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NORTHERN STATES POWER COMPANY
Minneapolis, Minnesota 55401

August 7, 1979

Mr.James G. Keppler
Director - Region III
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

The following information relative to the Monticello Nuclear Generating Plant is submitted in response to Items 1-10 of IE Bulletin 79-08 and the subsequent request for additional information from Thomas A. Ippolito dated July 20, 1979.

1. All licensed operators and plant management and supervisors with operational responsibilities have participated in a review of the Three Mile Island Accident. The review (1) included circumstances described in Enclosure 1 of IE Bulletin 79-05 and the chronology included in Enclosure 1 to IE Bulletin 79-05A, and (2) was directed toward understanding the subjects discussed in IE Bulletin 79-08, Item 1.a. The review sessions were conducted on regular scheduled retraining days and were completed May 31, 1979. Attendance has been documented.

   It must be recognized that several valid reasons exist for overriding automatic actions (e.g., see response to Item 5 below). Therefore, Administrative Directives have been issued to instruct operational personnel that automatic action of engineered safety features and isolation signals shall not be manually overrun unless:

   (a) Continued operation of the engineered safety features or isolation signals will result in unsafe plant conditions, or

   (b) It is known or positively determined that the automatic action was initiated by a spurious or erroneous signal and it is verified that operation of the engineered safety feature or isolation is not required, or

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(c) Approved procedures specifically allow manual override under specific conditions, and those conditions are verified to be satisfied.

Administrative Directives have also been issued to instruct operational personnel that all available information should be considered in decisions to manually initiate, terminate or control operation of safety systems. When one or more confirmatory indications are available, operational decisions on abnormal conditions shall not be based solely on a single plant parameter indication.

2. Containment isolation design and procedures have been reviewed. All lines penetrating containment that do not degrade needed safety features or cooling capability are automatically isolated upon reactor vessel low water level or high drywell pressure prior to or simultaneous with initiation of emergency core cooling systems.

3. The auxiliary heat removal systems provided to remove decay heat from the reactor core and containment following loss of the feedwater system are:

   High Pressure Coolant Injection (HPCI) System
   Reactor Core Isolation Cooling (RCIC) System
   Safety Relief Valves (SRV) and Auto Pressure Relief System (APRS)
   Low Pressure Core Spray System (LPCS)
   Residual Heat Removal (RHR) System

   The operation of systems needed to achieve initial core cooling, containment cooling and extended core cooling for long term plant shutdown is described below --

   (a) <u>Initial Core Cooling</u>

   Following a loss of feedwater and reactor scram, a low reactor water level signal will automatically initiate main steam line isolation valve closure. The safety relief valves (SRV) will automatically actuate to maintain reactor pressure. At the same time, the low level signal will automatically initiate the HPCI and RCIC Systems. These systems will continue to inject water into the vessel until a high water level signal automatically trips the systems. Following a high reactor water level trip, the HPCI system will automatically reinitiate when reactor water level decreases to the low water level setpoint. The RCIC System must be manually reset before it will automatically reinitiate after a high water level trip. The HPCI and RCIC Systems have redundant supplies of water. Normally they take suction from the condensate storage tanks (CST). The HPCI System suction will automatically transfer from the CST to the suppression pool if the CST water is depleted or the suppression pool water level increases to a high level. The RCIC System suction must be manually transferred from

the CST to the suppression pool using controls located in the main control room. This action would be taken when control room alarms indicate low CST or suppression pool high water level.

The operator can manually initiate the HPCI and RCIC Systems from the control room before the low level automatic initiation level is reached. The operator has the option of manual control after automatic initiation and can maintain reactor water level by throttling system flow rates. This would prevent a trip of the systems due to high water level. The operator can verify that these systems are delivering water to the reactor vessel by --

(1) Verifying reactor water level increases when systems initiate.

(2) Verify systems flow using flow indicators in the control room.

(3) Verify system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to the reactor.

The HPCI and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Core Spray (CS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

(b) Containment Cooling

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety relief valve (SRV) discharge to the suppression pool and RCIC and HPCI exhaust to the suppression pool. This would be accomplished by placing the RHR System in the containment (suppression pool) cooling mode; i.e., RHR suction from and discharge to the suppression pool.

The operator could verify proper operation of the RHR System containment cooling function from the control room by --

(1) Verifying RHR and Service Water (SW) System flow using system control room flow indicators.

(2) Verify correct RHR and SW System flow paths using control room position indication of motor-operated valves.

Even though the RHR is in the containment cooling mode, core cooling is its primary function. Thus, if a high drywell pressure signal is received at any time during the period when the RHR is in the containment cooling mode, the RHR System will automatically revert to the LPCI injection mode. The CS System would automatically initiate and both the LPCI and CS Systems would inject water into the reactor vessel if reactor pressure is below system discharge pressure. If there is a coincident low reactor water level and high drywell pressure, the Automatic Pressure Relief System (APRS) will automatically relieve reactor pressure to allow the LPCI and CS Systems to inject water into the vessel after a two-minute time delay.

(c) Extended Core Cooling

When the reactor has been depressurized, the RHR System can be placed in the long term shutdown cooling mode. The operator manually terminates the containment cooling mode of one of the RHR containment cooling loops and places the loop in the shutdown cooling mode as follows:

(1) trip the RHR pumps,

(2) close motor-operated valves in the suppression pool suction and discharge lines,

(3) open suction valves from and discharge valves to the reactor vessel, and

(4) restart the RHR pumps.

In this operating mode, the RHR System can cool the reactor to cold shutdown. Prior operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

4. Reactor vessel water is continuously monitored by nine indicators or recorders for normal and accident conditions. Level instruments used to provide automatic safety feature actuation are arranged in a redundant array of one or two instruments in each of two independent divisions.

A separate set of level instrumentation provides reactor level control by means of the feedwater system. This set includes two control room indicators and one recorder.

For a detailed description of the reactor vessel level monitoring system refer to FSAR Section 7.5.0, Reactor Vessel Instrumentation.

Application of the reactor vessel level signals is described in FSAR Sections 6.2.0, Emergency Core Cooling Systems, and 7.7.0, Plant Protection System.

Other instrumentation indications which provide information to the operator relative to possible changes in reactor coolant inventory include:

- Drywell High Pressure
- Drywell Atmosphere High Radioactivity
- Suppression Pool High Temperature
- Safety Relief Valve (SRV) Discharge High Temperature or Discharge Pressure Switch Actuated
- High Feedwater Flow Rates
- High Main Steam Flow
- High Containment and Equipment Area Temperatures
- Abnormal Reactor Pressure
- High Suppression Pool Water Level
- High Drywell Sump Fill and Pumpout Rate
- Decreased Hotwell Rate of Reject or Increased Hotwell Makeup Requirements
- Decrease in Electric Generation
- Increased Radiation Level at Off-Gas Monitor
- Increased Radiation Level at Main Steamline Monitors
- Increased Area Radiation Levels
- Imbalance of dP or Flow Between Steam Lines
- Main Steam Line Leakage Alarm

An example of the use of this additional information by the operator is as follows: Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High SRV discharge temperature and discharge pressure switch actuated would pinpoint loss of coolant via a particular open valve.

As noted in item 1) above, Directives have been issued to instruct operators to utilize all available information. Procedures for specific abnormal situations identify potential instrument indications which may be used by the operator to identify problems.

5. Considerable effort has been expended to assure that operating procedures and Administrative Directives provide complete information, such that no other instructions are required for training associated with operation of plant systems. Great care has been taken to assure that system descriptions (which are also used for training) provide descriptive material related to system design and theory only. Therefore, "training instructions" relevant to this item of the Bulletin do not exist.

   Plant operating procedures and Administrative Directives have been reviewed, and it has been determined that several valid reasons exist for allowing an override of an automatic initiation signal or shutdown of a system after it has been automatically initiated (see response to Item 1, above). For example,

   (a) If an automatic initiation of the HPCI System and RCIC System occurs, the operator is allowed to shutdown the HPCI System if the RCIC System is capable of maintaining vessel level. This is allowed to prevent an automatic trip of both systems due to high water level. As noted in response to Item 3 of this Bulletin, a trip of the RCIC System requires manual operator action to reset.

   (b) The procedures allow the operator to manually override automatic actuation of the APRS if it has been determined that adequate water level is being maintained by the high pressure injection systems. In this case, low pressure coolant injection is not required and therefore APRS actuation can be interrupted. This override is permitted to allow a controlled cooldown and depressurization of the reactor and to prevent injection of poor quality torus water into the reactor when it is not required.

   (c) The procedures allow transfer of part or all of the RHR System from the LPCI mode of operation to the containment cooling mode of operation when adequate reactor water level is maintained with part of the RHR System and/or other systems. This is allowed to assure that torus water temperatures and containment pressure limits are maintained.

   Other procedures which specifically discuss override or shutdown of an emergency system allow it only with Plant Management approval and would only be performed after consideration by plant technical staff personnel.

   Directives were issued, as stated in the response to Item 1, above, to assure operators consider all available information in decisions to take manual action. Operating procedures for specific events do describe expected parameter indications. However, it should be

recognized that events might occur such that vessel level indication might be the only immediately obvious parameter affected. We are reluctant to issue instructions which might be considered contrary to the longstanding directive for operators to believe and respond conservatively to instrument indications unless the indications are proven to be incorrect.

6. Review of all safety-related valve positions, positioning requirements, and positive controls to assure that valves remain positioned in a manner to ensure the proper operation of engineered safety features has been completed numerous times and will be completed periodically in the future as part of the ongoing Operational QA Program.

   Prior to a plant startup, valve checklists are completed to establish and verify all safety-related valve positioning. On plant restarts, following short outages when major maintenance is not performed, checklists may not be completed; however, they are reviewed by the Shift Supervisor to assure he understands the status of plant systems and equipment. The valve checklists are prepared and reviewed in accordance with Operational QA Program requirements.

   For maintenance and surveillance, most activities are governed by specific procedures which are also prepared and reviewed in accordance with Operational QA Program requirements. Activities not governed by specific procedures are controlled by the Work Request Authorization process which includes controls on valves for pre and post work valve positioning.

   Following preparation or revision of the operational checklists and maintenance, surveillance, and special procedures, they are reviewed by an independent person knowledgeable in the affected area and by the Plant Operations Committee, and they are approved by a member of plant management. In addition, all procedures, including checklists, are reviewed at least once every two years for required changes (or prior to use as may be the case for special procedures). A temporary change process is used to document and review situations when valves cannot or should not be positioned as specified in the procedure or checklist. This process includes considerations for any required permanent changes to procedures.

   In addition to specific procedure controls, control room operators check all control and annunciator panels for normal indications each shift. This include valve position indications. Valves critical to system operation but without position indication in the control room are locked in the required position. Their positioning is verified with completion of startup checklists and in most cases with periodic completion of surveillance testing.

In May, 1979, the Region III Office of Inspection and Enforcement conducted a special unannounced inspection during which the inspectors verified by independent examination of records, procedures and equipment that the reactor core isolation cooling, high pressure coolant injection, auto pressure relief, low pressure core spray, residual heat removal, standby liquid control, and emergency diesel generator systems were operable according to Technical Specification requirements and that Administrative Controls provide adequate assurance of continued operability. This inspection, which is documented in Report No. 50-263/79-06, included a comprehensive verification of valve/breaker/switch alignment procedures, checklists and Administrative Controls for system operation, testing and maintenance. It also involved verification of all accessible valve/breaker/switch alignments for the inspected systems.

The plant staff has completed additional reviews and inspections for safety-related valves, including Primary Containment Isolation, Standby Gas Treatment System, Residual Heat Removal Service Water System, Control Rod Drive System and Emergency Service Water System procedures and valves which were not covered in the NRC Inspection.

Proper positioning requirements and proper positioning for all safety-related valves has been verified. Valves located in inaccessible areas were verified by means of remote position indication, satisfaction of valve position interlocks and normal system pressures and flows.

The actions requested by the Bulletin are therefore completed.

7. All systems designed to transfer potentially radioactive gases and liquids out of the primary containment are provided with automatic isolation signals initiated by a variety of reactor, containment or system conditions. These systems are:

   Main Steam Lines

   Main Steam Line Drains

   Recirculation Loop Sample Line

   Drywell Equipment Drain Sump

   Drywell Floor Drain Sump

   Reactor Water Cleanup System

   Containment Vent and Purge System

   HPCI

   RCIC

   RHR

Containment Nitrogen Recirculation

Containment Atmosphere Sample

Automatic isolation valves and the signals which initiate isolation are listed in Technical Specification Table 3.7.1 (copy attached). As indicated in the Table there are five (5) different isolation logic groups. The logic system trip signal for each of the five groups "seals-in", i.e., it remains in effect until the operator manually resets the logic system, even though the instruments which initiated the logic system trip may have reset in the meantime. If the condition which initiated the logic trip persists, the system will remain tripped regardless of attempts to reset.

The final assurance against inadvertant transfer of radioactive liquids or gases outside of containment depends upon operator evaluation of reactor and containment conditions prior to resetting the isolation logic. As discussed in items 1 and 4, Administrative Controls require that reactor conditions be evaluated prior to resetting the isolation signals and opening the valves. Procedures and Administrative Directives do not allow such action unless it is positively determined that isolation is not required or that continued isolation would result in an unsafe condition.

Continued operability of these features is assured by periodically calibrating the sensors as well as testing the isolation functions to insure proper response. Additionally, Administrative Controls and checklists, as discussed in Item 6 above, are used to verify systems are lined up for automatic operations. The feasibility of installing additional isolation signals on high radiation will be evaluated.

8. Removal of equipment from service is controlled by a procedure or with a Work Request Authorization. In each instance, Administrative Controls currently require Technical Specification requirements for the operability of redundant safety-related systems be identified and verified. Administrative Controls will be issued by August 30, 1979, to require a visual check of safety system status to the extent practical prior to the removal of safety system equipment from service. These Administrative Controls will apply to the conduct of all procedures and WRAs.

   Upon completion of procedures or WRA activities, Administrative Controls require identification and verification of functional testing or other actions necessary to assure operability.

   Signed authorization to commence all maintenance and test activities must be obtained from the Shift Supervisor. In addition, plant operators are the only personnel who remove equipment from and

return it to service. Records of equipment tagging, installation of jumpers and bypasses, and the use of keylocked switches for maintenance or modification activities are maintained by the Shift Supervisors. This provides adequate notification and control by operating personnel of safety-related system status.

The procedures and WRAs are reviewed, approved and audited in accordance with requirements of the Operational Q.A. Program. This ensures that the above requirements are included and adhered to.

All oncoming operational personnel are explicitly notified about the status of systems removed from or returned to service at shift turnover by several formal methods which are detailed in Administrative Directives. Each on-duty operator is required to verbally communicate to the oncoming operator prior to turnover, a summary of the status of his areas of responsibility, including the following:

1) Any tests or special procedures in progress at shift change.

2) Work in progress and components isolated for work.

3) Any off-normal conditions in plant power supplies, ECCS, Reactor Pressure Control, Reactor Level Control, Nuclear Instrumentation, Recirc Flow Control, Reactor Protection System, Primary and Secondary Containment and Radiation Monitoring Systems.

4) Any off-normal conditions that are safety-related, or, which involve safety-related equipment.

Also, the oncoming shift personnel are required to make themselves aware of the plant operating status as follows:

1) All shift personnel must read their respective logs to become aware of operations that took place since they were last on duty.

2) The Shift Supervisor must check the Night Order Book. The Bypass Log, Shift Supervisor's Key Cabinet status and active Work Request Authorizations. Documentation of this check is entered in the Reactor and Control Room Log.

3) The control room operators are required to check all control panels for normal indication annunciators and radiation monitors. Documentation of this check is entered in the Reactor and Control Room Log.

Adherence to the above requirements is audited in accordance with the Operational Q.A. Program and has also been periodically inspected by the Region III Office of Inspection and Enforcement.

9. Emergency procedures have been revised to notify the NRC within one hour and establish a continuous open communication channel as rapidly as possible in the event that the reactor is not in a controlled or expected condition of operation. During normal daytime working hours, and for events which are not extremely complex, it is expected that such communication can be established within one hour. However, if a complex emergency were to occur during weekend, holiday or evening hours when the number of on-site personnel is reduced, conditions might require utilization of all on-site personnel for activities related to assessment of conditions and controlling or mitigating the situation. In that event, establishment of continuous communication would be delayed until arrival of additional staff personnel.

10. During normal operation, the containment is inerted with nitrogen gas to maintain oxygen concentration below five percent by volume. This minimizes the possibility of hydrogen combustion following an accident in which significant amounts of hydrogen are generated.

    In the event of an accident, samples of containment atmosphere may be obtained and analyzed in the laboratory for hydrogen. The containment may be purged and vented using non-safeguards systems to control hydrogen concentration.

    In the event of hydrogen that may remain inside the reactor vessel, these gases may be vented to the containment through the reactor head vent. The head vent isolation valves, while not safeguards equipment, can be operated remotely from the control room.

    In the October 27, 1978, Federal Register, the NRC published their final rule on combustible gas control systems for nuclear power plants. The rule, contained in Section 50.44 of 10CFR Part 50, establishes requirements for combustible gas control systems based on the date of the construction permit notice and on the results of accident dose calculations. We will comply with this rule and have initiated an engineering study to identify the plant modifications which are needed. This engineering study will include the feasibility of installing hydrogen recombiners.

If additional information is required, please communicate directly with plant management.

Yours very truly,

L. J. Wachter

L. J. Wachter
Vice President - Power Production
and System Operation

cc: Mr. G Charnoff
Office of Inspection and Enforcement
Washington, D. C.
Mr. T. A. Ippolito, Chief
Operating Reactors Branch #3

attachment

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Number of Valves		Maximum Operating Time (Sec)	Normal Position
		Inboard	Outboard		
1	Main Steam Line Isolation	4	4	$3 \leq T \leq 5$	Open
1	Main Steam Line Drain	1	1	60	Closed
1	Recirculation Loop Sample Line	1	1	60	Closed
2	Drywell Floor Drain		2	60	Open
2	Drywell Equipment Drain		2	60	Open
2	Drywell Vent		2	60	Closed
2	Drywell Vent Bypass		1	60	Closed
2	Drywell Purge Inlet		2	60	Open
2	Drywell and Suppression Chamber Air Makeup		1	60	Closed
2	Suppression Chamber to Drywell N_2 Recirculation		1	60	Open
2	Suppression Chamber Vent		2	60	Closed
2	Suppression Chamber Vent Bypass		1	60	Open
2	Shutdown Cooling System	1	1	120	Closed

TABLE 3.7.1, continued

PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Number of Valves		Maximum Operating Time (Sec)	Normal Position
		Inboard	Outboard		
2	Shutdown Cooling System		1	120	Closed
2	Shutdown Cooling System		1	120	Closed
2	Reactor Head Cooling Line	1	1	120	Closed
3	Cleanup Demineralizer System	1	1	40	Open
3	Cleanup Demineralizer System		1	40	Open
4	HPCI Turbine Steam Supply	1	1	40	Open
5	RCIC Turbine Steam Supply	1	1	30	Open

NOTE: Isolation Groupings are as follows:

Group 1: The valves in Group 1 are closed upon any of the following conditions:
1. Reactor low low water level
2. Main steam line high radiation
3. Main steam line high flow
4. Main steam line tunnel high temperature
5. Main steam line low pressure (RUN mode only)

Group 2: The actions in Group 2 are initiated by any one of the following conditions:
1. Reactor low water level
2. High drywell pressure

TABLE 3.7.1, continued

PRIMARY CONTAINMENT ISOLATION

Group 3: The actions in Group 3 are initiated by reactor low water level.

Group 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:
1. HPCI steam line high flow
2. HPCI steam line low pressure
3. High temperature in the vicinity of the HPCI steam line

Group 5: Isolation valves in the reactor core isolation cooling system (RCIC) are closed upon any one of the following signals:
1. RCIC steam line high flow
2. RCIC steam line low pressure
3. High temperature in the vicinity of the RCIC steam line