REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8511190027 DOC.DATE: 85/11/03 NOTARIZED: NO DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
AUTH.NAME AUTHOR AFFILIATION
MANGAN,C.V. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
VASSALLO,D.B. Operating Reactors Branch 2

SUBJECT: Application for amend to LIcense DPR-63,changing Tech Specs to allow operation during remainder of Cycle 8 w/one emergency cooling sys continuously inoperable.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 1+2
TITLE: OR Submittal: General Distribution

NOTES: 05000220
OL:08/22/69

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	NRR ORB2 BC	01	7	7				
INTERNAL:	ACRS	09	6	6	ADM/LFMB		1	0
	ELD/HDS3		1	0	NRR/DE/MTEB		1	1
	NRR/DL DIR		1	1	NRR/DL/ORAB		1	0
	NRR/DL/TSRG		1	1	~~NRR/DSI/METB~~		1	1
	NRR/DSI/RAB		1	1	REG FILE	04	1	1
	RGN1		1	1				
EXTERNAL:	24X		1	1	EG&G BRUSKE,S		1	1
	LPDR	03	1	1	NRC PDR	02	1	1
	NSIC	05	1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 28 ENCL 25

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

November 3, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactor Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Vassallo:

Enclosed are 40 copies of a proposed change to the Technical Specifications set forth in Appendix A to the above mentioned license. Pursuant to 10CFR170.12, a one hundred and fifty dollar ($150.00) application fee is being provided under separate cover.

The proposed Technical Specification contained herein represents revisions to Specifications 3.1.3 and 4.1.3 for the Emergency Cooling System. The proposed Technical Specification changes are set forth in Attachment A. The proposed changes would not authorize any change in the types of effluents or in the authorized power level of the facility. Supporting information and analysis which demonstrate that the proposed changes involve no significant hazards considerations pursuant to 10CFR50.92 are included as Attachment B.

It is requested that the proposed Technical Specification changes be considered an emergency situation as defined in 10CFR50.91. Information contained in Attachment C explains why this emergency situation occurred and why it could not be avoided. Attachment D presents the environmental considerations for the proposed Technical Specification changes.

A001
1/1

Pursuant to 10CFR50.91(b)(1), Niagara Mohawk has provided a copy of this license amendment request and the associated analysis regarding no significant hazard considerations to the appropriate state representative.

Very truly yours,

C. V. Mangan
Senior Vice President

PEF:bd
Attachments
cc: Mr. Jay Dunkleberger
Division of Policy Analysis and Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, NY 12223

Mr. Steve Hudson, Resident Inspector

Director
Office of Inspection and Enforcement
Washington, D.C.

ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Proposed Changes to Technical Specifications

Attached are revised pages 47 and 48.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE DPR-63

DOCKET NO. 50-220

Supporting Information and No Significant Hazards Considerations Analysis

Supporting Information

The proposed technical specification changes described in Attachment A will allow operation during the remainder of Cycle 8 with one emergency cooling system continuously inoperable. This proposed technical specification is similar to that which was in place during Cycle 5. At the conclusion of Cycle 8, modifications will be performed on the inoperable emergency cooling system to place it back in service.

Each of the two emergency cooling systems serve as an alternate heat sink during reactor isolation from the turbine condenser. During normal and rapid shutdown, the turbine condenser is used for cooling. Neither of the emergency cooling systems is used. If the turbine condenser is lost, one emergency cooling system can more than adequately remove heat until the shutdown cooling system is operational.

The two emergency cooling systems can provide a source of additional cooling during certain postulated accidents and transients. However, the applicable accident and transient analyses have been reviewed and it has been concluded that the continuous inoperability of one emergency cooling system will not change the results.

It has been determined that transient results remain applicable with one emergency cooling system continuously inoperable, since credit was taken for only one emergency cooling system in the analysis.

In order to assess the safety aspects of extended operation with one emergency condenser out of service, the transient which has the greatest potential for core uncovery was assumed to occur. This transient is the total loss of feedwater flow, i.e., Feedwater Controller Failure - Zero Demand. The transient results in a low water level scram, initiation of the emergency condenser, reactor isolation and turbine trip. The pressure rise due to reactor isolation will open the relief valves to turn the pressure transient. Calculated coolant loss to the suppression pool during this transient with one emergency condenser out of service is 13,000 pounds. This amount of steam is comparable to that expected to be discharged to the suppression pool during normal relief valve testing over the life of the plant.

The capacity of one emergency condenser system is 19.0×10^7 BTU/hr. This capacity is sufficient to handle the decay heat production at a calculated 725 seconds following the low water level scram. The water level reduces to a minimum of approximately 8 feet above the active fuel and stabilizes at that point.

The relief valves cycle open for 725 seconds after the scram and the energy loss to the suppression pool is 15.3 x 10^6 BTU. This results in a maximum rise in suppression pool bulk temperature to 3F above the initial Technical Specification limit (maximum of 93F).

In summary, the analysis shows that a single emergency condenser is adequate to handle the most severe transient, maintaining the water level well above the top of the fuel with a maximum suppression pool temperature of 96F.

Another safety analysis which may be affected by the continuous inoperability of one emergency cooling system is the emergency core cooling system analysis. The applicable emergency core cooling system analysis is contained in References 1 and 3. This analysis assumed the worst single failure was failure of an emergency cooling system. For conservatism, this analysis assumed a simultaneous worst single failure in the auto-depressurization system. The worst single failure in the auto-depressurization system would delay the initiation time from 120 to 125 seconds as described in Reference 2.

The emergency core cooling system analysis evaluated the effects of pipe breaks of three general types; recirculation line breaks, emergency cooling system steam line breaks and other line breaks.

For recirculation line breaks, the break was conservatively assumed to occur at the junction of the emergency cooling return line disabling that emergency cooling system. Also, the recirculation line break analysis assumed simultaneous failure of the other emergency cooling system and a worst single failure in the auto-depressurization system. The resulting operable systems were two core spray systems plus auto-depressurization (125 seconds delay). Since the original ECCS analysis assumed the loss of both emergency cooling systems, having one system inoperable does not change the results of the ECCS analysis.

For emergency cooling system steam line breaks, the break was assumed to disable one emergency cooling system. Also the analysis assumed a simultaneous failure of the other emergency cooling system and a worst single failure in the auto-depressurization system. Resulting operable systems were two core spray systems plus auto-depressurization (125 seconds delay). Therefore, the emergency cooling system steam line breaks previously analyzed is applicable for the condition where one emergency cooling system is continuously inoperable.

For other line breaks (main steam, feedwater and core spray), the analysis assumed a simultaneous failure of one emergency cooling system and a worst single failure in the auto-depressurization system. The resulting operable systems were two core spray systems and one emergency cooling system plus auto-depressurization (125 seconds delay). With a continuous inoperable emergency cooling system, the allowed operable systems assuming a simultaneous failure of the operable emergency cooling system and a worst single failure in the auto-depressurization system is two core spray systems plus auto-depressurization (125 seconds delay). When the Nine Mile Point Unit 1 analysis was performed, the main steam line break was analyzed for both sets of operability assumptions. The difference in uncovery

time of the core was 10 seconds and the difference in time for rated core spray was 1.5 seconds. This will cause a difference in calculated peak clad temperature of less than 50°F. The effect of a continuously inoperable emergency cooling system on the feedwater and core spray line breaks will be similar and since the peak clad temperatures of these breaks were previously calculated to be less than 1900°F, none of these breaks can be more limiting than the recirculation line breaks.

Therefore, the Nine Mile Point Unit 1 Appendix K analysis submitted previously is conservatively applicable for the continuous inoperability of one emergency cooling system.

The applicable accident and transient analyses have been reviewed and it has been concluded that the continuous inoperability of one emergency cooling system will not change the results.

It is proposed that during Cycle 8 the surveillance for operation with an inoperable emergency cooling system be performed on a weekly rather than a daily basis. Performing the surveillance on a daily basis for the next 4 months could result in unnecessary equipment wear. For Cycle 9 and beyond, additional surveillance would remain on a daily basis since operation with one emergency cooling system would only be allowed for a 7 day period.

Appendix R Analysis

Emergency Cooling loop 11 (which will be inoperable) was originally required to achieve safe shutdown in the event of a fire in Fire Area 5 (Turbine Building elevation 261'-0" and above). For the analysis, a total burnout of the Turbine Building above elevation 261'-0" was assumed to occur. Emergency cooling loop 11 was considered necessary because the analysis conservatively assumed that for a fire in Fire Area 5, emergency cooling loop 12 was rendered inoperable (at DC valve board 12 located on Turbine Building elevation 291'-0") and the diesel generators were rendered inoperable (cables located on Turbine Building elevation 261'-0"). Emergency cooling loop 11 can still be manually returned to service to provide hot shutdown capability, if required.

Additionally, a more realistic review involving a fire above elevation 261'-0" has been performed. This review has shown that emergency cooling loop 11 would probably not be required because it is highly unlikely that a single fire could cause the loss of emergency cooling loop 12 and both diesel generators. A fire at DC valve board 12 (which could cause the loss of emergency condenser loop 12) should be confined to that elevation and not impact the diesel generators because; 1) low fire loading, 2) detection throughout the turbine building, 3) full-time fire brigade, and 4) the diesel generator cables are located a horizontal distance of approximately 100' away and two elevations below DC valve board 12. Therefore, if emergency cooling loop 12 were lost and with emergency cooling loop 11 administratively inoperable, safe shutdown could be achieved using the diesel generators and associated systems. Conversely, a fire on elevation 261'-0" (which could cause the loss of both diesel generators) should be confined to that elevation for the same reasons stated above. Therefore, safe shutdown could be achieved using emergency condenser loop 12.

Based on the above, operation with emergency cooling loop 11 inoperable will not affect the Appendix R analysis.

In addition, this temporary Technical Specification was allowed for an 18 month cycle beginning in 1977. This request is for approximately 4 months.

No Significant Hazards Considerations Analysis

The proposed amendment in accordance with the operation of Nine Mile Point Unit 1 will not involve a significant increase in the probability or consequences of an accident previously evaluated:

The proposed change to allow operation for the remainder of Cycle 8 with one emergency cooling system loop inoperable would not increase the probability of any accident previously evaluated. Since the DC motor operated isolation valve in emergency cooling system loop no. 11 may not isolate under worst case conditions, an isolation valve will be closed and the system declared inoperable. This will ensure system isolation in the event of a pipe break in the emergency condenser steam line. It will not increase the probability of a pipe break.

The analysis summarized in the Supporting Information section above shows that the consequences of an accident previously evaluated (i.e., pipe breaks) would not be increased.

The proposed amendment, in accordance with the operation of Nine Mile Point Unit 1, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment is to allow continued operation with one emergency cooling system isolated from the reactor vessel pressure boundary for the remainder of Cycle 8. There is no other physical change in the plant other than to close an isolation valve which is normally opened. Therefore, there is no new or different kind of accident which could be created by this proposed amendment.

The proposed amendment, in accordance with the operation of Nine Mile Point Unit 1 will not involve a significant reduction in a margin of safety.

Although the Technical Specifications currently require two emergency cooling systems to be operable, the analysis summarized in the above Supporting Information section shows that continued operation for the remainder of Cycle 8 does not represent a significant reduction in a margin of safety.

Based on the above analysis, the proposed amendment involves no significant hazards consideration.

Reference 1: "Application for Amendment to Technical Specifications," dated October 31, 1975, and supplemented November 3, 1975.

Reference 2: Letter from James Bartlett (Niagara Mohawk) to George Lear (Nuclear Regulatory Commission), dated October 8, 1975.

Reference 3: Loss of Coolant Accident Analysis Report for Nine Mile Point Unit One Nuclear Power Station, NEDO 24348, August 1981.

ATTACHMENT C

NIAGARA MOHAWK POWER CORPORATION

LICENSE DPR-63

DOCKET NO. 50-220

Explanation of Why Emergency Situation Occurred and Why it Could Not Be Avoided

The emergency cooling system consists of two redundant systems or loops. Each loop has an AC and DC motor operated isolation valve in the steam supply line. These isolation valves and motor operators are being replaced during the Spring 1986 refueling outage. In conducting the preliminary engineering for this modification, it was found that the cables routed to the existing DC motor operators for isolation valves 39-07 and 39-08 are of too small a gauge for the application. This results in a voltage drop at the DC motor operator. This condition was reported to Niagara Mohawk Power Corporation Licensing Group for evaluation under 10CFR Part 21. The information initially provided indicated that if the valves did close it would do so "slowly." The preliminary evaluation indicated the condition was not reportable under Part 21 because a similar condition had been previously evaluated (i.e. emergency cooling system line rupture without isolation) and found to be within the design bases of the plant.

However, when the preliminary Part 21 evaluation was reported to management for approval, it was determined that this condition could result in a failure to meet Technical Specification requirements for valve closure in a maximum time of 38 seconds. Preliminary notification was made to the Nuclear Regulatory Commission Operations Center at this time. Subsequently, the valve motor operator vendor performed calculations which indicated that under worst case conditions with a differential pressure of 1250 psig across the valve, enough current could not be drawn to fully seat the valves (i.e. valves would not close beyond approximately 80% of full closure). At this time, valves 39-07 and 39-08 were declared inoperable. Discussions were then held with the Nuclear Regulatory Commission staff informing them of the status of valves 39-07 and 39-08.

There is currently cable on site of the correct gauge (with a low enough resistance/length) for DC motor operated valve 39-08 to assure it will close in the required time frame. However, there is no cable of the required gauge for DC motor operated valve 39-07 available at the site. Valve 39-07 requires a cable with a lower resistance/length as the cable run is approximately twice as long (1100 versus 500 ft.). Therefore, it is proposed to recable the controls to DC motor operated valve 39-08 and return to power with one emergency condenser cooling system operable.

Technical Specification 3.2.7.b indicates that "In the event any isolation valve becomes inoperable the system shall be considered operable provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition. However, if Technical Specification 3.2.7.b is complied with, this would make the emergency cooling system with DC motor operated valve 39-07 inoperable. Therefore, a Limiting Condition for Operation would be entered under Technical Specification 3.1.3.b which requires an inoperable emergency cooling system to be returned to an operable condition within 7 days or Specification 3.1.3.e requires a normal orderly shutdown to be initiated within the reactor to be in cold shutdown within ten hours.

Therefore, the proposed technical specification change is requested for Nine Mile Point Unit 1 to meet Specification 3.1.3. This condition could not be avoided as it was only discovered to be a potential Part 21 on September 30, 1985 and the evaluation was just completed on November 1, 1985. Also, there is not cable of the correct resistance to correct both emergency condenser DC motor operated valves.

This condition will be corrected during the Spring 1986 refueling outage. At that time, the valves and motor operators will be replaced and the existing cables or new cables will be provided which are compatible with the new motor operators.

ATTACHMENT D

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. Niagara Mohawk has determined that this amendment involves no significant hazards consideration. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), Niagara Mohawk has determined that no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.