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ComEd

March 28, 1997

JSPLTR: #97-0064

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attention: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3 Supplemental Response to NRC Generic Letter (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"
NRC Docket Numbers 50-237 and 50-249

Reference: (a) J. Hosmer Letter to USNRC, ComEd Response to Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS" dated January 28, 1997

This letter provides a supplemental response to NRC Generic Letter 96-06 for Dresden Units 2 and 3 as requested in a telephone conversation between Dresden and NRC personnel on March 6, 1997. The purpose is to provide additional details on the work done to evaluate isolable piping sections for potential thermal overpressurization, rationale for operability evaluations, and plans and status of work on implementing hardware changes. At this time we are aggressively pursuing various modifications for 7 out of 11 Dresden Unit 3 penetrations as described in the attachment. The other Dresden Unit 3 penetrations have a different plan as described in the attachment. However, because of the material availability and outage windows we cannot yet confirm if these aggressive modifications can be completed in D3R14. Remaining modifications will be completed in D2R15 and D3R15. The additional information is provided in the attachment to this letter.

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If there are any questions concerning this letter, please refer them to Frank Spangenberg, Dresden Station Regulatory Assurance Manager, at (815) 942-2920, extension 3800.

Sincerely,

for

J. Stephen Perry
Site Vice President
Dresden Station

Subscribed and Sworn to before me
on this 28 day of
March, 1997.
Sherry L. Butterfield
Notary Public

OFFICIAL SEAL
SHERRY L BUTTERFIELD
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:05/23/00

cc: A. Bill Beach, Regional Administrator, Region III
P. L. Hiland, Branch Chief, Division of Reactor Projects, Region III
J. F. Stang, Project Manager, NRR (Units2/3)
Senior Resident Inspector, Dresden Station

ATTACHMENT
Dresden GL 96-06 Supplemental Response

NRC REQUEST:

Within 120 days of the date of this generic letter, addressees are requested to submit a written summary report stating actions taken in response to the requested actions noted above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment, the basis for continued operability of affected systems and components as applicable, and corrective actions that were implemented or are planned to be implemented. If systems were found to be susceptible to the conditions that are discussed in this generic letter, identify the systems affected and describe the specific circumstances involved.

NRC REQUESTED ACTIONS:

"Addressees are requested to determine:

(1) if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions:

(2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

The Dresden response to the water hammer and two phase flow issues dated January 28, 1997 explained why the Dresden Reactor Building Closed Cooling Water (RBCCW) piping inside drywell will not develop any significant voids due to the post accident drywell environment and the water hammer and two phase flow issues identified in NRC GL 96-06 are not significant for Dresden Units 2 and 3. In subsequent telephone conversations on March 6, 1997 the NRC restated their concerns related to reestablishing RBCCW flow to the drywell following an accident.

The RBCCW containment isolation valves do not automatically close due to any isolation signal. During and after an accident, the RBCCW system continues to operate and supply water to the non-safety related cooling needs in the drywell. However, following an accident RBCCW may be manually isolated via controls in the control room in response to a RBCCW pump trip or low RBCCW expansion tank level. All applicable post accident operating procedures will be reviewed and updated, as needed, to include a warning on the potential for water hammer in the RBCCW system if and when water flow to the drywell is reinitiated after a LOCA.

ATTACHMENT
Dresden GL 96-06 Supplemental Response

RESPONSE TO ITEM 2:

A summary of the review performed to address the post LOCA thermally induced pressurization issue identified in NRC GL 96-06 was included in the Dresden response dated January 28, 1997. The systems reviewed, scope of the review, susceptible piping sections, basis of operability, long term resolutions under consideration and the implementation schedule are discussed below:

A. Thermally-induced over pressurization is applicable to liquid-filled piping systems which penetrate containment. The following systems were reviewed:

    Core Spray Piping System
    Control Rod Drive Hydraulic Piping System
    Demineralized Water Piping System
    HP Coolant Injection Piping System
    Isolation Condenser Piping System
    L.P. Coolant Injection Piping System
    Reactor Building Equipment Drain System
    Reactor Feed Piping System
    Recirculation Piping System
    Reactor Water Clean Up Piping System
    Shut Down Reactor Cooling Piping System
    Standby Liquid Control Piping System
    Reactor Building Cooling Water Piping System

B. A review was performed to identify the piping sections for which such pressurization could jeopardize the ability of accident mitigating systems to perform their safety functions or could lead to breach of containment or bypass leakage. The portions of the systems that are not susceptible to thermally induced pressurization due to system configuration, valve line-up, check valves, relief valves, and piping open to the reactor were excluded from further review.

C. Eight systems were found to be susceptible to thermally induced pressurization conditions discussed in Generic Letter 96-06. The penetrations and systems are as follows:

X-139B and X-144 (Unit 2), and X-139C (Unit 3)	- Control Rod Drive
X-119 (Unit 2 and 3)	- Demin Water Piping
X-109B (Unit 2) and X-109A (Unit 3)	- Isolation Condenser
X-117 and X-118 (Unit 2 and 3)	- Drywell Building Equipment Drain
X-122 (Unit 2 and 3)	- Reactor Recirc System Sample Line
X-113 (Unit 2 and 3)	- Reactor Water Cleanup
X-111A and X-111B (Unit 2 and 3)	- Shutdown Cooling
X-123 and X-124 (Unit 2 and 3)	- Reactor Building Closed Cooling Water

D. Each segment of piping system that could potentially be affected by post accident thermally induced pressurization including the specific circumstances involved, the basis for continued operability and immediate corrective actions taken are given below.

1. The isolated portion of the <u>control rod drive system (CRD)</u> pipe lines at Unit 2 penetration X-139B and the Unit 3 penetration X-139C are susceptible to thermally induced pressurization. The flow paths associated with these lines are non-safety related and the lines are not used during normal plant operation. The lines were partially drained to assure continued operability.

2. The isolated portion of the <u>control rod drive system (CRD)</u> pipe line at Unit 2 penetration X- 144 is susceptible to thermally induced pressurization. This line is non-safety related and is no longer used. The line was drained to assure continued operability.

3. The isolated portion of the <u>demineralized water system</u> line at Unit 2 and 3 penetration X-119 that supplies water to the drywell during plant outage is susceptible to post LOCA thermally induced pressurization. The flow path associated with this line is non-safety related. The piping at Unit 3 penetration was drained from inside containment prior to the drywell close-out (forced outage D3F23) to assure continued operability. Currently, Unit 2 is at full power and the line cannot be drained. The allowable design pressure of the piping at the containment penetration is three times that of the closed isolation valves. Consequently, the valves are considered to be the weak link in this isolated volume. One of the valves will develop a leak through either the

bolted bonnet assembly, stem packing or the valve disk/plug and depressurize the isolated volume. The other isolation valves will remain intact and assure containment integrity.

4. The isolation condenser system condensate return line at Unit 2 penetration X-109B and Unit 3 penetration X-109A has a normally closed isolation valve on the outside and a normally open containment isolation valve on the inside. The above penetration piping will be susceptible to thermally induced pressurization if the inboard isolation valve is closed due to a Group V isolation signal, which indicates a failure in the isolation condenser system. The isolation condenser piping at the penetration is insulated and therefore will heat up and pressurize at a significantly slower rate than uninsulated pipe. The continued operability of the above isolated piping section was based on the slower pressurization rate due to piping insulation and the measured air leakage rate from the latest local leakage rate tests.

5. The drywell floor drain (Reactor Building Equipment Drain System) sump pump discharge pipe at Unit 2 and 3 penetration X-117 is susceptible to thermally induced pressurization. The isolation valves are located outside containment. However, the isolated volume at penetration X-117 between the inboard isolation valve (located outside containment) and the check valves at the sump pump discharge is susceptible to thermally induced pressurization. Continued operability is assured by the non-safety related relief valve on connected high radiation sampling system branch lines.

6. The drywell equipment drain (Reactor Building Equipment Drain System) sump pump discharge pipe at Unit 2 and 3 penetration X-118 is susceptible to thermally induced pressurization. The isolation valves are located outside containment. However, the isolated volume at penetration X-118 between the inboard isolation valve (located outside containment) and the check valves at the sump pump discharge is susceptible to thermally induced pressurization. Continued operability is assured by the non-safety related relief valve on connected high radiation sampling system branch lines.

   There is a normally open air operated valve between the inboard isolation valve (located outside containment) and the check valves at the sump pump discharge. If the normally open air operated gate valve (AOV) is closed during the accident, the isolated volume between the AOV and the inboard isolation valve could become susceptible to thermally induced pressurization. The allowable design pressure of the piping at containment penetration is three times that of the air operated gate valve and therefore the air operated valve is the weak link in this isolated volume. One of the AOVs will develop a leak through either; the bolted bonnet assembly, stem packing, or the valve disk/plug and depressurize the isolated volume. The other isolation valves will remain intact and assure containment integrity.

7. The Reactor Recirculation system sample lines at Unit 2 and 3 penetration X-122 have normally open containment isolation valves and the lines are normally hot. However, if the lines are isolated prior to an accident, the lines could be cold and may be susceptible to thermally induced pressurization. The isolation valves are air operated. As thermally induced pressure increases between the isolation valves, the air operator spring force that keeps the valves closed will be overcome. The air operator on the outboard side of the drywell will lift, relieving the pressure buildup and subsequently reseating, assuring continued operability.

8. During normal plant operation, the reactor water clean-up (RWCU) system containment isolation valves at Unit 2 and 3 penetration X-113 are open, and the lines have continuous flow and are normally hot. However, if the line is isolated prior to the accident the containment penetration piping and the isolation valves at penetration X-113 will be susceptible to thermally induced pressurization. The RWCU piping inside drywell at penetration X-113 is insulated and will heat up and pressurize at a significantly slower rate than uninsulated pipe. The continued operability of the above isolated section is based on the slower pressurization rate due to pipe insulation and the measured air leakage rate from the latest local leakage rate tests.

9. The shutdown cooling system (SDC) at Unit 2 and 3 containment penetrations X-111A and X-111B is susceptible to thermally induced pressurization. Since the continuation piping outside containment is non-safety related, the valves are not required to operate after the accident. However, the shutdown cooling containment penetration piping and isolation valves are required to maintain structural integrity. The shutdown cooling piping inside drywell between the inboard isolation valves and the penetration are insulated and therefore will heat up and pressurize at a significantly slower rate than uninsulated pipe. The continued operability of the above isolated piping section is based on the slower pressurization rate due to insulation and the measured air leakage rate from the latest local leakage rate tests.

10. The reactor building closed cooling water (RBCCW) system supply to the containment enters the drywell through Unit 2 and 3 penetration X-123, supplies water to the non-safety related cooling needs in the containment and leaves the containment through Unit 2 and 3 penetration X-124. At the supply penetration there is a motor operated valve (MOV) on the outside and a check valve on the inside. At the return penetration there is a MOV on the inside and another MOV on the outside. All piping inside the drywell other than those between the containment isolation valves are non-safety related. The isolation valves and the safety related piping between them are required to maintain structural integrity. However, the valves are not required to open after an accident. The RBCCW containment isolation valves do not automatically close due to any isolation signal. During and after an accident the RBCCW continues to operate and supply water to the non-safety related cooling needs in the drywell. As long as the isolation valves remain open, the RBCCW piping will not be subject to thermally induced pressurization. However, following an accident the RBCCW may be manually isolated from the control room due to RBCCW pump trip or low RBCCW expansion tank level. If and when the containment isolation valves are manually closed, the RBCCW inside containment will become susceptible to thermally induced pressurization.

    (a) The non-safety related RBCCW piping inside containment form a closed volume that contains: seven drywell coolers, the Recirculation pump coolers, and the drywell equipment sump heat exchanger, that is susceptible to thermally induced pressurization. The above closed volume contains many non-safety related cooling units that are not designed for the high pressures associated with thermally induced pressurization, since the tube wall thickness is sized for heat transfer performance. The non-safety related cooling coils will develop a leak and depressurize the piping before significant pressures that challenge the integrity of the safety related inboard isolation valves at penetrations X-123 and X-124 are developed. Continued operability is assured by the above leakage through the non-safety related cooling coils.

    (b) The RBCCW containment penetration piping at the supply penetration X-123 has a motor operated valve on the outside and a check valve on the inside. The leak in the drywell cooling coils discussed above will relieve the thermally induced pressure in the non-safety related piping inside drywell. As the non-safety related closed loop depressurizes, the inboard check valve will open, relieve the pressure, and assure continued operability.

(c) The RBCCW containment penetration piping at drywell penetration X-124 has a motor operated valve on the inside and another motor operated valve on the outside. The RBCCW piping inside the drywell is not insulated and is susceptible to rapid heat-up due to the post accident hot drywell environment. The Unit 3 RBCCW piping between the penetration and the inboard isolation valve was insulated prior to (D3F23) drywell close-out and therefore will heat up and pressurize at a significantly slower rate than uninsulated pipe. The continued operability of the RBCCW return line at Unit 3 penetration is based on the slower pressurization rate due to pipe insulation and the measured air leakage rate from the latest local leakage rate tests. Unit 2 is now at power and therefore the line inside the drywell cannot be insulated. The allowable design pressure of the piping at containment penetration is approximately three times that of the valves and therefore the valves are expected to be the weak link in this isolated volume. Therefore, one of the valves is expected to develop a leak through either; the bolted bonnet assembly, stem packing, or the valve disk/plug and depressurize the isolated volume. The other isolation valves will remain intact and assure containment integrity.

E. Dresden intends to either install physical changes that provide over pressure protection, or use specific analytical considerations that clearly address and confirm that over pressurization is not a concern. Some of the detailed analysis that will support our engineering judgment requires an extensive amount of time. In order to assure that the correct resolution for each susceptible section is identified, additional time is necessary. Additionally, ComEd is actively working with EPRI, BWROG, and NEI in determining the best options with respect to resolution of the concerns identified in GL 96-06. The outcome of ongoing efforts by these industry groups will be used in determining the long term plans at Dresden Station.

The main concern associated with the hardware changes for thermal relief is their potential for interference with the normal operation of the plant. The over pressure protection options currently under consideration include: relief valves open to atmosphere, rupture disk open to atmosphere, a relief valve with a rupture disk at the valve inlet, drilling a small hole in a check valve, and upgrading existing non-safety related relief valves. The specific design selected for each application will depend on the system involved and the consequences associated with the malfunction of the over protection device.

The over pressure protection devices under consideration are classified as devices that are intended to mitigate the consequences of a service condition. Since these thermally induced pressures are caused by a LOCA, the applicable service level is the faulted condition. Consequently, the set pressures were chosen to limit the thermally induced pressures to the faulted condition allowable pressure. That is, the devices will be set to limit the maximum internal pressure to the allowable faulted condition pressure defined in the applicable codes. Based on USAS B31.1.0-1967, the Dresden faulted condition pressures will be limited to 1.2 times the design pressure. If inadvertent operation of the over pressure protection devices are found to interfere with the normal operation of the plant, Dresden may consider increasing the pressure setting of the over pressure protection devices to limit the faulted condition internal pressure to 1.5 times the design pressure as permitted under the ASME Code section III.

Following is a summary of the hardware changes under consideration for the piping sections that were identified to be susceptible to post LOCA thermally induced pressurization identified in NRC GL 96-06. Other solutions may also be considered if they are found to be suitable. The proposed hardware changes will not be finalized until the design process including the safety evaluation is completed. All Unit 2 and Unit 3 long term resolutions / hardware modifications will be completed by the refueling outages D2R15 and D3R15 respectively.

1. CRD pipe line at Unit 2 penetration X-139B and Unit 3 penetration X-139C

   The flow path associated with the line is non-safety related, the line is not used during normal plant operation and was partially drained to assure continued operability. Dresden intends to perform detailed heat transfer and stress analysis to demonstrate the adequacy of the above penetration, 3/4 inch Schedule 80 pipe and the manual isolation valves. If the analytical approach is unsuccessful, Dresden intends to install hardware changes to resolve the thermally induced pressurization issue at the next outage of sufficient duration.

2. CRD pipe line at Unit 2 penetration X-144

   The line is no longer used and it will be either blind flanged or disconnected to keep the piping at penetration X-144 empty.

3. Demineralized water system line at Unit 2 and 3 penetration X-119.

   Dresden intends to install hardware changes to provide over pressure protection at this penetration. The installation of the Unit 3 hardware modifications in D3R14, is contingent upon material availability and its impact on the D3R14 outage schedule.

4. Isolation condenser line at Unit 2 penetration X-109B and Unit 3 penetration X-109A

   Based on the initial review the piping at Unit 2 penetration X-109B and Unit 3 penetration X-109A were identified as susceptible to thermally induced pressurization, if the inboard isolation valve is closed due to a Group V isolation signal.

   The isolation condenser system condensate return line at Unit 2 penetration X-109B and Unit 3 penetration X-109A has a normally closed isolation valve on the outside and a normally open containment isolation valve on the inside. The piping inside drywell is connected to the reactor recirculation piping. Due to the piping configuration, the natural convective flow inside the pipe will keep the penetration piping hot. The temperature of the Unit 3 penetration piping outside drywell was measured through a small crack in the insulation and was found to be over 300°F. Since, the post LOCA drywell ambient temperature does not exceed 300°F, the Unit 3 penetration piping will not be susceptible to post LOCA thermally induced pressurization. Dresden intends to perform a thermal scan of the corresponding Unit 2 piping to confirm the same for Unit 2. Based on the above, no hardware changes are planned.

5. Drywell floor drain sump discharge at Unit 2 and 3 penetration X-117

   The flow path associated with this line is non-safety related and the line is required to maintain penetration integrity. Dresden intends to provide over pressure protection by either; upgrading the non-safety related relief valve on connected high radiation sampling system, installing an additional over pressure protection device, or drilling a small hole in the disk of the check valve at the pump discharge. The installation of the Unit 3 hardware modifications in D3R14 is contingent upon material availability and its impact on the D3R14 outage schedule.

6. Drywell equipment sump discharge at Unit 2 and 3 penetration X-118

   The flow path associated with this line is non-safety related and the line is required to maintain penetration integrity. Dresden intends to provide over pressure protection by either; upgrading the non-safety related relief valve on connected high radiation sampling system, installing an additional over pressure protection device, or drilling a small hole in the disk of the check valve at the pump discharge. The installation of the Unit 3 hardware modifications in D3R14 is contingent upon material availability and its impact on the D3R14 outage schedule.

7. The Reactor Recirculation system sample lines at Unit 2 and 3 penetration X-122

   Dresden intends to either install hardware changes to provide over pressure protection or make procedure revisions to ensure that one of the isolation valves is closed and disabled, and the other is opened and disabled per Technical Specification Section 3.7.D, whenever the subject line is taken out of service. The installation of the Unit 3 hardware modifications in D3R14, is contingent upon material availability and its impact on the D3R14 outage schedule.

8. The RWCU system pipe line at Unit 2 and 3 penetration X-113

   Dresden intends to either install hardware changes to provide over pressure protection or make procedure revisions to ensure that one set of the isolation valves are closed and disabled, and the other set is opened and disabled per Technical Specification Section 3.7.D, whenever the subject line is taken out of service. The installation of the Unit 3 hardware modifications in D3R14, is contingent upon material availability and its impact on the D3R14 outage schedule.

9. The SDC system pipe at Unit 2 and 3 containment penetrations X-111A/B

   The shutdown cooling piping inside drywell between the inboard isolation valves and the penetration are insulated and therefore will heat up and pressurize at a significantly slower rate than uninsulated pipe. Dresden intends to perform detailed heat transfer and stress analysis to demonstrate adequacy of the shutdown cooling piping at the above penetrations. If the analytical approach is unsuccessful, Dresden intends to install hardware changes to resolve the thermally induced pressurization issue at the next outage of sufficient duration.

10. The RBCCW lines at Unit 2 and 3 penetrations X-123 and X-124

    Dresden intends to install hardware changes to provide thermal relief for the closed volume between the inboard and outboard isolation valves at the return penetration X-124.

    Dresden also intends to install hardware changes to provide thermal relief for the closed volume between the inboard isolation valve at the supply penetration X-123 and the inboard isolation valve at the return penetration X-124.

    The installation of the Unit 3 hardware modifications in D3R14, is contingent upon material availability and its impact on the D3R14 outage schedule.

    If the portion of the RBCCW system inside drywell is manually isolated after a LOCA, post LOCA heat up will cause pressure build-up and the over pressure protection devices may open and relive a portion of the RBCCW water inventory. When the system cools down, the RBCCW system inside drywell may develop voids and reestablishing flow to the drywell may result in a water hammer. Dresden intends to review all applicable post accident operating procedures and update them, as needed, to include a warning on the potential for water hammer in the RBCCW system when the flow is reestablished to the drywell.

F. A follow-up response update will be provided by the end of May, 1997 as previously identified. At that time it is expected that we will have more answers on material availability, design package and implementation schedules for the subject changes, and further clarification from NEI, EPRI and BWROG efforts that will assure an effective and consistent approach with the rest of the nuclear industry.