UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.   TO PROVISIONAL OPERATING LICENSE NO. DPR-21

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-245

1.0 INTRODUCTION

By letter dated April 9, 1984 (Ref. 1) Northeast Nuclear Energy Company (NNECo) (the licensee) proposed Technical Specification changes to allow plant operation at rated conditions following completion of actions planned for the outage and refueling (Reload 9) which began on April 14, 1984. The core (Reload 9) for fuel cycle 10 operation includes 200 new (unirradiated) 8x8 prepressurized retrofit fuel assemblies fabricated by General Electric Company described in an attachment to NNECo letter dated March 27, 1984. The new fuel assemblies contain 3.0 w/o 235U with 7 Gadolinia rods at 5 percent. The increased number of new replacement fuel assemblies and the heavier 235U and Gd loadings are necessary to achieve the extended operating cycles desired by the licensee. The core for fuel cycle 10 has been calculated to produce 851 MWD/ST (in contrast to 8022 for cycle 9). This burnup is equivalent to 18 months operation at rated power. A NNECo letter dated May 15, 1984 (Ref. 12) relating to Reload 9/Cycle 10 operation concerned an additional change to those listed in Reference 3. The change correctly identifies new MAPLHGR Tables and has no significance relating to the staff SER.

It is noted that the Segmented Test Rod Fuel assembly described and evaluated in previous reload evaluations for Millstone-1 has been removed and is no longer listed in the core fuel inventory (see Table 1).

2.0 EVALUATION

The items identified below (Ref. 1 Attachment 4) include changes other than those described in General Electric Company proprietary topical report NEDE-24011-P and NNECo letter dated 3/27/84 dealing with GE fuel which has been reviewed and approved by the NRC. Each item is presented in the same order as reference 4.

2.1 FUEL DESIGN EVALUATION

The reload application involves three fuel-design related issues: (1) the replacement of 200 spent fuel assemblies with fresh BP8x8R fuel assemblies,

(2) the analysis of safety considerations involved in the determination of Cycle 10 operating limits, and (3) the incorporation of new and extended maximum average planar linear heat generation rate (MAPLHGR) limits.

2.1.1 Cycle 10 Core Inventory

The Millstone-1 Cycle 10 core contains 580 fuel assemblies of which 200 were replaced during the Reload-9 outage.

The Cycle 10 core composition is summarized in Table 1. The 200 replacement assemblies are new (BP8DRB300) prepressurized 8x8 retrofit fuel assemblies with an average enrichment of 3.00 w/o in U-235. Since (1) the prepressurized 8x8 retrofit with barrier has been previously approved (Ref. 5), (2) the average enrichment of the fresh fuel is less than that of the approved maximum enrichment stated in reference 5, and (3) the MAPLHGR limits are established for the fresh fuel to avoid violation of the peak cladding temperature limit (2200°F) during a loss-of-coolant accident, the staff has concluded that the fuel assemblies are acceptable for Millstone-1 Cycle 10 operation. The other types of fuel assemblies in the core as specified in Table 1 have been approved previously for use in Millstone-1. Therefore, the design aspects of those fuel assemblies require no further NRC review.

Table 1

MILLSTONE- CORE INVENTORY

Assembly Designation	Cycle Loaded	Number
8DB274H (Irradiated)	6	20
P8DRB265H (Irradiated)*	8	40
P8DRB282 (Irradiated)*	8	128
P8DRB282 (Irradiated)*	9	72
P8DRB283H (Irradiated)*	9	120
BP8DRB300 (NEW)*	10	200
		580

* All assemblies are drilled.

2.1.2 Cycle 10 Operating Limits

The licensee's analysis of other considerations involved in the determination of Cycle 10 operating limits is presented in the reload safety analyses (Ref. 2, 3). In all fuel-design-related areas except those separately identified, the reload report relies on the generic report, General Electric Standard Application for Reactor Fuel (Ref. 5). Reference 5 has been reviewed and approved by the NRC staff. Additional staff review of those portions of reference 5 concerning the standard fuel design for the Cycle 10 application is not justified.

2.1.3 MAPLHGR Limits and Extensions

The licensee's submittal provided new MAPHLGR limits for the fresh assemblies (BP8x8R). MAPLHGR limits for three fuel assembly types no longer in the Millstone-1 core have been removed. The limits were generated by methods previously approved (Ref. 6). Although the methodology used is generically applicable for MAPLHGR limit determination, the staff was concerned that the effects of enhanced fission gas release at high-burnup (i.e., greater than 20 GWd/MTU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (Refs. 7 and 8) that credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at higher burnups. This proposal was found acceptable (Ref. 9) provided that certain plant-specific conditions were met. The licensee has previously stated (Ref. 11) that the GE proposal is applicable to the Millstone-1 analysis. On the basis of this finding, the staff has concluded that the MAPLHGR Limits proposed for Cycle 10 operation of Millstone-1 are acceptable, i.e. figures 3.11.1a through 3.11.1e. Also, the staff concludes that the licensee has adequately described the Millstone-1 Cycle-10 fuel design and its predicted conformance to the applicable regulations and NRC staff positions.

2.1.4.0 THERMAL AND HYDRAULIC DESIGN EVALUATION

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, that the design provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and that the core is not susceptible to thermal-hydraulic instability.

The review is presented under the following headings: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, and (3) thermal-hydraulic stability. Discussion of the review concerning the thermal-hydraulic design for Cycle 10 operation follows:

2.1.4.1 Safety Limit MCPR

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core do not experience boiling transition during normal and anticipated operational transients. As stated in Reference 5, the NRC approved safety limit MCPR is 1.07. The safety limit MCPR of 1.07 used for the Millstone-1 Cycle 10 operation is therefore acceptable.

2.1.4.2 Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The Δ CPR values given in Section 10 of Reference 2 are plant specific values calculated by the approved methods including ODYN methods. The calculated Δ CPRs are adjusted to reflect the calculational uncertainties by employing the conversion methods described in Reference 10. The Δ CPRs are then added to the safety limit MCPR to obtain operating MCPR limits. Section 12 of Reference 2 presents both the cycle-10 MCPR values of the pressurization and non-pressurization transients. The maximum cycle MCPR values in Section 12 are specified as the operating limit MCPRs and incorporated into the Technical Specifications. The value of operating limit MCPR resulting from the limiting transient, the generator load rejection without bypass transient, is 1.48 for Cycle 10, which is the same value as that for Cycle 9. Since (1) the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients, and (2) the Cycle 10 operating limit MCPRs are the same as that for Cycle 9, which was previously approved, the staff concludes that these limits are acceptable.

Table 3.11.1 in Section 3/4.11-10 of the proposed Technical Specifications has included the operating limit MCPRs for operation of Cycle 10. The staff finds that the proposed operating MCPR limits have been established using approved thermal-hydraulic methods. These operating MCPR limits assure that the fuel clad temperature rise during anticipated operational occurrences is negligible. The staff, therefore, concludes that the Technical Specification changes related to the operating limit MCPRs as indicated in table 3.11.1 of Section 3/4.11-10 of the proposed Technical Specifications are acceptable.

2.1.4.3 Thermal-Hydraulic Stability

The results of thermal-hydraulic analysis (Ref. 2) show that maximum reactor core stability decay ratio is about 0.59 for Cycle 10 as compared to 0.61 for Cycle 9. Based on the evaluation results that (1) the calculated decay ratio for Cycle 10 is less than that for Cycle 9, and (2) the calculated decay ratio compares favorable with the calculated values for several operating reactors which have been previously approved. The staff concludes that the thermal-hydraulic stability results are acceptable for Cycle 10 operation.

The staff has concluded that the licensee has adequately described the Millstone-1 Cycle-10 thermal-hydraulic design and its predicted conformance to the applicable regulations and therefore, the thermal-hydraulic characteristics of the reload core for Millstone-1 Cycle-10 are acceptable.

2.1.5 NUCLEAR DESIGN

The Millstone-1 Cycle-10 reload will consist of 580 fuel bundles as shown in Table 1. The initial core loading had a maximum average enrichment of 1.87 w/o U-235. The reload fuel is similar in physical design to the initial core load fuel, but it has a maximum average enrichment of 3.00 w/o U-235. All fuel bundles consist of 62 fuel rods and 2 water rods. The active fuel length is 150 inches.

The shutdown margin of the new core meets the Technical Specification requirement that the core be at least .25%ΔK subcritical in the most reactive condition when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For Millstone-1 Cycle 10, GE calculated that the K_{eff} under cold conditions with the strongest rod out is equal to .987 resulting in a shutdown margin of 1.3%ΔK.

The standby liquid control system is capable of bringing the reactor from full power to a cold shutdown condition assuming none of the withdrawn control rods is inserted. The 600 ppm boron concentration will bring the reactor subcritical to K_{eff} = .961 at 20°C xenon free conditions (Ref. 2).

The licensee proposes to begin replacing the standard control rod blades with the new Hybrid I blades. These blades are designed to have the same worth and weight as the existing blades. The differences in design are in the cladding and absorber materials and serve to improve blade lifetime. The staff finds this use acceptable in Millstone-1. It should be noted that the Design Features-Section 5 of the Techncial Specifications should be revised prior to the next outage to include these blades.

Based on our review of the licensee submittal (Ref. 1) and the plant specific analysis (Ref. 2), the staff determined that the nuclear characteristics and the expected performance of the reload core for Millstone-1 Cycle 10 are acceptable.

2.2 Core and Containment Cooling Systems

The licensee, NNECo, has requested that the feedwater coolant injection (FWCI) system, the automatic pressure relief (APR) system and the isolation condenser system Technical Specifications be changed such that the operability requirements are for temperatures greater than 330°F in lieu of the present specification of pressures greater than 90 psig. The purpose of this request is to permit additional flexibility in performing reactor vessel hydrostatic testing.

The 330°F requirement is based upon the fact that it is the saturation temperature for 90 psig, therefore, the operability requirements are unchanged for normal operating conditions. The temperature indication will be taken from the recirculation loop temperature indicators during hydrostatic testing. These are not safety-grade and the licensee has stated that loss of the temperature indication will, by procedure, result in discontinuing the testing. During the normal plant operation, the temperature indication will be derived from the safety-grade pressure indication by use of steam tables.

Since Millstone operates at saturation, the requested change should not affect plant safety and is therefore acceptable. Minor editorial changes to the proposed changes have been made to pages 3/4 5-6 and 5-7 as shown in attached approved page changes. NNECo representatives agreed to these changes.

2.3 Turbine Bypass System

The proposed change to Technical Specifications 2.1.2.F and B2.1.2.F would allow the 260 millisecond scram delay in the generator load reject logic to be changed to 280 milliseconds. All transients that depend upon this scram delay were run with the revised value. The licensee has stated that the resultant change in the MCPR was very small (less than .01). According to the licensee, the actual delay time will be set to 260 milliseconds, leaving a 20 millisecond margin to the allowed value of the 280 milliseconds. This leaves room for calculational error and drift between measurements.

The proposed request is acceptable because it is consistent with the transient analyses.

2.4 Reactor Protection System

The purpose of the change is to permit reactor protection system (RPS) and refueling interlocks maintenance, without affecting other plant maintenance and refueling activities.

For the Millstone design, placing the Mode Switch in the SHUTDOWN position initiates a scram and applies a control rod withdraw block on all rods. Placing the Mode Switch in the REFUEL position enables the following rod blocks: (1) the refueling platform is blocked from passing over the core if it is carrying a fuel bundle and any control rods are withdrawn, (2) a rod block is applied whenever the refueling platform is carrying a fuel bundle over the core, and (3) withdrawal of more than one control rod at a time is prohibited. In lieu of maintaining the RPS and refueling interlocks operable at all times when the Mode Switch is in the SHUTDOWN and REFUEL position, as is currently required by technical specification, the licensee has proposed alternate technical specification requirements to ensure all control rods remain fully inserted. These alternate technical specification requirements include measures to isolate mechanically and electrically control rods after they are fully inserted and to verify daily that all control rods remain isolated. The control rods will be electrically disarmed by removing the four amphenol type plug connectors from the drive insert and withdrawal solenoids. This electrically isolates the drive from any signal that could move it. The control rod will be valved out by closing the inlet isolation valves to the underpiston and over-piston areas on the hydraulic drive unit. Subsequent movement of a rod would require leakage through two closed valves in series. In addition, due to the design of the collet fingers that mechanically latch the rod in the fully inserted position, leakage would have to be through multiple valves in a proper sequence (i.e., causing a slight insertion prior to withdrawal) to unlatch the rod.

Based on its review, the staff finds that the proposed technical specification change provides sufficient requirements to ensure that the control rods will remain in their full-in position without reliance on the RPS or refueling interlocks. Therefore, the staff finds the proposed technical specification change acceptable.

2.5 Isolation Condenser

The proposed Technical Specifications change to 3.5.E.2 and the applicable bases section adds the requirement that operation shall remain restricted to 40% of full power until such time the Isolation Condenser is returned to service as long as all active components of the core spray and LPCI subsystems are operable.

The modification to the Millstone, Unit 1, Technical Specifications is acceptable because it is more conservative than the present specification and it brings the specification into conformance with the equipment assumed available in the isolation condenser analyses.

2.6 Recirculation Pumps

The proposed Technical Specification change would allow the operator to attempt to restart the recirculation pumps following a trip of both pumps. NNECo and NRC representatives agreed that more evaluation is required and that the change is not required for fuel cycle 10 (Ref 13). Therefore, the T.S. changes proposed by NNECo for restarting coolant recirculation following a trip of both pumps are not approved.

2.7 Turbine Bypass System

This Technical Specification change is a result of an NRC request of the licensee to provide surveillance requirements for the turbine bypass system. As a result, NNECo has submitted new specifications, 3.14 and 4.14, to require operability of the turbine bypass system while in the run mode and to functionally test the system once per cycle. This modification to the Millstone, Unit 1, Technical Specifications is acceptable. Most BWRs today do not have any Technical Specifications on their turbine bypass system. In addition, by procedures, the bypass valves are moved every week to demonstrate their linkage to the turbine control valves and thus their operability.

2.8 Reactor Protection System Power Supply

The staff concerns regarding the deficiencies in the existing design of RPS power monitoring in BWRs were transmitted to the licensee by NRC generic letter dated September 24, 1980. In response, by letters dated August 18, 1982; November 2, 1982; April 5, 1983; May 2, 1983 and November 17, 1983, NNECo proposed design modifications and changes to the technical specifications. A detailed review and technical evaluation of these proposed modifications and changes to the technical specifications was performed by Lawrence Livermore Laboratory (LLL) under contract to the NRC, and with general supervision by NRC staff. This work is reported in LLL report UCID-19986, "Technical Evaluation of the Monitoring of Electric Power to the Reactor Protection System," dated January 1984. The staff has reviewed this technical evaluation report and concurs in its conclusion that the proposed design modifications and technical specification changes are acceptable.

The following design modifications and technical specification changes were proposed by NNECo for Millstone Unit 1:

1. Installation of two Class 1E detection and isolation assemblies, similar to GE designed protection assemblies, in each of the three sources of power to the RPS (RPS M-G sets A and B and the one alternate source). Each assembly includes a circuit breaker and a monitoring module consisting of an under-voltage, an overvoltage and an underfrequency sensing relay. In conjunction with these relays, there is an auxiliary relay to provide the proposed time delay for an underfrequency, an undervoltage and an overvoltage trip.

2. NNECo also proposed the addition of trip setpoints, limiting condition for operation and surveillance requirements in the technical specification associated with the design modifications cited above.

   The criteria used by LLL in its technical evaluation of the proposed changes include GDC-2, "Design Basis for Protection Against Natural Phenomenon," and GDC-21, "Protection System Reliability and Testability," of Appendix A to 10 CFR 50; IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations; "and NRC memorandum from F. Rosa to J. Stolz, T. Ippolito and G. Lainas dated February 19, 1979.

The staff has reviewed the LLL Technical Evaluation Report and concurs in its findings that (1) the proposed modifications will provide automatic protection to the RPS components from sustained abnormal power supply and (2) the proposed changes to the technical specifications include acceptable Limiting Conditions for Operation (LCO) and periodic testing in accordance with the standard technical specifications for BWRs. Therefore, the staff concludes that NNECo's proposed design modifications and changes to technical specifications are acceptable.

2.9 Safety and Relief Valves

The changes to the Technical Specifications for safety and relief valves (3/4.6.E) and its associated Bases (B3/4.6.E) were proposed so as to clarify safety and relief valve operability requirements, incorporate provisions for vessel hydrostatic testing and to account for a change in the kind of valve position indication installed at the plant.

Sections 3.6.E.1, and 3.6.E.2 have been amended so as to delete the temperature conditions for the operability requirements for the safety and relief valves. The pressure specifications, which is the saturation pressure for that temperature, remains. Since the valves provide pressure protection for the reactor coolant system, the temperature conditions are not necessary. Thus, these changes are acceptable.

As a result of removing the temperature condition from 3.6.E.1, a provision had to be made in the Technical Specifications to permit hydrostatic testing since this will take place with pressures above the pressure limit of 3.6.E.1, although at low temperature. As such, 3.6.E.1 has been additionally amended to exclude this limiting condition for operation for hydrostatic testing. Thus, this change to 3.6.E.1 is acceptable.

Section 3.6.E.5 has been added so as to present limiting conditions for operation during hydrostatic testing. The present Technical Specifications do not specifically address the issue of a required number of operable safety and relief valves during hydrostatic testing. Under the present Technical Specifications, with temperature below 320°F, it is possible to perform hydrostatic testing without any operable safety/relief valves. The proposed specification will require a minimum of two of the valves to be operable.

This new limiting condition for operation is acceptable. Overpressurization of the vessel during plant hydro testing can come from only two CRD pumps and the feedwater pumps. For the latter case, there exists a safety-grade trip of the pumps on water level in the vessel in addition to closure of the control valves. It would therefore take a multiple failure to result in feedwater injection. For the case of the CRD pumps, each valve has a relieving capacity in excess of the combined injection of the CRD pumps. Thus, there is sufficient vessel protection considering single failures.

Additional changes to the Technical Specifications for the safety and relief valves (3.6.E.3, 3.6.E.4, 4.6.E.4 and B3/4.6.E) involve the valve position indication system. The licensee has replaced the acoustic monitors on the valves with qualified pressure sensors to indicate valve position. As a result, the Technical Specification phrase "Acoustic Valve Position Indication" has been changed to "Valve Position Indication". Use of pressure sensors to indicate valve position is acceptable and thus these changes to the Technical Specifications are also acceptable.

The remaining change to the section on safety and relief valves was to extend the functional test period of the Valve Position Indiction System from once per three months to once per operating cycle. The basis for the change was that a replacement of control room equipment would not permit more frequent testing. By telecon, the licensee was informed that the system should be as testable at power as possible. As a result, NNECo retracted this request and by telecopy transmitted a new page of the Technical Specifications on May 17, 1984. The staff found the revised Technical Specification acceptable.

2.10 Degraded Grid Protection (N.B. The vertical lines in the margin indicate changes to the SER enclosed with NRC letter dated June 23, 1982)

Introduction and Summary

The criteria and staff positions pertaining to degraded grid voltage protection were transmitted to (NNECo) by NRC Generic Letter dated June 3, 1977. In response to this, by letters dated August 1, 1977, August 10, 1979, April 29, 1980, July 16, 1980, August 20, 1980, April 21, 1982 and drawings informally provided to the LPM, J. Shea, on May 15, 1982, the licensee proposed certain design modifications to the degraded grid voltage protective system for Millstone Unit 1. A detailed review and technical evaluation of the proposed modifications was performed by Lawrence Livermore Laboratory (LLL) under contract to the NRC, and with general supervision by NRC staff. This work was reported by LLL in "Degraded Grid Protection for Class 1E Power System Millstone Nuclear Power Station, Unit 1." The staff reviewed the technical evaluation report and the licensee submittals and found that additional information was required to complete the review of this issue.

By letters dated January 17, 1984 and April 26, and May 24, 1984 NNECo submitted the design details, and preliminary technical specification changes to cover set-points and tolerances of the degraded grid undervoltage protection relays. The licensee stated in the April 26, 1984 submittal that the final technical specification changes, limiting condition for operation and the associated surveillance testings of the undervoltage protection relays would be provided to the NRC for review and approval after testing of the relays are completed. The staff reviewed the information submitted in the January 17, 1984 letter and found the undervoltage protection system design acceptable. The staff will review the final specification changes and limiting conditions for operation and the surveillance testing of the degraded grid undervoltage relays when this information is available.

Evaluation Criteria

The criteria used by LLL in its technical evalution of the proposed changes include GDC-17 (Electric Power Systems) of Appendix A to 10 CFR 50; IEEE Standard 279-1971 (Criteria for Protection Systems for Nuclear Power Generating Stations); IEEE Standard 308-1974 (Class 1E Power Systems for Nuclear Power Generating Stations); and staff positions defined in NRC Generic Letter to NNECo dated June 3, 1977.

Proposed Changes, Modifications and Discussion

The licensee has proposed to relocate the first level (loss of voltage) and second level (degraded grid voltage) relays, presently connected to the bushing potential devices, to the 4160 volt Class 1E buses. By letter dated April 26, 1984 the licensee stated that due to a delay incurred in obtaining the first and second level relay panels, the necessary time required to install, monitor, and test the relays during the current April 1984, outage was not available. However, the relay panels will be installed during the April-June 1984 refueling outage but the trip logic of the relays will not be connected to the safety buses until necessary testing and monitoring of the relays are accomplished. The relay alarm circuits will be activated during the current outage. The licensee has committed to complete the circuitry connections and implementation of the degraded voltage protection relays no later than the next refueling outage which is presently scheduled for December 1985. This is acceptable to the staff.

The proposed modifications for the second level undervoltage protection system retain the feature that will permit automatic separation of the Class 1E power system from offsite power only if a degraded grid exists coincident with a safety injection signal (LOCA). This approach provides protection to the Class 1E equipment needed to mitigate the consequences of an accident and is acceptable. For a degraded grid condition without a LOCA an alarm will be actuated and operator action will be taken to restore the grid to an acceptable level. In the event operator action is not successful in restoring grid voltage within an acceptable time period, the operator will manually start the onsite diesel/gas turbine generator and separate the Class 1E buses from the offsite power system. The Class 1E loads would then be seqenced onto the onsite emergency generators. This approach deviates from the staff position that requires automatic isolation of the offsite power system from such undervoltage after a time delay. Acceptability of this alternate approach requires demonstration by the licensee that adequate safety systems will be available for safe shutdown of the reactor for these conditions and that appropriate plant operating procedures are developed and available to the operator for the required manual operator action. In response to the above concerns, the licensee in a submittal dated April 21, 1982 provided a list of systems, that would not be exposed to degraded grid voltage, and which will be available to bring the plant to a safe shutdown under non-LOCA conditions. The staff reviewed this listing and concurred with the licensee's approach. By letter dated October 14, 1982, the licensee committed to make the Isolation Condenser make-up fill valve AC independent during the 1982 refueling outage and noted that there is no need to make the condensate transfer pump AC independent, since make-up for the Isolation Condenser can be provided by the diesel-driven fire water pump, thus making the Isolation Condenser System AC independent. By a letter dated February 16, 1983 the licensee stated that the Isolation Condenser make-up fill valve was modified to operate with DC power only. The staff finds that this modification provides a redundant capability to shut the plant down and, maintain it at safe shutdown under non-LOCA conditions. The use of the diesel-driven fire water pump to effect safe shutdown in place of the condensate transfer pump has been evaluated by the staff. The staff concurs with the licensee that the diesel-driven fire water pump can be substituted for the condensate transfer pump when the latter pump is not available. The staff finds that the isolation condenser systems is not dependent on AC power and therefore acceptable.

A redundant means of providing safe shutdown is provided by the Automatic Depressurization System (ADS), which uses N_2 operated valves, with backup bottled N_2 supply, to depressurize the system and allow placing the Low Pressure Coolant Injection (LPCI) and Core Spray (CS) system(s) in operation.

On the basis of the above and that protection devices i.e., circuit breakers, fuses, relays,...etc., are provided to prevent damage to the equipment required for plant safe shutdown and that an alarm is installed to alert the operator to this abnormal condition, the staff finds the licensee's alternate approach for manual operator action under degraded grid condition without an accident acceptable. Acceptability of this approach is subject to institution of adequate procedures covering actions to be taken by the operator during a degraded grid under non-accident conditions.

In addition, we require the following information on the proposed undervoltage protection relaying system:

1. The licensee has committed to bypass load shedding while Class 1E buses are powered from the onsite power system. Provide details on how the load shedding feature will be bypassed and how reinstatement of load shedding and load sequencing on an emergency generator breaker trip will be accomplished.

   The proposed load shedding circuit will be bypassed once the emergency diesel generator or gas turbine is supplying power to the safety loads. The load shedding feature will be reinstated if the diesel generator or gas turbine breakers should trip.

2. Provide design details and a description of operation of the first level (loss of power) and second level (degraded grid) undervoltage relaying systems.

   The proposed undervoltage protection circuitry at Millstone 1 consists of two power train undervoltage trip circuits S1 and S2. The S1 train provides voltage sensing for 4160 V safety buses 14A and 14C and 480 V safety bus 12E. Emergency ac power to the above buses is provided by a gas turbine. The S2 train provides voltage sensing for 4160 V safety bus 14F and 480 V safety bus 12F. Emergency ac power to the above buses is provided by an emergency diesel generator. Both S1 and S2 trip circuit trains provide first level and second level undervoltage protection. The first level (loss of power) undervoltage protection consists of six ITE-27HS relays on each 4.16 kV Class 1E bus (two per phase) with one out-of-two taken thrice coincident logic. These relays are set at 2828 ± 28 volts (approximately 71% of 4 kV rated Class 1E equipment) with associated time delay of 1.5±.015 second The second level (degraded grid) undervoltage protection consists of three ITE-27NS relays (one per phase) on each 4.16 kV Class 1E bus with two-out-of-three coincident logic. These relays are set at 3654± 35 volts (approximately 90% of 4 kV rated Class 1E equipment). The time delays associated with the S1 and S2 trains second level undervoltage protection are 12 ± .12 seconds and 8 ± .08 seconds respectively.

Both emergency power supplies (gas turbine and diesel generator) start simultaneously with an accident signal after three seconds time delay. Under degraded grid condition coincident with an accident, the S2 train will automatically separate from the offsite source (Station Service Transformer (SST) after eight seconds time delay and after an additional two seconds loads will reconnect to its respective onsite emergency diesel generator. Under the above condition the S1 train will automatically disconnect from the offsite source (SST), after 12 seconds time delay and after an additional 33 seconds loads will reconnect to its respective onsite emergency gas turbine.

The licensee's voltage analysis shows that the earlier separation of the S2 train from the offsite power will reduce the loading on the SST, common source of offsite power for both trains. With the grid voltage between 332 kV and 338 kV the studies show that the S1 degraded voltage relays may reset before its associated time delay expires thereby allowing the feedwater system and the ECCS motors which are supplied by the S1 train to continue to perform their safety functions without interruption. However, while the shutdown loads on the S1 train are being supplied by the SST, the gas turbine continues to operate at no load. Further degradation of grid voltage will result in disconnection of offsite power from the S1 buses and transfer of the shutdown loads onto the gas turbine.

The Millstone 1 design implements the plant feedwater system (FWS) as a high pressure safety injection system under a LOCA condition. The FWS is designed to Category 1 Class 3 requirements and all electrical circuitry and pump motor drives are Class 1E.

The longer time delay associated with the S1 train degraded voltage relays does not delay the loading of the safety buses onto the gas turbine as long as this delay does not exceed the time required for the gas turbine to commence loading.

The staff discussed and evaluated longer time delay of the S1 buses relays to maintain the continuous operation of the feedwater system on the offsite power and concurs with the licensee that the additonal four seconds time delay associated with the S1 degraded voltage relays is beneficial and does not adversely affect the plant safe shutdown during accident condition. (NNECo letter dated 5/24/84).

Actuation of anyone of the six loss of power relays or anyone of the three degraded grid relays will activate an alarm to notify the operator of the degraded voltage condition after 10 seconds when voltage on the 4160 V safety buses reduces to less than or equal to 90% of 4 kV.

With degraded voltage under non-accident conditions operator action is necessary to restore the voltage to an acceptable level. In the event operator action fails to restore the voltage to an acceptable level within a reasonable time period, the operator will manually start the onsite ac emergency power supply (diesel or gas turbine generator or both) and separate the Class 1E buses from the offsite power. These buses are then connected to the emergency ac supply and safety loads are subsequently sequenced onto these buses. However, with degraded grid voltage under accident condition, the proposed second level undervoltage protection system is activated to provide automatic starting of diesel/gas turbine generator, separation of Class 1E buses from offsite power source, initiation of load shedding and sequencing of loads onto these buses.

Voltage sensing of both S1 and S2 trains at 480 V bus level provides degraded grid voltage annunciation only. Each 480 V safety bus 12E and 12F is equipped with one relay which activates an alarm to notify the operator of degraded voltage condition when voltage on these buses reduces to 92% or less of 460 V after one minute time delay. The licensee states in the April 26, 1984 letter that the setpoint for the degraded voltage relays is based on the most limiting conditions of 90% terminal voltage on the 4 kV motors. The licensee also states that installation of regulating transformers on the 120 V level during the current refueling outage will provide 120 V ± 1% at 120 V Class 1E buses over a wide range of input voltage (73% to 107%).

3. Technical specifications to cover relay setpoints and tolerances, limiting conditions for operation, and surveillance testing for the undervoltage relaying system.

   By letter dated April 26, 1984, NNECo provided the preliminary information on the technical specification changes to cover setpoints and tolerances of the degraded grid voltage relays. The licensee stated in the above letter that the final setpoints and tolerances will be provided to NRC for review and approval after the testing of the relays are completed.

CONCLUSION

The staff finds that:

1. The proposed degraded grid modifications will protect the Class 1E equipment from sustained degraded voltages of the offsite power system.

2. The proposed load shedding circuit will block load shedding once the emergency generator (diesel or gas turbine) is supplying the safety loads. The load shedding feature will be reinstated if the gas turbine or diesel generator breaker should trip.

3. The preliminary technical specification information supplied by the licensee is conditionally acceptable. However, the staff requires a formal submittal of the changes and additions to technical specifications after the necessary testing of the degraded grid protection relays are completed

4. The licensee's proposal to use operator action instead of automatic disconnection of the Class 1E buses from a degraded offsite power source under non-accident conditions does not meet the staff's position. To justify this alternate approach, the licensee has shown that redundant safety systems,

which are not exposed to degraded voltage, are available to bring the plant to a safe shutdown condition. The staff has reviewed the licensee's proposal and finds that these systems will be available to effect a safe plant shutdown under non-accident conditions. Based on the above, the staff finds the licensee's alternate approach acceptable.

5. The licensee states in the April 26, 1984 letter that due to unanticipated delays incurred in receiving the relay panels for the degraded grid undervoltage protection system, there is insufficient time to monitor and test the relays during the current April 1984 outage. However, the relay panels will be installed during the current outage but only the alarm logic will be connected to the safety buses. The licensee will incorporate the tripping logic and complete the installation after testing of the relays is completed. The licensee has committed to complete the installation of the degraded grid protection system during a scheduled outage but not later than the next refueling outage scheduled for December 1985. The staff finds the existing protection system and the use of the proposed alarm logic sufficient as an interim measure until relay testing and installation is completed.

The staff, therefore, finds the Millstone Nuclear Power Station, Unit 1 degraded grid protection system design acceptable and that revised technical specification changes should be delayed until the undervoltage protective circuitry (i.e. connecting the trip logic of the relays to the safety buses) installation is completed. These changes are not required for Cycle 10 operation.

3.0 ENVIRONMENTAL CONSIDERATION

The staff has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, the staff has further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ACKNOWLEDGEMENT

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Dated: June 14, 1984

REFERENCES

1. W. G. Counsil (NNECo) letter to D. M. Crutchfield (NRC), April 9, 1984.

2. Attachment 1 of Reference 1, "Supplemental Reload Licensing Submittal for Millstone Unit 1, Reload 9 (Cycle 10), "GE Report, Revision 0, January 1984.

3. Attachment 2 of Reference 1, "Loss-of-Coolant Accident Analysis Report for Millstone Unit 1 Nuclear Power Station", July 1980.

4. Attachment 4 of Reference 1, "Proposed Technical Specification Changes for Millstone Unit No. 1, Reload 9."

5. "General Electric Standard Application for Reactor Fuel," GE Report NEDE-24011-P-A-6, April 1983.

6. D. G. Eisenhut (NRC) letter to E. D. Fuller (GE), June 30, 1977.

7. R. E. Engel (GE) letter to T. A. Ippolito (NRC), May 6, 1981.

8. R.E. Engel (GE) letter to T. A. Ippolito (NRC), May 28, 1981.

9. L. S. Rubenstein (NRC) memorandum for T. Novak, "Extension of General Electric Emergency Core Cooling Systems Performance Limits." June 25, 1981.

10. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," GE Report NEDE-24154-P, October 1978.

11. W. G. Counsil (NNECo) letter to D. M. Crutchfield (NRC) dated November 2, 1982.

12. W. G. Counsil (NNECo) letter to D. M. Crutchfield (NRC) dated May 15, 1984.

13. W. G. Counsil (NNECo) letter to D. M. Crutchfield (NRC) dated May 31, 1984.