ATTACHMENT C

PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS, OF FACILITY OPERATING LICENSES DPR-29 AND DPR-30

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e. Core spray header Δp instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

f. Logic system functional test — Once/Each refueling outage

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that during such 7 days all active components of the other core spray subsystem and the LPCI mode of the RHR system and the diesel generators required for operation of such components if no external source of power were available shall be operable.

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~~2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI mode of the RHR system shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.~~

3. The LPCI mode of the RHR system shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

$\boxed{2.}$

~~3.~~ 2. LPCI mode of the RHR system testing shall be as specified in Specifications 4.5.A.1.a, b, c, d, and f, except that each LPCI division (two RHR pumps per division) shall deliver at least 9000 gpm against a system head corresponding to a reactor vessel pressure of 20 psig, with a minimum flow valve open.

4. From and after the date that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days the remaining active components of the LPCI mode of the RHR, containment cooling

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~~4. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI mode of the RHR, containment cooling mode of the RHR, and both core spray subsystems shall be demonstrated to be operable immediately and the operable RHR pumps daily thereafter.~~

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mode of the RHR, all active components of both core spray subsystems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment cooling mode of the RHR (including two RHR pumps), and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated, and the reactor shall be in the cold shutdown condition within 24 hours.

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~~5. When it is determined that the LPCI mode of the RHR system is inoperable, both core spray subsystems, the containment cooling mode of the RHR shall be demonstrated to be operable immediately and daily thereafter.~~

B. Containment Cooling Mode of the RHR System

1. a. Both loops of the containment cooling mode of the RHR system, as defined in the bases for Specification 3.5.B, shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

B. Containment Cooling Mode of the RHR System

Surveillance of the containment cooling mode of the RHR system shall be performed as follows:

1. RHR service water subsystem testing:

Item	Frequency
a. Pump and valve operability	Once/3 months

1. b. From the effective date of this amendment until November 1, 1989, the "B" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 1 "C" and "D" RHR service water pumps using cross tie line 1/2-10509-16"-D. Consequently, the requirements of Specifications 3.5.B.2 and 3.5.B.3 will impose the corresponding surveillance testing of equipment associated with both reactors if the shared RHR service water pump or pumps, or the cross tie line, are made or found to be inoperable.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days all other active components of the containment cooling mode of the RHR system are operable.

3. From and after the date that one loop of the containment cooling mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that all active components of the other loop of the containment cooling mode of the RHR system, both core spray subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

b.	Flow rate test - each RHR service water pump shall deliver at least 3500 gpm against a pressure of 198 psig	After pump maintenance and every 3 months
c.	A logic system functional test	Each refueling outage

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~~2. When it is determined that one RHR service water pump is inoperable, the remaining components of that loop and the other containment cooling loop of the RHR system shall be demonstrated to be operable immediately and daily thereafter.~~

~~3. When one loop of the containment cooling mode of the RHR system becomes inoperable, the operable loop shall be demonstrated to be operable immediately, and daily thereafter.~~

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4. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F and prior to reactor startup from a cold condition. Continued reactor operation is permitted provided that a maximum of one drywell spray loop may be inoperable for 30 days when the reactor water temperature is greater than 212°F.

5. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours.

C. HPCI Subsystem

1. The HPCI subsystem shall be operable whenever the reactor pressure is greater than 150 psig and fuel is in the reactor vessel.

2. During startup following a refuel outage or an outage in which work was performed that directly affects HPCI system operability, if the testing requirements of 4.5.C.3.a cannot be met, continued reactor startup is not permitted. The HPCI subsystem shall be declared inoperable, and the provisions of Specification 3.5.C.4 shall be implemented.

3. Except for the limitations of 3.5.C.2, if the HPCI subsystem is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 14 days unless such subsystem is sooner made operable, provided that during such 14 days the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable. Otherwise, the provisions of Specification 3.5.C.4 shall be implemented.

4. During each 5-year period, an air test shall be performed on the drywell spray headers and nozzles and a water spray test performed on the torus spray header and nozzles.

C. HPCI Subsystem

Surveillance of HPCI subsystem shall be performed as specified below with the following limitations. For item 4.5.C.3, the plant is allowed 12 hours in which to successfully complete the test once reactor vessel pressure is adequate to perform each test. In addition, the testing required by item 4.5.C.3.a shall be completed prior to exceeding 325 psig reactor vessel pressure. If HPCI is made inoperable to perform overspeed testing, 24 hours is allowed to complete the tests before exceeding 325 psig.

	Item	Frequency
1.	Valve Position	Every 31 days
2.	Flow Rate Test- HPCI Pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 1150 psig when steam is being supplied to the turbine at 920 to 1005 psig.	Every 92 days

3.5 LIMITING CONDITIONS FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

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

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDC-31345P core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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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. ~~Under these limiting conditions of operation, increased surveillance testing of the remaining ECCS systems provides assurance that adequate cooling of the core will be provided during a loss-of-coolant accident.~~

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Quad-Cities 1 and 2, 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 cooling of the reactor core before the internal pressure has fallen to 90 psig.

The LPCI mode of the RHR system 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 cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.05 ft^2 up to and including 4.26 ft^2, the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops closed 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 3. Using the results developed in this reference, the repair period is found to be less than

half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. ~~For multiple failures, a shorter interval is specified; to improve the assurance that the remaining systems will function, a daily test is called for.~~ Although it is recognized that the information given in Reference 1 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

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Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. ~~To assure that the remaining core spray and the LPCI mode of the RHR system are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves.~~ Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

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Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, ~~which will be demonstrated to be operable;~~ a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The Containment Cooling mode of the RHR System consists of two loops. Each loop consists of 1 Heat Exchanger, 2 RHR Pumps, and the associated valves, piping, electrical equipment, and instrumentation. The "A" loop on each unit contains 2 RHR Service Water Pumps. Until November 1, 1989, the "B" loop on each unit may utilize the "C" and "D" RHR Service Water Pumps from Unit 1 via a cross-tie line. After November 1, 1989, each "B" loop will contain 2 RHR Service Water Pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. ~~The operable system is demonstrated to be operable each day when the above condition occurs.~~ Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, ~~which is tested daily,~~ a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem and LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystem and LPCI mode of the RHR system. The core spray subsystem and the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The surveillance requirements bases described in this paragraph apply to all core and containment cooling systems except HPCI and RCIC. The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems. ~~With components or subsystems out of service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., causes the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.~~

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The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. ~~With a cooling system out of service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling systems. The verification of operability, as used in this context, for the remaining cooling systems means to administratively check by examining logs or other information to verify that the remaining systems are not out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the remaining systems. However, if a failure, design deficiency, etc., causes the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining systems. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test.~~ Following a refueling outage or an outage in which work was performed that directly affects system operability, the HPCI and RCIC pumps are flow rate tested prior to exceeding 325 psig and again at rated reactor steam pressure. This combination of testing provides adequate assurance of pump performance throughout the range of reactor pressures at which it is

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With a system, subsystem, loop, or equipment out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining systems, subsystems, loops, or equipment. The verification of operability, as used in this context, for the remaining cooling systems means to administratively check by examining logs or other information to verify that the remaining systems are not out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the remaining systems. However, if a failure, design deficiency, etc., causes the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining systems. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test.

e. Core spray header Δp instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

f. Logic system functional test — Once/each refueling outage

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that during such 7 days all active components of the other core spray subsystem and the LPCI mode of the RHR system and the diesel generators required for operation of such components if no external source of power were available shall be operable.

3. The LPCI mode of the RHR system shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

4. From and after the date that one of the RHR ; umps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days the remaining active components of the LPCI mode of the RHR, containment cooling mode of the RHR, all active components of both core spray subsystems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI mode of the RHR system is made or found to be inoperable for any reason,

2. LPCI mode of the RHR system testing shall be as specified in Specifications 4.5.A.1.a, b, c, d, and f except that each LPCI division (two RHR pumps per division) shall deliver at least 9000 gpm against a system head corresponding to a reactor vessel pressure of 20 psig, with a minimum flow valve open.

continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment cooling mode of the RHR (including two RHR pumps), and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated, and the reactor shall be in the cold shutdown condition within 24 hours.

B. Containment Cooling Mode of the RHR System

1. a. Both loops of the containment cooling mode of the RHR system, as defined in the bases for Specification 3.5.B, shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

1. b. From the effective date of this amendment until Nov. 1, 1989, the "B" loop of the containment cooling mode of the RHR system for each reactor may share the Unit 1 "C" and "D" RHR service water pumps using cross tie line 1/2-10509-16"-D. Consequently, the requirements of Specifications 3.5.B.2 and 3.5.B.3 will impose the corresponding surveillance testing of equipment associated with both reactors if the shared RHR service water pump or pumps, or the cross tie line, are made or found to be inoperable.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such 30 days all other active components of the containment cooling mode of

B. Containment Cooling Mode of the RHR System

Surveillance of the containment cooling mode of the RHR system shall be performed as follows:

1. RHR service water subsystem testing:

	Item	Frequency
a.	Pump and valve operability	Once/3 months
b.	Flow rate test - each RHR service water pump shall deliver at least 3500 gpm against a pressure of 198 psig	After pump maintenance and every 3 months
c.	A logic system functional test	Each refueling outage

the RHR system are operable.

3. From and after the date that one loop of the containment cooling mode of the RHR system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such subsystem is sooner made operable, provided that all active components of the other loop of the containment cooling mode of the RHR system, both core spray subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

During the time period from April 17, 1978 through April 30, 1978 while the 2A Containment Cooling Loop of the RHR System is made inoperable for heat exchanger repair, continued reactor operation is permissible beyond the above 7-day limitation, unless such loop is sooner made operable, provided that during the time the 7-day limit is exceeded, a visual inspection is performed daily to assure that proper valve alignment and system integrity is maintained in the "B" RHR loop.

4. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F and prior to reactor startup from a cold condition. Continued reactor operation is permitted provided that a maximum of one drywell spray loop may be inoperable for 30 days when the reactor water temperature is greater than 212°F.

5. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in a cold shutdown condition within 24 hours.

2. During each 5-year period, an air test shall be performed on the drywell spray headers and nozzles and a water spray test performed on the torus spray header and nozzles.

3.5 LIMITING CONDITIONS FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available.

Based on the loss-of-coolant analyses included in References 1 and 2 and in accordance with 10 CFR 50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit the calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact to limit the corewide cladding metal-water reaction to less than 1% and to limit the calculated local metal-water reaction to less than 17%.

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 3. Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. Although it is recognized that the information given in Reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The Containment Cooling mode of the RHR System consists of two loops. Each loop consists of 1 Heat Exchanger, 2 RHR Pumps, and the associated valves, piping, electrical equipment, and instrumentation. The "A" loop on each unit contains 2 RHR Service Water Pumps. Until Nov. 1, 1989, the "B" loop on each unit may utilize the "C" and "D" RHR Service Water Pumps from Unit 1 via a cross-tie line. After Nov. 1, 1989, each "B" loop will contain 2 RHR Service Water Pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. Based on the fact that when one system of the containment cooling mode of the RHR system becomes inoperable, only one system remains, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystems are a backup to the HPCI subsystem. They enable the core spray subsystem and LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems and LPCI mode of the RHR system. The core spray subsystem and/or the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

Analyses have shown that only four of the five valves in the automatic depressurization system are required to operate. Loss of one of the relief valves does not significantly affect the pressure relieving capability, therefore continued operation is acceptable. Loss of two relief valves significantly reduces the pressure relief capability of the ADS: thus, a 7 day repair period is specified with the HPCI available, and a 24 hour repair period with the HPCI unavailable.

E. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. Under these conditions the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therfore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

F. Emergency Cooling Availability

The purpose of Specification 3.5.F is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only two RHR pumps would be available. Likewise, if two RHR pumps were out of service and two RHR service water pumps on the opposite side were also out of service no containment cooling would be available. It is during the refueling outages that major maintenance is performed and during such time that all low-pressure core cooling systems may be out of service. This specification 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 systems. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Quad Cities Units 1 and 2 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and

4.5 SURVEILLANCE REQUIREMENTS BASES

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

With a system, subsystem, loop or equipment out of service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining systems, subsystems, loops or equipment. The verification of operability, as used in this context, for the remaining cooling systems means to administratively check by examining logs or other information to verify that the remaining systems are not out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the remaining systems. However, if a failure, design deficiency, etc., causes the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining systems. For example, if an out-of-service period is caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test.

The surveillance requirements bases in this paragraph apply to all core and containment cooling systems except RCIC and HPCI. The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

The surveillance requirements bases described in this paragraph apply only to the RCIC and HPCI systems. Following a refueling outage or an outage in which work was performed that directly affects system operability, the HPCI and RCIC pumps are flow rate tested prior to exceeding 325 psig and again at rated reactor steam pressure. This combination of testing provides adequate assurance of pump performance throughout the range of reactor pressures at which it is required to operate. The low pressure limit is selected to allow testing at a point of stable plant operation and also to provide overlap with low pressure ECC systems. A time limit is provided in which to perform the required tests during startup. This time limit is considered adequate to allow stable plant conditions to be achieved and the required tests to be performed. Flow rate testing of the HPCI and RCIC pumps is also conducted every 92 days at rated reactor pressure to demonstrate system operability in accordance with the LCO provisions and to meet inservice testing requirements for the HPCI system. Applicable valves are tested in accordance with the provisions of the inservice testing program. In addition, monthly checks are made on the position of each manual, power operated or automatic valve installed in the direct flowpath of the suction or discharge of the pump or turbine that is not locked, sealed, or otherwise secured in position. At each refueling outage, a logic system functional test and a simulated automatic actuation test is performed on the HPCI and RCIC systems. The tests and checks described above are considered adequate to assure system operability.

The verification of the main steam relief valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the pilot valves during the surveillance test. By

ATTACHMENT D

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

As discussed in Attachment A (Description of Proposed Amendment) of this submittal, the proposed amendment to the Technical Specifications for Quad Cities Station involves removal of the requirement to operate the Core Spray and RHR pumps and valves when part of the RHR or Core Spray systems is inoperable, as reflected in the surveillance requirements of later operating BWR plants and the Standard Technical Specifications. Commonwealth Edison company has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

- Involve a significant increase in the probability of consequences of an accident previously evaluated; or
- Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- Involve a significant reduction in a margin of safety.

Each of these considerations is discussed below.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes involve deletion of the redundant testing of the ECCS pumps and valves when part or all of the RHR system or Core Spray system is inoperable. The present testing methods represent requirements beyond those necessary to adequately assure that remaining core cooling systems are operable and capable of performing their design intent. The proposed deletion of this multiple system testing requirement will bring Quad Cities station in line with the testing requirements of current BWR plants, as well as the BWR Standard Technical Specifications and BWR Improved Technical Specifications (NUREG 1433).

The proposed changes do not affect any accident precursors and, therefore, do not increase the probability of an accident previously evaluated.

The proposed changes do not increase the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed changes do not modify RHR system or Core Spray system design, or reduce the capability of these systems to perform their design intent. The plant will not be placed in an operational condition which has not been previously evaluated, nor does this change introduce any new operating modes or new equipment. Therefore, the possibility of a new or different kind of accident has not been created.

The proposed changes do not involve a significant reduction in a margin of safety because:

The proposed amendment will not significantly reduce the availability of RHR system or Core Spray system when required to mitigate accident conditions. Excessive testing of systems and components can reduce rather than increase reliability. An acceptable level of testing to demonstrate operability currently being used at later BWR plants does not include multiple testing of redundant equipment when the RHR system or Core Spray system is inoperable. Therefore, these changes do not involve a significant reduction in the margin of safety.

ATTACHMENT E

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison Company has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed amendment meets the criteria for a categorical exclusion as provided under 10 CFR 51.22(c)(9). This conclusion has been determined because the proposed amendment: 1) involves no significant hazards consideration (see Attachment 'D' of this submittal); 2) involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and 3) involves no significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT F

NRC Safety Evaluations

Dresden Station Units Two and Three
Amendments 107 and 102
August 10, 1989

Quad Cities Station Units One and Two
Amendments 130 and 124
March 8, 1991

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO PROPOSED LICENSE AMENDMENT TO REVISE TESTING REQUIREMENTS

OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) AND STANDBY GAS

TREATMENT SYSTEMS (SGTS)

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated December 21, 1988, Commonwealth Edison Company (CECo) proposed to amend Appendix A of Provisional Operating License (POL) No. DPR-19 for Dresden Unit 2 and Facility Operating License No. DPR-25 for Dresden Unit 3 to: revise the testing requirements for other systems or subsystems of the Emergency Core Cooling System (ECCS) or Standby Gas Treatment Systems (SGTS) when one system or subsystems is inoperable; revise the operability requirements of several ECCS systems; and incorporate some administrative changes. By letter dated May 4, 1989, CECo provided supplemental information to support the proposed amendment and included two additional changes. These proposed changes which are part of the Dresden Station improvement program action plan, are consistent with similar technical specifications approved for more recently licensed BWRs and the BWR Standard Technical Specifications.

2.0 EVALUATION

A. Multiple Testing of ECCS and SBGT Systems

Present Dresden Units 2 and 3 Technical Specification Surveillance Requirements for ECCS and SBGT provide for demonstrating the operability of redundant systems or subsystems when one system or subsystem is inoperable. These requirements are as follows:

(1) One Core Spray subsystem inoperable-demonstrate operability immediately of the operable core spray subsystem and the LPCI subsystem. Demonstrate daily thereafter operability of the operable core spray subsystem.

(2) One LPCI inoperable-demonstrate operability immediately of the remaining LPCI subsystem, containment cooling subsystem, and both core spray subsystem. Demonstrate daily the operability of the operable LPCI pumps.

(3) The LPCI subsystem is inoperable-demonstrate operability immediately and daily thereafter of both core spray subsystems and the containment cooling subsystem.

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(4) One containment cooling subsystem service water pump is inoperable-demonstrate operability immediately and daily thereafter the remaining components of that subsystem and the other containment cooling subsystem.

(5) One containment cooling subsystem is inoperable-demonstrate operability immediately and daily thereafter of the operable containment cooling subsystem.

(6) The HPCI subsystem is inoperable-demonstrate operability immediately of the LPCI subsystem, both core spray subsystems, the automatic pressure relief subsystem and the motor operated isolation valves and shell side make-up system for the isolation condenser. Demonstrate operability daily of the motor operated isolation valves and shell side make-up system of the isolation condenser. Daily demonstration of the operability of the automatic pressure relief subsystem may be required depending on plant power level and the number of operating feedwater pumps.

(7) One of the five relief valves of the automatic pressure relief subsystem is inoperable-demonstrate the operability immediately and weekly thereafter of the HPCI subsystem.

(8) More than one relief valve of the automatic pressure relief subsystem is inoperable-demonstrate operability immediately of the HPCI subsystem.

(9) The isolation condenser system is inoperable-demonstrate operability immediately and daily thereafter of the HPCI subsystem.

(10) The unit or shared diesel generator is inoperable-demonstrate operability immediately and daily thereafter of all low pressure core cooling, the containment cooling subsystems, and the operable diesel generator.

(11) One SBGT subsystem is inoperable-demonstrate operability within 2 hours and daily thereafter of the operable SBGT subsystem.

The purpose of this proposed amendment change is to remove the redundant system testing requirements from the ECCS and SGTS sections of the Technical Specifications (Sections 4.5 and 4.7) while maintaining adequate assurance of system operability needed for accident mitigation.

The requirement for demonstrating operability of the redundant systems identified above for Dresden Units 2 and 3 was originally chosen because there was a lack of plant operating history and a lack of sufficient equipment failure data. Since that time, plant operating experience has demonstrated that testing of the redundant ECCS and SGTS when one system is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the redundant system out of service for testing creates the risk of the second system also failing and in some instances it has been observed that failures of the redundant system are related to the test itself and not an

i ation that the system would have failed should it have been needed. Operability of these systems can be shown by checking records to verify that valve lineups, electrical lineups and instrumentation requirements have not been changed since the last time the system was verified to be operable.

The current Standard Technical Specifications (STS) and more specifically all the technical specifications approved for recently licensed BWR's accept the philosophy of system operability based on satisfactory performance of monthly, quarterly, refueling interval, post maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing). The staff reviewed CECo's December 21, 1988 submittal and requested additional information primarily to confirm that the testing requirements for the redundant systems or subsystems contained in the existing Technical Specifications, as modified by the proposed amendments, were consistent with the requirements contained in the Standard Technical Specifications. In Attachment 2 to CECo's May 4, 1989 submittal, a comparison between the Dresden Technical Specifications and the Standard Technical Specifications was provided. The staff has reviewed this submittal and determined the proposed Technical Specifications for Dresden are consistent with the Standard Technical Specifications and those of recently licensed BWR's with regard to the testing requirements for redundant systems.

On this basis, the fact that testing of the redundant system creates the risk of the second system failing and past operational experience, the staff has determined that the revised testing requirements for the ECCS and SGTS systems and subsystems are acceptable.

In addition, other changes to Section 3.5 of the Technical Specifications have been proposed which are administrative in nature. Since these changes either clarify present requirements or promote consistency in location of requirements within the Technical Specifications (i.e. relocating all diesel generator operability requirements in one section of the Technical Specifications), the staff finds them acceptable.

During the review, a need to revise a footnote in Table 4.5.1, which waived the applicability of Specification 4.0.D and would have permitted the plant to enter into the Startup/Hot Standby Mode provided the required surveillances were successfully completed with 12 hours after reactor steam pressure is adequate to perform the test, was identified by the staff. The wording of the footnote presumed prior approval of Section 4.0.D which is also part of the Dresden Technical Specification improvement program but has not yet been submitted. CECo's May 4, 1989 submittal eliminated any reference to Section 4.0.D and included an additional footnote pertaining to entry into the Run Mode which is the same as that required for entry into the Startup/Hot Standby Mode. However, to assure that reactor operation does not continue during startup when the HPCI system testing requirements contained in Table 4.5.1 cannot be met, a proposed action statement 3.5.C.2.b has been added. The staff recognizes that some systems cannot be tested until the plant operational mode has been entered and therefore an exception to the normal Technical Specification surveillance requirements is needed for a limited time to permit the testing. These types of exceptions have been granted in the past and the staff finds them acceptable.

B. HPCI Operability Requirements

The present Technical Specification Sections 3.5.C/4.5.C require the HPCI subsystems to be operable whenever the reactor pressure is greater than 90 psig. If the HPCI is inoperable and cannot be restored within the time limits of Section 3.5.C, then the plant must be shut down and reactor pressure reduced to 90 psig. However, this present LCO requirement of 90 psig for operability of HPCI is not based on HPCI subsystem design or testing requirements. The present Surveillance Requirement in Section 4.5.C.1 requires HPCI subsystem testing to demonstrate that HPCI can deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 to 150 psig. Since the HPCI system is designed to pump 5600 gpm into the reactor vessel within a reactor pressure range of about 1120 psig to 150 psig, the operability of the HPCI system cannot be tested at 90 psig in accordance with the current Technical Specification requirements (at pressures below 150 psig it is estimated that the flow decreases linearly to zero at 50 psig). In addition, one of the HPCI automatic isolation signals is low steam line pressure (less than 100 psig). Since the HPCI system is isolated below a steam line pressure 100 psig, the present LCO requirement of 90 psig for operability is impractical.

CECo has proposed changing the HPCI operability requirement to 150 psig to support system design flow and pressure requirements of Section 4.5.C.1 of the Technical Specifications and to provide an adequate margin to the present setpoint for system automatic isolation on low steam line pressure. The staff has reviewed this proposed change and determined it is acceptable since it corrects inconsistencies in the current Technical Specifications related to HPCI operability requirements and does not result in a decrease in safety.

CECo has also proposed to change the Surveillance Requirements in Section 4.5.C.1 to include the HPCI testing requirements (Table 4.5.1) rather than provide a reference to these requirements in the Core Spray and LPCI subsystem testing (Section 4.5.A.1). To be consistent with the standard Technical Specifications and current BWR industry practice, CECo has added a second low reactor steam pressure flow rate test to the HPCI pump flow rate testing. This second test requirement is also identified in Table 4.5.1. A test is performed every 3 months to demonstrate HPCI operability when steam is being supplied to the turbine at rated reactor pressure. The added second low pressure test is performed approximately every 18 months to demonstrate ECCS design flow when steam is being supplied to the turbine at low pressure. This proposed low pressure test will be run at a pump discharge pressure of 50 psig over reactor pressure when steam is being supplied to the turbine at 300 psig. The 350 psig upper allowable limit for testing was selected to conform with the approximate reactor pressure corresponding to the shutoff head of the low pressure coolant injection pump.

The staff has reviewed these proposed changes and determined that both the administrative changes and the additional low pressure HPCI operability test are improvements over the existing Technical Specifications and are, therefore, acceptable.

C. Automatic Pressure Relief and Isolation Condenser Operability Requirements

The present Technical Specification Sections 3.5.D (Automatic Pressure Relief) and 3.5.E (Isolation Condenser) require their respective systems to be operable whenever the reactor pressure is greater than 90 psig. CECo has proposed a Technical Specification change that would not require the Automatic Pressure Relief and the Isolation Condenser to be operable until the reactor pressure is greater than 150 psig. These changes have been proposed to preserve the consistency between the Technical Specifications for the HPCI, Automatic Depressurization System and the Isolation Condenser. Although the operability requirement is being increased from 90 to 150 psig, sufficient overlap with the low pressure systems to assure adequate core cooling will still be provided since the injection interlock for the low pressure systems is set between 300 to 350 psig. On this basis and to provide consistency between the operability requirements for these systems, the staff has concluded the proposed changes are acceptable.

D. Standby Gas Treatment System (SGTS)

The proposed changes to the SGTS Section of the Technical Specifications (Sections 3.7.B and 4.7.B) in addition to the elimination of the testing of the redundant train discussed in Section A of this Safety Evaluation are: replacing the word "circuits" with the word "subsystems;" deletion of outdated requirements for special tests in Section 4.7.B.4; and changing the test frequency for performing Surveillance Requirements 4.7.B.2a and 4.7.B.2.b.

The first two proposed changes are administrative in nature and are acceptable. The word change is editorial. The special tests are no longer required because the equipment modifications needed to allow verification of the system performance requirements are complete. The frequency of performing Surveillance requirements is presently stated as "once per operating cycle but not to exceed 18 months." The not to exceed 18 months requirement excludes allowances for use of the allowable standard accepted interval extensions permitted for other systems in the Technical Specifications (Definition CC). The proposed change would use the Terminology "or every 18 months whichever occurs first" which would permit the use of these interval extensions. The staff has reviewed this proposed change and, since it is consistent with current standard acceptable practices, finds it acceptable.

E. Secondary Containment Integrity Requirements

The proposed changes to Technical Specification Section 3.7.C on Secondary Containment integrity are: inclusion of a time frame for restoration of Secondary Containment Integrity; clarification of Definition Z on Secondary Containment Integrity; elimination of completed preoperational and first cycle operating tests and a one-time exemption which was used in 1979; and the relocation of core spray and LPCI subsystem operational requirements to Specification 3.5.A.

The first proposed change will allow 4 hours to restore Secondary Containment integrity and, if not restored, an orderly shutdown is required to at least hot

shutdown within the next 12 hours and to cold shutdown within the following 24 hours. The staff has determined that these times are consistent with those of other operating nuclear plants including those that have been recently licensed and that operating experience has demonstrated these times support safe operation. The proposed orderly reactor shutdown is also consistent with the requirements of present Specification 3.0.A. The staff therefore finds this proposed change acceptable. The remaining three changes are administrative in nature and are acceptable.

F. Additional Proposed Changes in Supplemental Submittal

In CECo's May 4, 1989 submittal, two additional changes were proposed. One change, related to the Containment Cooling Service Water (CCSW) system, would add a surveillance requirement to verify that each manual, power operated or automatic valve in the flow path that is not locked, sealed or otherwise secure must be verified to be in its correct position. Since this proposed change is the same as one of the requirements to demonstrate operability of the ECCS contained in the STS and is a safety enhancement, the staff finds this acceptable. The second change, which is purely administrative adds the words "not used" next to Section 3/4.5.G and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to surveillance and operability requirements for ECCS equipment located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9). Pursuant to 10 CFR Part 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: Byron L. Siegel

Dated: August 10, 1989

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated May 25, 1989, as supplemented January 25, 1991, Commonwealth Edison Company (CECo) proposed to amend Appendix A of the Quad Cities Facility Operating Licenses, DPR-29 and DPR-30. These amendments revise certain Limiting Conditions for Operation (LCO) and surveillance requirements associated with the High Pressure Core Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The changes are consistent with similar Technical Specifications (TSs) approved for more recently licensed BWRs and the BWR Standard Technical Specifications (STS). The January 25, 1991 letter provided an additional surveillance requirement that did not significantly alter the proposed action or change the initial proposed no significant hazards consideration determination published in the Federal Register on August 9, 1989.

2.0 EVALUATION

The current Quad Cities, Units 1 and 2, TSs require that other Emergency Core Cooling System (ECCS) subsystems be demonstrated to be operable whenever HPCI or RCIC are inoperable. The main purpose of the proposed amendments is to remove the requirement to demonstrate operability while still maintaining adequate assurance of system operability.

The requirement for demonstrating operability of other ECCS subsystems was originally needed because there was a lack of plant operating history and equipment failure data. However, plant operating history now shows that testing of other ECCS subsystems when one subsystem is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the other subsystem out of service for testing creates the risk of the second system also failing; in some instances, it has been observed that subsystem failures are related to the test itself and not an indication that the subsystem would have failed should it have been needed to actually

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mitigate an accident. Operability of these subsystems can be shown by checking records to verify that valve lineup, electrical lineups and instrumentation requirements have not been changed since the last time the subsystem was verified to be operable.

The current BWR STS and the TSs approved for more recently licensed BWRs accept the philosophy that testing subsystems to demonstrate operability is not required when another subsystem is inoperable (except for diesel generator testing). Instead, operability is based on satisfactory performance of monthly, quarterly, refueling interval, post-maintenance or other specified performance tests.

Therefore, based on the risk of the other subsystem failing, past operational experience, and the similarity to the BWR STS and other BWR TSs, we conclude that it is acceptable to eliminate the requirements to test other ECCS subsystems when either HPCI or RCIC is inoperable. (A similar change was approved for HPCI for Dresden on August 10, 1989. Dresden does not have a RCIC system.)

The proposed amendments also increase the required reactor pressure for HPCI operability from 90 psig to 150 psig. Since the HPCI system is designed to pump 5000 gpm into the reactor vessel within a reactor pressure range of about 1150 psig to 150 psig, the operability of the HPCI system cannot be tested at 90 psig (at pressures below 150 psig). Since the HPCI system is isolated below a steam line pressure of 100 psig, the present LCO requirement of 90 psig for operability is impractical. Therefore, because this change corrects inconsistencies in the current TSs and does not decrease safety, we find the increase in HPCI operability from 90 psig to 150 psig to be acceptable. (The same change was approved for Dresden on August 10, 1989.)

The amendments also increase the allowable outage time for HPCI and RCIC from 7 days to 14 days. This is consistent with the BWR STS and more recently licensed BWRs and reflects the availability of low and high pressure core cooling systems for mitigating an accident.

The amendments delete the requirement for HPCI and RCIC to be operable prior to reactor startup. These systems cannot be considered operable until reactor pressure is adequate for system operation. Thus, we find this change acceptable. (This is consistent with the BWR STS, more recently licensed BWRs, and the Dresden TSs for HPCI.)

In reviewing the licensee's May 25, 1989 amendment request, we compared the proposal to the BWR STS, NUREG-0123, Revision 3. It is our policy that when a licensee wishes to adopt provisions of the STS for a particular subsystem or TS section (in this case, HPCI and RCIC) then the licensee must adopt all provisions of the STS for that subsystem or section unless there is technical justification for not doing so. We noted that the licensee's May 25, 1989 submittal had not included the STS requirement to vent from the high point every month to verify that the system piping is filled with water. We informed the licensee of this omission and, by

letter dated January 25, 1991, the licensee proposed the appropriate TS changes to include this provision.

Therefore, we find the amendment request submitted on May 25, 1989, as supplemented January 25, 1991, to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of this amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: L. Olshan

Date: March 8, 1991