UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 14, 1994

Docket No. 50-461

Clinton Power Station
ATTN: Mr. Richard F. Phares
Director - Licensing
Post Office Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Phares:

SUBJECT: REVISIONS TO TECHNICAL SPECIFICATION BASES SECTIONS - CLINTON POWER STATION, UNIT NO. 1 (TAC NO. M88373)

Your letter of December 10, 1993 (U-602192), requested the following changes to the Clinton Power Station Technical Specifications Bases.

Section 3/4.3.2, "Containment and Reactor Vessel Isolation Control System"

Amendment No. 86 to the Clinton Power Station Operating License, issued on November 5, 1993, extended the out-of-service time for an inoperable leak detection differential temperature instrument channel(s) provided a sufficient number of associated ambient temperature instrument channels remained operable. Following our review, the staff recommended that Bases Section 3/4.3.2 be modified to indicate that the extended time is permitted because no loss of function exists during the time that the differential temperature instrument channel(s) is(are) inoperable. Your letter proposed this modification.

Section 3/4.5.1 and 3/4.5.2, "ECCS - Operating and Shutdown"

By letter dated September 13, 1993, the staff approved Relief Request No. 2036 relative to ASME Section XI Inservice Testing. This action expanded the acceptance criteria for the safety relief valve (SRV) safety-mode lift setpoint from ±1% to ±3%. Accordingly, a plant specific analysis assuming a higher SRV lift setpoint increased the peak pressure at which the high pressure makeup systems are required to operate by approximately 23 psi. Modifications to Bases Section 3/4.5.1 and 3/4.5.2 were proposed to address this higher lift setpoint.

Section 3/4.6.5, "Drywell Post-LOCA Vacuum Relief Valves"

Amendment No. 84 to the Clinton Power Station Operating License, issued on September 20, 1993, revised three Action Statements involving inoperable drywell post-LOCA vacuum relief valves. In the safety evaluation, the staff requested a revised Bases Section 3/4.6.5 to accurately describe the post-accident function of the relief valves. Modifications to Bases Section 3/4.6.5 were proposed.

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The staff has reviewed these proposed modifications to the Bases sections and finds them acceptable. The appropriate Bases pages have been modified and are included for your use.

The staff is also reissuing Bases page 3/4 8-3. This page was modified by changes made in our letter of May 21, 1993, but was inadvertently omitted when issued. This page is also included for your use.

Sincerely,

Original signed by Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
Bases Pages B 3/4 3-3, B 3/4 5-2,
B 3/4 6-8, B 3/4 6-8a and B 3/4 8-3

cc w/enclosure:
See next page

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Mr. Richard F. Phares
Illinois Power Company

Clinton Power Station
Unit No. 1

cc:

Mr. J. S. Perry
Senior Vice President
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

Mr. J. A. Miller
Manager Nuclear Station
Engineering Department
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

Resident Inspector
U.S. Nuclear Regulatory Commission
RR#3, Box 229 A
Clinton, Illinois 61727

Mr. R. T. Hill
Licensing Services Manager
General Electric Company
175 Curtner Avenue, M/C 481
San Jose, California 95125

Regional Administrator, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Chairman of DeWitt County
c/o County Clerk's Office
DeWitt County Courthouse
Clinton, Illinois 61727

Mr. Robert Neumann
Office of Public Counsel
State of Illinois Center
100 W. Randolph, Suite 11-300
Chicago, Illinois 60601

Mr. J. W. Blattner
Project Manager
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Illinois Department
of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

3/4.3.2 CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (Continued)

each case which in turn determines the valve speed in conjunction with the 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

Differential and ambient temperature instrumentation is provided for certain equipment rooms or areas to effect automatic isolation of the affected systems in response to a 25-gpm equivalent steam leak. The ACTIONs specified in Technical Specification 3.3.2 for inoperable differential temperature instrumentation permit an extension of the allowed outage time for up to 24 hours when a sufficient number of ambient temperature channels remain OPERABLE in the affected area to maintain the capability for automatic isolation in response to a steam leak.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

The emergency core cooling system (ECCS) pump minimum flow instruments are provided to ensure that ECCS pump minimum flow paths are preserved to prevent pump damage in the event that ECCS pumps are started without reactor or test line flow paths. The minimum flow instruments are not part of ECCS actuation instrumentation.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979, and Section 15.8 of the USAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is an essential safety supplement to the Reactor Protection System. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of the turbine control valves provides input to the four RPS logic divisions of the EOC-RPT system. Similarly, a position switch for each of the turbine stop valves provides input to the four logic divisions of the EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-4 logic for the fast closure of turbine control valves and a 2-out-of-4 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 140 ms. Included in this time are: the time from initial valve movement to reaching the trip setpoint; the response time of the sensor; the response time of the system logic and the time alloted for breaker arc suppression.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel.

Operation with a trip set less conservative than its Trip Satpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING AND SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 467/1400/5010 gpm at differential pressures of 1200/1147/200 psid. Initially, water from the reactor core isolation cooling (RCIC) tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the RCIC tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days, as specified in the corresponding ACTION statement, is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

The surveillance requirements for the ADS include a requirement to manually open each ADS valve. This requirement includes an exception to the provisions of Specification 4.0.4. This exception allows reactor steam conditions to be established which are adequate to open the ADS valves without resulting in unnecessary wear on the valves and to ensure that proper reactor pressure control can be maintained while opening and reclosing the valves. Reactor steam conditions which are considered adequate to perform the test thus include the establishment of sufficient reactor pressure as well as sufficient

3/4.6.5 DRYWELL POST-LOCA VACUUM RELIEF VALVES

Drywell vacuum relief valves are provided on the drywell to pass sufficient quantities of gas from the containment to the drywell to prevent an excess negative pressure from developing in the drywell following a large-break LOCA. In addition, this function controls rapid weir wall overflow (following a large-break LOCA) to minimize drag and impact loadings on essential equipment and systems in the drywell above the weir wall. OPERABILITY for opening of the drywell vacuum relief valves is also required to support operation of the drywell-containment atmosphere mixing system.

The drywell post-LOCA vacuum relief valve penetrations are required to be closed in order to prevent steam bypass of the suppression pool in the event of a LOCA.

3/4.6.6 SECONDARY CONTAINMENT

The secondary containment completely encloses the primary containment, except for the upper personnel hatch. It consists of the fuel building, gas control boundary, and portions of the auxiliary building enclosed by the extension of the gas control boundary and the ECCS cubicles and areas as described in USAR Figure 6.2-132. The standby gas treatment system (SGTS) is designed to achieve and maintain a negative 1/4" W.G. pressure within the secondary containment following a design basis accident. This design provides for the capture within the secondary containment of the radioactive releases from the primary containment, and their filtration before release to the atmosphere.

Establishing and maintaining a vacuum in the secondary containment with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. The inleakage values are not verified in the surveillances since no credit for dilution was taken in the dose calculation. As noted however, adequate drawdown is verified once per 18 months. The acceptance criteria specified in Figure 4.6.6.1-1 for the drawdown test is based on a computer model, verified by actual performance of drawdown tests, in which the drawdown time determined for accident conditions is adjusted to account for performance of the test during normal plant conditions. The acceptance criteria indicated per Figure 4.6.6.1-1 is based on conditions corresponding to power operation (with the turbine building ventilation system in operation) and wind speeds less than or equal to 10 mph. The acceptance criteria for plant conditions other than those assumed will be adjusted as necessary to reflect the conditions which exist during performance of the surveillance test.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. The specified heater dissipation is based on a bus voltage of 460 volts. Heater test results shall be adjusted to account for actual bus voltage.

3/4.6.7 ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the containment below its flammable limit during post-LOCA.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance. The low-frequency motor generator set electrical power supply to the reactor recirculation pumps is provided with one overcurrent protection circuit breaker since the generator's maximum output under fault conditions is less than the penetration's design rating. The surveillance requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The bypassing of the motor-operated valves thermal overload protection continuously ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection continuously are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, March 1977.

The reactor protection system (RPS) electric power monitoring assemblies provide protection to the RPS and other systems which receive power from the RPS buses by acting to disconnect the RPS from the power source in the presence of an electrical fault in the power supply.

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