Nebraska Public Power District

GENERAL OFFICE
P. O. BOX 499, COLUMBUS, NEBRASKA 68601
TELEPHONE (402) 564-8561

January 30, 1979

Director, Nuclear Reactor Regulation
Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Proposed Changes to the Radiological Technical Specifications/Reactor Start-ups for Training
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

Dear Mr. Ippolito:

Pursuant to page 3 of I&E Inspection Report No. 50-298/78-07, the Nebraska Public Power District proposes several minor Technical Specification changes to resolve a minor discrepancy noted in the report. The discrepancy is as follows:

1. The status of the LPCI System is not clearly defined during reactor cool-down or during evolutions such as shutdown margin testing, low power core physics testing or reactor start-ups for training. (Unresolved Item 7801-1).

   To resolve this item several minor changes to the Technical Specifications are proposed and copies of the affected pages are attached.

This request for a license amendment is considered to be exempt from any fee because it is of minor safety significance and is being submitted to clarify the Technical Specifications at the request of the Commission.

Should you have any questions or require additional information regarding these proposed changes, please do not hesitate to contact me.

In addition to three signed originals, 37 copies of the proposed changes are also submitted.

Sincerely yours,

Jay M. Pilant
Director of Licensing and
Quality Assurance

JMP/jdw:srs29/18
Enclosure

7902020238

STATE OF NEBRASKA)
) ss
PLATTE COUNTY)

Jay M. Pilant, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to execute this request on behalf of Nebraska Public Power District; and that the statements in said application are true to the best of his knowledge and belief.

Jay M Pilant
Jay M. Pilant

Subscribed in my presence and sworn to before me this 30th day of January, 1979.

Marilyn R. Hohndorf
NOTARY PUBLIC

My Commission expires Oct. 14, 1980.

GENERAL NOTARY - State of Nebraska
MARILYN R. HOHNDORF
My Comm. Exp. Oct. 14, 1980

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.F (cont'd)	4.5.F (cont'd)
h. A special flange, capable of sealing a leaking control rod housing, is available for immediate use.	
i. The control rod housing is blanked following the removal of the control rod drive.	
j. No work is being performed in the vessel while the housing is open.	
6. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.	
7. The LPCI System is not required to be operable while performing training startups at atmospheric pressure at power levels less than 1% of rated thermal power.	
G. Maintenance of Filled Discharge Pipe	G. Maintenance of Filled Discharge Pipe
Whenever core spray subsystems. LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.	The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:
	1. Whenever the Core Spray, LPCI, HPCI or RCIC systems are made operable, the discharge piping shall be vented from the high point of the system and water flow observed initially and on a monthly basis.
	2. The pressure switches which monitor the LPCI, core spray, HPCI and RCIC lines to ensure they are full shall be functionally tested and calibrated every three months.

3.5.A BASES

Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analysis included in General Electric Topical Report NEDO-10329 and the sensitivity studies given in Supplement 1 thereto and subsection 6.5 of the FSAR and in accordance with the AEC's "Interim Acceptance Criteria for Emergency Core Cooling Systems" published on June 19, 1971, any of the following cooling systems provides sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel clad temperature to less than 2300°F, to assure that core geometry remains intact, and to limit clad metal-water reaction to less than 1%; the two core spray subsystems; or either of the two core spray subsystems and three RHR pumps operating in the LPCI mode with operable LPCI injection valves.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. During reactor shutdown when the residual heat removal system is realigned from LPCI to the shutdown cooling mode, the LPCI System is considered operable.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Cooper Nuclear Station, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psig.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem and the core spray subsystem provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in reference (1). Using the results developed in this reference, the repair period is found to be 1/2 the test interval. This assumes that the

---

(1) Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Co. A.P.E.D., April, 1969 (APED 5736).

ment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. Specification 3.5.F.4 provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.5.F.5 recognizes that, concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.2.h. In this case, if excessive control rod housing leakage occurred, three levels of protection against loss of core cooling would exist. First, a special flange would be used to stop the leak. Second, sufficient inventory of water is maintained to provide, under worst case leak conditions, approximately 60 minutes of core cooling while attempts to secure the leak are made. This inventory includes water in the reactor well, spent fuel pool, and condensate storage tank. If a leak should occur, manually operated valves in the condensate transfer system can be opened to supply either the core spray system or the spent fuel pool. Third, sufficient inventory of water is maintained to permit the water which has drained from the vessel to fill the torus to a level above the core spray and LPCI suction strainers. These systems could then recycle the water to the vessel. Since the system cannot be pressurized during refueling, the potential need for core flooding only exists and the specified combination of the core spray or the LPCI system can provide this. This specification also provides for the highly unlikely case that both diesel generators are found to be inoperable. The reduction of rated power to 25% will provide a very stable operating condition. The allowable repair time of 24 hours will provide an opportunity to repair the diesel and thereby prevent the necessity of taking the plant down through the less stable shutdown condition. If the necessary repairs cannot be made in the allowed 24 hours, the plant will be shutdown in an orderly fashion. This will be accomplished while the two off-site sources of power required by Specification 3.9.A.1 are available. Specification 3.5.F.7 provides for the performance of training startups without realigning the residual heat removal system from the shutdown cooling mode to the LPCI mode. Power levels during training startups are kept below the level of significant heat addition.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. If a water hammer were to occur at the time at which the system were required, the system would still perform its design functions. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

H. Engineered Safeguards Compartments Cooling

The unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.7.A (cont'd)	4.7.A (cont'd)
f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.	e. During physics testing or training startups, when primary containment integrity is not required, the thermal power and reactor coolant temperature shall be verified to be within the limits at least once per hour.
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above	2. Integrated Leak Rate Testing a. Integrated leak rate tests (ILRT's)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.7.A (cont'd)	4.7.A (cont'd)
212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests or training start-ups at atmospheric pressure at power levels not to exceed 1% of rated thermal power.	shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the leakage rate does not exceed the equivalent of 0.635 percent of the primary containment volume per 24 hours at 58 psig. b. Integrated leak rate tests may be performed at either 58 psig or 29 psig the leakage rate test period, extending to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period the agreed-upon shorter period may be used. Prior to initial operation, integrated leak rate tests must be performed at 58 and 29 psig (with the 29 psig test being performed prior to the 58 psig test) to establish the allowable leak rate, L_t (in percent of containment volume per 24 hours) at 29 psig as the lesser of the following values: (L_a is 0.635 percent) $L_t = 0.635 \frac{L_{tm}}{L_{am}}$ for $\frac{L_{tm}}{L_{am}} \leq 0.7$ where L_{tm} = measured ILR at 29 psig L_{am} = measured ILR at 58 psig, and $\frac{L_{tm}}{L_{am}} \leq 1.0$

3.7.A & 4.7.A BASES

Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. Exceptions are made to this requirement while low power physics tests or low power training startups are being conducted to allow ready access to the primary containment. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 58 psig which is below the maximum of 62 psig. Maximum water volume of 91,000 ft^3 results in a downcomer submergence of 5' and the minimum volume of 87,650 ft^3 results in a submergence approximately 12 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only