 20.407 of 10 CFR Part 20. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

6.6.2 Annual Radiological Environmental Operating Report*

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1, 1985.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls as appropriate, and with environmental surveillance reports from the previous 5 years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.6.22.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.6.21; discussion of all deviations from the sampling schedule of Table 3.6.20-1; and discussion of all analyses in which the LLD required in Table 4.6.20-1 was not achievable.

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* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.6.3 Semi-annual Radioactive Effluent Release Report**

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin on January 1, 1985.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the Offsite Dose Calculation Manual.

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* In lieu of submission with the Semi-annual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

** A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.6.3 Semi-annual Radioactive Effluent Report (cont'd)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement).

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.6.20.

6.6.4 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15^{th} of each month following the calendar month covered by the report.

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specifications 3.1.7.a and 3.1.7.e.

2. The K_f core flow adjustment factor for Specification 3.1.7.c.

3. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.1.7.c and 3.1.7.e.

4. The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5. The Power/Flow relationship for Specifications 3.1.7.d and 3.1.7.e.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in NEDE-24011-P-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (Latest approved revision as specified in the COLR).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

6.6.6 <u>Special Reports</u>

Special reports shall be submitted within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.2.2.(b) (12 months).

b. Safety Class 1 Inservice Inspection, Specification 4.2.6 (Three months).

c. Safety Class 2 Inservice Inspections, Specification 4.2.6 (Three months).

d. Safety Class 3 Inservice Inspections, Specification 4.2.6 (Three months).

e. Primary Containment Leakage Testing, Specification 3.3.3 (Three months).

f. Secondary Containment Leakage Testing, Specification 3.4.1 (Three months).

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months).

h. Calculate Dose from Liquid Effluent in Excess of Limits, Specification 3.6.15.a.(2)(b) (30 days from the end of the affected calendar quarter).

i. Calculate Air Dose from Noble Gases Effluent in Excess of Limits, Specification 3.6.15.b.(2)(b) (30 days from the end of the affected calendar quarter).

j. Calculate Dose from I-131, H-3 and Radioactive Particulates with half lives greater than eight days in Excess of Limits, Specification 3.6.15.b.(3)(b) (30 days from the end of the affected calendar quarter).

k. Calculated Doses from Uranium Fuel Cycle Source in Excess of Limits, Specification 3.6.15.d (30 days from the end of the affected calendar year)

l. Inoperable Gaseous Radwaste Treatment System, Specification 3.6.16.b (30 days from the event).

m. <u>Environmental Radiological Reports</u>. With the level of radioactivity (as the result of plant effluents) in an environmental sampling medium exceeding the reporting level of Table 6.6.6-1, when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission, within thirty (30) days from the end of the calendar quarter a special report identifying the cause(s) for exceeding the limits, and define the corrective action to be taken.

TABLE 6.6.6-1
REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/l)	Airborne Particulate Or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95, Nb-95	400				
I-131	2**	0.9		3	100
Cs-134	30	10.0	1,000	60	1,000
Cs-137	50	20.0	2,000	70	2,000
Ba/La-140	200			300	

* For drinking water samples. This is a 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 20 pCi/liter may be used.

6.7 High Radiation Area

6.7.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection manager or designate in the Radiation Work Permit.

6.7.2 In addition to the requirements of 6.7.1 areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem** shall be provided with locked doors to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Station Shift Supervisor or designate on duty and/or the radiation protection manager or designate. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify in accordance with site approved procedures accordingly, the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the drywell, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

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* by accessible passage and permanently fixed ladders

** measurement made at 18" from source of radioactivity

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ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Current Technical Specification Markup and Discussion of Changes

INTRODUCTION

Section 6.0 of the Nine Mile Point Unit 1 (NMP1) Current Technical Specifications (CTS) delineates the Administrative Controls required at NMP1. Section 6.0 includes a discussion of plant management responsibilities, station organization, staff qualifications and training, review and audit activities, procedures, reporting requirements, record retention, high radiation areas, and various plant programs. Recently, Nine Mile Point Unit 2 (NMP2) converted to the Improved Standard Technical Specifications (ITS) in License Amendment No. 91. Section 5.0 of the NMP2 ITS delineates the Administrative Controls required at NMP2. Niagara Mohawk Power Corporation (NMPC) proposes to revise the format and content of Section 6.0 of the NMP1 Technical Specifications (TS) in a manner similar to NMP2 ITS Section 5.0; however, changes to incorporate the recommendations of Generic Letter 89-01 regarding radiological effluent technical specifications (RETS) are not included. Changes associated with RETS and updates to 10 CFR Part 20 references are the subject of a separate submittal. Consistency between the NMP1 and NMP2 Administrative Controls TS is necessary to avoid confusion and improve efficiency, since many of the processes and programs described are common to both units.

EVALUATION

The proposed changes to CTS Section 6.0 are organized by individual subsection. For each subsection, a markup of the NMP1 CTS and a Discussion of Changes (DOC) are provided (Attachments B.1 through B.16). The corresponding No Significant Hazards Considerations (NSHC) evaluations are provided in Attachment C. This method of presentation is similar to that employed in the original NMP2 ITS submittal. Associated changes to the TS Table of Contents and other miscellaneous pages (e.g., to correct TS section cross-references) are also included.

The NMP1 CTS pages are annotated to show the disposition of the existing requirements into the NMP1 Revised TS. The annotated copy of the NMP1 CTS pages is marked with sequentially numbered "clouds" that provide a cross-reference to a Discussion of Changes (DOC) between the NMP1 CTS and the NMP1 Revised TS. The Revised TS number is noted in the top right corner of each CTS page, identifying the Revised TS section where the CTS requirement is located. Items on the CTS page that are located in one or more Revised TS sections have the appropriate location(s) noted adjacent to the items. When the Revised TS requirement differs from the CTS requirement, the CTS being revised is annotated with an alpha-numeric designator. This designator relates to the appropriate DOC. Each DOC provides a justification for the proposed change. The DOC for each Revised TS subsection immediately follows the marked-up CTS pages. The alpha-numeric designator also relates the proposed change to the applicable NSHC analysis (Attachment C).

The alpha-numeric designator is based on the category of the change and a sequential number within that category. The changes to the NMP1 CTS are categorized as follows:

A ADMINISTRATIVE - associated with restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement. There is a single NSHC evaluation for this category.

M TECHNICAL CHANGES – MORE RESTRICTIVE - changes to the TS being proposed that result in added restrictions or eliminating flexibility. There is a single NSHC evaluation for this category.

L TECHNICAL CHANGES – LESS RESTRICTIVE - "Specific" changes where requirements are relaxed, relocated, eliminated, or new flexibility is provided. Each "Specific" LESS RESTRICTIVE change has a corresponding unique NSHC analysis.

LA TECHNICAL CHANGES – LESS RESTRICTIVE - "Generic" changes consisting of relocation of details out of the TS and into the TS Bases, Updated Final Safety Analysis Report (UFSAR), Quality Assurance Manual, or other plant controlled documents. There is a single NSHC evaluation for this subcategory of "Generic" LESS RESTRICTIVE changes.

CONCLUSION

Section 6.0 of the NMP1 TS delineates the administrative controls required at NMP1. NMPC proposes to revise Section 6.0 of the NMP1 TS to be consistent with NMP2 ITS Section 5.0, as revised by License Amendment No. 91. The revised administrative controls will continue to assure operation of the facility will be conducted in compliance with applicable NRC regulations.

Based on the evaluation and associated conclusions stated in the Discussion of Changes (Attachments B.1 through B.16) and the NSHC evaluations (Attachment C), NMPC believes there is reasonable assurance that the proposed TS changes will not adversely affect the health and safety of the public and will not be inimical to the common defense and security.

ATTACHMENT B.1

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS TABLE OF CONTENTS

<u>Current Technical Specification Markup and Discussion of Changes</u>

A.1

Contents

A.1

Contents

DISCUSSION OF CHANGES
REVISED TS: TABLE OF CONTENTS

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

None

"Specific"

None

ATTACHMENT B.2

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.1
RESPONSIBILITY

Current Technical Specification Markup and Discussion of Changes

A.1

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

LA.1

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

A.2

Insert 6.1-A

(SSS) A.3

M.1

6.1.2 The Station Shift Supervisor - Nuclear ~~or during his absence from the control room, a designated individual~~ shall be responsible for the control room command function. ~~A management directive to this effect, signed by the Chief Nuclear Officer shall be re-issued to station personnel on an annual basis.~~

A.4

Insert 6.1-B

M.1

6.2 Organization

Onsite and Offsite Organization

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the position for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions or in equivalent forms of documentation. The organization charts shall be documented in the Final Safety Analysis Report, and the functional descriptions of departmental responsibilities and relationships and job descriptions for key personnel positions are documented in procedures.

b. The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to assure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant Manager shall have responsibility for overall unit operation and shall have control over those resources necessary for safe operation and maintenance of the plant.

See Discussion of Changes for Revised TS: 6.2, "Organization"

Insert 6.1-A A.2 LA.1 L.1

The plant manager or a designee shall approve, prior to implementation, each proposed test and experiment not addressed in the UFSAR or Technical Specifications, and each modification to systems or equipment that affect nuclear safety.

Insert 6.1-B M.1

During any absence of the SSS from the control room while the unit is in the power operating or hot shutdown conditions, an individual with an active Senior Reactor Operator license shall be designated to assume the control room command function. During any absence of the SSS from the control room while the unit is in the cold shutdown or refueling conditions, an individual with an active Senior Operator License or Reactor Operator license shall be designated to assume the control room command function.

6.5.2.3 Proposed modifications to unit structures, systems and components that affect nuclear safety shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to structures, systems and components ~~and the safety evaluations~~ shall be approved prior to implementation by the plant manager or ~~the Manager Technical Support as previously designated by the Plant Manager.~~ a designee A.2 L.1 LA.1

Move to 6.1.1

6.5.2.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the Plant Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary such review shall be performed by the appropriate designated station review personnel. A.2

~~6.5.2.5~~ Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications ~~and their safety evaluations~~ shall be reviewed by the plant manager, or ~~the Manager Technical Support as previously designated by the Plant Manager.~~ a designee L.1 LA.1

Move to 6.1.1

6.5.2.6 The Plant Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President - Nuclear Generation.

6.5.2.7 The facility security program, and implementing procedures, shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

6.5.2.8 The facility emergency plan, and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

DISCUSSION OF CHANGES
REVISED TS: 6.1 - RESPONSIBILITY

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 The requirements of CTS 6.5.2.3 and CTS 6.5.2.5 regarding Plant Manager reviews and approvals of proposed tests, experiments, and modifications to systems or equipment that affect nuclear safety are proposed to be moved to Revised TS 6.1, "Responsibility," except that the phrase "and their safety evaluations" would be deleted rather than relocated. Approval of the safety evaluation is inherent in the approval of the modification, test, or experiment; therefore, a separate requirement to approve the safety evaluation is not necessary. This change is consistent with NUREG-1434, Revision 1.

A.3 The acronym "SSS" has been added for the Station Shift Supervisor-Nuclear position title. This is strictly an editorial change.

A.4 CTS 6.1.2 requires a management directive to be reissued annually to all station personnel stating that the Station Shift Supervisor-Nuclear is responsible for the control room command function. This management directive requirement is being deleted. CTS 6.1.2 and Revised TS 6.1.2 state who is responsible for the control room command function. This requirement appears to serve only as a "reminder" to personnel as to who is in charge. Nowhere else in TS is a management directive required to remind personnel of a TS requirement, and this requirement is not considered to be one of the more important requirements (as it does not directly impact a safety margin). Since the TS responsibility requirement is not being changed, this deletion is considered administrative.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

M.1 CTS 6.1.2 identifies the Station Shift Supervisor – Nuclear (or during his absence, a designated individual) as responsible for the control room command function. The proposed change would delete the phrase "(or during his absence from the control room, a designated individual)," and add a requirement that an individual with either an active Senior Reactor Operator license or Reactor Operator license (depending on the unit operating condition) shall be designated to assume the control room command function. This change more clearly specifies the qualifications of the individual designated to assume the control room command function. This is an additional restriction on plant operation and is consistent with NUREG-1434, Revision 1.

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.1.1 uses the title "Plant Manager." This specific title is replaced with the generic title "plant manager." The specific title is proposed to be relocated to UFSAR Section XIII-A, which is where the organizational chart and description of this specific title is currently located. Relocation of specific titles out of the TS is consistent with the NRC letter from C. Grimes to the Owners Group Technical Specification Committee Chairman, dated November 10, 1994, as documented in NRC-approved TSTF-65, Revision 1. The various requirements of the individuals are still retained in the Revised TS. In addition, Revised TS 6.2.1 requires the organization chart to be documented in the UFSAR. Therefore, the relocated specific titles are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

"Specific"

L.1 CTS 6.5.2.3 and CTS 6.5.2.5 currently identify the Manager Technical Support as the designated alternate to the Plant Manager for the approval of proposed modifications, tests, and experiments. In Revised TS 6.1.1, the phrase "the Manager Technical Support as previously designated by the Plant Manager" that is currently contained in CTS 6.5.2.3 and CTS 6.5.2.5 is replaced with "a designee." This change provides additional flexibility while maintaining plant manager (changed to the generic title by Discussion of Change LA.1 above) control over the designation of personnel performing these activities. This is consistent with CTS 6.1.1, which states that the Plant Manager is responsible for overall unit operation, and which allows the Plant Manager to designate an individual to take over this responsibility during the Plant Manager's absence. Since the plant manager is still maintaining this control, the removal of a specific titled individual to whom the plant manager delegates responsibility does not impact plant safety.

ATTACHMENT B.3

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.2
ORGANIZATION

Current Technical Specification Markup and Discussion of Changes

See Discussion of Changes for Revised TS: 6.1, "Responsibility"

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Station Shift Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Chief Nuclear Officer shall be re-issued to station personnel on an annual basis.

6.2 Organization

A.1

Onsite and Offsite Organization

6.2.1 ~~An~~ onsite and ~~an~~ offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be defined and established ~~and defined from the~~ throughout highest management levels, ~~through~~ intermediate levels, ~~to~~ and ~~including~~ all operating organization positions. These ~~Those~~ relationships shall be documented and updated, as appropriate, in ~~the form of~~ organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions or in equivalent forms of documentation. The organization charts shall be documented in the UFSAR ~~Final Safety Analysis Report~~. ~~and~~ The functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions ~~are~~ shall be documented in procedures. LA.1

b. A specified corporate officer ~~The Chief Nuclear Officer~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure ~~assure~~ acceptable performance of the staff in operating, maintaining, and providing technical support ~~in~~ to the plant to ensure ~~so that continued~~ nuclear safety. ~~is assured.~~

c. LA.1 The Plant Manager shall be responsible ~~have responsibility~~ for overall safe ~~unit~~ operation of the plant and shall have control over those onsite activities ~~resources~~ necessary for safe operation and maintenance of the plant.

and the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications LA.1

A.1

Specification 6.2

individuals who train the operating staff, carry out

radiation protection, A.3

or perform

the

d. The ~~persons responsible for the training, health physics~~ ~~and~~ quality assurance functions may report to ~~an~~ appropriate manager ~~onsite,~~ but shall have direct access to responsible corporate management at a level where action appropriate to the mitigation of training, health physics and quality assurance concerns can be accomplished.

onsite

; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

Unit
~~Facility~~ Staff

A.4

6.2.2 The unit organization shall be subject to the following:

Insert 6.2-A

a. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~ LA.2

Insert 6.2-B

b. ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation, this licensed operator shall be present at the controls of the facility.~~ LA.2

A.5

~~c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.~~ LA.3

c. ~~d.~~ An individual qualified ~~in~~ to implement radiation protection ~~*~~ procedures shall be on site when fuel is in the reactor. A.2

~~e. A licensed Senior Reactor Operator shall be required in the Control Room during power operations, hot shutdown,~~ LA.2 ~~and when the emergency plan is activated. This may be the Station Shift Supervisor - Nuclear or the Assistant Station Shift Supervisor - Nuclear or another Senior Reactor Operator during power operations or hot shutdown. When the emergency plan is activated during normal operations or hot shutdown, the Assistant Station Shift Supervisor - Nuclear becomes the Shift Technical Advisor and the Station Shift Supervisor - Nuclear is restricted to the control room until an additional licensed Senior Reactor Operator arrives.~~ LA.4

~~f. A licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling shall be responsible for all movement of new and irradiated fuel within the site boundary. All core alterations shall be directly supervised by a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.~~ LA.5

~~* The requirement for a Radiation Protection qualified individual may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.~~

The position may be vacant for not more than 2 hours, in order to provide for unexpected absence of on-duty personnel, provided immediate action is taken to fill the required position.

A.1

Specification 6.2

~~g. DELETED~~

personnel LA.6

6.2.2.d Administrative procedures shall be developed and implemented to limit the working hours of ~~facility staff~~ personnel who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, ~~health physicists~~ key radiation protection personnel, auxiliary operators and key maintenance personnel). A.3

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an 8 to 12 hour day, nominal 40-hour week while the facility is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications on a temporary basis, the following guidelines shall be followed:~~ LA.6

~~1) An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).~~

~~2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period (all excluding shift turnover time).~~

~~3) A break of at least 8-hours should be allowed between work periods (including shift turnover time).~~

~~4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~ L.1

Insert 6.2-C

LA.6

in advance

the plant manager's designee

approved administrative

Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or ~~higher levels of management~~ the plant manager's designee, in accordance with ~~established~~ approved administrative procedures, ~~and~~ with documentation of the basis for granting the deviation. ~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Vice President - Nuclear Generation or designee to assure that excessive hours have not been assigned.~~ LA.1

Insert 6.2-D

Routine deviation from the ~~above~~ working hour guidelines ~~is not~~ shall not be authorized. LA.6

Working hour

shall not be

6.2.2.e As a minimum, either the Manager Operations or the General Supervisor Operations shall hold a senior reactor operator license. ~~The Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear shall hold senior reactor operator licenses. Only licensed individuals may direct licensed activities.~~ LA.7

6.2.2.f Insert 6.2-E M.1

Insert 6.2-A A.4

a. At least two non-licensed operators shall be assigned when the unit is in the power operating condition; and at least one non-licensed operator shall be assigned when the unit is in the hot shutdown, cold shutdown, or refueling conditions. In addition, if the process computer is out of service for greater than 8 hours, at least three non-licensed operators shall be assigned when the unit is in the power operating, hot shutdown, cold shutdown, or refueling conditions.

Insert 6.2-B A.5

b. The Shift Crew Composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specification 6.2.2.a for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements.

Insert 6.2-C LA.6

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

Insert 6.2-D LA.6

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

Insert 6.2-E M.1

A.9

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift supervision in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A.8

LA.3

LA.2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION[1][6]

License	Normal Operation	Shutdown Condition	Operation[3] W/O Process Computer	Reactor Startups
Senior Operator	1	1[5]	1	1
Operator	2	2[4]	2	3
Unlicensed[2]	2	1	3	2
Asst. Station Shift Supervisor (Shift Technical Advisor Function) (Senior Operator License)[7]	1	1[4]	1	1

A.4

To 6.2.2.a

LA.2

Notes:

(1) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, refuel outages, etc. A.7

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License. A.4

To 6.2.2.a

(3) For operation longer than eight hours without process computer.

LA.3

(4) Hot shutdown condition only. For cold shutdown and refueling conditions, only one senior operator and one operator are required to be on shift.

(5) An additional Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities shall supervise all core alterations. LA.2

To 6.2.2.b A.5

(6) The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent. A.6

(7) The Assistant Station Shift Supervisor performs the Shift Technical Advisor function when the emergency plan is activated during normal operations or hot shutdown and shall hold a senior reactor operator license. LA.4 Normally, the Assistant Station Shift Supervisor is a combined Assistant Station Shift Supervisor/Shift Technical Advisor, however, there may be instances when a shift may be staffed by two Senior Reactor Operators plus a dedicated Shift Technical Advisor.

DISCUSSION OF CHANGES
REVISED TS: 6.2 - ORGANIZATION

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 CTS 6.2.2.d uses the phrase "qualified in" as it relates to radiation protection procedures. In Revised TS 6.2.2.c, this phrase is replaced with "qualified to implement," consistent with the NMP2 ITS. This is an administrative change that does not alter the qualifications of the identified individual.

A.3 CTS 6.2.1.d uses the term "health physics." In Revised TS 6.2.1.d this term is replaced with "radiation protection." CTS 6.2.2.h uses the term "health physicists." In Revised TS 6.2.2.d this term is replaced with "key radiation protection personnel." The change in terminology is considered administrative and is consistent with Revised TS 6.2.2.c and the current organization.

A.4 CTS Table 6.2-1, including Notes (2) and (3), contains requirements for unlicensed operating personnel. These requirements are moved to Revised TS 6.2.2.a and presented in text form rather than the tabular form of CTS Table 6.2-1. For the case where the process computer is out of service for greater than 8 hours, the specific operating conditions for which three unlicensed operators shall be assigned are listed. Also, the term "unlicensed" is replaced with "non-licensed." These administrative changes do not alter the existing requirements, and are consistent with NUREG-1434, Revision 1.

A.5 CTS Table 6.2-1, Note (6), allows the shift crew composition to be less than the minimum requirements of CTS Table 6.2-1 under certain conditions. This requirement is moved to Revised TS 6.2.2.b. In addition, since CTS Table 6.2-1 is not being retained in the Revised TS, the reference to Table 6.2-1 is replaced with "10 CFR 50.54 (m)(2)(i) and Specification 6.2.2.a." This is consistent with the changes described in DOC LA.2 below and with NUREG-1434, Revision 1. These are administrative changes that do not alter the existing requirements.

A.6 Note (6) of CTS Table 6.2-1 does not allow any shift crew position to be unmanned upon shift change because an oncoming crewman scheduled to come on duty is late or absent. Revised TS 6.2.2.b allows a period of time not to exceed two hours in order to accommodate unexpected absence of "on-duty" shift crew members. The term "on-duty" implies that the absence refers to on-duty shift crew members and not the oncoming crew. If anyone in the oncoming crew is not present, the "on-duty" person may not leave. Therefore, the requirement of this footnote is covered in Revised TS 6.2.2.b. Since the minimum shift crew requirements continue to be maintained in Revised TS 6.2.2.b, deletion of this portion of the footnote is an administrative change.

A.7 Note (1) of CTS Table 6.2-1, which states that more operators can be assigned if needed, is deleted. The CTS table specifies the requirements of the minimum shift crew composition and thus it is not necessary to specify whether the requirements may be exceeded.

A.8 The specific qualification requirements of the Shift Technical Advisor (STA) contained in CTS 6.3.1 have been moved to Revised TS 6.2.2.f and have been modified to reference the Commission Policy Statement on Engineering Expertise on Shift. This change is consistent with NUREG-1434, Revision 1. Since the policy statement encompasses the current requirements, this change is considered administrative.

A.9 The person to whom the STA provides advisory technical support has incorporated a more generic statement than is indicated in NUREG-1434, Revision 1. In the NUREG, the STA is required to provide advisory technical support to the Shift Supervisor. This term for whom the STA supports was derived from the generic term in NUREG-0737, Item I.A.1.1. At NMP1, both an Assistant Station Shift Supervisor (ASSS) and a Station Shift Supervisor (SSS) are on the operating shift, and both hold senior operator licenses. As noted in CTS Table 6.2-1, Note (7), normally the ASSS is a combined ASSS/STA; however, there may be instances when a shift may be staffed by two Senior Reactor Operators plus a dedicated STA. This dedicated STA would normally provide support to the ASSS, since the ASSS is normally the control room supervisor. However, when the ASSS is not in the control room, the SSS would assume control room supervisor duties. Thus, the dedicated STA could provide support to either the SSS or the ASSS at the start of an event. To provide a more generic, but technically accurate, statement as to whom the STA provides technical support, the words "Shift Supervisor" used in NUREG-1434 have been replaced with "shift supervision." This change is consistent with the NMP2 ITS.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

M.1 Revised TS 6.2.2.f is added to the TS to describe the duties of the Shift Technical Advisor. This is an additional restriction on plant operation and is consistent with NUREG-1434, Revision 1.

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.2.1.c and CTS 6.2.2.h use the title "Plant Manager." In Revised TS 6.2.1.c and Revised TS 6.2.2.d, this specific title is replaced with the generic title "plant manager." CTS 6.2.1.b uses the title "Chief Nuclear Officer." This specific title is replaced with the generic term "a specified corporate officer." The specific titles are proposed to be relocated to UFSAR Section XIII-A, which is where the organizational chart and description of these specific titles is currently located. Relocation of specific titles out of the TS is consistent with the NRC letter from C. Grimes to the Owners Group Technical Specification Committee Chairman, dated November 10, 1994, as documented in NRC-approved TSTF-65, Revision 1. The various requirements of the individuals are still retained in the Revised TS. In addition, Revised TS 6.2.1 requires the organization chart to be documented in the UFSAR. Therefore, the relocated specific titles are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

LA.2 Details of the minimum shift crew requirements located in CTS Table 6.2-1 are proposed to be relocated to the UFSAR (Section XIII-A). The requirements of 10 CFR 50.54(k), (l), and (m) adequately provide for shift manning. In 10 CFR 50.54(m)(2)(iii), it is required that "when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 10 CFR 50.54(k) requires "An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during operation of the facility." The minimum shift crew requirements for licensed operators and senior operators contained in CTS 6.2.2.a, CTS 6.2.2.b, CTS 6.2.2.e and CTS Table 6.2-1 will be met through compliance with these regulations and do not need to be repeated in the TS. This is consistent with NRC-approved TSTF-258, Revision 4. The minimum shift crew requirements for non-licensed plant equipment operators are transferred from CTS Table 6.2-1 to Revised TS 6.2.2.a. In addition, Revised TS 6.1.2 contains requirements for the control room command function, and Revised TS 6.2.2.f contains requirements for the Shift Technical Advisor (STA). The relocation of the details of the minimum shift crew requirements to the UFSAR is acceptable considering the controls provided by regulations, the remaining requirements in the TS, and the control of changes to procedures governing the conduct of operations under 10 CFR 50 Appendix B programs. Therefore, the relocated requirements are not required to be in the TS to provide adequate protection of the public health and safety.

LA.3 CTS 6.2.2.c requires two licensed Operators in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips. In addition, CTS Table 6.2-1, including Note (4), requires two licensed Operators for the hot shutdown condition. These requirements are proposed to be relocated to the UFSAR. The requirement specifying the minimum number of operators in the control room is adequately controlled by the requirements of 10 CFR 50.54(k), (l), and (m), as discussed in DOC LA.2 above. The requirement for location of these operators is also already specified in current administrative procedures. Therefore, the relocated requirement is not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

LA.4 CTS 6.2.2.e and CTS Table 6.2-1, Note (7) specify staffing requirements when the emergency plan is activated. These requirements are proposed to be relocated to the Site Emergency Plan. Staffing requirements when the emergency plan is activated are documented in the Site Emergency Plan and in administrative procedures. As discussed in DOC LA.2 above, the regulations provide the staffing requirements during the power operating and hot shutdown conditions and are adequate since the personnel required during emergencies are specified. Therefore, the relocated requirement is not required to be in the TS to provide adequate protection of the public health and safety. Changes to the Site Emergency Plan are controlled by the provisions of 10 CFR 50.54(q).

LA.5 Details contained in CTS 6.2.2.f that require all Core Alterations to be supervised by either a licensed Senior Reactor Operator or a licensed Senior Reactor Operator Limited to Fuel Handling are proposed to be relocated to the UFSAR. These CTS requirements are contained in 10 CFR 50.54(m)(2)(iv) and do not need to be repeated in the TS to provide adequate protection of the public health and safety. In addition, CTS 6.2.2.f requires that all fuel moves be directly monitored by a member of the reactor analyst group. This requirement is also proposed to be relocated to the UFSAR. In 10 CFR 50.54(m)(2)(iv), the minimum requirements for moving reactor fuel are specified. It does not require a non-licensed member of the reactor analyst group (or any other type of engineer) to monitor fuel movement. This is an additional administrative requirement that does not need to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

LA.6 CTS 6.2.2.h contains requirements for working hour limits for facility staff who perform safety-related functions. CTS Section 6.2.2.h is proposed to be revised from specific working hour limits to administrative procedures to control working hours, consistent with NRC-approved TSTF-258, Revision 4. The proposed changes will provide reasonable assurance that impaired performance caused by excessive working hours will not jeopardize safe plant operation. Specific working hour limits are not otherwise required to be in the TS under 10 CFR 50.36(c)(5). Specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision-making process to minimize the potential for impaired personnel performance, and established procedure control processes will provide sufficient control of changes to that procedure. These changes are consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis, as documented in NRC-approved TSTF-258, Revision 4. Additionally, the statement "Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Vice President-Nuclear Generation or designee to assure that excessive hours have not been assigned." is being deleted. There is no guidance in Generic Letter 82-12 that discusses these additional controls. The additional requirement to have the Plant Manager (or his designee) review individual overtime on a monthly basis is unnecessary since sufficient administrative controls and policies exist, as well as the role of the individuals' supervisors in supervising personnel prevent excessive use or abuse of overtime. Therefore, the working hour limits are not required to be in the TS to provide adequate protection of the public health and safety.

LA.7 Details of the operator license requirements in CTS 6.2.2.i for the specific positions of Station Shift Supervisor Nuclear and Assistant Station Shift Supervisor Nuclear, and the CTS requirement that only licensed individuals may direct licensed activities, are proposed to be relocated to the UFSAR (Section XIII-A). This level of detail is not necessary in the TS to provide adequate protection of the public health and safety. These details are adequately addressed by the requirements contained in 10 CFR 50.54(i), (j), (k), (l), and (m) and by the qualification requirements in Revised TS 6.3.1. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

"Specific"

L.1 CTS 6.2.2.h currently provides a description of the individuals who can be designated by the Plant Manager to approve modifications to overtime requirements. The proposed change to CTS 6.2.2.h would replace the phrase "higher levels of management" with "the plant manager's designee." This change provides additional flexibility while maintaining plant manager (changed to the generic title by DOC LA.1 above) control over the designation of personnel who can approve this activity. This is consistent with CTS 6.1.1, which states that the Plant Manager is responsible for overall unit operation, and which allows the Plant Manager to designate an individual to take over this responsibility during the Plant Manager's absence. Since the plant manager is still maintaining this control, the change does not impact plant safety. Therefore, this change is considered acceptable.

ATTACHMENT B.4

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.3
UNIT STAFF QUALIFICATIONS

<u>Current Technical Specification Markup and Discussion of Changes</u>

A.1

6.3 ~~Facility~~ Unit Staff Qualifications

radiation protection manager — LA.1

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for; the Manager Operations who, in lieu of meeting the senior reactor operator license requirements of ANSI N18.1-1971, shall 1) hold a senior reactor operator license at the time of appointment, or 2) have held a senior reactor operator license at Nine Mile Point Nuclear Station Unit 1 or at a similar unit, or 3) have been certified for equivalent senior reactor operator knowledge; and the ~~Manager Radiation Protection~~ who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~,~~. and the Shift Technical Advisor who shall have a bachelor's degree in a physical science or engineering or a professional engineer license issued by examination and shall have received specific training in plant design, and response and analysis of the plant for transients and accidents.

A.2

Moved to Specification 6.2.2

Insert 6.3-A — M.1

6.4 Training

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Manager Training and shall meet or exceed the recommendations and requirements of Section 5.5 of ANSI N18.1-1971 and of 10CFR Part 55, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager Training and Supervisor-Fire Protection, Nuclear and shall meet or exceed the requirements of Appendix R to 10CFR50.

See Discussion of Changes for CTS: 6.4, "Training"

6.5 Review and Audit

6.5.1 Station Operations Review Committee (SORC)

Function

6.5.1.1 The Station Operations Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

Composition

6.5.1.2 The SORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman/Member:	Manager Operations
Vice Chairman/Member:	Manager Technical Support
Member:	Manager QA Operations
Member:	Manager Maintenance
Member:	Manager Chemistry
Member:	Manager Radiation Protection

See Discussion of Changes for CTS: 6.5, "Review and Audit"

Insert 6.3-A M.1

6.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed Reactor Operator are those individuals who, in addition to meeting the requirements of Specification 6.3.1, perform the functions described in 10 CFR 50.54(m).

DISCUSSION OF CHANGES
REVISED TS: 6.3 – UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 The requirements in CTS 6.3.1 regarding the Shift Technical Advisor (STA) are proposed to be moved to Revised TS 6.2, "Organization." Technical changes to these requirements are addressed in the Discussion of Changes for Revised TS 6.2.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

M.1 Revised TS 6.3.2 is added to clarify the qualification requirements for licensed Senior Reactor Operators and licensed Reactor Operators. Definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that…" Adding TS 6.3.2 ensures that there is no misunderstanding when complying with 10 CFR 55.4 requirements. This change is consistent with the recommendations in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI), as documented in NRC-approved TSTF-258, Revision 4.

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.3.1 uses the title "Manager Radiation Protection." This specific title is replaced with the generic title "radiation protection manager." The specific title is proposed to be relocated to UFSAR Section XIII-A, which is where the organizational chart and description of this specific title are currently located. Relocation of specific titles out of the TS is consistent with the NRC letter from C. Grimes to the Owners Group Technical Specification Committee Chairman, dated November 10, 1994, as documented in NRC-approved TSTF-65, Revision 1. The various requirements of the individuals are still retained in the Revised TS. In addition, Revised TS 6.2.1 requires the organization chart to be documented in the UFSAR. Therefore, the relocated specific titles are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

"Specific"

None

ATTACHMENT B.5

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.4
PROCEDURES

Current Technical Specification Markup and Discussion of Changes

A.1

Specification 6.4

6.6 Reportable Occurrence Action

<See Discussion of Changes for CTS: 6.6, "Reportable Occurrence Action">

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.

6.7 Safety Limit Violation

<See Discussion of Changes for CTS: 6.7, "Safety Limit Violation">

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

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear Generation and the SRAB shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, within 30 days of the violation, and to the SRAB, and the Vice President - Nuclear Generation within 14 days.

6.4 ~~6.8~~ Procedures

6.4.1 ~~6.8.1~~ Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and ~~Appendix "A" of USAEC Regulatory Guide 1.33~~ ~~except as provided in 6.8.2 and 6.8.3 below.~~ cover the following activities: A.2 LA.1

Insert 6.4-A

a. Written procedures shall be established, implemented, and maintained for activities involving the Fire Protection Program implementation. A.1

LA.1

~~6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation by the branch manager for the functional area of the procedure or higher levels of management as governed by administrative procedures. Each procedure and administrative policy of 6.8.1 above shall be reviewed periodically as set forth in administrative procedures.~~

Insert 6.4-A

a. The applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972; A.2

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; M.1

c. Quality assurance for radioactive effluent and radiological environmental monitoring;

d. Fire Protection Program implementation; and From CTS 6.8.1.a

e. All programs specified in Specification 6.5. M.1

6.8.3 ~~Temporary changes to procedures of 6.8.1 above may be made provided:~~

a. ~~The intent of the original procedure is not altered.~~

b. ~~The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.~~

c. ~~The change is documented, reviewed and approved within 14 days of implementation by the branch manager for the functional area of the procedure or higher levels of management as governed by administrative procedures.~~

LA.1

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

See Discussion of Changes for Revised TS: 6.6, "Reporting Requirements"

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 CTS 6.8.1 requires that written procedures and administrative policies be established, implemented, and maintained that meet or exceed the requirements and recommendations of Appendix A of Regulatory Guide 1.33. This requirement is proposed to be moved to a specific sub-item (Item a) within Revised TS 6.4.1. The specific version of the Regulatory Guide 1.33 (i.e., dated November 3, 1972) is also identified, which is consistent with NMPC statements of conformance contained in Amendment No. 1 to Application to Convert Provisional Operating License to Full-Term Operating License (November 1973) and in NMPC letter to the AEC dated November 16, 1973. Since the requirements remain unchanged, this is considered to be a format change only, and therefore is considered an administrative change.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

M.1 Revised TS 6.4.1, Items b, c, and e are added to the TS. This change will assure proper procedure control for emergency operating procedures, quality assurance for radioactive effluent and radiological environmental monitoring, and the programs list in Revised TS 6.5. This is an additional restriction on plant operation and is consistent with NUREG-1434, Revision 1.

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.8.2 describes details of procedure reviews and approvals, and CTS 6.8.3 describes requirements relating to temporary changes to procedures. The proposed change would relocate these requirements to the Nine Mile Point Nuclear Station Quality Assurance Topical Report (QATR). These changes are consistent with the guidance of AL 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995, and NUREG-1434, Revision 1. The administrative letter concluded that TS administrative quality assurance-related requirements may be relocated to licensee-controlled quality assurance programs. For NMP1, these requirements would be relocated in their entirety to the QATR, with changes only to the format. Requirements for the processes related to review and approval of procedures and changes to procedures are contained in 10 CFR 50 Appendix B, Criterion II and Criterion V; ANSI N18.7-1972; and ANSI/ASME NQA-1-1983, including 1983 Addenda. Relocation of these TS provisions to the QATR will provide adequate controls over procedure review and approval activities for NMP1. Thus, the relocated details are not necessary to be in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

"Specific"

None

ATTACHMENT B.6

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.5
PROGRAMS AND MANUALS

<u>Current Technical Specification Markup and Discussion of Changes</u>

Changes to the ODCM

6.5.1 ~~Changes to the~~ Offsite Dose Calculation Manual (ODCM) ~~s~~ Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

f. CORE OPERATING LIMITS REPORT

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.1.7.a and 3.1.7.e.

2) The K_f core flow adjustment factor for Specification 3.1.7.c.

3) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.1.7.c and 3.1.7.e.

4) The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5) The Power/Flow relationship for Specification 3.1.7.d and e.

and shall be documented in the CORE OPERATING LIMITS REPORT.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

See Discussion of Changes for Revised TS: 6.6, "Reporting Requirements"

Page 1 of 6

6.5.3 Insert 6.5-B

See Discussion of Changes for CTS: 6.13, "Fire Protection Inspection"

6.13 Fire Protection Inspection

6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

6.5.2 ~~6.14~~ ~~Systems Integrity~~ Primary Coolant Sources Outside Containment Insert 6.5-A A.3

~~Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.~~

See Discussion of Changes for CTS: 6.15, "Iodine Monitoring"

6.15 Iodine Monitoring

Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.8.c of NUREG 0578.

6.5.4 ~~6.16~~ 10 CFR 50 Appendix J Testing Program Plan

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel ~~Topic~~ Topical Report BN-TOP-1, and

2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

b. The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

c. The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

d. Leakage Rate Surveillance Test acceptance criteria are:

1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.

2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.

3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

(A.1)

6.5.4 (Cont'd)

4. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a minimum pathway basis, at all times when containment integrity is required.

e. The provisions of Specification 4.0.1 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

6.5.5 Radiation Protection Program

(Next page)

<See Discussion of Changes for CTS: 6.10, "Record Retention"> A.1 Specification 6.5

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

6.5.5 ~~6.11~~ Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 High Radiation Area

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

<See Discussion of Changes for Revised TS: "6.7, "High Radiation Area">

Insert 6.5-A A.3

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, Containment Spray, Emergency Cooling, Shutdown Cooling, Reactor Cleanup, Vacuum Relief, Reactor Water Sampling, Containment Atmosphere Dilution (CAD) H_2-O_2 Monitor, Drywell Containment Atmosphere Monitoring (CAM), Post Accident Sampling, Radioactive Gaseous Effluent Monitoring (RAGEMS), Offgas Effluent Stack Monitoring (OGESMS), and Post Accident Vent to Reactor Building Emergency Ventilation. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. System leak test requirements for each system at 24 month intervals.

A.4

The provisions of Specification 4.0.1 are applicable to the 24 month frequency for performing system leak test activities.

Insert 6.5-B M.1

6.5.3 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to the Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or A.2

2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of 6.5.3.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

DISCUSSION OF CHANGES
REVISED TS: 6.5 – PROGRAMS AND MANUALS

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 NUREG-1434, Revision 1, and the NMP2 ITS state that licensees may make changes to the TS Bases without prior NRC approval provided the changes do not involve "A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59." The proposed change would revise the quoted phrase to "A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." This change is consistent with the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999, as noted in NRC-approved TSTF-364, Revision 0. The final rule clarifies the specific types of changes, tests, and experiments conducted at a licensed facility that require evaluation, and revises the criteria that licensees must use to determine when NRC approval is needed before such changes, tests, or experiments can be implemented. The final rule also adds definitions of terms that have been subject to differing interpretations, and reorganizes the rule language for clarity. This change to Revised TS 6.5.3 is administrative in nature.

A.3 CTS 6.14, "Systems Integrity", contains a brief statement indicating that the requirements and recommendations of Section 2.1.6.a of NUREG-0578 will be met or exceeded. In Revised TS 6.5.2, this statement is replaced with a more descriptive paragraph that outlines the elements of the program, and lists the systems to which the program applies. The revised program description is consistent with Section 2.1.6.a of NUREG-0578 and NUREG-1434. These are administrative changes that do not alter the existing requirements.

A.4 A statement of applicability of Specification 4.0.1 has been added to CTS 6.14 (Revised TS 6.5.2). This statement is needed to maintain allowances for Surveillance Interval extensions contained in the TS, since Specification 4.0.1 is not normally applied to intervals identified in the Administrative Controls section of the TS. Since this change is a clarification required to maintain provisions that would be allowed in the Limiting Conditions for Operation sections of the TS, it is considered administrative in nature.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

M.1 This change proposes to add new Section 6.5.3, Technical Specifications (TS) Bases Control Program, to Revised TS Section 6.0. This program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the TS Bases. The proposed program is identical to NMP2 TS Section 5.5.10, which was issued by the NRC in NMP2 License Amendment No. 91, except as noted in Discussion of Change A.2 above. This change does not revise any safety limits, limiting conditions for operation or surveillance test requirements for the plant. TS Bases are not considered part of the TS as documented in 10 CFR 50.36.

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

None

"Specific"

None

ATTACHMENT B.7

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.6
REPORTING REQUIREMENTS

Current Technical Specification Markup and Discussion of Changes

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved within 14 days of implementation by the branch manager for the functional area of the procedure or higher levels of management as governed by administrative procedures.

6.6 ~~6.9~~ Reporting Requirements

A.1

~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations,~~ the following ~~identified~~ reports shall be submitted in accordance with 10 CFR 50.4.

LA.1

~~6.9.1 Routine Reports~~

~~a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

Radiation

d

6.6.1 ~~Annual~~ Occupational Radiation Exposure Report. A tabulation shall be submitted on an annual basis which includes the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,(1) e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

L.1

6.6.4 Monthly Operating Report. Routine reports of operating statistics and shutdown experience ~~including documentation of challenges to the safety relief valves or safety valves,~~ shall be submitted on a monthly basis, ~~which will include a narrative of operating experience,~~ ~~in accordance with 10 CFR 50.4,~~ no later than the 15th of each month following the calendar month covered by the report.

A.3

~~(1) This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.~~

6.6.2 d. Annual Radiological Environmental Operating Report*.

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1, 1985.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls as appropriate, and with environmental surveillance reports from the previous 5 years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.6.22.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.6.21; discussion of all deviations from the sampling schedule of Table 3.6.20-1; and discussion of all analyses in which the LLD required in Table 4.6.20-1 was not achievable.

* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.6.3 ⓑ Semi-annual Radioactive Effluent Release Report**

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin on January 1, 1985.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the Offsite Dose Calculation Manual.

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* In lieu of submission with the Semi-annual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

** A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.6.3 (Cont'd) The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and,

f. Solidification agent or absorbent (e.g., cement)

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.6.20.

~~Changes to the Process Control Program (PCP) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:~~

~~a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;~~

~~b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and~~

~~c. Documentation of the fact that the change has been reviewed and found acceptable.~~

LA.2

See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals" A.1

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

6.6.5 f. CORE OPERATING LIMITS REPORT (COLR)

a. 1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specifications 3.1.7.a and 3.1.7.e.

2) The K_f core flow adjustment factor for Specification 3.1.7.c.

3) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.1.7.c and 3.1.7.e.

4) The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

5) The Power/Flow relationship for Specifications 3.1.7.d and ~~e.~~ 3.1.7.e.

and shall be documented in the ~~CORE OPERATING LIMITS REPORT.~~ COLR.

b. 2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in ~~the following documents.~~

NEDE-24011-P-A, "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (Latest approved revision as specified in the COLR). A.2

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A.2

1) NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL" (Latest approved revision).

2) NEDE-30966-P-A "SAFER MODEL FOR EVALUATION OF LOSS-OF-COOLANT ACCIDENTS FOR JET PUMP AND NON-JET PUMP PLANTS" (Latest Approved Revisions)

Vol I "SAFER LONG TERM INVENTORY MODEL FOR BWR LOSS-OF-COOLANT ACCIDENT ANALYSIS"

Vol II "SAFER APPLICATION METHODOLOGY FOR NON-JET PUMP PLANTS"

3) NEDO-20566-P-A "GENERAL ELECTRIC COMPANY ANALYTICAL MODEL FOR LOSS-OF-COOLANT ACCIDENT ANALYSIS IN ACCORDANCE WITH 10CFR50 APPENDIX K". (Latest approved revision)

4) NEDO-32465-A, "REACTOR STABILITY DETECT AND SUPPRESS SOLUTIONS LICENSING BASIS METHODOLOGY FOR RELOAD APPLICATIONS," August 1996.

c. 3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. 4 The COLR ~~CORE OPERATING LIMITS REPORT~~, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC. ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~

A.3

6.9.2 Fire Protection Program Reports

Noncompliances with the Fire Protection Program (as described in the Final Safety Analysis Report) that adversely affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10CFR50.72 and 10CFR50.73.

LA.3

6.6.6
~~6.9.3~~ Special Reports

A.3

Special reports shall be submitted ~~in accordance with 10 CFR 50.4 Regional Office~~ within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.2.2(b) (12 months).

b. Safety Class 1 Inservice Inspection, Specification 4.2.6 (Three months).

c. Safety Class 2 Inservice Inspections, Specification 4.2.6 (Three months).

d. Safety Class 3 Inservice Inspections, Specification 4.2.6 (Three months).

e. Primary Containment Leakage Testing, Specification 3.3.3 (Three months).

f. Secondary Containment Leakage Testing, Specification 3.4.1 (Three months).

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months).

h. Calculate Dose from Liquid Effluent in Excess of Limits, Specification 3.6.15.a(2)(b) (30 days from the end of the affected calendar quarter).

i. Calculate Air Dose from Noble Gases Effluent in Excess of Limits, Specification 3.6.15.b(2)(b) (30 days from the end of the affected calendar quarter).

j. Calculate Dose from I-131, H-3 and Radioactive Particulates with half lives greater than eight days in Excess of Limits, Specification 3.6.15.b(3)(b) (30 days from the end of the affected calendar quarter).

k. Calculated Doses from Uranium Fuel Cycle Source in Excess of Limits, Specification 3.6.15.d (30 days from the end of the affected calendar year).

l. Inoperable Gaseous Radwaste Treatment System, Specification 3.6.16.b (30 days from the event).

m. <u>Environmental Radiological Reports</u>. With the level of radioactivity (as the result of plant effluents) in an environmental sampling medium exceeding the reporting level of Table 6.6.6-1 ~~6.9.3-1~~, when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within thirty (30) days from the end of the calendar quarter a special report identifying the cause(s) for exceeding the limits, and define the corrective action to be taken.

6.6.6-1

TABLE 6.9.3-1
REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95, Nb-95	400				
I-131	2**	0.9		3	100
Cs-134	30	10.0	1,000	60	1,000
Cs-137	50	20.0	2,000	70	2,000
Ba/La-140	200			300	

* For drinking water samples. This is a 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 20 pCi/liter may be used.

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 CTS 6.9.1.f, Item 2, identifies specific analytical methods used to determine the core operating limits that are documented in the COLR. The proposed change deletes the references to three (3) of the identified reports (NEDE-30966-P-A, NEDO-20556-P-A, and NEDO-32465-A), and retains only the reference to NEDE-24011-P-A. NEDE-24011-P-A now contains all of the methods reviewed and approved by the NRC for the NMP1 Loss of Coolant Accident analysis and for the Stability Analysis. Therefore, the references to the other three reports (NEDE-30966-P-A, NEDO-20556-P-A, and NEDO-32465-A) are redundant. This change is administrative in nature. There are no changes to the actual analytical methods being used.

A.3 CTS 6.9.1.f, Item 4, requires that the Core Operating Limits Report (COLR) shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, CTS 6.9.1.c requires that monthly operating reports be submitted in accordance with 10 CFR 50.4, and CTS 6.9.3 requires that special reports be submitted "in accordance with 10 CFR 50.4 Regional Office." Revised TS 6.6 contains a single statement that requires submittal of reports in accordance with 10 CFR 50.4. The TS do not need to give report submittal details since this material is subject to change and would require a change to the TS. The Revised TS submittal requirements are sufficient without including unnecessary duplication or details.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.9.1.a requires that a startup report be submitted detailing plant startup and power escalation testing following receipt of an operating license, an increase in licensed power level, installation of nuclear fuel with a different design of manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The proposed change would relocate this requirement to the UFSAR. The startup report required by CTS 6.9.1.a provides the NRC with a mechanism to review the appropriateness of licensee activities after-the-fact, but there is no requirement for the NRC to approve the report. The quality assurance requirements of 10 CFR 50, Appendix B, and the Startup Test Program provisions contained in the UFSAR provide assurance that the listed activities will be adequately performed and that appropriate corrective actions, if required, are taken. Also, given that the report may be submitted to the NRC up to 90 days following completion of the respective milestone, report completion and submittal is clearly not necessary to assure operation of the unit in a safe manner for the interval between completion of the startup testing and submittal of the report. Thus, the startup report is not required to be in the TS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LA.2 The details contained in CTS 6.9.1.e regarding changes to the Process Control Program (PCP) are proposed to be relocated to the UFSAR. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the NMP1 operating license and, as such, relocation of the requirements regarding changes to the PCP from the TS does not affect the safe operation of the facility. Therefore, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LA.3 The details contained in CTS 6.9.2, "Fire Protection Program Reports," are proposed to be relocated to the UFSAR (Appendix 10A), where the program requirements currently reside. This program is required by an NMP1 commitment to Branch Technical Position APCSB 9.5-1, Appendix A, as stated in the UFSAR, Appendix B. Revised TS 6.4.1 will continue to require that procedures shall be established to implement and maintain the Fire Protection Program. This is consistent with Generic Letter 88-12, which allowed the Fire Protection Program requirements to be relocated to plant controlled documents. Therefore, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Fire Protection Program changes are controlled by the provisions of Paragraph 2.D(7) of the Operating License.

"Specific"

L.1 The reporting of safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, Revised Contents of the Monthly Operating Report," requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the safety and relief valves. As noted in NRC-approved TSTF-258, Revision 4, an NRC staff member (AEOD) was contacted and he indicated that this information was not required for the Performance Indicator Program and therefore would not need to be reported. Based on this information, it is acceptable to delete the requirement to provide documentation of all challenges to safety relief valves or safety valves.

ATTACHMENT B.8

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

REVISED TS 6.7
HIGH RADIATION AREA

Current Technical Specification Markup and Discussion of Changes

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

6.11 Radiation Protection Program

<See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals">

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.7 ~~6.12~~ High Radiation Area

~~6.12.1~~ 6.7.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~Manager Radiation Protection~~ radiation protection manager or designate in the Radiation Work Permit. LA.1

6.7.2 ~~6.12.2~~ In addition to the requirements of 6.7.1 ~~6.12.1~~ areas accessible* to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem** shall be provided with locked doors to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Station Shift Supervisor or designate on duty and/or the ~~Manager Radiation Protection~~ radiation protection manager or designate. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify in accordance with site approved procedures accordingly, the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the drywell, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* by accessible passage and permanently fixed ladders
** measurement made at 18" from source of radioactivity

DISCUSSION OF CHANGES
REVISED TS: 6.7 – HIGH RADIATION AREA

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.7.1 and CTS 6.7.2 use the title "Manager Radiation Protection." This specific title is replaced with the generic title "radiation protection manager." The specific title is proposed to be relocated to UFSAR Section XIII-A, which is where the organizational chart and description of this specific title are currently located. Relocation of specific titles out of the TS is consistent with the NRC letter from C. Grimes to the Owners Group Technical Specification Committee Chairman, dated November 10, 1994, as documented in NRC-approved TSTF-65, Revision 1. The various requirements of the individuals are still retained in the Revised TS. In addition, Revised TS 6.2.1 requires the organization chart to be documented in the UFSAR. Therefore, the relocated specific titles are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

"Specific"

None

ATTACHMENT B.9

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

MISCELLANEOUS REVISED TS PAGE CHANGES

<u>Current Technical Specification Markup and Discussion of Changes</u>

1.28 Ventilation Exhaust Treatment System

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

1.29 Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification ~~6.9.11~~ 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point (5 feet below minimum normal water level). The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operation during the major maintenance with the water level below the low-low level set point.

In addition to the ~~Facility Staff~~ requirement~~s~~ ~~given in Specification 6.2.2.b,~~ that at least one licensed Operator be in the control room when fuel is in the reactor, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

A.2

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux for IRM range 9 or lower.

The IRM scram trip setting shall not exceed 38.4% of rated neutron flux for IRM range 10.

c. The reactor high pressure scram trip setting shall be ≤ 1080 psig.

d. The reactor water low level scram trip setting shall be no lower than -12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be ≥850 psig when the reactor mode switch is in the run position or the IRMs are on range 10.

g. The main-steam-line isolation valve closure scram setting shall be ≤10 percent of valve closure (stem position) from full open.

A.1

Miscellaneous

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10 CFR 100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

To assure that periodic surveillances of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Specification:

Whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required, the primary containment leakage rate shall be limited to:

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. The primary containment leakage rates shall be demonstrated at test schedules and in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan as described in Specification ~~6.16~~ 6.5.4.

b. The provisions of Specification 4.0.1 are not applicable, and the surveillance interval extensions are in accordance with the 10 CFR 50 Appendix J Testing Program Plan.

Page 3 of 13

Miscellaneous

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

(2) Dose

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas (see Figures 5.1-1) shall be limited:

(a) During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

(b) During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification ~~6.9.3~~ 6.6.6 a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENT

(2) Dose

Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual, prior to each release of a batch of liquid waste.

Page 4 of 13

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

(2) Air Dose

The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 5 milliroentgen for gamma radiation and less than or equal to 10 mrads for beta radiation and,

(b) During any calendar year: Less than or equal to 10 milliroentgen for gamma radiation and less than or equal to 20 mrads for beta radiation.

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification ~~6.9.3~~ 6.6.6, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

(2) Air Dose

Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

LIMITING CONDITION FOR OPERATION

(3) Tritium, Iodines and Particulates

The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

(b) During any calendar year: Less than or equal to 15 mrems to any organ.

With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.6.6 ~~6.9.3~~, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENT

(3) Tritium, Iodines and Particulates

Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

NOTES FOR TABLE 4.6.15-2

(a) The LLD is defined in notation (a) of Table 4.6.15-1.

(b) Purge is defined in Section 1.23.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report pursuant to Specification ~~6.9.1~~. 6.6.3

(d) Sampling and analysis shall also be performed following shutdown, startup or an increase on the recombiner discharge monitor of greater than 50 percent, factoring out increases due to changes in thermal power level or dilution flow; or when the stack release rate is in excess of 1000 μCi/second and steady-state gaseous release rate increases by 50 percent.

(e) The sample flow rate and the stack flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.6.15.b.(1).(b) and 3.6.15.b.(3).

(f) When the release rate is in excess of 1000 μCi/sec and steady state gaseous release rate increases by 50 percent. The iodine and particulate collection device shall be removed and analyzed to determine the changes in iodine-131 and particulate release rate. The analysis shall be done daily following each change until it is shown that a pattern exists which can be used to predict the release rate; after which it may revert to weekly sampling frequency. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(g) When RAGEMS is inoperable the LLD for noble gas gross gamma analysis shall be 1×10^{-4}.

(h) Tritium grab samples shall be taken weekly from the station ventilation exhaust (stack) when fuel is offloaded until stable tritium release levels can be demonstrated.

A.1

Miscellaneous

LIMITING CONDITION FOR OPERATION

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.6.15.a(2)(b), 3.6.15.b(2)(b) and 3.6.15.b(3)(b), calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above listed 40CFR190 limits have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification ~~6.9.3~~ 6.6.6, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report.

SURVEILLANCE REQUIREMENT

Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual. This requirement is applicable only under conditions set forth in Specification 3.6.15.d.

Page 8 of 13

LIMITING CONDITION FOR OPERATION

With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification ~~6.9.3~~ 6.6.6, Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.

c. Solid

The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

SURVEILLANCE REQUIREMENT

c. Solid

The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms).

(1) If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program.

(2) If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification.

A.1

Miscellaneous

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting levels of Table ~~6.9.3-1~~ 6.6.6-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification ~~6.9.3~~ 6.6.6. The Special Report shall identify the cause(s) for exceeding the limit(s) and define the corrective action(s) to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3). When more than one of the radionuclides in Table ~~6.9.3-1~~ 6.6.6-1 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots\ldots \geq 1.0$$

When radionuclides other than those in Table ~~6.9.3-1~~ 6.6.6-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specification 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3).

Page 10 of 13

NOTES FOR TABLE 4.6.20-1

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification ~~6.9.1.d~~. 6.6.2

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N.545 (1975), Section 4.3. Allowable exceptions to ANSI N.545 (1975), Section 4.3 are contained in the Nine Mile Point Unit 1 Offsite Dose Calculation Manual (ODCM).

(c) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\text{LLD} = \frac{4.66\, S_b}{E \bullet V \bullet 2.22 \bullet Y \bullet \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, where applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

NOTES FOR TABLE 4.6.20-1

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for the particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification ~~6.9.1.d~~ 6.6.2.

A.1

Miscellaneous

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If the D/Q value at a new milk sampling location is significantly greater (50%) than the D/Q value at an existing milk sampling location, add the new location to the radiological environmental monitoring program within 30 days. The sampling location(s) excluding the control station location, having the lowest calculated D/Q may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification ~~6.9.1.e~~ 6.6.3 identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).	

DISCUSSION OF CHANGES
REVISED TS: MISCELLANEOUS PAGE CHANGES

ADMINISTRATIVE (A)

A.1 Editorial changes, reformatting, and revised numbering have been adopted to make the Revised TS consistent with the Nine Mile Point Unit 2 Improved Technical Specifications (which are consistent with the BWR Standard Technical Specifications, NUREG-1434, Revision 1).

A.2 Details of the minimum shift crew requirements located in CTS 6.2.2.b are proposed to be relocated to the UFSAR. The reference to CTS 6.2.2.b on TS Page 11 is replaced by stating the CTS 6.2.2.b requirement; i.e., that at least one licensed Operator be in the control room when fuel is in the reactor. Technical changes to minimum shift crew requirements are addressed in the Discussion of Changes for Revised TS 6.2.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

None

"Specific"

None

ATTACHMENT B.10

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.4
TRAINING

Current Technical Specification Markup and Discussion of Changes

See Discussion of Changes for Revised TS: 6.3, "Facility Staff Qualifications"

Current Specification 6.4

6.3 Facility Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for; the Manager Operations who, in lieu of meeting the senior reactor operator license requirements of ANSI N18.1-1971, shall 1) hold a senior reactor operator license at the time of appointment, or 2) have held a senior reactor operator license at Nine Mile Point Nuclear Station Unit 1 or at a similar unit, or 3) have been certified for equivalent senior reactor operator knowledge; the Manager Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; and the Shift Technical Advisor who shall have a bachelor's degree in a physical science or engineering or a professional engineer license issued by examination and shall have received specific training in plant design, and response and analysis of the plant for transients and accidents.

LA.1

~~6.4 Training~~

~~6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Manager Training and shall meet or exceed the recommendations and requirements of Section 5.5 of ANSI N18.1-1971 and of 10CFR Part 55, and shall include familiarization with relevant industry operational experience.~~

~~6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager Training and Supervisor-Fire Protection, Nuclear and shall meet or exceed the requirements of Appendix R to 10CFR50.~~

LA.2

6.5 Review and Audit

6.5.1 Station Operations Review Committee (SORC)

Function

6.5.1.1 The Station Operations Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

Composition

6.5.1.2 The SORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman/Member:	Manager Operations
Vice Chairman/Member:	Manager Technical Support
Member:	Manager QA Operations
Member:	Manager Maintenance
Member:	Manager Chemistry
Member:	Manager Radiation Protection

See Discussion of Changes for CTS: 6.5, "Review and Audit"

ADMINISTRATIVE (A)

None

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.4.1 discusses the training and replacement training program for the facility staff. The proposed change would relocate the details of this training program to the UFSAR. These training provisions are adequately addressed by other proposed TS Section 6.0 provisions and by regulations. Revised TS 6.3, "Facility Staff Qualifications," provides requirements to assure adequate, competent staff in accordance with ANSI/ANS N18.1-1971 and Regulatory Guide 1.8, September 1975. Revised TS 6.2 details facility staff requirements. Revised TS 6.2.2.a and 10 CFR 50.54 state minimum shift crew requirements. Training and requalification of licensed positions is contained in 10 CFR Part 55. Thus, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

LA.2 CTS 6.4.2 discusses the training program for the Fire Brigade. The proposed change would relocate the details of this training program to the Fire Hazards Analysis (UFSAR Appendix 10A). The Fire Protection requirements have previously been relocated to the UFSAR in accordance with Generic Letter 88-12; therefore, the fire brigade requirement with respect to training is not needed in the TS. The relocated requirements will assure an adequate training program is maintained in accordance with NMP1 commitments and regulations. As such, these relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the procedures governing the conduct of operations, including the areas of organization, position titles, responsibilities, shift staffing, personnel qualifications and training programs, are controlled under 10 CFR 50 Appendix B programs.

"Specific"

None

ATTACHMENT B.11

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.5
REVIEW AND AUDIT

Current Technical Specification Markup and Discussion of Changes

⟨See Discussion of Changes for Revised TS: 6.3, "Facility Staff Qualifications"⟩

Current Specification 6.5

6.3 Facility Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for; the Manager Operations who, in lieu of meeting the senior reactor operator license requirements of ANSI N18.1-1971, shall 1) hold a senior reactor operator license at the time of appointment, or 2) have held a senior reactor operator license at Nine Mile Point Nuclear Station Unit 1 or at a similar unit, or 3) have been certified for equivalent senior reactor operator knowledge; the Manager Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975; and the Shift Technical Advisor who shall have a bachelor's degree in a physical science or engineering or a professional engineer license issued by examination and shall have received specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 Training

⟨See Discussion of Changes for CTS: 6.4, "Training"⟩

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Manager Training and shall meet or exceed the recommendations and requirements of Section 5.5 of ANSI N18.1-1971 and of 10CFR Part 55, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager Training and Supervisor-Fire Protection, Nuclear and shall meet or exceed the requirements of Appendix R to 10CFR50.

6.5 Review and Audit

LA.1

6.5.1 Station Operations Review Committee (SORC)

Function

6.5.1.1 The Station Operations Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

Composition

6.5.1.2 The SORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman/Member:	Manager Operations
Vice Chairman/Member:	Manager Technical Support
Member:	Manager QA Operations
Member:	Manager Maintenance
Member:	Manager Chemistry
Member:	Manager Radiation Protection

Alternates

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman or Vice-Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SORC activities at any one time.

Meeting Frequency

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman, Vice-Chairman, or a designated alternate.

Quorum

6.5.1.5 The quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, or a Vice-Chairman, and four members, including alternates.

Responsibilities

6.5.1.6 The SORC shall be responsible for:

a. Review of all REPORTABLE EVENTS.

b. Review of unit operations to detect potential safety hazards.

c. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Safety Review and Audit Board.

d. Investigation of violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

LA.1

Authority

6.5.1.7 The SORC shall:

a. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6 (a) through (d) above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President - Nuclear Generation and the Safety Review and Audit Board of disagreement between the SORC and the Plant Manager; however, the Plant Manager shall have the responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Records

6.5.1.8 The SORC shall maintain written minutes of each meeting that, at a minimum, document the result of all SORC activities performed under the responsibilities and authority provisions of this section. ~~and~~ copies shall be provided to the Vice President - Nuclear Generation and the Safety Review and Audit Board.

6.5.2 Technical Review and Control

Activities

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto. Approval of procedures and programs and changes thereto and their safety evaluations, shall be controlled by administrative procedures.

6.5.2.2 Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Plant Manager.

LA.1

6.5.2.3 ~~Proposed modifications to unit structures, systems and components that affect nuclear safety shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification.~~ Proposed modifications to structures, systems and components and the safety evaluations shall be approved prior to implementation by the Plant Manager; or the Manager Technical Support as previously designated by the Plant Manager.

A.1

Moved to Specification 6.1.1

LA.1

~~6.5.2.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the Plant Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary such review shall be performed by the appropriate designated station review personnel.~~

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications and their safety evaluations shall be reviewed by the Plant Manager, or the Manager Technical Support as previously designated by the Plant Manager.

Moved to Specification 6.1.1

A.1

~~6.5.2.6 The Plant Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President - Nuclear Generation.~~

~~6.5.2.7 The facility security program, and implementing procedures, shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.~~

~~6.5.2.8 The facility emergency plan, and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the Plant Manager and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.~~

LA.1

6.5.2.9 The Plant Manager shall assure the performance of a review by a qualified individual/organization of changes to the Radiological Waste Treatment systems.

6.5.2.10 Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

6.5.2.11 Review of changes to the Process Control Program and the Offsite Dose Calculation Manual. Approval of any changes shall be made by the Plant Manager or his designee before implementation of such changes.

6.5.2.12 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Vice President - Nuclear Generation and the Safety Review and Audit Board.

6.5.2.13 The Plant Manager shall assure the performance of a review by a qualified individual/organization of the Fire Protection Program and implementing procedures at least every 12 months and submittal of recommended changes to the Safety Review and Audit Board.

6.5.3 Safety Review and Audit Board (SRAB)

Function

6.5.3.1 The Safety Review and Audit Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices
i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

LA.1

Composition

6.5.3.2 The Safety Review and Audit Board shall be composed of the:

Chairman: Staff Engineer or Manager or Vice President
Member: Plant Manager or Designee
Member: Staff Engineer - Nuclear
Member: Staff Engineer - Mechanical or Electrical
Member: Consultant (See 6.5.3.4)

Alternates

6.5.3.3 Alternate members shall be appointed in writing by the SRAB Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SRAB activities at any one time.

Consultants

6.5.3.4 Consultants shall be utilized as determined by the SRAB Chairman to provide expert advice to the SRAB.

Meeting Frequency

6.5.3.5 The SRAB shall meet at least once per six months.

Quorum

6.5.3.6 The quorum of the SRAB necessary for the performance of the SRAB review and audit functions of these Technical Specifications shall consist of not less than a majority of the members, including alternates. The quorum requires the presence of the Chairman or the Chairman's designated alternate and no more than a minority of the quorum shall have line responsibility for operation of the facility.

LA.1

Review

6.5.3.7 The SRAB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

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

g. All REPORTABLE EVENTS.

h. All recognized ~~Any~~ indications of an unanticipated deficiency in some aspect of design or operation of ~~safety related~~ structures, systems, or components that could affect nuclear safety.

i. Reports and meeting minutes of the SORC.

LA.1

Current Specification 6.5

Audits

~~6.5.3.8 Audits of facility activities shall be performed under the cognizance of the SRAB. These audits shall encompass:~~

~~a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.~~

~~b. The performance, training and qualifications of the entire facility staff at least once per year.~~

~~c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.~~

~~d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10CFR50, at least once per two years.~~

~~e. The Facility Emergency Plan and implementing procedures at least once every 12 months.~~

~~f. The Facility Security Plan and implementing procedures at least once every 12 months.~~

~~g. The Facility Fire Protection Program and implementing procedures at least once per two years.~~

~~h. Any other area of facility operation considered appropriate by the SRAB or the Vice President - Nuclear Generation.~~

~~i. The radiological environmental monitoring program and the results thereof at least once per 12 months.~~

~~j. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.~~

~~k. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.~~

LA.1

Current Specification 6.5

Authority

6.5.3.9 The SRAB shall report to and advise the Chief Nuclear Officer on those areas of responsibility specified in Section 6.5.3.7 and 6.5.3.8.

Records

6.5.3.10 Records of SRAB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRAB meeting shall be prepared, approved and forwarded to the Chief Nuclear Officer within ~~30~~ 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 b, e,f,g and h above, shall be prepared, approved and forwarded to the Chief Nuclear Officer within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Chief Nuclear Officer and to the management positions responsible for the areas audited within ~~90~~ 30 days following completion of the ~~review~~ audit by the auditing organization.

LA.1

DISCUSSION OF CHANGES
CTS: 6.5-REVIEW AND AUDIT

ADMINISTRATIVE (A)

A.1 The requirements of CTS 6.5.2.3 and CTS 6.5.2.5 regarding Plant Manager reviews and approvals are proposed to be moved to Revised TS 6.1, "Responsibility." Technical changes to these requirements are addressed in the Discussion of Changes for Revised TS 6.1.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.5.1 describes the Station Operations Review Committee (SORC) review and audit requirements; CTS 6.5.2 describes technical review and control requirements; and CTS 6.5.3 describes the Safety Review and Audit Board (SRAB) review and audit requirements. The proposed change would relocate these requirements to the QATR, which is contained in the UFSAR as Appendix B. These changes are consistent with the guidance of Administrative Letter (AL) 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995, and NUREG-1434, Revision 1. The administrative letter concluded that TS administrative quality assurance-related requirements may be relocated to licensee-controlled quality assurance programs. For NMP1, these requirements would be relocated in their entirety (except as noted below) to the QATR, with only minor wording and formatting changes. Requirements relating to review and audit activities described in CTS 6.5 are contained in 10 CFR 50.54(p); 10 CFR 50.54(t); 10 CFR 50 Appendix B, Criterion XVIII; 10 CFR 73; ANSI/ANS 3.2-1982; ANSI N18.7-1972; and ANSI/ASME NQA1-1983, including 1983 Addenda. Relocation of these TS provisions to the QATR will provide adequate controls over review and audit activities for NMP1. Thus, the provisions are not necessary to be in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

Minor Differences Between CTS and the QATR

There are several minor differences between the current wording of CTS 6.5 and the existing wording in the QATR, as described below (*italics* added to highlight differences). These minor differences, which are shown on the marked-up TS pages, will be evaluated in accordance with the NMPC administrative procedures that implement 10 CFR 50.54(a).

1. CTS 6.5.1.8 – The TS states that SORC shall maintain written minutes of each meeting, whereas the QATR specifies that "The SORC shall maintain written minutes

of each meeting *that, at a minimum, document the result of all SORC activities performed under the responsibilities and authority provisions of the Technical Specifications and this section.*"

2. CTS 6.5.3.7.h –The TS states that SRAB shall review "Any indication of" an unanticipated deficiency, whereas the QATR specifies that SRAB shall review "*All recognized indications of*" an unanticipated deficiency. Also, the scope of the TS requirement covers "safety related structures, systems, or components," whereas the QATR specifies "structures, systems, or components *that could affect nuclear safety.*"

3. CTS 6.5.3.10.a – The TS requires that minutes of each SRAB meeting shall be prepared, approved and forwarded to the Chief Nuclear Officer within 30 days following each meeting, whereas the QATR requires *14* days for completion of these activities.

4. CTS 6.5.3.10.b – The TS requires that reports of certain SRAB reviews be prepared, approved and forwarded to the Chief Nuclear Officer within 14 days following completion of the review. The QATR includes one additional SRAB review within the scope of this requirement, that being: "*Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.*"

5. CTS 6.5.3.10.c – The TS states that SRAB audit reports shall be forwarded to the Chief Nuclear Officer within 90 days following completion of the review, whereas the QATR requires that SRAB audit reports be forwarded to the Chief Nuclear Officer *and to the management positions responsible for the areas audited* within *30* days following completion of the *audit by the auditing organization.*"

"Specific"

None

ATTACHMENT B.12

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.6
REPORTABLE EVENT ACTION

Current Technical Specification Markup and Discussion of Changes

Current Specification 6.6

A.1

6.6 ~~Reportable Occurrence Action~~

6.6.1 ~~The following actions shall be taken for REPORTABLE EVENTS:~~

a. ~~The Commission shall be notified and a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10 CFR Part 50, and~~

b. ~~Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.~~

LA.1

6.7 Safety Limit Violation

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

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear Generation and the SRAB shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, within 30 days of the violation, and to the SRAB, and the Vice President - Nuclear Generation within 14 days.

See Discussion of Changes for CTS: 6.7, "Safety Limit Violation"

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

a. Written procedures shall be established, implemented, and maintained for activities involving the Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation by the branch manager for the functional area of the procedure or higher levels of management as governed by administrative procedures. Each procedure and administrative policy of 6.8.1 above shall be reviewed periodically as set forth in administrative procedures.

See Discussion of Changes for Revised TS: 6.4, "Procedures"

DISCUSSION OF CHANGES
CTS: 6.6 – REPORTABLE EVENT ACTION

ADMINISTRATIVE (A)

A.1 CTS 6.6.1.a delineates NRC notification and report submittal requirements for Reportable Events. The proposed change would delete CTS 6.6.1.a. The notification and report submittal requirements of CTS 6.6.1.a are contained in 10 CFR 50.72 and 10 CFR 50.73. There is no need to repeat these requirements in the TS. Since these requirements are contained in the regulations, and since the NMP1 Operating License requires compliance with 10 CFR 50, deletion of this requirement from the TS is considered administrative in nature.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.6.1.b describes SORC responsibilities regarding the review of Reportable Events and submittal of the results of the reviews to the SRAB and the Vice President-Nuclear Generation. The proposed change would relocate the requirements of CTS 6.6.1.b to the QATR. The requirements of CTS 6.6.1.b duplicate the SORC responsibilities given in CTS 6.5.1.6.a and CTS 6.5.1.6.d, which are proposed for relocation to the QATR. These activities are required following the event without a specified completion time. As such, the proposed relocated requirements are not necessary to assure operation of the facility in a safe manner, and are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

"Specific"

None

ATTACHMENT B.13

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.7
SAFETY LIMIT VIOLATION

Current Technical Specification Markup and Discussion of Changes

See Discussion of Changes for CTS: 6.6, "Reportable Occurrence Action

Current Specification 6.7

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.

~~6.7 Safety Limit Violation~~

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated:~~

A.1

~~a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.~~

LA.1

~~b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear Generation and the SRAB shall be notified within 24 hours.~~

LA.1

~~c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

A.1

A.1

~~d. The Safety Limit Violation Report shall be submitted to the Commission, within 30 days of the violation, and to the SRAB, and the Vice President - Nuclear Generation within 14 days.~~

LA.1

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

a. Written procedures shall be established, implemented, and maintained for activities involving the Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved prior to implementation by the branch manager for the functional area of the procedure or higher levels of management as governed by administrative procedures. Each procedure and administrative policy of 6.8.1 above shall be reviewed periodically as set forth in administrative procedures.

See Discussion of Changes for Revised TS: 6.4, "Procedures"

DISCUSSION OF CHANGES
CTS: 6.7 – SAFETY LIMIT VIOLATION

ADMINISTRATIVE (A)

A.1 The proposed change would delete the Safety Limit Violation requirements of CTS 6.7 as they relate to NRC notification (CTS 6.7.1.a, and portions of CTS 6.7.1.b, CTS 6.7.1.c, and CTS 6.7.1.d). These requirements are contained in and based upon the requirements located in 10 CFR 50.36(c)(1), 10 CFR 50.72, and 10 CFR 50.73. Since the NMP1 Operating License requires compliance with 10 CFR 50, there is no need to repeat these requirements in the TS. Deletion of these requirements from the TS is considered administrative in nature.

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 The CTS 6.7.1.b requirement for notification of the Vice President – Nuclear Generation and the SRAB in the event of a Safety Limit Violation; the CTS 6.7.1.c requirement for SORC to review the Safety Limit Violation Report; and the CTS 6.7.1.d requirement to submit the Safety Limit Violation Report to the SRAB and the Vice President – Nuclear Generation are proposed to be relocated to the QATR. Given that the notification occurs following the Safety Limit Violation and that the Safety Limit Violation Report is an after-the-fact report, the proposed relocated requirements are clearly not necessary to assure operation of the unit in a safe manner and are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

"Specific"

None

ATTACHMENT B.14

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.10
RECORD RETENTION

Current Technical Specification Markup and Discussion of Changes

6.10 Record Retention

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. REPORTABLE EVENT REPORTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

LA.1

LA.1

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

6.11 Radiation Protection Program

See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals"

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 High Radiation Area

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

See Discussion of Changes for Revised TS: 6.7, "High Radiation Area"

DISCUSSION OF CHANGES
CTS: 6.10 – RECORD RETENTION

ADMINISTRATIVE (A)

None

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.10 delineates records retention requirements, including those records that are to be retained for at least five years (CTS 6.10.1) and those records that are to be retained for the duration of the facility operating license (CTS 6.10.2). The proposed change would relocate the requirements of CTS 6.10 to the QATR. These changes are consistent with the guidance of AL 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995, and NUREG-1434, Revision 1. The administrative letter concluded that TS administrative quality assurance-related requirements may be relocated to licensee-controlled quality assurance programs. For NMP1, these requirements would be relocated in their entirety to the QATR, with changes only to the format. Records retention requirements related to activities affecting quality are contained in 10 CFR 50 Appendix B, Criterion XVII, and other sections of 10 CFR 50 that are applicable to NMP1 (e.g., 10 CFR 50.71, 10 CFR 73, etc.). These records retention requirements provide a record of certain activities important to plant safety, but the records themselves do not assure safe operation of the facility since review of these records is a post-compliance review. Relocation of these TS provisions to the QATR will provide adequate controls over records retention requirements for NMP1. As such, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

"Specific"

None

ATTACHMENT B.15

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.13
FIRE PROTECTION INSPECTION

Current Technical Specification Markup and Discussion of Changes

LA.1

6.13 ~~Fire Protection Inspection~~

~~6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.~~

~~6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.~~

6.14 Systems Integrity

<See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals">

Procedure shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.

6.15 Iodine Monitoring

<See Discussion of Changes for CTS: 6.15, "Iodine Monitoring">

Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.8.c of NUREG 0578.

6.16 10 CFR 50 Appendix J Testing Program Plan

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topic BN-TOP-1, and
2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

Leakage Rate Surveillance Test acceptance criteria are:

1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.
2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.
3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

<See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals">

DISCUSSION OF CHANGES
CTS: 6.13 – FIRE PROTECTION INSPECTION

ADMINISTRATIVE (A)

None

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.13 requires performance of inspections and audits of the fire protection and loss prevention program, to be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm (CTS 6.13.1), and at intervals no greater than 3 years by an outside qualified fire consultant (CTS 6.13.2). The proposed change would relocate the requirements of CTS 6.10 to the QATR as activities performed under the cognizance of the Safety Review and Audit Board (SRAB). These changes are consistent with the guidance of AL 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995, and NUREG-1434, Revision 1. The administrative letter concluded that TS administrative quality assurance-related requirements may be relocated to licensee-controlled quality assurance programs. For NMP1, the requirements of CTS 6.13 would be relocated in their entirety the QATR, with changes only to the format. Requirements relating to review and audit activities of the SRAB are contained in 10 CFR 50 Appendix B, Criterion XVIII; ANSI/ANS 3.2-1982; ANSI N18.7-1972; ANSI/ASME NQA1-1983, including 1983 Addenda; and Branch Technical Position ASCSB 9.5-1. Relocation of these TS provisions to the QATR will provide adequate controls over inspection and audit activities relating to the fire protection program for NMP1. As such, the provisions are not necessary to be included in the TS to provide adequate protection of the public health and safety. Changes to the QATR are controlled by the provisions of 10 CFR 50.54(a).

"Specific"

None

ATTACHMENT B.16

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

PROPOSED REVISED TECHNICAL SPECIFICATION SECTION 6.0
ADMINISTRATIVE CONTROLS

CTS 6.15
IODINE MONITORING

Current Technical Specification Markup and Discussion of Changes

See Discussion of Changes for CTS: 6.13, "Fire Protection Inspection"

6.13 Fire Protection Inspection

6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals"

6.14 Systems Integrity

Procedure shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.

~~6.15 Iodine Monitoring~~

~~Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.c of NUREG 0578.~~

LA.1

6.16 10 CFR 50 Appendix J Testing Program Plan

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topic BN-TOP-1, and
2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

Leakage Rate Surveillance Test acceptance criteria are:

1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.
2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.
3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

See Discussion of Changes for Revised TS: 6.5, "Programs and Manuals"

ADMINISTRATIVE (A)

None

TECHNICAL CHANGES – MORE RESTRICTIVE (M)

None

TECHNICAL CHANGES – LESS RESTRICTIVE (L, LA)

"Generic"

LA.1 CTS 6.15 discusses the iodine monitoring program. The proposed change would relocate the details of this program to the UFSAR. This program is required by the NMP1 commitment to NUREG-0737, Item III.D.3.3 (NUREG-0578, Section 2.1.8.c). This program contains controls to assure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions, and is designed to minimize radiation exposure to plant personnel post-accident. The training aspect of the program is accomplished as part of the continuing training for personnel in the cognizant organizations, as well as during the training for those individuals responsible for implementing the Radiological Emergency Planning procedures. Provisions for monitoring and performing maintenance of the sampling and analysis equipment are addressed in chemistry and radiation protection procedures. Therefore, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

"Specific"

None

ATTACHMENT C

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

No Significant Hazards Consideration Analysis

INTRODUCTION

10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in 10 CFR 50.92 concerning the issue of no significant hazards consideration (NSHC). According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

Niagara Mohawk Power Corporation (NMPC) has evaluated this proposed amendment pursuant to 10 CFR 50.91 and has determined that it involves no significant hazards consideration. A single NSHC analysis has been performed for each of the Administrative (A), More Restrictive (M), and Less Restrictive-Generic (LA) change categories. For each Less Restrictive-Specific (L) change, there is a corresponding unique NSHC analysis that is identified by an alpha-numeric designator relating the marked-up CTS and Discussion of Change (DOC) to the applicable NSHC analysis.

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussion)

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES – MORE RESTRICTIVE
("M.x" Labeled Comments/Discussion)

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to assure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive administrative requirements has no impact on the margin of plant safety. As provided in the discussion of the change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, the change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO UFSAR OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussion)

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the UFSAR or other plant controlled documents. The UFSAR and other plant controlled documents containing the relocated information will be maintained in accordance with the applicable change control process (e.g., 10 CFR 50.59, 10 CFR 50.54(a), etc.). The UFSAR is also subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the UFSAR or other plant controlled documents will be evaluated per the applicable change control process requirements, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the UFSAR or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these details in the UFSAR or other plant controlled documents will be evaluated per the requirements of the applicable change control process, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, these details proposed for relocation do not have a specific margin of safety to evaluate. Since the proposed change is consistent with the NMP2 ITS and with the BWR Standard Technical Specifications, NUREG-1434, Revision 1, both approved by the NRC staff, revising the Technical Specifications to reflect the approved level of detail assures no significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the actual title of an individual designated by the plant manager and replaces it with the term "designee." The approval of modifications or of proposed tests and experiments is not considered as an initiator of any previously evaluated accident. The proposed change will not impact the correctness of the modification or proposed test or experiment. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the Revised TS continues to assure that the plant manager is responsible for the safe operation of the unit. Therefore, this change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change deletes the actual title of an individual designated by the plant manager and replaces it with the term "designee." The Revised TS continues to assure that the plant manager is responsible for the safe operation of the unit. Thus, while the plant manager can delegate the authority to approve modifications or proposed tests and experiments, the plant manager cannot delegate the responsibility for safe operation of the unit (except in the plant manager's absence); the plant manager maintains responsibility. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the description of the individuals designated by the plant manager to approve modifications to overtime requirements, and replaces it with the term "designee." The approval of modifications to the overtime requirements is not considered an initiator of any previously evaluated accident. The proposed change will not impact the correctness of modifying the requirements. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the Revised TS continues to assure that the plant manager is responsible for the safe operation of the unit. Therefore, this change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change deletes the description of the individuals designated by the plant manager to approve modifications to overtime requirements, and replaces it with the term "designee." The Revised TS continues to assure that the plant manager is responsible for the safe operation of the unit. Thus, while the plant manager can delegate the authority to approve modifications to the overtime requirements, the plant manager cannot delegate the responsibility for safe operation of the unit (except in the plant manager's absence); the plant manager maintains responsibility. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
<u>("L.x" Labeled Comments/Discussion)</u>

There are no plant specific less restrictive changes identified for this specification.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

There are no plant specific less restrictive changes identified for this specification.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

There are no plant specific less restrictive changes identified for this specification.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specification change and has determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change alters the content of the Monthly Operating Report by deleting the requirement to document challenges to the pressure relief valves or safety valves. This change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. The proposed change does not physically alter the valves, change their functions or performance characteristics, or affect requirements for maintaining the valves. Thus, accidents previously evaluated (Inadvertent Actuation of One Solenoid Relief Valve, UFSAR Section XV-B.3.11) will be no more likely to occur, and performance of the valves assumed in accidents previously evaluated (Main Steam Isolation Valve Closure (With Scram), UFSAR Section XV-B.3.5; Safety Valve Actuation (Overpressurization Analysis, UFSAR Section XV-B.3.12; and Loss-of-Coolant Accident, UFSAR Section XV-C.2.0), are not affected. Therefore, this change does not involve a significant increase in the probability or consequences of accidents previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not physically alter the pressure relief valves or safety valves, change the valve functions or performance characteristics, or affect requirements for maintaining the valves. The change does not introduce new modes of plant operation or eliminate any actions required to prevent or mitigate accidents. Deletion of the requirement to report challenges to the pressure relief valves or safety valves in the Monthly Operating Report is administrative in nature. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change alters the content of the Monthly Operating Report by deleting the requirement to document challenges to the pressure relief valves or safety valves. This change is administrative in nature. It does not physically alter the valves, change their functions or performance characteristics, or affect requirements for maintaining the valves. There is no effect on the assumptions of design basis accidents, and no impact on safe operation of the plant. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

There are no plant specific less restrictive changes identified for this specification.

TECHNICAL CHANGES - LESS RESTRICTIVE
("L.x" Labeled Comments/Discussion)

There are no plant specific less restrictive changes identified for these miscellaneous pages.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.4 – TRAINING

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.5 – REVIEW AND AUDIT

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.6 – REPORTABLE EVENT ACTION

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.7 – SAFETY LIMIT VIOLATION

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.10 – RECORD RETENTION

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.13 – FIRE PROTECTION INSPECTION

There are no plant specific less restrictive changes identified for this specification.

NO SIGNIFICANT HAZARDS EVALUATION
CTS: 6.15 – IODINE MONITORING

There are no plant specific less restrictive changes identified for this specification.

ATTACHMENT D

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Eligibility for Categorical Exclusion from Performing an Environmental Assessment

10CFR51.22 provides criteria for, and identification of, licensing and regulatory actions eligible for exclusion from performing an environmental assessment. Niagara Mohawk Power Corporation has reviewed the proposed amendment and determined that it does not involve a significant hazards consideration, and there will be no significant change in the types or a significant increase in the amounts of any effluents that may be released offsite; nor will there be any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10) and, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment is required to be prepared in connection with this license amendment application.