MEMO FOR YOUR FILES Oct 27, 1992

TO: U.S. Nuclear Regulatory Commisson
Executive Director for Operations
Public Document Room
1717 H Street
Washington, DC 20555

FROM: Ben L. Ridings
P.O. Box 1101
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Ref: Petition pursuant 10CFR2.206

Dear Sirs:

Enclosed for filing PETITION FOR EMERGENCY ENFORCEMENT ACTION AND REQUEST FOR PUBLIC HEARING.

Respectfully submitted,

Ben L Ridings

Ben L. Ridings

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UNITED STATED OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

PETITION FOR EMERGENCY ENFORCEMENT ACTION AND REQUEST FOR PUBLIC HEARING

I. INTRODUCTION

I, BEN L. RIDINGS (hereinafter "Petitioner") hereby petition the Commissioners of the Nuclear Regulatory Commission ("NRC" or "Commission") for emergency enforcement action against Niagra Mohawk's Nine Mile (Unit One) Nuclear power plant, which is operating in violation of both the NRC and Federal requirements for availability of Emergency Core Cooling (ECCS) high pressure core injection. As an ECCS system, the Nine Mile plant also fails to provide the mandatory emergency backup power to the high pressure core injection (HPCI) system . Over the twenty years the Nine Mile One plant has been allowed to operate, no safety related pumps have ever been available to inject water into the vessel at reactor pressure. At the same time this plant was allowed to operate at full power, there are many postulated accidents assumed in the Final Safety Analysis Report (that are capable of draining the reactor vessel) and specifically rely on the ECCS HPCI Pumps to maintain reactor water level. These pumps have never been installed and the current administrative controls allowed this plant to operate outside the minimum federal requirement. This specific type of plant operation outside the known minimum federal requirements greatly endangers health and property risk to the public.

As discussed in detail below, the responsible utility, its Quality Assurance group and the NRC have routinely failed in their responsibility to ensure the operation of nuclear power plants within the license agreement. Even when problems are identified, documented and brought to the attention of the responsible parties, various safety concerns are routinely dismissed, ignored or

administratively eliminated. Even issues which obviously endanger public safety have been routinely dismissed, not only by the utility but such actions authorized and approved by the independent quality assurance groups and by the NRC. Any and all of these organizations have the authority to stop the operation of plants outside the minimum safety requirements, and not one have come forward to fulfill its duty and protect the public. Instead, each organization has reviewed the enclosed safety concerns and contrary to any practical justification, have remained silent and allowed this manner of plant operation to take place with their approval, giving evidence that these groups have also failed to remain independent of each other. Independent review by not only the government agency but the quality assurance review groups is the basic premise which allowed congress to grant operation of commercial nuclear power plants with limited liability for damages. The current administrative controls used today failed to ensure the plant operate within the minimum federal guidelines. It is Congress's duty to protect public safety and its current administrative controls have failed.

Because the Nine Mile Point Unit One Reactor violates both federal law and the Commissions's requirements for HIGH PRESSURE CORE INJECTION, the Commission can make no finding that there is resonable assurance of no undue risk to public health and safety. Petitioner therefore request that the Commission issue immediately an effective order directing the licensee to cease power operation and place the reactor in a cold shutdown condition. The plant should not be permitted to continue or resume operation unless and until subsequent tests and inspections are shown to provide the requisite reasonable assurance of no undue risk to public health and safety. Moreover, Petitioners seek a public hearing before the plant is allowed to operate again.

II. DESCRIPTION OF PETITIONER

I, Ben L. Ridings, am a technical consultant for commercial nuclear power plants. Over a span of some fifteen years, while working at some twenty four sites, I have specialized in reviewing of licensing agreement (FSAR, Technical Specifications, Federal Codes and Regulations, ASME Codes, etc.), establishing administrative controls to meet these requirements and test programs to ensure compliance at all times. My test programs and administrative controls established while under contract to various utilities are still in use today at many facilities.

III. THE COMMISSION SHOULD EXERCISE ITS SUPERVISORY JURISDICTION OVER THIS PETITION

A. The Commission has an Inherent Supervisory Jurisdiction over the Safety of Operation of the Niagra Mohawk Nine Mile Plant.

This petition is brought before the Commission pursuant to the authority granted to it in 42USC 2233(d), 2236(a),2237 and 10CFR 2.204, 2.206(c)(1), 50.54, 50.57, 50.100 and 50.109. It invokes the inherent supervisory authority of the Commission to oversee all aspects of the regulatory and licensing process and its "overriding responsibility for assuring public health and safety in the operation of nuclear power facilities." Consolidated Edison Coo. of N.Y., Inc. (Indian Point, Units 1,2 and 3). CLI-75-8, 2 NRC 173 (1975). As the Commission has previously observed, its supervisory powers include the power to order immediate shutdown of a facility "if the public health or safety so requires." Petition for Emergency and Remedial Action, CLI-78-6, 7 NRC 400, 405 (1978), citing 5 USC 558(c), 42 USC 2236(b), 10CFR 2.202(f), 2.204.

The Commission has exercised its inherent authority on a number of occasions. In addition to the ceases cited above, see Petition for Emergency and Remedial Action, CLI-80-21, 11 NRC 700(1980); U.S. Energy Research and Development Administration (Clinch River Breeder Reactor Project), CLI-76-13, 4 NRC 67, 75-76(1976); Consumers Power Co. (Midland Units 1 and 2), CLI-73-38, 6 AEC 1084 (1973); Public Service Co. of New Hampshire (Seabrook Nuclear Power Station, Units 1 and 2), CLI-77-8, 5 NRC 503, 515-517(1977).

B. Exercise of the Commissions's Independent Jurisdiction is Appropriate in This Case.

NRC regulations at 10CFR2.206 provide that under ordinary circumstances, enforcement petitions are to be lodged with the NRC Staff, and that the Commission may take discretionary review of Staff denials of such petitions. However, the Commissions's reviewing power "does not limit in any way" its "supervisory power over delegated Staff actions", 10CFR2.206(c)(1).

It is appropriate for the commission to exercise its supervisory powers and take jurisdiction in this case because the NRC Staff has acquiesced to Niagra Mohawks' violations for more than two years. In Jan 1990, Niagra Mohawk Compliance Supervisor was given written notice of HPCI and other inadequacies which effect public safety. After no apparent action, the Nine Mile Quality First Team was also given notice. Petitioner was later notified by the Quality First Team that the NRC had been contacted and made aware of the problem as well. Petitioner was later contacted by the Quality First Team and told that the NRC had exempted the plant from the HPCI requirement and its need for backup power in the event of loss of power. Petitioner has yet to hear directly from the NRC on this matter.

IV. GROUNDS FOR ENFORCEMENT ACTION

A. Federal Requirements for having radioactive fuels on site

In accordance with 10CFR50.10, the utility Niagra Mohawk entered into contractual agreement with the federal government under the provisions of public document 50-220, on file with the federal register. Now under the jurisdiction of 10CFR50, App. A (General Design Criteria), establish the minimum requirements for the principal design for water cooled nuclear power plant. Criterion 33 and 35 (Attachment 2) specify the minimum need that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core and must have suitable redundancy in components and on site electric power system (assuming offsite power is not available) which will enable the safety function to be accomplished. Also (Criterion 33), a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. Criterion 37 provides the testing requirements of the emergency core cooling system. 10CFR70 details the utility and NRC responsibility for testing and inspection of these systems and 10CFR50 App. B (Quality Assurance Criteria) details the Quality Assurance Program and the administrative requirements for Inspections, Test Control, Operating Status, Corrective Action and Records.

B. A Study of Contractual Agreement (docket 50-220)

In accordance with 10CFR50.34, the technical specification shall perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC). This investigation found the Nile Mile Point Technical Specification in compliance with this requirement. Technical Specification 4.1.8(Attachment 3) gives positive proof that the ECCS

requirement for the HPCI system was anticipated by the designers. Secondly, the corresponding Limiting Condition for Operation (LCO) 3.1.8.c (Attachment 3) view this system as so critical that if "the utility fails to verify HPCI operability it will demand an orderly shutdown be initiated within one hour. When only one HPCI component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter (as opposed to monthly demonstration)."

In accordance with the Bases for Technical Specification 3.1.8, the HPCI system is provided to ensure adequate core cooling in the unlikely event of a reactor coolant line break (also a federal requirement-design criterion 33). The HPCI system is required for line breaks which exceed the capability of the Control Rod Drive pumps and which are not large enough to allow fast enough depressurization for core spray to be effective (core spray ~350 psi as opposed to HPCI ~2200 psi).

In accordance with the Final Safety Analysis Report (FSAR), Chapter VII (Attachment 4), the Design Bases for HPCI is discussed. Although several revision have been implemented by the utility in order to fabricate the existence of a ECCS system to satisfy the HPCI federal requirement, its primary safety function is listed; (1)provide adequate cooling of the reactor core under abnormal and accident conditions, (2)remove the heat from radioactive decay and residual heat from the reactor core at such a rate that fuel clad melting would be prevented, (3)provide for continuity of core cooling over the complete range of postulated break sizes in the primary system process barrier. Once the safety functions are understood it becomes obvious as to why this system is a minimum requirement of the federal guidelines.

The following paragraph of FSAR Chapter VII gives the reader an indication of the lack of proper review that exists. At Nine Mile Point, unlike every other nuclear facility, "HPCI is not an engineered safeguards system and is not considered in any Loss of Coolant Accident Analysis." As stated in the FSAR (in layman terms) this feedwater system does not pretend to meet the 10CFR50 Appendix A (Criterion 33, 35, 36, 37) requirements of the minimum federal requirements. In fact, Nine Mile Point has no system meeting these minimum federal requirements.

Next, reviewing the Design Evaluation portion of FSAR Chapter VII, (Attachment 4) a paradox occurs in design philosophy. "During a loss-of-coolant accident within the drywell, high drywell pressure due to a line break will cause a reactor scram. The automatic scram will cause a turbine trip after a five-second delay. In order to <u>prevent cladding temperature from exceeding their maximum limit</u> for the entire spectrum of breaks, the 3800 gpm (from one train of the HPCI pumps) <u>would have to be available immediately</u>."

Obviously, the HPCI system is absolutely necessary to ensure critical heat flux (CHF) is not exceeded. Without the coolant water to transfer the heat from the fuel to the coolant, the fuel rod would then heat up rapidly and fuel cladding would take place and cause a possible melt down unless the reactor were shutdown quickly. Further, once the critical heat flux was exceeded, the departure from nucleate boiling ratio (DNBR) would exceed its 1.25 limit. These limits are Technical Specification requirements as well but it gives an indication of the interdependence of the ECCS systems. To make a statement in a license that "HPCI has not been considered in any Loss of Coolant Accident Analyses" is a another indication of the lack of proper review that exists at Nine Mile Point. Every safety limit assumed

at the Nine Mile Point plant is jeopardized without the assurance that the fuel will remain covered at all times. The NRC has approved the non-safety related feedwater system as an appropriate substitute for an ECCS HPCI federal requirement. What at first seems like a quibble about a single pump is in actuality a valid argument that every bases assumed by this license is null and void. At Nine Mile Point, standard basic thermal reactor design has been significantly altered in several ECCS systems. There are no HPCI or RCIC system to transfer heat from the reactor core. There is no way of taking steam away from the reactor and using this energy to drive a high prossure pump. Normally the HPCI pumps return the condensed steam (water) back into the vessel to maintain water level. At Nine Mile Point, there is no HPCI or RCIC systems. At Nine Mile Point, unlike normal reactor design, electrically driven, non-quality related feedwater pumps are considered. These non-quality related feedwater pumps supposedly fulfill the HPCI safety function and yet do not meet the electrical backup requirements. It must be noted that the size of these electrical pumps make it impossible to have on-site power available in the event of loss of off-site power. On-site power availability is assumed in the bases of the FSAR. It is therefore impossible for this plant to fulfill the minimum safety obligation as dictated by federal statute of the known postulated accidents.

This same feedwater system (being non-quality related) was purchased as a non-quality related system. In this same system; piping, valves, instrumentation, wiring, electrical components and control systems were all purchased and installed under non-quality related contractual provisions. HPCI automatically initiates on a Loss Coolant Accident (LOCA) signal from the NSSS logic. The NSSS logic performs the ECCS safeguard functions and

always installed under strict contractual mandates, which include training, quality assurance reviews, certified skilled craftsmen, etc. Secondly, the piping system, welding, hanger restraints and maintenance considerations were installed and maintained under non-quality related provisions as well. Again, ECCS safeguard systems are purchased, constructed and maintained under much stricter guidelines. The feedwater system was never designed, purchased, built, maintained nor capable of fulfilling the HPCI requirements of the federal guidelines. At Nine Mile Point the HPCI system simply does not exist. The administrative controls which allowed acceptance of such a non-quality related system to fulfill this mandatory ECCS federal requirement is not acceptable.

C. Knowledge of Existing Concerns

The need for an operable ECCS HPCI System is mandatory as evidenced from the grounds for relief in this report. At Nine Mile Point, the Utility, Quality Assurance personnel and the NRC were well aware of this requirement. For what ever reason, this plant was licensed by the NRC and allowed to operate without this mandatory requirement installed. Attempts by these same parties to substitute non-quality related feedwater equipment to fulfill this mandatory safeguard function supports the fact the need for requirement was understood. Even if non-quality related equipment was acceptable to support ECCS functions (and its not), there is no onsite electric power system that will support the safety function of a feedwater/HPCI system. This electric system is another mandatory minimum requirement (Attachment 2-Criterion 35). To prove the collaboration between all parties mentioned, the licensee attempts to take credit for onsite power availability from the Benton Dam, some 100 miles away. Obviously the reviewers are aware of these mandatory requirements but there

resolution to the safety concerns is not acceptable. The possibility of a tornado destroying the switchyard is a known postulated accident that can occur. Without this power availability, the HPCI function cannot possibly be assumed, as stated in the FSAR Chapter VII (Attachment 4).

Every time the feedwater procedures were revised this issue would have to be reviewed. Everytime the FSAR (Chapter VII) was revised, the Technical Specifications revised or containment integrity was questioned this issue had to be reviewed in accordance with administrative requirements set out by the federal guidelines. Everytime the Quality Assurance groups and NRC performed their independent audits and inspections this issue had to be reviewed. Everytime this plant was operated at modes 1 or 2, the responsible Senior Reactor Operator (SRO), who is specifically trained (10CFR50 App E) on these issues would have to question the validity of the current HPCI system. Every time the HPCI surveillance (monthly) was performed to ensure operability, the responsible SRO would have to question the validity of a non quality related feedwater system fulfilling the HPCI system. Taking credit for non-quality related equipment to fulfill the requirements of a ECCS safety function is not acceptable and it would be the SRO's responsibility to question the feedwater ability to perform this HPCI safety function. Of course, that is the another problem to consider, it would be the SRO's job.

Although previously aware of the problem, on Jan 18, 1990, the Utility was served notice of these and other safety concern. If the non-quality related feedwater system was to supposedly fulfill the HPCI safety function, it failed to met the onsite electrical requirements and many of

the main flow path valves had never been included in the Inservice Test Program (10CFR50.55). Some 44 out of 47 valves were currently not identified in the Inservice Test Program(ECCS Surveillance violation). With such knowledge, the Utility, Quality Assurance group and the NRC allowed the plant to start up and continue into full operating (mode 1) condition. No pumps, no valves yet Technical Specification 4.1.8 (Attachment 3) demands if one valve is not demonstrated operable a daily surveillance is required to be performed. This is just another lack of administrative control in which the review groups have failed to audit or review properly.

Unfortunately, this dilemma is not unique to Nine Mile Point. Other plants were also somehow licensed without this mandatory HPCI capability. That is another indicator of the type of review that has taken place at other facilities as well but eventually these plants installed the mandatory system. The most stunning fact of this investigation shows that after literally thousands of technical reviews performed by hundreds of "qualified personnel" working in different shifts, separate departments, sites or regions, have all failed to stop this facility from operating outside the minimum federal guidelines. Every month during full power operation, the HPCI system is verified operable by a "qualified" Senior Reactor Operator and a sworn affidavit submitted each month by the Utility to the NRC attesting that all requirements have been fulfilled. Obviously, the current system of checks and balances cannot stop this plant from operating outside these mandatory federal guidelines, an assumption falsely made by congress.

D. Responsibilities

10CFR50 App. B details the administrative requirements for Test Control, Inspections, Operating Status, Corrective Action, Records and Independent Audits. These requirements are addressed in both the Technical Specifications and FSAR. Site specific administrative procedures detail utility and quality assurance staff position responsibilities. 10CFR50.70 detail the NRC inspections while 10CFR50.72 detail report notification responsibilities for all parties. The NRC have their own administrative procedures which detail staff responsibilities. NUREG-0800 details the USNRC standard review plan for inservice testing of pumps and valves.

All parties mentioned were required to have knowledge of the HPCI requirements at the level of review for which each individual was involved. These reviews require mandatory action. Despite all mentioned reviews this requirement was not met. On Jan 18, 1990 the Niagra Mohawk, Nine Mile Point Nuclear Regulatory Compliance Group were served notice of this and many other known safety concerns. On July 31, 1990 the Niagra Mohawk Quality First Team were served written notice. The NRC was notified and on and the Quality First Team notified petitioner that the NRC exempted the utility from the requirement.

V. STATEMENT OF THE LAW

1. There is a minimum requirement for a High Pressure Core Injection ECCS Safeguard System at the Nine Mile Point Unit One facility. This requirement comes from the federal guidelines, Technical Specifications and FSAR minimum mandates.

2. No High Pressure Core Injection System meeting the safeguard federal guidelines exists at Nine Mile Point, Unit One.

3. If the non-quality related feedwater system was to supposedly fulfill the HPCI safety function, it failed to met the onsite electrical requirements and many of the main flow path valves had never been included in the Inservice Test Program.

4. If the HPCI System is not a safeguard system and is not considered in any Loss of Coolant Accident Analyses as stated in the FSAR Chapter VII, then no assumption can be made that the fuel will remain covered by the moderator and related safety limits set in the current license are null and void. Obviously unreviewed safety questions exist.

5. Congress made an assumption of the current checks and balances that would never allow a plant to operate outside the minimum safety requirements set out in federal guidelines. On this assumption, unlike any other industry, the nuclear industry has been allowed to operate under limited liability. The utility, Quality Assurance Groups, NRC and Chief Executive Officer have received written notice of their failure to comply with the minimum federal guidelines and have administratively failed to comply with this issue.

As discussed above, the Nine Mile Unit One Plant fails to comply with both the minimum federal and NRC's requirements for HPCI ECCS System. This has been acknowledged by the NRC Staff and is demonstrated unequivocally by the evidence in the public record. Moreover, the Staff has performed no valid analysis that meets the Commission's narrow criteria for continuing to operate in the absence of compliance. Compliance with both Federal and NRC safety regulations is a prerequisite to safe operation of a nuclear power plant. In fact, as the NRC's Appeal Board has observed, regulatory

compliance is the "sine qua non of adequate protection to the public health and safety." Maine Yankee Atomic Power Company ALAB-161, 6 AEC 1003, 1009(1973). Compliance may not be avoided

> by arguing that, although an applicable regulation is not me, the public health and safety will still be protected. For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety.

Vermont Yankee Nuclear Power Corp. ALAB-138, 6 AEC 520, 528(1973)(emphasis added). The Commission's essential safety standards must be met, without regard to the cost or inconvenience of achieving compliance. 10CFR50.109 See also Union of Concerned Scientists v NRC, 824 F .2d 108(DC Cir 1987).

VI. REQUEST FOR RELIEF

For the reasons enumerated above, petitioner states that the following relief is required:

A. Immediate Shutdown Pending Demonstration of Regulatory Compliance.

As discussed abo, the Nine Mile Point nuclear plant fails to comply with an array of fundamental requirements for HPCI ECCS mandatory requirements. No exemptions to this requirement can possibly be justified without undue risks to public safety. Consistent with the requirements of the Atomic Energy Act, Federal mandatory requirements and NRC regulations, Petitioner therefore seeks immediate shutdown of the Nine Mile Point unit one reactor pending full compliance with the regulations.

In seeking this relief, Petitioner notes that maintaining ECCS systems necessary to metigate loss of coolant accidents is a regulatory goal that warrants the most immediate and stringent enforcement action. Nine Mile Point's noncompliance with the federal minimum design criteria and the "cover up" activities of all responsible parties which poses a safety risk

of commensurate, if not graver, dimension than the suspicion of ECCS pipe cracking that caused the commission to order 23 plant shutdowns in 1975. See Petition for Emergency and remedial Action, CLI-78-6, 7 NRC 400, 405(1978). Like the ECCS pipe cracking, this plant doesn't even have the pipes, valves or pumps necessary to metigate a known postulated accident that effects known safety limits of the FSAR. This system is necessary for the cooling of the core during an accident and this system (which does not exist) is the only means to prevent a meltdown. Again, unlike normal ECCS systems which have redundant components and can therefore withstand a single failure, this system does not exist and cannot be compensated for by any other system. Simply put, a small break described in the FSAR bases as a postulated accident will in all likelihood meltdown the reactor for lack of cooling. Because the containment is not designed to withstand a meltdown, such an event would probably lead to an uncontained release of radioactivity to the public environment. This utility is not insured for such an accident.

B. **Public Hearing**

The issues raised by the Nine Mile Point's noncompliance with federal requirements raises grave safety questions of tremendous public importance. Petitioner therefore request that before allowing the Nine Mile Point plant to continue operating, the Commission provide for public hearing, with rights of discovery and cross examination, to determine whether Nine Mile Point is in full compliance with all federal minimum requirements revelant to HPCI and public safety.

Secondly, congress be notified that the administrative controls relied upon to grant the nuclear industry the immunity of liability have failed to ensure public safety. After literally thousands of reviews by "qualified

personnel" from different disciplines, departments, sites and regions completed their review, not one came forward and demand this plant operate within the law as laid out by act of congress. Should noncompliance be found, many of these reviews demand mandatory action on the part of the reviewer. The petitioner has notified all responsible parties and after two years Nine Mile Point Unit One continues to operate outside the federal guidelines at a tremendous risk to public safety. A congressional investigation of this matter be initiated immediately.

The petitioner's services were contracted by Niagra Mohawk to review and ensure administrative compliance to Technical Specification prior to Start-Up. A qualified group of ten began a laborious review and when enormous problems began to immerge. This group was disbanded immediately. In Jan 1990, the Niagra Mohawk's Nuclear Regulatory Compliance Staff was given a detailed memo (Attachment 5) giving evidence that 45% of the containment isolation valves had administrative deficiencies. Two weeks later the review group was disbanded prior to completion of their review. Along with HPCI concerns, containment isolation valves as found in the FSAR Table VI-3 had deficiencies with corresponding Technical Specification Tables 3.3.4 & 3.2.7. This plant had operated for twenty years and yet the license failed to even correspond to itself, let alone actual plant conditions. These valves are required by federal guidelines to protect the public yet almost half had deficiencies. Petitioner alleges that when concerns are identified, the concerns are routinely "covered up", dismissed or administratively exempted. A proper review of the Nine Mile Point Unit One Technical Specification 4.0.5 requirements and the comliance of the

test programs will show that the utility simply hired another review group that (for whatever reason) failed to document the deficiencies that truly exist. Nine Mile Point Unit One resumed full power operations even after the safety concerns were identified and documented. This type of cover up is not unique to this plant and a congressional investigation of this matter be initiated immediately.

IN SUMMARY

There can be no justification for the operation of nuclear power plants outside the minimum requirements specified by act of congress. These are the minimum requirements deemed necessary by act of congress to grant the immunity of liability currently assumed by the utility. When public safety is jeopardized by known postulated accidents, there can be no justification for the lack of action by the responsible parties in this instance. Simply put, this utility is not insured to operate in this manner.

Respectfully submitted,

Ben L Ridings

Ben L. Ridings
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2. 10CFR50.10, "Requirement of License."

3. 10CFR50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors."

4. 10CFR50.55a, "Codes and Standards."

5. 10CFR50.59, "Changes, tests and experiments."

6. 10CFR50.70, "Inspection, Records, Reports, Notifications."

7. 10CFR50, Appendix A, General Design Criterion 33, "Reactor coolant makeup."

8. 10CFR50, Appendix A, General Design Criterion 35, "Emergency core cooing."

9. 10CFR50, Appendix A, General Design Criterion 36, "Inspection of emergency core cooing system."

10. 10CFR50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling systems."

11. 10CFR50, Appendix B, II. "Quality Assurance Program"

12. 10CFR50, Appendix B, III. "Design Control."

13. 10CFR50, Appendix B, VI. "Document Control."

14. 10CFR50, Appendix B, X. "Inspections."

15. 10CFR50, Appendix B, XI. "Test Control."

16. 10CFR50, Appendix B, XIV. "Inspection, Test and Operating Status."

17. 10CFR50, Appendix B, XVI. "Corrective Action."

18. 10CFR50, Appendix B, XVII. "Quality Assurance Records."

19. 10CFR50, Appendix E, F. "Training."

20. Federal Register, Public Docket: 50-220, Niagra Mohawk, Unit One, Nine Mile Point Thermal Nuclear Reactor.

UNITED STATED OF AMERICA
BEFORE THE NUCLEAR REGULATORY COMMISSION

AFFIDAVIT OF BEN L. RIDINGS

I, Ben L. Ridings do make oath and say:

1. My name is Ben L. Ridings. I am a technical consultant for commercial nuclear power plants. Over a span of some fifteen years, while working at some twenty four sites, I have specialized in reviewing of licensing agreement (FSAR, Technical Specifications, Federal Codes and Regulations, ASME Codes, etc.), establishing administrative controls to meet these requirements and test programs to ensure compliance at all times. My test programs and administrative controls established while under contract to various utilities are still in use today at many facilities.

2. I have reviewed all of the relevant publicly available correspondence between the Nuclear Regulatory Commission and Niagra Mohawk during the relvant time span. I am familar with NRC regulations and regulatory guidance governing High Pressure Core Injection.

3. The factual statement made in the attached Petition for Emergency Action and Request for public Hearing are true and correct to the best of my knowlege and belief.

Ben L Ridings
Ben L. Ridings

Subscribed and sworn to before me this 28th day of Oct, 1992.

Mary Duke

My commision expires: 5-21-95

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core

Attachment - 2

cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Attachment-2

Attachment E-3

LIMITING CONDITION FOR OPERATION

3.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the operational status of the high pressure coolant injection system.

Objective:

To assure the capability of the high pressure coolant injection system to cool reactor fuel in the event of a loss-of-coolant accident.

Specification: 1,2H**

a. During the power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature greater than saturation temperature, the high pressure coolant injection system shall be operable except as specified in Specification "b" below.

b. If a redundant component of the high pressure coolant injection system becomes inoperable the high pressure coolant injection shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed.

SURVEILLANCE REQUIREMENT

4.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the high pressure coolant injection system.

Objective:

To verify the operability of the high pressure coolant injection system.

Specification:

The high pressure coolant injection surveillance shall be performed as indicated below:

a. At least ~~once per operating cycle~~ -

Automatic start-up of the high pressure coolant injection system shall be demonstrated.

b. At least once per quarter -

Pump operability shall be determined.

Attachment - 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specification "a" and "b" are not met, a normal orderly shutdown shall be initiated within one hour and reactor coolant pressure and temperature shall be reduced to less than 110 psig and saturation temperature within 24 hours.	c. <u>Surveillance with Inoperable Component</u> When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

e High Pressure Coolant Injection System (HPCI) is provided to ensure adequate core cooling in the unlikely event of all reactor coolant line break. The HPCI System is required for line breaks which exceed the capability of the ntrol Rod Drive pumps and which are not large enough to allow fast enough depressurization for core spray to be fective.

e set of high pressure coolant injection pumps consists of a condensate pump, a feedwater booster pump and a motor iven feedwater pump. One set of pumps is capable of delivering 3,800 gpm to the reactor vessel at reactor essure. The performance capability of HPCI alone and in conjunction with other systems to provide adequate core oling for a spectrum of line breaks is discussed in the Fifth Supplement of the FSAR.

ı determining the operability of the HPCI System, the required performance capability of various components shall be onsidered.

Attachment-3

. The HPCI System shall be capable of meeting its pump head versus flow curve.

. The motor driven feedwater pump shall be capable of automatic initiation upon receipt of either an automatic turbine trip signal or reactor low-water-level signal.

. The Condenser hotwell level shall not be less than 57 inches (75,000 gallons).

. The Condensate storage tanks inventory shall not be less than 105,000 gallons.

. The motor-driven feedwater pump will automatically trip if reactor high water level is sustained for ten seconds and the associated pump downstream flow control valve and low flow control valve are not closed.

uring reactor start-up, operation and shutdown, the condensate and feedwater booster pumps are in operation. At eactor pressures up to 450 psig, these pumps are capable of supplying the required 3,800 gpm. Above 450 psig a otor-driven-feedwater pump is necessary to provide the required flow rate.

he capability of the condensate, feedwa·· booster and motor driven feedwater pumps will be demonstrated by their peration as part of the feedwater supply during normal station operation. Stand-by pumps will be placed in service t least quarterly to supply feedwater during station operation. An automatic system initiation test will be erformed at least once per operating cycle. This will involve automatic starting of the motor driven feedwater pumps nd flow to the reactor vessel.

I. HIGH-PRESSURE COOLANT INJECTION

1.0 Design Bases

The high-pressure coolant injection (HPCI) system is an operating mode of the feedwater system available in the event of a small reactor coolant line break which exceeds the capability of the control rod drive pumps (0.003 ft^2). HPCI along with one emergency cooling system has the capability of keeping the swollen reactor coolant level above the top of active fuel for small reactor coolant boundary breaks up to 0.07 ft^2 for at least 1000 seconds. The HPCI system with one of the two emergency cooling systems and two core spray systems, will provide core cooling for the complete spectrum of break sizes up to the maximum design basis recirculation discharge line break (5.446 ft^2). Its primary purpose is to: Rev. 7

a. provide adequate cooling of the reactor core under abnormal and accident conditions.

b. remove the heat from radioactive decay and residual heat from the reactor core at such a rate that fuel clad melting would be prevented.

c. provide for continuity of core cooling over the complete range of postulated break sizes in the primary system process barrier.

HPCI is not an engineered safeguards system and is not considered in any Loss of Coolant Accident Analyses. It is discussed in this section because of its capability to provide makeup water at reactor operating pressure. Rev. 7

2.0 System Design

The HPCI system utilizes the two condensate storage tanks, the main condenser hotwell, two condensate pumps, condensate demineralizers, two feedwater booster pumps, feedwater heaters, two motor-driven feedwater pumps, an integrated control system and all associated piping and valves. The system is capable of delivering 7600 gpm into the reactor vessel at reactor pressure when using two trains of feedwater pumps. The condensate and feedwater booster pumps are capable of supplying the required 3,800 gpm at approximately reactor pressures up to 270 psig. Above 270 psig a motor-driven feedwater pump is necessary to provide the required flow rate. Rev. 7 Rev. 7

Attachment - 4

The feedwater system pumps have recirculation lines with air operated flow control valves to prevent the pumps from operating against a closed system. In the event of loss of air pressure, these valves open recycling part of the HPCI flow to the hotwell. HPCI flow would be reduced to approximately 3,000 gpm at a reactor pressure of 1,150 psig and 3,800 gpm at a reactor pressure of 940 psig. Rev. 7

Condensate inventory is maintained at an available minimum volume of 180,000 gallons.

3.0 Design Evaluation

During a loss-of-coolant accident within the drywell, high drywell pressure due to a line break will cause a reactor scram. This automatic scram will cause a turbine trip after a five-second delay. In order to prevent cladding temperatures from exceeding their maximum limit for the entire spectrum of breaks, the 3800 gpm (from one train of HPCI/feedwater pumps) would have to be available immediately. Feedwater flow would be available for considerable time from the shaft-driven feedwater pump. The shaft-driven feedwater pump would coast down while the electric motor-driven condensate pumps and feedwater booster pumps would continue to operate. The coast down time to reach 3,800 gpm delivery to the core is approximately 3.2 minutes (Figure VII-17), since both the condensate and feedwater booster pumps will continue to operate on off-site power. Rev. 7

The turbine trip will signal the motor-driven feedwater pump to start. The signal will be simultaneous with the start of the shaft pump coast down. The motor-driven feedwater pump will be up to speed and capable of supplying 3,800 gpm in about ten seconds. As a backup, low reactor water level will also signal the motor-driven pump to start. The initiation signal transfers control from the normal feedwater to the HPCI instrumentation and controller which has been continuously tracking the normal feedwater control signal. Thus there will be a continuous supply of feedwater to the reactor.

The HPCI single element control system will attempt to maintain reactor vessel water level at 65 inches or 72 inches (depending upon which pump, 11 or 12 respectively, is in service) with a maximum feedwater flow limit of 3800 gpm.

Attachment-4

A sustained high reactor water level reactor protection system signal coincident with an open feedwater flow control valve will selectively trip the associated feedwater pump. The clutch of the shaft-driven pump will also be disengaged immediately upon high reactor water level.

Should the reactor water level reach the low level scram setpoint the motor driven pump that tripped on high reactor water level will restart. Necessary feedwater pump recirculation is provided to allow for continued pump operation with the flow control valve closed.

As feedwater is pumped out of the condenser hotwell, through the selected equipment of the condensate and feedwater systems and into the reactor, the condenser hotwell level will fall. Since condensed steam from the turbine no longer replenishes the condenser hotwell, condensate will be transferred from the condensate storage tanks to the hotwell for makeup.

The feedwater system pumps operate on 4160 v. When the plant is in operation, the power is supplied from the main generator through the station service transformer when the generator is on-line and connected to the grid. When the main generator is off-line, the feedwater pumps are supplied with normal off-site power from the 115 KV system through the reserve transformers. If a HPCI initiation signal should occur, all HPCI/feedwater system pumps would start immediately with two feedwater pump trains available for HPCI injection using the single element feedwater control system for reactor vessel level control. If a major power disturbance were to occur that resulted in loss of the 115 KV power supply to the Nine Mile Point 115 KV bus, power would be restored from a generator located at the Bennetts Bridge Hydro Station. This generator would have the capacity of supplying approximately 6,000 KVA which is sufficient to operate one train of HPCI/feedwater system pumps. If HPCI initiation were to occur, the preferred feedwater train pumps (feedwater pump 12, feedwater booster pump 13, condensate pump 13) would start. The non-preferred train pumps would be locked out on loss of off-site power and not start until the operator manually reset the lock out. If a preferred train pump had been locked out prior to the loss of off-site power, it would remain locked out and the non-preferred train backup pump would automatically start on HPCI initiation. If both the preferred and backup pumps are running, the preferred pump would remain in service and the backup pump will trip. The

Rev. 7

Attachment -4

use of a Bennetts Bridge hydro generator, while not equivalent to an on-site emergency power source, provides a highly reliable alternate off-site power supply for the HPCI function of the feedwater system. Rev. 7

4.0 Tests and Inspections

Tests and inspections of the various components are described in Section XI - Steam to Power Conversion.

Attachment -4

COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	INDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTE
05-03	Y	CLOSE	5	N	—	—	*	9
05-02	Y	CLOSE	5	N	—	—	*	9
39-12	Y	CLOSE	5	N	—	—	Y	9
39-11	Y	CLOSE	5	N	—	—	Y	9
39-13	Y	CLOSE	5	N	—	—	Y	9
39-14	Y	CLOSE	5	N	—	—	Y	9
40-12	Y	OPEN	—	N	—	—	Y	1
40-01	Y	OPEN	—	N	—	—	Y	1
40-09	Y	OPEN	—	N	—	—	Y	1
40-11	Y	OPEN	—	N	—	—	Y	1
40-10	Y	OPEN	—	N	—	—	Y	1
40-02	Y	OPEN	—	N	—	—	Y	1
58.1-01	N	—	—	Y	—	60	Y	10
80-118	N	—	—	N	—	—	—	2
30-31	N	—	—	N	—	—	HPCI	13
30-32	N	—	—	N	—	—	HPCI	13
201.1-14	N	—	—	N	—	—	Y	11
201.1-16	N	—	—	N	—	—	Y	11
201.1-09	N	—	—	N	—	—	Y	11
201.1-11	N	—	—	N	—	—	Y	11
63-04	Y	CLOSE	30	N	—	—	Y	12
63-05	Y	CLOSE	30	N	—	—	Y	12
80-15	Y	—	60	Y	OPEN	60	Y	3/1
80-16	Y	—	60	Y	OPEN	60	Y	3/1
80-35	Y	—	60	Y	OPEN	60	Y	3/1
80-36	Y	—	60	Y	OPEN	60	Y	3/1

* - see note 9

Attachment - 5

COMPONENT I.D.	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
201.2-109	Y	CLOSE	60	N	—	—	Y	11
201.2-112	Y	CLOSE	60	N	—	—	Y	11
44.2-15	Y	CLOSE	18	Y	CLOSE	10	Y	4
44.2-16	Y	CLOSE	18	Y	CLOSE	10	Y	4
40-30	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-31	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-32	Y	CLOSE	30	Y	CLOSE	30	Y	5
40-33	Y	CLOSE	30	Y	CLOSE	50	Y	5
122-03	Y	CLOSE	30	N	—	—	Y	12
110-127	Y	CLOSE	20	N	—	—	Y	9
110-128	Y	CLOSE	20	N	—	—	Y	9
202-07	Y	CLOSE	60	N	—	—	Y	12
202-08	Y	CLOSE	60	N	—	—	Y	12
02-35	Y	CLOSE	60	N	—	—	Y	12
02-36	Y	CLOSE	60	N	—	—	Y	12
30-114	Y	CLOSE	90/70	N	—	—	Y	6
30-115	Y	CLOSE	90/70	N	—	—	Y	6
201.7-08	N	—	—	—	—	—	Y	7
201.7-09	N	—	—	—	—	—	Y	7
201.2-25	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-26	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-27	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-28	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-29	Y	CLOSE	60	Y	CLOSE	60	Y	8
0.2-30	Y	CLOSE	60	Y	CLOSE	60	Y	8
01.2-23	Y	CLOSE	60	Y	CLOSE	60	Y	8
201.2-24	Y	CLOSE	60	Y	CLOSE	60	Y	8

COMPONENT I #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
201.7-01	Y	CLOSE	60	N	—	—	Y	11
01.7-02	Y	CLOSE	60	N	—	—	Y	11
01.7-03	Y	CLOSE	60	N	—	—	Y	11
01.7-04	Y	CLOSE	60	N	—	—	Y	11
01.7-10	Y	CLOSE	60	N	—	—	Y	11
01.7-11	Y	CLOSE	60	N	—	—	Y	11
01.2-110	Y	CLOSE	60	N	—	—	Y	11
01.2-111	Y	CLOSE	60	N	—	—	Y	11
34-01	Y	—	30	Y	—	30	—	10
29-51	N	—	—	N	—	—	HPCI	13
29-51	N	—	—	N	—	—	HPCI	13
1P-1	Y	CLOSE	60	N	—	—	**	14
1P-2	Y	CLOSE	60	N	—	—	**	14
1P-3	Y	CLOSE	60	N	—	—	**	14
1P-4	Y	CLOSE	60	N	—	—	**	14
0-94	Y	—	30	Y	—	30	N	15/16
0-92	Y	—	30	Y	—	30	N	16
70-93	Y	—	—	Y	—	—	N	16
70-94	Y	—	—	Y	—	—	N	16
80-17	Y	—	—	Y	—	—	N	17
80-18	Y	—	—	Y	—	—	N	17
80-37	Y	—	—	Y	—	—	N	17
80-38	Y	—	—	Y	—	—	N	17
80-65	Y	—	—	Y	—	—	N	17
80-66	Y	—	—	Y	—	—	N	17
80-67	Y	—	—	Y	—	—	N	17
80-68	Y	—	—	Y	—	—	N	17

Attachment-5

(4)

COMPONENT ID #	IDENTIFIED IN FSAR	FSAR SIGNAL	FSAR ST	IDENTIFIED IN TS	TS SIGNAL	TS ST	P&ID PRINTS ID RPS LOGIC	NOTES
80-01	Y	–	70	Y	–	70	N	17
80-02	Y	–	70	Y	–	70	N	17
80-21	Y	–	70	Y	–	70	N	17
80-22	Y	–	70	Y	–	70	N	17
81-01	Y	–	70	Y	–	70	N	18
81-02	Y	–	70	Y	–	70	N	18
81-21	Y	–	70	Y	–	70	N	18
81-22	Y	–	70	Y	–	70	N	18

Attachment-5

NOTES:

1. FSAR section VII requires these valves to go open within 20 sec Hi Drywell or Lo Rx Level RPS signal and this items fails to appear in either TS Table 3.3.4 or FSAR Table VI-3a. Also, these valves are 10CFR50 Appendix A Criterion 55 valves and are not being tested accordingly.

2. Containment Spray Test Line currently does not recieve RPS signal to go closed. The effectiveness of one containment spray pump is lost until operator response manually closes valve should the accident occur during testing of containment spray pumps. Also, this is a criterion 56 valve and is not being tested accordingly and should appear in TS 3.2.7 and FSAR Table VI-3b.

3. FSAR Table VI-3b shows these valves recieve NO RPS signal. TS Table 3.3.4 shows these valves recieve signal to open. P&ID C18012C shows RPS logic to these valves. Also, these are Criterion 56 valves values and are not being tested accordingly.

4. FSAR Table VI-3a shows a close stroke time of 18 seconds while TS Table 3.2.7 shows 10 second closure. Eventhough this is more conservative, the discrepancy came about as an error because components are not individually listed in tables.

Notes cont.

(6)

(5) FSAR Table 3a shows RPS logic to close with core spray actuation while TS Table 3.2.7 does not.

(6) FSAR Table 3b shows these values with a 70 second and 90 second stroke time. These values should also appear on TS Table 3.3.4.

(7) P&ID 18014C sht 2 shows these valves recieve an RPS signal however FSAR Table VI-3b and TS Table 3.3.4 fail to include these penetrations and stroke times.

(8) These value are Criterion 56 valves which appear in FSAR Table VI-3b. These valves may or may not (see note 12) appear in TS Table 3.3.4. TS as written, it is impossible to distinguish however these values are identified in surveillance test (N1-ST-Q5) as TS acceptance criteria.

(9) FSAR Table VI-3a shows RPS logic to close however TS Table 3.2.7 does not identify these values. Also, values (*) appear on P&ID C18006C with no RPS logic while they are identified with RPS logic on P&ID C18017C.

(10) These values are deactivated and the TS and appropriate FSAR sections should be revised to reflect this change.

NOTES
CONT.

(11) These valves are identified on P&ID C18014C sht 1 as recieving RPS logic yet do not appear in FSAR Table VI-36 or TS Table 3.3.4.

(12) FSAR Table VI-36 show RPS logic to close however TS Table 3.3.4 does not identify these valves. Effects Surveillance Program and procedure revision,

(13) P&ID C18005C Sht 1 show HPCI logic to close yet are not identied in TS or FSAR. Also Not identified in IST Program.

(14) FSAR Table VI-36 show RPS logic to close however TS Table 3.3.4 does not identify these valves. Also, tested IAW N1-ST-Q5, current procedure 5 sec TS acceptance criteria that does not exist. Also these valves do not appear on drawings C18014C as identified in IST Plan.

(15) Currently tested IAW N1-ST-Q7 with IST acceptance criteria of 60 sec. No FSAR or TS stroke times identified.

(16) FSAR VI-3c identifies these valves as criterion 57 valves. TS Table 3.3.4 identifies these valves as both criterion 56 and 57 valves. This is physically impossible. Secondly, these valves are not tested to either criterion.

NOTES
CONT

(8)

18 FSAR Table VI-3b and TS Table 3.3.4 identify these values as criterion 56 valves however are not being tested accordingly.

Attachment-5

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Original Signed By:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
10 CFR 2.206 Petition dated
October 17, 1992 from
Ben L. Ridings

cc w/enclosure:
See next page

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