Docket
TSC3
50-237/249

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

January 28, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS TO RESOLVE AN UNREVIEWED SAFETY QUESTION RELATED TO EMERGENCY CORE COOLING SYSTEM SUCTION STRAINER PRESSURE DROP (TAC NO. M97696)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 152 to Facility Operating License No. DPR-19 and Amendment No. 147 to Facility Operating License No. DPR-25 for Dresden, Units 2 and 3. The amendments are in response to your application dated January 13, 1997, as resubmitted January 17, 1997, and supplemented January 22, 1997.

By letter dated January 15, 1997, the NRC staff requested additional information (RAI) regarding the January 13, 1997, application. In addition, the staff indicated that the proposed application was not submitted under oath or affirmation as required by 10 CFR 50.30(b) and requested the licensee to resubmit the entire application under oath or affirmation including the response to the RAI. By letter dated January 17, 1997, the licensee resubmitted the license amendments in accordance with 10 CFR 50.30(b) under oath or affirmation and included the response to the RAI. By letter dated January 22, 1997, at the request of the NRC staff, the licensee submitted additional information which supplemented the January 17, 1997, application.

The proposed amendments would: evaluate the Unreviewed Safety Question (USQ) associated with the operation of Dresden, Units 2 and 3, with the recently discovered error in the head loss across the Emergency Core Cooling System (ECCS) suction strainers; change the Technical Specification (TS) values by lowering the allowable water temperature in the suppression chamber and ultimate heat sink; change the basis of the TS to allow credit for two psig of containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA); and add a license condition to allow the licensee to change the Updated Final Safety Analysis Report to reflect the use of two psig of containment pressure to compensate for the deficiency in NPSH during the first 10 minutes following a DBA.

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The licensee requested that these amendments be processed on an emergency basis. The emergency exists in that failure of the Commission to act in a timely manner would result in the prevention of the resumption of operation of Dresden, Unit 3. The licensee was unable to make a more timely application because of the recent discovery of the calculation error on December 20, 1996. A prompt 10 CFR 50.59 evaluation by the licensee of the error resulted in the discovery of the USQ. In accordance with the NRC Inspection Manual Chapter 9900, the licensee made the decision that resumption of operation of Dresden, Unit 3, could not take place until the resolution of the USQ by the NRC staff. The NRC elected to publish the proposed no significant hazards consideration determination associated with the license amendment in the local newspaper to provide reasonable notice to the public in the area surrounding the Dresden Nuclear Power Station.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance of these amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 152 to DPR-19
2. Amendment No. 147 to DPR-25
3. Safety Evaluation

cc w/encl: see next page

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I. Johnson
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 13, 1997, resubmitted January 17, 1997 and supplemented January 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license* amendment and paragraphs 2.C.(2) and 2.C.(6) of Facility Operating License No. DPR-19 are hereby amended to read as follows:

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*License pages 3 and 3a are provided, for convenience, for the composite license to reflect this change.

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(6) By Amendment No. 152, the license is amended to allow the licensee to change the Updated Final Safety Analysis Report to allow credit for two psig containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License pages 3 and 3a
2. Changes to the Technical Specifications

Date of Issuance: January 28, 1997

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated February 2, 1983, September 28, 1987, July 6, 1989, and August 15, 1989.

In addition, the facility has been granted certain exemptions from Sections II and III of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to the licensee in a letter dated June 25, 1982.

(5) ComEd, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2527 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

(4) The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

(5) The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

(6) By Amendment No. 152, the license is amended to allow the licensee to change the Updated Final Safety Analysis Report to allow credit for two psig containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA).

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 13, 1997, resubmitted January 17, 1997 and supplemented January 22, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license* amendment and paragraphs 3.B. and 3.O. of Facility Operating License No. DPR-25 are hereby amended to read as follows:

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*License page 6 is provided, for convenience, for the composite license to reflect this change.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 147, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

O. By Amendment No. 147, the license is amended to allow the licensee to change the Updated Final Safety Analysis Report to allow credit for two psig containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License page 6
2. Changes to the Technical Specifications

Date of Issuance: January 28, 1997

L. Deleted. [Amdt. 87, 7-24-86]

M. Deleted. [Amdt. 85, 12-12-85]

N. By Amendment No. 144, the license is amended to allow, on a one time temporary basis, operation of Dresden, Unit 3, with the corner room structural steel members in the Low Pressure Coolant Injection Corner Rooms outside the Updated Final Safety Analysis Report (UFSAR) design parameters. Operation under these conditions is allowed up to and including the next scheduled refueling outage (D3R14).

The repairs to Dresden, Unit 3, corner room structural steel shall restore the steel design margins to the current UFSAR (updated through Revision 1A) design criteria. The design of the modifications to the Dresden, Unit 3, corner room structural steel members will be based on use of elastic section modules and the structural steel stresses will be limited to 1.6 of the American Institute of Steel Construction (AISC allowables). The modifications to Dresden, Unit 3, corner room structural steel will be implemented during the upcoming D3R14 refueling outage.

During this interim period of operation, should vibratory ground motion exceeding the UFSAR Operating Basis Earthquake (OBE) design parameters, Dresden, Unit 3, will be shut down for inspection and will not start up without prior NRC approval.

O. By Amendment No. 147, the license is amended to allow the licensee to change the Updated Final Safety Analysis Report to allow credit for two psig containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA).

4. This license is effective as of the date of issuance and shall expire at Mid-night January 12, 2001.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed By:

Peter A. Morris, Director
Division of Licensing

Enclosures:
Appendix A - Technical Specifications

Date of Issuance: January 12, 1971

ATTACHMENT TO LICENSE AMENDMENT NOS. 152 AND 147

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4.7-16	3/4.7-16
3/4.7-17	3/4.7-17
B 3/4.7-6	B 3/4.7-6
3/4.8-5	3/4.8-5

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 6.5" and 14' 10.5",

2. A suppression pool maximum average water temperature of ≤75°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a. ≤85°F during testing which adds heat to the suppression pool.

   b. ≤100°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

   c. ≤110°F with the main steam line isolation valves closed following a scram.

3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤75°F, except:

   a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤85°F.

   b. At least once per hour when suppression pool average water temperature is ≥ 75°F, by verifying:

      1) Suppression pool average water temperature to be ≤ 100°F, and

      2) THERMAL POWER to be ≤ 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 75°F for more than 24 hours.

   c. At least once per 30 minutes with the main steam isolation valves closed following a scram and suppression pool average water temperature >75°F, by verifying suppression pool average water temperature to be ≤110°F.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >75°F, except as permitted above, restore the average temperature to ≤75°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >85°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤75°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >100°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >110°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown to the suppression chamber with an initial water temperature of 95°F results in a water temperature of approximately 145°F. This peak temperature is low enough to provide complete condensation via T-quencher devices. However, a maximum average suppression pool temperature of 75°F and approximately 2 psi of containment pressure is required to assure adequate net positive suction pressure for the ECCS pumps during the first 10 minutes following certain analyzed accidents. No positive containment pressure is required to assure adequate net positive suction pressure for the ECCS pumps after the first 10 minutes.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the low pressure coolant injection (LPCI)/containment cooling system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the LPCI/containment cooling system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3.8 - LIMITING CONDITIONS FOR OPERATION

C. Ultimate Heat Sink

The ultimate heat sink shall be OPERABLE with:

1. A minimum water level at or above elevation 500 ft Mean Sea Level, and

2. An average water temperature of ≤75°F.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, 5 and *.

ACTION:

With the requirements of the above specification not satisfied:

1. In OPERATIONAL MODE(s) 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. In OPERATIONAL MODE(s) 4 or 5 declare the diesel generator cooling water system inoperable and take the ACTION required by Specification 3.8.B.

3. In OPERATIONAL MODE *, declare the diesel generator cooling water system inoperable and take the ACTION required by Specification 3.8.B. The provisions of Specification 3.0.C are not applicable.

4.8 - SURVEILLANCE REQUIREMENTS

C. Ultimate Heat Sink

The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential to drain the reactor vessel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated January 13, 1997, as resubmitted January 17, 1997, and supplemented January 22, 1997, Commonwealth Edison Company (ComEd, the licensee) submitted a license amendment requesting review and approval to allow credit for two psig of containment pressure to compensate for a slight increase in the amount of Net Positive Suction Head (NPSH) deficiency during the first 10 minutes following a design basis accident (DBA). The proposed amendment also requested changes to the Technical Specifications (TS) to lower the allowable water temperature in the suppression chamber and ultimate heat sink (UHS). In addition, the licensee also proposed to change the basis of the TS to reflect the allowance of containment pressure to compensate for the deficiency in NPSH.

On December 20, 1996, the licensee discovered an error in a calculation concerning NPSH for the Emergency Core Cooling System (ECCS) pumps. The calculation specified an actual 5.8 foot of head loss across the ECCS suction strainers. The Updated Final Safety Analysis Report (UFSAR) and original installation drawings identified a 1-foot head loss across the strainers. As a result of this discrepancy, the licensee performed a prompt 10 CFR 50.59 evaluation and found that the error in the calculation resulted in an Unreviewed Safety Question (USQ). In accordance with 10 CFR 50.59 and pursuant to 10 CFR 50.90, the licensee requested a license amendment to evaluate the USQ and approve associated TS changes and TS basis changes.

2.0 BACKGROUND

Dresden Station's original design basis as identified in the UFSAR and on vendor drawings included a 1-foot head loss across the ECCS suction strainers located in the suppression pool. This pressure drop is utilized in the calculations which demonstrate that adequate NPSH is available to support the operation of the ECCS pumps during DBA conditions. The design basis for the ECCS has been under review by the licensee. The licensee determined that the 1-foot head loss drop across the suction screen which was previously utilized is not representative of the actual pressure drop which could exist.

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As a part of the design basis review, the licensee has concluded that the original design basis of Dresden Station assumed an elevated pressure in the containment following a postulated DBA. Many similar vintage Boiling Water Reactors (BWR) were constructed with ECCS designs which utilize ECCS pumps and pump locations which do not provide as much NPSH margin as later designs. Dresden is an early vintage plant and the design does not include the additional margin which is available in later designs.

The assumption of an elevated post-accident pressure in the suppression chamber was not fully credited in the Dresden, Units 2 and 3, licensing basis, although a limited discussion is included in the UFSAR, Section 6.3.3.4.3. This section of the UFSAR describes an analysis performed to verify the NPSH available for the ECCS pumps. The description of the analysis indicates that for at least one of the analyzed cases, the presence of a two psig pressure in the drywell is adequate to offset the calculated deficiency in the available NPSH. This implies that the over pressure is a required design basis assumption of the facility.

However, the design and licensing basis for the Dresden Station also contains a number of statements which indicate that the facility does not require containment pressure to assure adequate NPSH is available to the ECCS pumps, including the TS basis. The licensee has concluded that these discrepancies and inconsistencies, when taken together, do not support a clear basis for assuming the availability of the two psig pressure following a postulated DBA. Following the discovery of the calculation error, the licensee performed a prompt 10 CFR 50.59 evaluation on the change in head loss across the ECCS suction strainers and discovered a USQ existed. In summary, the UFSAR states that two psig of containment pressure will make up for the 3 feet of NPSH deficiency to prevent ECCS pump cavitation or ECCS pump cavitation will occur. The new analyses indicates that even with two psig of pressure, limited cavitation and reduced ECCS pump flow will occur. This is the reason the licensee concluded that the error in the calculation resulted in a USQ.

The licensee has performed calculations which include the increased head loss across the ECCS suction strainers. The calculations indicated that to regain NPSH margin, the initial accident analysis assumptions regarding the UHS and suppression pool average water temperature must be reduced. The current TS, in Limiting Condition for Operation (LCO) Sections 3.7.K and 3.8.C, limit these water temperatures to less than or equal to 95 degrees Fahrenheit. The licensee has concluded that these temperatures should be limited to less than or equal to 75 degrees Fahrenheit to assure that the DBA analyses results are consistent with the existing licensing basis.

3.0 EVALUATION

3.1 Evaluation of the USQ

The proposed amendment requested review of the USQ to allow credit for a nominal amount of containment pressure following a DBA during the short-term accident injection phases, and would revise the TS, and the TS Bases.

3.1.1 Credit for Containment Overpressure

The crediting of post-accident containment pressure to satisfy the NPSH requirements for the Low Pressure Core Injection (LPCI) and Core Spray (CS) pumps at Dresden, Units 2 and 3, is not explicitly addressed in the licensing Safety Evaluation Reports (SER) for the plants. However, the design basis of Dresden, Units 2 and 3, as described in the UFSAR (Reference 1), identifies the accident conditions and pump configuration for which the NPSH available to the LPCI and CS pumps is minimized. These worst-case NPSH conditions occur for a double-ended break of a recirculation pump suction line with three LPCI pumps operating and pumping directly to the break, and two CS pumps providing flow to the core. This scenario results in an NPSH deficit of 3 feet for the LPCI pumps. The UFSAR states that while the presence of two pounds per square inch gauge (psig) in the drywell following a double-ended recirculation line break would offset the 3 foot deficiency, vendor-supplied tests for the residual heat removal (RHR) pumps run at flow rates of 4000-6000 gpm showed no significant effect on pump internals after an hour of pump operation with a 3 foot deficit in NPSH. The UFSAR did not explicitly credit containment overpressure to satisfy NPSH requirements.

The staff did explicitly address the issue of NPSH and containment overpressure for the LPCI pumps at Dresden and Quad Cities in an SER dated January 4, 1977 (Reference 2). The staff's SER addressed an NPSH analysis corresponding to the case of a break in the recirculation pump suction piping (i.e., the aforementioned DBA as described in the UFSAR) and also addressed pump test data which demonstrated that for a 10 minute period following the accident, LPCI pump damage due to cavitation would not occur. Although the Dresden UFSAR states that two psig overpressure would be expected in containment following a loss-of-coolant accident (LOCA), the staff's SER stated explicitly that no credit was given for containment overpressure to satisfy LPCI pump NPSH requirements. Rather, the staff found a limited amount of LPCI pump cavitation for a short period of time acceptable based on pump test data.

As a result of the higher head loss across the ECCS suction strainers than was previously thought to exist, the licensee has determined that two psig containment overpressure is required to meet the NPSH requirements of the LPCI and CS pumps. In their analysis of peak suppression pool temperature, the licensee has also found it necessary to assume a lower initial temperature in the suppression pool as stated in the UFSAR. Because containment overpressure is not credited in Dresden's current licensing basis, the licensee has determined that the need for containment overpressure constitutes an unresolved safety question. By crediting two psig of containment overpressure and by lowering the initial suppression pool temperature, the licensee has determined that the worst-case NPSH deficit for the LPCI pumps 10 minutes following an accident is approximately 3.3 feet; a .3 foot NPSH deficit increase over what is stated in the UFSAR. For the CS pumps, the NPSH deficit is approximately 9.5 feet. The NPSH deficit for the CS pumps had not been previously analyzed. The worst-case NPSH scenario for which this deficit occurs corresponds to a double-ended break in the recirculation pump suction

line with four LPCI pumps pumping to the broken line and two CS pumps providing flow to the core. This combination includes one more LPCI pump than the worst-case NPSH scenario described in the UFSAR.

The licensee has, therefore, proposed a lower initial suppression pool temperature to be used in their suppression pool temperature analysis and has proposed the crediting of two psig of containment overpressure to meet LPCI and CS pump NPSH requirements. It should be noted that the licensee's current request is different than the situation approved in 1977 in one respect. In 1977, two psig would have been necessary to avoid cavitation of the LPCI pumps, whereas now, limited cavitation would still be expected to occur for the LPCI and the CS pumps when two psig of overpressure is credited. The acceptability of the limited cavitation resulting from the LPCI and CS pump NPSH deficit has been found acceptable by the staff, and is discussed in Section 3.1.2 of this evaluation.

With regard to the proposal to credit two psig of containment overpressure for the first 10 minutes following a DBA, the licensee has presented in the application a comparison of Dresden to Quad Cities. Like Dresden, Units 2 and 3, Quad Cities, Units 1 and 2, are General Electric Company (GE) BWR/3 designs with Mark 1 containments, and are considered "sister" plants to the Dresden units. Given this similarity in plant designs, the licensee has relied upon the explicit crediting of overpressure in the staff's original licensing SER for Quad Cities to technically justify the request for the use of two psig overpressure at Dresden. Specifically, the staff's licensing SER for Quad Cities states, relative to the LPCI pumps, that "a few psi is needed for about 8 hours following a design-basis loss-of-coolant accident," (Reference 3).

To support the applicability to Dresden of the overpressure credited for Quad Cities, the licensee submitted a comparison between the two plants of the key design and operating parameters which affect the containment pressure response. The licensee's comparison is as follows:

Parameter	Dresden	Quad Cities
Core licensed power	2527 MWt	2511 MWt
Drywell free volume	158,236 cu. ft.	158,236 cu. ft.
Wetwell free volume	120,097 cu. ft.	119,963 cu. ft.
Wetwell water volume	112,000 cu. ft.	111,500 cu. ft.
Total downcomer area	301.6 sq. ft.	301.6 sq. ft.
Vent system path loss coefficient	5.17	5.17
Vacuum breaker flow area	18.84 sq. ft.	18.85 sq. ft.
Vacuum breaker full open pressure	0.5 psid	0.5 psid
LPCI/RHR pump flow rate	4500 gpm rated	4500 gpm rated
CS pump flow rate	4500 gpm rated	4500 gpm rated

CCSW/RHRSW pump flow	3500 gpm/pump	3500 gpm/pump
LPCI/RHR HX original design condition	105 MBTU at 10,700 gpm LPCI/7000 gpm CCSW 165 degrees F 95 degrees F service water	105 MBTU at 10,700 gpm RHR/7000 gpm RHRSW 165 degrees F 95 degrees F service water
Long-term limiting case pump combinations	1 LPCI/2 CCSW	1 RHR/1 RHRSW

In its comparison of the parameters between the two plants, the licensee stated that the parameters of containment mass, volume, and non-condensable mass are virtually identical between Dresden and Quad Cities and that any differences that are present are less than one percent. Where differences exist, the licensee concluded that no significant difference in containment pressure response is anticipated. The licensee further stated that the recirculation suction piping that defines the worst-case (from an NPSH perspective) DBA is the same size at Dresden and Quad Cities. Based on the similarity of the design features and operating parameters that affect the containment pressure response, the licensee concluded that the containment overpressure credited in the Quad Cities licensing SER would also be present at Dresden following a DBA.

The staff has reviewed the licensee's comparison of the Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2. With respect to core licensed power, the power at Dresden is higher than at Quad Cities, but by less than one percent. While the effect of a higher power would be to add more energy to the suppression pool, thereby decreasing the NPSH available, the containment pressure could also be expected to increase marginally, creating a higher overpressure. Since one effect would tend to offset the other, the net effect would be expected to be negligible, given the small difference in power. The staff, therefore, finds that no significant difference in containment pressure would be expected as a result of these differences.

The staff reviewed the licensee's comparison of the containment volume and downcomer/vent system. The ratio of air in the drywell to air in the wetwell air space determines the compression of the wetwell airspace, thereby affecting the pressure of both the drywell and wetwell air space and consequently the available overpressure for NPSH. The downcomer flow area and loss coefficients, and the vacuum breaker flow area and setpoints, affect, respectively, the flow characteristics of the drywell atmosphere into the wetwell and the purging of non-condensables from the wetwell airspace back to the drywell. Consequently, the overall containment pressure response would be affected. The staff notes that the aforementioned parameters are the same for Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2, and that any differences are so small as to be negligible. The staff, therefore, finds that no significant differences in containment pressure response due to differences in these parameters would be expected.

The staff also considered differences in wetwell water volume. The volume of water in the wetwell affects the post-LOCA temperature rise of the wetwell and the water height above the LPCI and CS pump suction, both of which affect the NPSH available. While the difference in wetwell volume between the two plants is small, the slightly larger wetwell volume at Dresden would be expected to produce a smaller suppression pool temperature rise and a higher column of water above the pump suction, both of which would increase the NPSH available. The staff finds that this small difference would not be anticipated to cause a significant difference in the pressure response between the two plants and further finds that if there were a noticeable effect on the pressure response, it would tend to increase the NPSH available.

The most significant differences between Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2, are the long-term suppression pool cooling pump configuration and heat exchanger performance. Because the Containment Cooling Service Water (CCSW) pumps at Dresden require less power to operate than the analogous RHR Service Water (RHRSW) pumps at Quad Cities, both of the two installed CCSW pumps at Dresden are assumed to operate for long-term (i.e., after 10 minutes) suppression pool cooling versus only one RHRSW pump at Quad Cities. Additionally, recent calculations performed by the licensee indicate that the heat transfer rate across the LPCI heat exchanger (used for suppression pool cooling) at Dresden is 98.6 MBTU/hr, versus 105 MBTU/hr originally assumed at Dresden and currently assumed at Quad Cities. However, because suppression pool cooling is manually initiated at Dresden after 10 minutes, any differences would not occur in the 10 minute period for which the licensee is requesting two psig containment overpressure credit. The staff, therefore, finds that the differences in suppression pool cooling pump and heat exchanger performance do not constitute a difference between the Quad Cities and Dresden plant designs for the 10 minute period under consideration.

Finally, the staff also reviewed the peak pressure response given in the Dresden, Units 2 and 3, UFSAR. This pressure, which was calculated for the purpose of containment loading and leakage considerations, indicates that a peak pressure of 20-35 psig would be present in the first 10 minutes following a LOCA. Although this pressure is a peak value and, therefore, does not correspond to the minimum pressure calculation which would typically be conducted to determine the most conservative overpressure that would exist for NPSH purposes, the 20-35 psig that would be present does provide a useful reference point from which to deduce that a pressure of two psig would likely be present in the first 10 minutes following a LOCA.

Based on the similarities in the key design features and operating parameters between Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2, the staff finds that the results of the analysis conducted for Quad Cities are applicable to Dresden, Units 2 and 3, for the purpose of crediting two psig containment overpressure to help satisfy LPCI and CS pump NPSH requirements.

3.1.2 LPCI and CS Pump Cavitation

The staff performed a review of the licensee's determination of the capacity of the LPCI and CS pumps while performing under cavitating flow conditions due to inadequate NPSH. Proper engineering design of pumping systems, such as these ECCS systems, would normally require that the pumps never be expected to operate under inadequate NPSH conditions associated with any normal operating or accident conditions. Therefore, the staff review encompassed several considerations to obtain assurance that the licensee could reasonably expect these pumps to reliably deliver adequate flow to meet ECCS requirements under the postulated cavitating flow conditions.

The licensee's calculation indicated that the cavitating CS pump flow under limiting conditions would be 5300 gpm. It should be noted that the licensee has also calculated that the expected CS pump flow would be 5800 gpm for the limiting accident conditions and under adequate NPSH conditions (i.e., with no cavitation occurring). The licensee provided test information supplied by the CS and LPCI pump manufacturer (Bingham Pump Company) for a pump representative of both the CS and LPCI pumps. The pump was tested under cavitating conditions for periods of time significantly greater than the required 10 minute period during which the pump would be required to operate while cavitating. No damage resulted from operating the pump during these tests.

The licensee took the manufacturer's test data for pump tests at various flows and NPSH conditions. These tests identified the points of maximum cavitation for the pump flow and NPSH conditions. The test data brackets the flow conditions that would occur for the limiting accident conditions at Dresden. The staff and its contractor (Oak Ridge National Laboratory) reviewed the test data and the reduced NPSH required curve and found them acceptable. The licensee determined the pump flow from the reduced NPSH required curve which would correspond to the available NPSH of 31 feet. The licensee, thus, estimated the cavitating pump flow to be 5300 gpm under the limiting accident conditions.

The staff was able to perform, with assistance provided by personnel at the Oak Ridge National Laboratory, a confirmatory calculation which verifies the licensee's calculation of pump discharge using the manufacturer's pump cavitation test information, the available NPSH at the CS pumps, and the licensee's CS system hydraulic resistance characteristic. A locus of reduced head and capacity points were determined for the CS pump curve for the cavitating conditions, thus, allowing the staff to estimate the resulting flow to be 5300 gpm. The staff considered the following relevant information in reviewing the licensee's estimate and in performing the confirmatory calculation.

1. The pump manufacturer's pump test information may have been performed with deaerated water since the guidance published by the Hydraulic Institute recommends such conditions for performing pump flow tests. However, the actual plant suppression pool water is expected to contain a significant amount of dissolved gas, since the pool is exposed to

either air or nitrogen. The effects of this dissolved gas could include a beneficial effect of actually reducing the pump impeller wear during cavitation because some of the voiding inside the pump would not collapse, but could also further reduce the average fluid density which would also reduce pump capacity. However, on balance, neither of these effects are considered to significantly affect the estimated pump flow.

2. The possibility of suppression pool temperatures higher than those assumed would result in higher fluid vapor pressures which would reduce available NPSH. However, the licensee has conservatively considered the increase in water temperature which would occur during the limiting accident and is limiting the initial suppression pool temperature to 75 degrees Fahrenheit to assure that the temperature effect on NPSH available does not exceed that assumed in the calculation.

3. The various system pumps represented by the manufacturer's cavitation flow tests have slightly different impeller diameters. However, based on knowledge of the range of impeller sizes and the fact that the suction eyes of all of the impellers are identical, the staff believes that the test data provides a conservative representation of all of the pumps being addressed. Also, the staff believes that minor variation in the rotative speeds of the various pumps could have some effect on individual pump NPSH requirements and cavitation behaviors, but these effects are expected to be relatively small.

4. The manufacturer's test loop piping for performing the cavitation flow tests may have been less prone to inducing cavitation (i.e., by having straightening vanes and no fittings close to the test pump inlet) than the plant piping configuration where the pumps are required to operate. However, the staff believes this consideration is offset by the conservative method in which the licensee performed the suction friction loss calculations which assumed complete blockage of the single most hydraulically limiting suction strainer rather than assuming a smaller and equal blockage of all strainers. Also, since the LPCI and CS pumps share some of the suction header piping, it is expected that since the LPCI pumps would also be cavitating at the assumed runout condition, the actual LPCI runout flow would be reduced. Therefore, the resulting NPSH available to the CS pumps would increase some amount greater than that assumed.

5. The assumed frictional characteristics of the suction strainer and piping may vary from that assumed. The staff performed a sensitivity analysis in the vicinity of the calculated cavitating CS pump flow of 5300 gpm associated with an available NPSH of 31 feet. The result indicates that for 1 foot less available NPSH, the pump flow would drop by 125 gpm, which is a significant change. However, because the staff agrees that the licensee has made a conservative estimate of the available NPSH, the staff believes there is reasonable assurance that the calculated flow of 5300 gpm is an acceptable estimate.

Therefore, the staff agrees that the licensee has performed an adequate assessment of the reduced CS pump flow which could be conservatively expected for the NPSH available during the limiting accident condition.

3.1.3 LPCI and CS NPSH Calculations

The licensee provided evaluations of post-LOCA NPSH for CS and LPCI pumps. The evaluations were divided into two portions as follows:

Short-Term: 0 to 600 seconds (10 minutes), no operator action credited, vessel injection phase, peak clad temperature (PCT) reached at 200 seconds (3.33 minutes)

Long-Term: 600 seconds to completion of event, operator actions credited, containment cooling phase

Section 6.2.1.3.3 in the UFSAR established the 600 second mark for operator action and the time at which credit for manual initiation of containment cooling can be taken. Therefore, for the long-term case, operator action is credited at the 600 second mark.

3.1.3.1 Short-Term NPSH Requirements

The bounding NPSH case for LPCI and CS pumps for short-term evaluation was determined to be four LPCI and two CS pumps running, with all LPCI pumps injecting into a broken reactor recirculation suction loop as a result of the LPCI loop select logic failure. Only CS flow is injecting into the reactor. This event was described in Generic Electric (GE) Service Information Letter (SIL) 151 (Reference 4) which postulates a failure of the LPCI Loop Select logic. This SIL primarily focused on the potential for loss of long-term containment cooling due to damage to the LPCI pumps under single failure assumptions. The concern was that operation in cavitation conditions could cause loss of the LPCI pumps and subsequent loss of the containment heat removal function. The licensee evaluated this event in 1976 with a known strainer head loss of 1 foot per 10,000 gpm. The evaluation concluded that a 3 foot NPSH deficit existed for the LPCI pumps. The staff found this limited amount of LPCI pump cavitation for a short period of time acceptable, based on pump test data, and documented this in a letter to the licensee dated January 4, 1977 (Reference 2).

Currently, the known head loss across the clean strainers is 5.8 feet at 10,000 gpm. With the same bounding event, a minimum CS system flow of 10,552 gpm (5276 gpm per pump) is required for the first 200 seconds post-accident to ensure the PCT remains below 2200 degrees Fahrenheit under the current licensing basis. The licensee has requested that the current licensing basis be changed to account for the increased head loss. To accomplish this, the licensee requested credit for two psig overpressure and cavitation of the LPCI and CS pumps for the first 10 minutes post-LOCA in order to ensure sufficient flow to the core from the CS pumps. The use of two psig containment overpressure, as discussed in Section 3.1.1 of this

evaluation, is acceptable. Based on this information, the following assumptions were made:

1. LPCI and CS pump friction losses were developed using clean, commercial steel pipe, and were increased by 15 percent to account for the effects of aging.

2. One of the four torus strainers was assumed to be 100 percent blocked while the others remained clean. This is consistent with Dresden's current licensing basis. The strainer closest to the break was assumed blocked. The licensee stated that blocking the strainer closest to the break provided more conservatism than blocking one strainer further from the break.

3. A suppression chamber pressure of two psig was assumed to exist. As discussed above, the containment analysis has shown that a suppression chamber pressure of at least two psig will be present during the first 600 seconds post-accident.

4. The initial suppression pool temperature is assumed to be 75 degrees Fahrenheit, per TS 3.7.K.2 which is discussed later. The corresponding suppression pool temperatures at 200 and 600 seconds are 129 degrees Fahrenheit and 132 degrees Fahrenheit, respectively. This is based on representative temperature profiles for Quad Cities as shown in Quad Cities UFSAR Figure 6.2-18.

5. The maximum LPCI and CS flow were assumed to be 5150 gpm (20,600 gpm total) and 5800 gpm (11,600 gpm total), respectively at the beginning of the event. This corresponds to NPSH Required (NPSHR) of 31.5 feet and 38.5 feet for LPCI and CS based on the manufacturer's pump curve.

6. The minimum suppression pool level, including drawdown of 2.1 feet, was used. This resulted in a static head of 13.3 feet.

Based on the above assumptions, the licensee evaluated the minimum suppression pool pressure (i.e., containment pressure) required for pump protection, assuming NPSH Available (NPSHA) was equal to NPSHR using the following equation.

$$P_{t,\min} = \frac{(NPSHR - Z + h_{total})}{144xV} + P_V$$

where:

P_t = suppression pool pressure in psia
P_V = saturation pressure in psia
V = specific volume in ft^3/lb
h_{total} = head loss across strainer in feet plus suction friction losses in feet
Z = static head of water above pump inlet in feet

The licensee's analysis showed that with all six ECCS pumps running and two psig minimum suppression pool pressure, a deficit of 2.9 feet for LPCI and 9.1 feet for CS existed at the 200 seconds mark (129 degrees Fahrenheit suppression pool temperature) and a deficit of 3.3 feet for LPCI and 9.5 feet for CS at 600 seconds. These results are based on maximum flow conditions with adequate NPSH. Since the NPSHR is greater than the NPSHA, the LPCI and CS pumps could cavitate, resulting in reduced flows.

As stated before, the PCT occurs at the 200 second mark and CS flow of at least 5276 gpm is limiting at this point. Using the manufacturer's pump curves which represent the point at which a three percent reduction in pump developed head has occurred, the CS flow at the point where NPSHR is equal to NPSHA is calculated, via linear interpolation of the licensee data, to be 5151.5 gpm. This is below the limiting CS flow for the 200 second mark. However, the licensee stated that cavitation tests performed on the pump model by the vendor at various flow rates indicates that the pump remains stable when NPSH is reduced several feet below the manufacturer's pump curve allowing for three percent reduction in pump developed head.

The licensee developed a reduced NPSHR curve that represents the point at which full cavitation has been achieved. This was based on the manufacturer's pump cavitation test. As discussed in Section 3.1.2 of this evaluation, the use of the reduced NPSHR curve is acceptable.

Using the reduced NPSHR curve, maximum flow conditions of 5800 gpm per pump, and two psig minimum suppression pool pressure, a deficit of 6.3 feet for CS existed at the 200 seconds mark (129 degrees Fahrenheit suppression pool temperature). This deficit could cause the CS pumps to cavitate, resulting in reduced flows. The point at which the NPSHA is equal to NPSHR (i.e., no deficit exists), as depicted on the reduced NPSHR curve, yields a CS flow of approximately 5333.3 gpm at the 200 second mark. This corresponds to an available NPSH of approximately 31 feet.

As described above, the limiting CS flow for a PCT of 2163 degrees Fahrenheit was calculated to be 5276 gpm per pump at the 200 second mark. The short term NPSH scenario predicted that a CS flow of 5333.3 gpm per pump would be achieved. Since the PCT will decrease after the 200 second mark the CS flow and NPSHA at the 600 second mark are bounded by this PCT analysis. The staff notes that margin is accounted for in this calculation based on the following:

1. The limiting CS flow of 5276 gpm per pump is for a PCT of 2163 degrees Fahrenheit which is lower than the allowable PCT of 2200 degrees Fahrenheit.

2. The PCT evaluation is being performed on a recirculation suction piping break basis. The licensee stated that discharge piping breaks are less limiting than the suction side breaks due to more restrictive blowdown flowpath; however, the licensee considers that only the discharge piping breaks are a concern in the runout flow conditions. Using the bounding case of suction piping breaks, it is anticipated that the bounding case

yields approximately 100 degrees Fahrenheit greater PCT than the discharge piping case.

3. The licensee used LPCI and CS pump friction losses developed based on clean, commercial steel pipe, and increased by 15 percent to account for the effects of aging.

4. The strainer closest to the break was assumed to be completely blocked. The blocking of the strainer closest to the break provides more conservatism than blocking one strainer further from the break.

Based on the above analysis, the staff concludes that with two psig of containment overpressure and some pump cavitation, NPSH for the ECCS pumps will be available to meet the short-term worst case scenario. The staff concludes that there is reasonable assurance that plant operation in this manner poses no undue risk to the health and safety of the public.

The staff issued NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," (Reference 5) identifying that the buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers is highly likely to occur, creating the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. The staff has requested that all BWR licensees implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. NRC Bulletin 96-03 also requested all licensee's to implement these actions by the end of the first refueling outage starting after January 1, 1997.

This timeframe for implementation was considered appropriate by the staff based on recent cleaning of suppression pools, operator training and appropriate emergency operating procedures, alternate water sources, and a low probability of the initiating event. In the case of Dresden, consideration of pump cavitation in conjunction with containment overpressure of two psig restores the ECCS capability to meet the requirements of 10 CFR 50.46(a)(1)(i) with the original licensing basis. The staff notes that this conclusion is based on the licensee's analysis of only one strainer completely blocked and does not take into account the potential for additional blockage as identified in NRC Bulletin 96-03. Appropriate corrective actions, if any, resulting from the licensee's evaluation of NRC Bulletin 96-03 will be implemented in accordance with 10 CFR Part 50 Appendix B. This action will resolve the staff's outstanding questions relative to ECCS performance and will provide long-term assurance that the requirements of 10 CFR 50.46 are met. The resolution of NRC Bulletin 96-03 will be addressed under separate cover.

3.1.3.2 Long-Term NPSH Requirements

The bounding NPSH case for LPCI and CS pumps for long-term evaluation was determined to be a DBA LOCA with atmospheric pressure in the torus. The evaluation is performed at 600 seconds following the accident at peak

suppression pool temperature. The effects of throttled LPCI pumps and reduced peak suppression pool temperature (160 degrees Fahrenheit versus 170 degrees Fahrenheit) were examined.

Currently, the known head loss across the clean strainers is 5.8 feet at 10,000 gpm. Under the same bounding event, the licensee evaluated the long-term NPSH for LPCI and CS crediting operator actions and accounting for the new head loss.

Based on this information, the following assumptions were made:

1. LPCI and CS pump friction losses were developed using clean, commercial steel pipe and were increased by 15 percent to account for the effects of aging.

2. One of the four torus strainers was assumed to be 100 percent blocked while the others remained clean. This is consistent with Dresden's current licensing basis. The strainer closest to the break was assumed block. The licensee stated that blocking the strainer closest to the break provided more conservatism than blocking one strainer further from the break.

3. Operator action will be taken at the 600 second mark to reduce LPCI and CS to their nominal rated flows of 5000 gpm and 4500 gpm, respectively.

4. The peak suppression pool temperature post-LOCA is not provided in the original Dresden UFSAR for any LPCI/CCSW combinations. However, a value of 170 degrees Fahrenheit was estimated for the one LPCI / two CCSW case based on representative temperature profiles for Quad Cities and Dresden.

5. The minimum suppression pool level, including drawdown of 2.1 feet and a recovery of 1.1 foot, was used. This resulted in a static head of 14.4 feet.

Based on the above assumptions, it was shown that reduced peak suppression pool temperature was only needed for the one LPCI pump and two CS pumps running case. In all other cases, operator actions to further reduce LPCI flow was sufficient to maintain the long-term NPSH requirements. The reduced peak suppression pool temperature, as described in the TS amendment below, will benefit all of the long-term pump combinations.

3.1.4 Conclusion

Based on the above evaluation, the staff finds that use of two psig containment pressure to compensate for a slight increase in the amount of NPSH deficiency during the first 10 minutes following a DBA is acceptable. In addition, the staff finds it acceptable for the licensee to change the UFSAR to reflect these conditions.

3.2 Technical Specification Changes

Based on its review and findings as discussed in sections 3.1.1, 3.1.2 and 3.1.3 of the SE, the staff concludes that it is acceptable for Dresden, Units 2 and 3, to credit two psig of containment overpressure to help satisfy LPCI and CS pump NPSH requirements for the first 10 minutes following a LOCA. The staff further notes that with two psig of overpressure, limited pump cavitation will still occur, but finds this limited cavitation acceptable based on the staff's 1977 SER and the above evaluation.

In order to limit the temperature rise of the suppression pool following a design-basis LOCA, the licensee has proposed to change the TS which governs the maximum suppression pool temperature and maximum service water temperature. Changes to the following TS LCO, Surveillance Requirements (SR), and Action statements related to suppression pool and service water temperature have been proposed:

Current TS LCO 3.7.K.2 specifies a maximum suppression pool temperature of 95 degrees Fahrenheit during Operational Modes 1 and 2. The licensee has proposed lowering this temperature to 75 degrees Fahrenheit.

Current TS LCO 3.7.K.2.a specifies a maximum suppression pool temperature of 105 degrees Fahrenheit during testing which adds heat to the suppression pool. The licensee has proposed lowering this temperature to 85 degrees Fahrenheit.

Current TS LCO 3.7.K.2.b specifies a maximum suppression pool temperature of 110 degrees Fahrenheit during operation at a power less than or equal to one percent of rated thermal power. The licensee has proposed lowering this temperature to 100 degrees Fahrenheit.

Current TS LCO 3.7.K.2.c specifies a maximum suppression pool temperature of 120 degrees Fahrenheit with the main steam isolation valves closed following a plant trip. The licensee has proposed lowering this temperature to 110 degrees Fahrenheit.

Current TS LCO 3.8.C.2 specifies an average UHS water temperature of 95 degrees Fahrenheit. The licensee has proposed to change this value to 75 degrees Fahrenheit.

Any changes to the SRs and Action statements that correspond to the LCOs being changed are to make them consistent with the LCOs. The actual actions and SR and/or frequencies will not be changed.

The licensee's submittal indicates that by lowering the maximum TS allowable values of these temperatures, the initial suppression pool temperature assumed in the licensee's suppression pool temperature analysis is effectively controlled, and the predicted suppression pool temperature following a LOCA is lowered. The licensee submitted calculations which indicate that for an initial suppression pool temperature of 75 degrees Fahrenheit, versus the current TS value of 95 degrees Fahrenheit, and a maximum service water

temperature of 75 degrees Fahrenheit versus the current TS value of 95 degrees Fahrenheit, the maximum suppression pool temperature rise following a double-ended recirculation suction line break would be limited to a maximum of 160 degrees Fahrenheit in the long-term, versus the current peak value of 170 degrees Fahrenheit. With a lower post-LOCA suppression pool temperature and credit for two psig of containment overpressure, an NPSH deficit of approximately 3.3 feet and 9.5 feet, will exist for the LPCI pumps and CS pumps, respectively, for the first 10 minutes, as discussed in the previous sections.

The technical basis for the suppression pool maximum temperature limit is to ensure that the suppression pool is capable of absorbing the energy of a LOCA blowdown such that containment design limits are not exceeded and the water supplied to the ECCS from the suppression pool does not exceed a temperature above which adequate core cooling would not be possible. Similarly, since the UHS water temperature (i.e., service water temperature) cools the suppression pool by removing heat via the LPCI heat exchanger, the service water temperature must be low enough to ensure sufficient heat transfer across the LPCI heat exchanger. The staff points out that the TS limits are maximums, and that any lowering of the suppression pool or service water temperatures, excluding lowering to those temperatures to which freezing would occur, will tend to assist the suppression pool pressure suppression function by creating a more effective heat sink. With respect to ECCS performance, lower temperature water supplied to the ECCS would tend to enhance ECCS performance. Because the changes to the maximum suppression pool and service water temperatures only tend to increase the margin of safety inherent in the current TS temperature limits, the staff finds these proposed TS changes acceptable for the purpose of ensuring lower post-LOCA suppression pool temperature rise.

Based on the above, the staff finds that the proposed changes to decrease the TS maximum suppression pool and service water temperatures from 95 degrees Fahrenheit to 75 degrees Fahrenheit are acceptable for the purpose of ensuring lower post-LOCA suppression pool temperature rise and, thus, ensuring adequate NPSH to the LPCI and CS pumps. However, because 75 degrees Fahrenheit maximum service water and suppression pool temperature limits will become difficult to achieve in 6-8 weeks due to warmer outdoor temperatures, the staff expects the licensee to submit a more thorough NPSH and containment pressure analysis for review. The staff expects that this analysis will be conducted explicitly for Dresden, Units 2 and 3, and will address the need for increased overpressure as a result of higher expected service water temperatures. Finally, the staff expects that this analysis will be submitted 2-3 weeks from the date of issuance of this amendment has elapsed.

The Bases for TS 3.7.K.2 will also be changed to reflect that an initial suppression pool temperature and two psig containment overpressure are necessary to ensure adequate NPSH is available to the ECCS pumps for the first 10 minutes following a LOCA, and that no positive containment pressure is required to ensure adequate NPSH for the ECCS pumps after the first

10 minutes. The TS bases and the UFSAR will be consistent. Therefore, the staff finds the proposed bases change acceptable.

4.0 EXIGENT CIRCUMSTANCES

On December 20, 1996, the licensee discovered that a calculation that had been performed by a vendor in 1983 was in error. The 1983 calculation was identified during design reviews in support of the installation of new ECCS Suction Strainers (resulting from actions which are being taken in response to NRC Bulletin 96-03). The calculation was identified as a reference in another design document. The 1983 calculation was prepared to assess the structural adequacy of the strainers as part of the Mark I containment program. The calculation specified an actual 5.8 foot head loss across the ECCS Suction Strainers. The UFSAR and original installation drawings identify a 1-foot head loss across the strainers. The 1983 calculation was not turned over to the licensee and could only be accessed through the vendor. The calculation is identified in the Primary Containment Design Basis Document in reference to the structural adequacy of the ECCS Suction Strainers. The licensee was not aware of the discrepancy between the 1983 calculation and the UFSAR, nor its impact on ECCS NPSH until December 20, 1996.

The exigency exists in that time does not permit the Commission to publish a Federal Register Notice allowing 30 days for prior public comment without preventing the resumption of operation of Dresden, Unit 3. The licensee was unable to make a more timely application because of the recent discovery of the calculation error on December 20, 1996. A prompt 10 CFR 50.59 evaluation by the licensee of the change in the facility due to the error resulted in the discovery of the USQ. In accordance with the NRC Inspection Manual Chapter 9900, the licensee made the decision that resumption of operation of Dresden, Unit 3, could not take place until the resolution of the USQ by the NRC staff. In accordance with 10 CFR 50.59 and pursuant to 10 CFR 50.90, the licensee submitted license amendments and requested the NRC staff's review and approval of the USQ and associated TS changes. The staff finds that the exigent situation occurred without prior indication and that the licensee has used its best effort to make timely application.

Dresden, Unit 3, is currently ready to return to service after a forced outage and approval of this amendment is required prior to resumption of power operation. Accordingly, the Commission has determined that exigent circumstances exist pursuant to 10 CFR 50.91(a)(6) and that the licensee did not create the exigency.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration of operation of the facility in accordance with the proposed amendment would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant hazards consideration because operation of Dresden, Units 2 and 3, in accordance with the proposed changes would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

The proposed changes to the TS limits on suppression pool and UHS average water temperature are required to assure that the safety analyses assumptions regarding containment function following a DBA remain representative of the facility. Therefore, the consequences of accidents previously evaluated are not affected by the proposed change. The proposed changes to the average water temperature limits do not affect the probability of an accident previously evaluated because these water temperature limits have not been identified as causes or contributors to any previously evaluated DBA.

In addition, the license amendment will allow the plant safety analyses to credit nominal containment pressure in its determination of the adequacy of NPSH for the ECCS pumps. The consequences of previously analyzed accidents are not significantly affected by this proposed license amendment. Containment pressure is described in UFSAR Section 6.3.3.4.3 for an evaluation of the adequacy of the NPSH available to the ECCS pumps during DBA conditions. The amendment requests clarification that two psig of containment pressure is an assumption utilized in the design basis safety evaluations applicable to Dresden. This change will be implemented by changes to the applicable Technical Specifications Bases and the UFSAR which clarify the inconsistencies with Section 6.3.3.4.3 of the UFSAR.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden Station; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. No modes of operation are introduced by the proposed changes such that adverse consequences are observed for Dresden Station.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed license amendment for Dresden Station does not create the possibility of a new or different kind of accident previously evaluated for Dresden Station. No new modes of operation are introduced by the proposed changes. This change merely restricts the average water temperatures of the

suppression pool and the UHS, and resolves discrepancies regarding use of two psig of containment pressure as an input assumption for facility safety analyses. Resolution of the USQ only allows the licensee to take full credit for the original plant design basis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety because:

The proposed license amendment does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes will preserve the existing margin of safety.

The proposed changes and subsequent revised analytical assumptions and calculation results demonstrate that adequate containment heat removal remains available and that ECCS pump NPSH availability is maintained. The proposed changes maintain existing levels of system and component reliability and do not involve a significant reduction in the margin of safety. Finally, the proposed license amendment for Dresden Station will not reduce the availability of systems required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Accordingly, the Commission has made a final determination that the amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 and change surveillance requirements. The NRC 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 made a final no significant hazards consideration determination with respect to these amendments. Accordingly, the amendments meet 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: January 28, 1997

9.0 REFERENCES

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2. D. L. Ziemann, USNRC to R. L. Bolger, Commonwealth Edison Company, "Evaluation of the Potential for Low Pressure Coolant Injection (LPCI) Pump Damage Due to Operation in Excess of Design Flow During a Postulated Loss of Coolant Accident (LOCA)," January 4, 1977.

3. Quad Cities Licensing Safety Evaluation Report, page 9, August 25, 1971.

4. General Electric Service Information Letter (SIL) 151.

5. NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers By Debris in Boiling-Water Reactors," May 6, 1996.