U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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Results: See Executive Summary

EXECUTIVE SUMMARY

Nine Mile Point Unit 1
50-220/96-05
February 17 to March 11, 1996

The purpose of this special inspection was to review concerns identified during the review of the Nine Mile Point Unit 1 (Unit 1) Licensee Event Report (LER) 95-05, "Building Blowout Panels Outside the Design Basis Because of Construction Error," dated November 30, 1995. These concerns involved NMPC's application of their safety assessment processes, design control measures, and reportability requirements, and corrective action measures.

For the time period from initial operations (December 1969) to October 1993, a safety problem existed, in that, the reactor and turbine building blowout panels would not have relieved until a pressure in excess of the structural design pressure for the buildings stated in the Updated Final Safety Analysis Report (UFSAR). This condition was caused by an original plant construction error involving the installation of oversized blowout panel fasteners (i.e., bolting) that occurred in a period that preceded the use of quality measures required to be implemented under the requirements of 10 CFR 50, Appendix B. However, the inspection determined that the safety problem continued from October 1993 until March 1995 due to inadequate implementation of 10 CFR 50, Appendix B design control measures that resulted in a calculation error and an inadequate design review of that calculation that allowed the oversized fasteners to remain in place. This condition was identified as an apparent violation of 10 CFR 50, Appendix B, Criterion III "Design Control."

Furthermore, the inspection identified concerns with the safety assessment process that: (1) allowed a change to the facility as described in the UFSAR (i.e., leaving the oversized fasteners in place in lieu of correcting the condition) to exist in the plant for approximately an 18-month period without the conduct of a required 10 CFR 50.59 safety evaluation; and (2) upon determining that the facility was being operated with blowout panel relief capabilities in excess of the structural design value prescribed in the UFSAR, altered the design of structures described in the UFSAR (i.e., removing every other fastener) without the conduct of a 10 CFR 50.59 safety evaluation. This condition was identified as an apparent violation of 10 CFR 50.59.

Regarding reportability of events to the NRC, the inspection identified a concern with NMPC's process that resulted in two occasions (October 1993 and March 1995) where NMPC should have identified that the plant was operated outside of it's design basis, and in fact did not perform the required reporting. This failure on two occasions to perform the requisite reporting was identified as an apparent violation of 10 CFR 50.72 and 50.73. Also, the inspection identified that when NMPC ultimately reported the matter to the NRC in LER 95-05, the submitted report was weak in describing corrective actions and significance of conditions.

The inspection identified a concern involving procedural adherence, which involved the failure of NMPC to enter the design control measure inadequacies identified in March 1995 into their corrective action system (i.e., the

Deviation/Event Report). This condition is an apparent violation of the Nine Mile Point Nuclear Station Unit 1 Technical Specification 6.8.1, which requires procedures to be implemented.

The above four (4) apparent violations are being considered for escalated enforcement.

The inspection identified inconsistencies within and between the UFSAR and the Individual Plant Examination that involved the stated value of the pressure relief capabilities of the blowout panels. There were also inconsistencies within the UFSAR regarding the design basis for the blowout panels and specific high energy line breaks. The NRC Staff plans to discuss this matter further at the Enforcement Conference.

In addition, while the inspection had not identified any immediate safety concerns with the manner in which the March 1995 modification of the blowout fasteners had resolved NMPC's safety problem, the NRC staff has initiated actions to perform a confirmatory independent review of NMPC's calculations that formed the basis for this modification.

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1.0 INTRODUCTION

1.1 Purpose of Inspection

The purpose of this special inspection was to review concerns identified by the NRC staff during the review of the Nine Mile Point Unit 1 (Unit 1) Licensee Event Report (LER) 95-05, "Building Blowout Panels Outside the Design Basis Because of Construction Error," dated November 30, 1995.

1.2 Event Overview

On October 25, 1993, with the reactor at 100% power, Unit 1 engineering staff determined that the safety-related blowout panels in the reactor and turbine buildings would not blowout at the design relief pressure of 45 pounds per square foot (psf). The purpose of the blowout panels is to provide pressure relief to prevent collapse of the superstructure due to a break of an emergency cooling system, or other primary coolant system line in the reactor building, and a steam line break in the turbine building. Unit 1 found that the existing shear bolts on the blowout panels were larger than those identified on the design drawings. This was documented in a Deviation/Event Report (DER 1-93-2526). The initial engineering evaluation indicated that the turbine and reactor building blowout panels would relieve at a pressure greater than the design basis value, but less than the design basis fail pressure for the reactor and turbine buildings. The 1993 engineering evaluation recommendation and NMPC resolution was to leave the as-found condition in place.

On March 27, 1995, during the completion of the recommended actions included in the DER, an engineering review determined that the blowout panels would actually not relieve until pressures in excess of the structural design of 80 psf for both buildings. At this time, Niagara Mohawk Power Corporation (NMPC) completed a design change to bring the relief pressures back in conformance with the UFSAR. During subsequent reviews of the issue, NMPC determined that the condition was reportable under Title 10 of the Code of Federal Regulations Part 50.73, (10 CFR 50.73) "Licensee Event Report System," and issued LER 95-05.

1.3 Scope of Inspection

During this inspection, the inspectors reviewed LER 95-05, applicable Technical Specifications (TSs), Updated Final Safety Analyses Report (UFSAR) and Individual Plant Examination (IPE) sections, portions of related calculations, procedures, DERs, Station Operations Review Committee (SORC) meeting minutes and other licensee documentation. The inspectors also conducted interviews with various members of the NMPC staff and management, and conducted walkdowns of the applicable areas of the facility. The inspectors focused their review on the following aspects of the issue:

- Engineering Support/Design Control
- Reportability/LER Adequacy
- Safety Review and Assessment
- Proper Procedure Implementation

Additionally, the inspector assessed the accuracy of the applicable UFSAR sections.

2.0 DETAIL DESCRIPTION OF THE EVENT

Attachment 1 to this report is a time line of events for this review.

During NMPC's effort to resolve contradictions identified in the UFSAR regarding the blowout panel relief pressure, Calculation S7-RX340-W01, dated August 23, 1993, was generated. Due to a lack of documentation regarding the material properties of the bolts identified on the design drawings, NMPC determined to test a sample of the installed bolts to obtain actual material properties. After initial calculations were performed, a number of bolts were replaced with new 3/16" diameter American Society for Testing and Materials (ASTM) A-307 bolts. The previously installed bolts were tested to determine their strength. Upon receipt of the test results, the structural engineer identified that the bolts were 1/4", and not 3/16" as specified on plant drawings (C-18713-C). Furthermore, the test results indicated that the strength of the bolts to be higher than that used in Calculation S7-RX340-W01. Therefore, the structural engineer initiated Revision 1 to the calculation, which indicated that the relief pressure for the reactor building to be 53 psf, and 60 psf for the turbine building.

On October 28, 1993, DER 1-93-2526 was written to address the difference between the size of the installed blowout panel bolts and the size indicated on the plant drawings. When the DER was reviewed by the SSS on November 1, 1993, an operability determination was attached that indicated the relief pressures of blowout panels for both the reactor and turbine buildings would exceed the value described in the UFSAR. Because the calculated relief pressures were less than the buildings internal failure pressure, engineering recommended that the blowout panels still be considered operable. The SSS accepted this recommendation. Additionally, the SSS did not consider the condition to be reportable.

As part of DER 1-93-2526 Action Plan, Unit 1 was to complete a calculation to identify exactly which bolts were required to be replaced to restore the blowout panels in conformance with the UFSAR relief pressure of 45 psf. This calculation was scheduled to be completed by June 30, 1995. The calculation was completed on March 27, 1995, during the Unit 1 refueling outage 13. Based upon the results of this calculation, the licensee determined different relief pressures for the as-installed configurations. These new relief pressures were in excess of the fail pressure for both the reactor and turbine buildings as stated in the UFSAR.

According to the engineering supervisor, there was an error in the assumptions used during the October 1993 calculation that caused the previous incorrect results. Particularly, loading of the panels was assumed to be equally distributed in both the horizontal and vertical directions. Therefore, the engineer incorrectly concluded that a failure of the sheet metal at the top of the panels would be sufficient to relieve pressure as required. To provide sufficient pressure relief, a failure of the bolts connecting the sides of the panels to the supports would have been required. Based on the new

calculations the correct relief pressure for the reactor building was 91 psf, and 89 psf for the turbine building. No DER was written at this time to address the human performance issues associated with the design control deficiencies inherent in the calculation error and the independent design review (this area is described further in Section 3.3). During the re-disposition of the DER in March 1995, NMPC indicated that the event was not reportable under 10 CFR 50.72 or 50.73 for the following reasons:

- UFSAR Section XVI.D.2.0 states that Unit 1 was designed prior to 10 CFR 50 Appendix A, General Design Criterion 4, and that it was not designed for the dynamic effects of a double-ended guillotine pipe rupture, and that the probability of this kind of occurrence is extremely low.

- The above assumption was substantiated by a 1984 leak-before-break analysis. This analysis concluded that a full double-ended pipe break need not be postulated as a design basis for defining loads at Unit 1. The results of this study were used to define the Unit 1 design basis for masonry walls at Unit 1, and were submitted to the NRC via letter dated June 8, 1984. The leak-before-break analysis eliminates the need for the blowout panels, because the high energy line break (HELB) event would be preventable by detection of the leak, and timely shutdown would follow. Therefore, the condition was not outside the Unit 1 design bases.

During the 1995 refueling outage, Unit 1 evaluated the situation and decided to remove every other bolt used to hold each of the blowout panels in place due to the higher relief pressures. This would provide a relief pressure of approximately 45 psf as per the UFSAR. DER 1-93-2526 and associated operability determinations were updated and the blowout panels were declared inoperable, on March 27, 1995. Modification N1-95-001 LG329 was initiated, and the bolts were removed prior to plant restart from the refueling outage in 1995. The Structure Engineering Supervisor determined that because NMPC was completing the design change to place the relief pressure back in accordance with the UFSAR, no additional analyses were needed; such as analyses to determine the actual internal fail pressures of the reactor and turbine buildings, the events that would exceed these fail pressures or the subsequent consequences of exceeding these failure pressures.

During the closeout SORC review of DER 1-93-2526, on June 22, 1995, the SORC questioned the reportability of the issue, and requested that engineering re-evaluate the reportability of the events. Nuclear Engineering confirmed the bases for the earlier decision not to report the condition. This information was documented in NMPC Memorandum ESB1-S95-0039 to file, and presented to SORC on July 6, 1995.

On October 31, 1995, DER 1-95-3012 was written to prompt another evaluation of the condition for reportability. This DER was generated as a result of a non-required review of original condition performed by NMPC personnel. As a result, NMPC determined to report the condition under 10 CFR 50.73(a)(2)(ii). LER 95-05 was submitted to the NRC on November 28, 1995. According to LER 95-05, the use of engineering judgement was improperly credited in the previously

concluding that this condition was not reportable and that the UFSAR inferred that the relief panels were credited with functioning for certain events.

DER 1-95-3012 identified the failure to properly report the condition to be caused by inadequate engineering reviews, and inadequate investigation of the reportability requirements when the error was identified. Based on these causes, NMPC initiated the following corrective actions:

- Emphasis to be given to the Structural Engineering Group to perform an adequate review of documentation and verification of assumptions used in calculation before final issue. (Completed shortly after the end of Refueling Outage 13.)

- Engineering to write a lessons learned transmittal to address the responsibility of engineering staff to promptly inform operations when new information is identified that could affect reportability. (Scheduled for completion October 1996.)

- Provide a training session to certain engineering and plant personnel regarding reportability (NUREG-1022, and 10 CFR 50.72, and 50.73), and include this training in the NMPC continuous training cycle. (Scheduled for completion October 1996.)

3.0 ASSESSMENT OF THE EVENT

Based on the inspectors' review, concerns were identified in the following areas:

- Design Control;
- 50.59 safety evaluations;
- Reportability;
- Procedure compliance; and
- Root Cause and Corrective actions as described in the LER.

3.1 Design Control

The inspectors reviewed the following facets of design control:

- the root cause for the installation of the oversized bolts during construction;

- the October 1993 calculational error; and

- the corrective actions to address the calculational error.

3.1.1 Root Cause of the Installation of Oversized Bolts during Construction

The cause of the oversized bolts used to install the reactor and turbine building blowout panels was documented in LER 95-05 to be inadequate quality control measures in place during construction. The corrective action credited in the LER is additional quality control and quality assurance requirements that have been implemented for the design and construction activities since

initial construction, which should prevent similar deficiencies. NMPC reported that, Unit 1 was designed and constructed prior to the implementation of 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The inspectors acknowledged NMPC statement in this regard, but did not focus on the performance problems in the construction period. Those same quality assurance requirements referenced to by NMPC were in effect during the performance problems since 1993. The inspectors focused their review on the time period from 1993.

3.1.2 October 1993 Calculational Error

In October 1993, after the oversize bolts were identified, the engineer made an error in his assumptions leading to the determinations of the incorrect relief pressure for the safety-related reactor and turbine building blowout panels. Specifically, the assumption that the loading of the panels was equally distributed in both the horizontal and vertical directions resulted in the incorrect determination that the reactor building blowout panels would relieve at 60 psf and the turbine building blowout panels would relieve at 53 psf. This error was not caught by the checker or the approver of the calculation as part of design review. As determined in March 1995, the actual relief pressures would be 91 psf for the reactor building and 89 psf for the turbine building. According to the UFSAR, the reactor building and turbine building blowout panels were designed to relieve at 45 psf to prevent failure of the building superstructures at pressures in excess of 80 psf.

This is an apparent violation of 10 CFR 50 Appendix B Criterion III, "Design Control," in that Calculation S7-RX340-W01 incorrectly determined the blowout panel relief pressures to be less than the 80 psf failure pressure of the reactor and turbine buildings, and the calculation was inadequately design reviewed.

3.1.3 Corrective Actions to Address the Calculational Error

The inspectors reviewed Modification N1-95-001 LG329, which initiated the removal of every other blowout panel bolt, and identified no immediate concerns. The inspectors verified that the applicable drawings and calculations were updated. Additionally, the inspectors also "walked down" the blowout panels for the reactor building and verified through sampling that the installed configuration was consistent with the plant drawings. However, the NRC did not complete a detail review of the calculations as of the close of the inspection period. Region I is performing a confirmatory independent review of the calculations. This is considered an unresolved item pending the completion of NRC staff's review. **(URI 50-220/96-05-01)**

3.2 10 CFR 50.59 Safety Evaluations

The inspectors evaluated the licensee's implementation of 10 CFR 50.59 "Changes, Tests and Experiments," for the relief pressures exceeding the values stated in the UFSAR identified in October 1993, and for the modification to remove every other bolt used to install the blowout panels completed in March 1995.

3.2.1 10 CFR 50.59 Safety Evaluation - October 1993

In October 1993, during NMPC's evaluation of the installed oversized bolts, NMPC determined the blowout panels to be operable. With respect to the original relief pressure of 45 psf and the new calculated relief pressures of approximately 60 psf both being less than the structural design value (80 psf), clearly the safety margin was reduced in these facts. Unit 1 decided to leave the oversized bolts installed until the completion of their corrective actions, scheduled to be completed June 30, 1995. When this decision was made, no 10 CFR 50.59 evaluation was completed. A delay or partial correction of a condition adverse to safety or quality for a structure, system, or component described in the UFSAR is considered by the NRC staff to be a change in the facility, which is subject to a 10 CFR 50.59 review. Additionally, the above facts indicated a reduced safety margin that needed to be evaluated in accordance with 10 CFR 50.59.

The inspectors considered the failure to complete a 10 CFR 50.59 evaluation, to allow for the approximately one and a half year delay in resolving the differences between the UFSAR stated design relief pressures and the installed/calculated relief pressures for the reactor and turbine building blowout panels, an apparent violation on 10 CFR 50.59.

3.2.2 10 CFR 50.59 Safety Evaluation - March 1995

During the evaluation of Modification N1-95-001 LG329, an applicability review was completed by the Unit 1 staff indicating no need for the conduct of a 10 CFR 50.59 safety evaluation. The inspectors noted that the NMPC documented basis in the applicability review for not completing a 10 CFR 50.59 safety evaluation was because the proposed change would bring the facility back onto compliance with the UFSAR. The inspectors acknowledged this basis in the applicability review, and also verified that the size and spacing of the blowout panel bolting were not described in the UFSAR. However, changes in the facility as described in the UFSAR are considered by the NRC staff (NRC Manual Chapter Part 9900) to pertain to any changes in the facility which alter the design, function, or method of performing the function of a component, system or structure described in the UFSAR. Accordingly, the NMPC Modification N1-95-001 LG329 made in March 1995, which consisted of the removal of every other blowout panel bolt, is considered by the NRC staff to be an alteration to the design of a structure described in the UFSAR. Therefore, the failure of NMPC to perform a safety evaluation for the subject modification is considered another example of an apparent violation of 10 CFR 50.59.

3.3 10 CFR 50.72 and 50.73 Reportability

The inspectors evaluated the licensee's implementation of 10 CFR 50.72 and 50.73, "Immediate notification requirements for operating nuclear power reactors," and "Licensee event report system," for the relief pressures exceeding the design values stated in the UFSAR identified in October 1993, and for the relief pressures exceeding the structural design pressures of the reactor and turbine buildings values stated in the UFSAR identified in March 1995.

The inspectors reviewed the applicable revisions of the licensee's procedure regarding reportability and determined it to provided appropriate requirements to ensure that 10 CFR 50.72 and 50.73 reportability regulations related to conditions that are outside of the design basis of the plant.

3.3.1 10 CFR 50.72 and 50.73 Reportability - October 1993

The inspectors discussed with the licensee their reasoning for not declaring the event reportable under 10 CFR 50.72, 50.73 in October 1993, and was informed, that even though the relief pressure of the blowout panels exceeded the design basis values stated in the UFSAR, the structural design basis pressure of the reactor and turbine buildings would not be exceeded. Additionally, NMPC reviewed their design basis and determined that HELBs were outside their design basis that there was no credible postulated event that would cause pressures to challenge the failure pressure of the reactor or turbine buildings. Based on these reasons, NMPC determined that they were not outside the design basis; therefore, the condition was not reportable.

The inspectors evaluated the reportability decision made by NMPC. Based on the definition of Design Basis as provided in 10 CFR 50.2, "Design bases means that information which identifies specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design...." Since the specific values chosen for controlling the relief pressures of the reactor and turbine building blowout panels were exceeded as stated in the UFSAR, the inspectors considered Unit 1 to be in a condition outside their design basis.

The inspectors concluded that the Unit 1 failure to complete the reports, as required in October 1993, is an apparent violation on 10 CFR 50.72 and 50.73.

3.3.2 10 CFR 50.72 and 50.73 Reportability - March 1995

During the period of March through July 1995, NMPC evaluated the reportability of the blowout panel relief pressures on several occasions. Through discussions with NMPC staff and management, the inspectors ascertained that NMPC understood that they were outside their design basis relief pressures of 45 psf as stated in the UFSAR. Furthermore, they understood that the calculated relief pressures exceeded the 80 psf failure pressure for the reactor and turbine buildings, as stated in UFSAR Sections VI-C.1.2 and III-A.1.2, respectively. However, NMPC decided that the condition was not outside the Unit 1 design bases, since the UFSAR Sections XVI.D.2.0 describes that HELB events are not part of the Unit 1 design basis; and, therefore, no postulated pressure-related event would challenge the reactor building blowout panels.

The inspectors evaluated the reportability decision made by NMPC for this time period. Unit 1 UFSAR Sections VI-C.1.2, "Pressure Relief Design," describes specifically, that the reactor building pressure relief is provided to prevent collapse of the superstructure due to a break of an emergency cooling system, or other primary coolant system line in the reactor building. Further, the pressure relief function is provided by the blowout panels, that were designed

to fail at an internal pressure of approximately 45 psf, to prevent excessive internal pressure on the superstructure walls, roof, and their supports, which would fail at an internal pressure in excess of 80 psf. A similar description is provided for the turbine building in UFSAR Section III-A.1.2, specifying the initiating event as a steam line break.

Since the specific values chosen for controlling the relief pressures of the reactor and turbine building blowout panels as stated in the UFSAR were exceeded, and the specific values chosen for the reactor and turbine building failure pressure as stated in the UFSAR were exceeded, the inspectors considered Unit 1 to be in a condition outside their design basis. Furthermore, the detailed description of the emergency cooling system line break and the steam line break provided in the UFSAR as the events for which the blowout panels were installed to protect against, indicated that there are analyzed pressure-related events would challenge the reactor and turbine building blowout panels.

The inspectors concluded that Unit 1 failure to complete the reports, as required in March 1995, is an apparent violation on 10 CFR 50.72 and 50.73.

3.4 Procedural Compliance

The inspectors reviewed the applicable revisions of the licensee's DER procedure to verify the completion of DER 1-93-2526 was in accordance with the procedure. Although the inspectors found the DER completed as required by the procedure, the inspectors noted that the length of time taken between the identification of the problem until the review of the SSS was longer than expected considering the potential significance of the condition. Particularly, the condition was identified on October 25, 1993, the DER was initiated on October 28, 1993, and signed by the SSS on November 1, 1993. Discussion with NMPC management indicated that the length of time taken for the condition to receive SSS review did not meet their expectations.

During the engineering review of the design control deficiencies involving the calculational error and the independent design review that allowed the oversized bolts to remain in place from October 1993 until March 1995, the inspectors identified that no DER was written in March 1995 to document and initiate root cause and corrective actions. NMPC Procedure NIP-ECA-01, "Deviation Event Report," Revision 8, requires a DER be written to address human performance and personnel performance problems adverse to quality. The failure to write a DER is an apparent violation of the Nine Mile Point Nuclear Station Unit 1 Technical Specification 6.8.1, which requires procedures to be implemented.

The inspectors noted that even though NMPC did not write a DER to address the human/personnel performance concerns related with the design control deficiencies, they did initiate some corrective actions by provided emphasis to the Structural Engineering Group on the need to perform an adequate review of documentation and verification of assumptions used in calculation before final issue. According to NMPC, this discussion was completed shortly after the completion of Refueling Outage 13, prior to the initiation of DER 1-95-3012. However, the inspectors did not consider this review of the design

control deficiencies to be a sufficient evaluation of the root cause to develop appropriate corrective actions to preclude recurrence.

4.0 REVIEW OF LER 95-05, "BUILDING BLOWOUT PANELS OUTSIDE THE DESIGN BASIS BECAUSE OF CONSTRUCTION ERROR"

The inspectors reviewed LER 95-05, and found it to accurately describe the event associated with the oversized bolts installed in the reactor and turbine building blowout panels. However, the following weaknesses were identified:

- the LER did not adequately address the failure to report the condition in October 1993, or in March 1995;
- the LER did not provide sufficient details regarding the 1993 calculational error to allow for an adequate assessment of the licensee's corrective actions to prevent recurrence; and
- the LER did not address the potential for, and the significance of a reactor building failure.

Based on these weaknesses, LER 95-05 will remain open, pending further NRC staff review.

5.0 SAFETY SIGNIFICANCE

5.1 Review of the Unit 1 Individual Plant Examination

The inspectors reviewed the applicable sections of the Unit 1 Individual Plant Examination (IPE) related to the reactor building pressure relief design. The information provided in Section 4.1, "NMP1 Containment Design Description and Data," of the IPE was consistent with that provided in UFSAR Section VI.C.1.2 "Pressure Relief Design" for the reactor building. These sections from the UFSAR and the IPE basically described that pressure relief is provided to prevent collapse of the superstructure due to a break of an emergency cooling system, or other primary coolant system line in the reactor building. The relief is provided by the blowout panels that were designed to fail with an internal pressure of approximately 45 psf. Relief of pressure through the panels in case of an energy release will prevent excessive internal pressure on the superstructure walls, roof, and their support that would fail at an internal pressure in excess of 80 psf.

The inspectors also noted that the Unit 1 IPE describes in Table 4.5-1, "Overall Level 2 Success Criteria," the reactor building integrity and effectiveness to be successful, if either one of the following two criteria are met:

1) Reactor building integrity is maintained if the reactor building pressurizes to no more than 36 psf. If not, then the reactor building blowout panels are to assumed to have opened.

2) The reactor building is assumed effective in removing radionuclides if the following criteria are met:

- No structural breach to the reactor building, allowing free communication with the environment, caused by events such as hydrogen detonation in the reactor building.

- No natural circulation paths with chimney effects are established within the reactor building that could drastically reduce residence time and retention within the reactor building.

Additionally, Section 4.1.2, "Summary of Secondary Containment Features," contains a list of the safety design basis for the secondary containment system. Included in this list as Item 5, "The reactor building is designed to contain a maximum positive internal pressure of 80 pounds per square foot (0.56 psig). Blowout panels in the refuel floor are used to release at a pressure of approximately 40 pounds per square foot (0.28 psig) to relieve internal reactor building pressure."

The information pertaining to the Unit 1 design basis as described in the IPE substantiated the information provide in the UFSAR, indicating that Unit 1 was outside their design basis for the reactor building blowout panel relief pressure. Additionally, the inspectors noted inconsistencies within the IPE with respect to the relief pressure of the blowout panels. These inconsistencies indicate blowout panel relief pressures of 36 psf, 40 psf and 45 psf. Some of these inconsistencies are similar to those originally identified in the UFSAR by NMPC. The inspectors considered the resolution of these inconsistencies an inspector follow item (IFI) to be reviewed during a future inspection. (IFI 50-220/96-05-02)

5.2 Review of the UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for additional verification that licensees were complying with UFSAR commitments. During an approximate two month time period, February through March 1996, all reactor inspections will provide additional attention to UFSAR commitments and their incorporation into plant practices, procedures and procedures.

While performing the inspection, which are discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. During discussions with NMPC personnel, the following apparent inconsistency was noted in the UFSAR between Section XVI.D.2.0 "Plant Design for Protection Against Postulated Piping Failure in High Energy Lines," and Sections VI.C.1.2, and III.A.1.2 "Pressure Relief Design" for the reactor and turbine buildings respectively:

Specifically, Section XVI.D.2.0 states that Unit 1 was design and constructed prior to 10 CFR 50 Appendix A, "General Design Criteria," (GDC) Criterion 4, and was not designed accordance with this criterion dealing with the effects of pipe whips from HELBs, vs, Section VI.C.1.2, which states that "breaks in all primary coolant systems piping has been analyzed since accidents of this type result in the highest pressure, temperature and humidity condition in the building. A break in the emergency cooling system is the most serious since

it releases the most coolant at the highest rate." Furthermore, Technical Specification Basis Section 5.4 "Containment," states, "Pressure relief is provided to prevent damage to the superstructure due to the break of any primary system line the reactor building. In this event, blowout panels will fail, relieving pressure in the event of a major line rupture."

Also Section III.A.1.2, states that "to prevent failure of the superstructure due to a steam line break, a wall area of 1800 square feet has been attached with bolts [blowout panel] will fail due to an internal pressure of approximately 45 psf; thus relieving the internal pressure."

The inspectors used the most conservative design basis for their review. These inconsistencies regarding the design basis for Unit 1 with respect to specific high energy line breaks will be discussed further during the enforcement conference pertaining to the apparent violations described in this report. Furthermore, the resolution of these inconsistencies is considered part of the inspector follow item identified in Section 5.1. (IFI 50-220/96-05-02)

Additionally, the licensee identified that there was a discrepancy between the UFSAR Sections III.A.1.2 and VI.C.1.2, which state that the pressure relief panels in the turbine and reactor buildings blow out at 45 psf. Contrary to this, UFSAR Table XVI-31 and discussion on the subsequent pages state that the pressure relief panels blow out at 40 psf. This contradiction led to Unit 1 identifying the discrepancy between the design and installed configuration. NMPC has initiated a change to the UFSAR to correct this discrepancy.

5.3 Safety Assessment and Summary

The NMPC organization has concluded that their March 1995 modification of the blowout panel bolting had resolved their safety problem. However, while the NRC has no immediate safety concerns pertaining to the supporting calculations for this modification, confirmatory independent review of these calculations is currently being performed by the NRC staff.

During this review, the inspectors noted that a general safety objective for the reactor building and the turbine building blowout panels is to relieve internal building pressure prior to structural failure for anticipated transients/challenges inside the building. The inspectors' focus was on the reactor building since it houses safety structures, systems and components. If such an event were to occur, radiation doses at the site boundary appear to be accounted for and are of minimal consequences. However, two areas remain unclear as a result of this review:

- How much design margin existed for the structural design value of 80 psf or what actual internal building pressure would result in failure (collapse) and the obvious impact on safety related equipment such as emergency core cooling systems being used in response to the anticipated transients/challenges.

- The highest pressure in the reactor building for design basis anticipated transients/challenges.

For the time period from initial operations to March 1995, a vulnerability existed, the significance of which is dependent on resolution of the above two areas. Notwithstanding the fact that no actual challenges occurred to the reactor building internal pressure relieving system, the potential existed which is not addressed in the related LER. More importantly, NMPC resolution of the issue from October 1993 to March 1995, was weak in thoroughly establishing, understanding, and evaluating the safety design basis for the reactor building internal pressure relieving system. The weak safety assessment coupled with the calculation error, inadequate design review of that calculation, and apparent failure to follow the DER procedure led to the untimely resolution of this problem commensurate with its safety significance. Although this problem was eventually reported to the NRC, the event report does not fully address the safety significance of the potential for (in distinction to actual) challenges to the internal building internal relieving system for the time period from initial operations to March 1995. As a result, licensee corrective actions do not address apparent weaknesses in their technical and safety review process.

6.0 MANAGEMENT MEETINGS

At periodic intervals and at the conclusion of the inspection period, meetings were held with senior station management to discuss the scope and findings of this inspection. The final exit meeting occurred on March 11, 1996. During the exit meeting, Richard Abbott, Vice President and General Manager, Nuclear for NMPC questioned the inspector's statements regarding NMPC basis for not reporting the condition under 10 CFR 50.73, in March 1995. Subsequent conversations between the inspectors and NMPC management, clarified NMPC basis, and was considered in this report. Based on the NRC Region I review of this report, and discussions held with NMPC representatives, it was determined that this report does not contain safeguards or proprietary information.

Attachment 2 to this report is listing key personnel contacted during this review.

ATTACHMENT 1

TIME LINE

October 20, 1993 . . .	During the removal of bolts for testing, technicians identified that 1/4" bolts used to install the blowout panels.
≈October 22-27, 1993 .	Test results were available to the structural engineer.
August 23, 1993	Calculation S7-RX340-W01, Revision 0 was approved.
October 25, 1993 . . .	The identification of reactor and turbine building blowout panel bolts stronger than required by design drawings as documented on DER 1-93-2526.
October 28, 1993 . . .	Originator signed DER 1-93-2526.
October 29, 1993 . . .	Supervisor signed DER 1-93-2526.
November 1, 1993 . . .	Station Shift Supervisor signed DER 1-93-2526.
November 5, 1993 . . .	Calculation S7-RX340-W01, Revision 1 was approved.
February 8, 1995 . . .	Refueling outage 13 began.
March 27, 1995	Calculation error was identified.
March 30, 31, 1995 . . .	Every other bolts was from the blowout panels removed in accordance with Modification N1-95-001 LG329.
April 4, 1995	Unit 1 connected to the grid following refueling outage 13.
≈April 5, 1995	Structure engineers training on the verification of assumptions.
June 22, 1995	SORC meeting discussing the safety evaluation associated with the UFSAR change to correct the originally identified UFSAR contradiction, and the need to document the basis for not reporting the condition in DER 1-93-2526.
July 6, 1995	SORC meeting approving the implementation of DER 1-93 -2526, and the issuance of NMPC memorandum documenting bases for not reporting the condition.
October 31, 1995 . . .	Origination of DER 1-95-3012, "Reportability Review of DER 1-93-2526."
November 28, 1995 . . .	SORC meeting approving the disposition of DER 1-95-3012, and LER 95-05.
November 30, 1995 . . .	NMPC Issued LER 95-05.

ATTACHMENT 2

PERSONS CONTACTED

Niagara Mohawk Power Corporation

R. Abbot, Vice President and General Manager, Nuclear
M. Alvi, Supervisor, Structural Design, Unit 1
D. Baker, Engineer, Licensing
M. Balduzzi, Operations Manager, Unit 1
C. Beckham, Manager, Quality Assurance
G. Corell, Manager, Chemistry, Unit 1
K. Dahlberg, General Manager, Projects
M. McCommick, Vice President, Nuclear Safety Assessment & Support
N. Rademacher, Plant Manager, Unit 1
K. Sweet, Manager, Technical Support, Unit 1
C. Terry, Vice President, Nuclear Engineering
G. Wierzbowski, Supervisor, Technical Support, Unit 1
G. Wilson, Counsel
D. Wolniak, Manager, Licensing
W. Yaeger, Manager, Engineering, Unit 1
A. Zallnick, Licensing Engineer

U.S. Nuclear Regulatory Commission

* R. Conte, Chief, Reactor Projects Branch (RPB) No. 5
* H. Eichenholz, Project Engineer, RPB No. 5
* M. Hartzman, Mechanical Engineering Branch, NRR
* D. Hood, Project Manager, NRR
* W. Rothman, Structural Engineering Branch, NRR
S. Sanchez, Resident Inspector

All of the above personnel were present at the exit meeting on March 11, 1996.

* Telephonic presence at the exit meeting.

Enclosure 2

DESIGN/LICENSING BASIS QUESTIONS ON NMP-1 REACTOR BUILDING BLOWOUT PANELS

1. The LER discusses use of blowout panels to protect the secondary containment structure. However, as discussed in "GE Design Specifications for Reactor Containment," blow-out panels are also sometimes used to protect primary containment from excessive reverse differential pressure that could occur in the event of a high energy line break in a compartment adjacent to the primary containment.

   Is this generic design objective applicable to the NMP-1 facility and what is the documented basis if it is?

2. With respect to page 11 of Nine Mile Point 1 License Event Report (LER), titled, "Building Blowout Panels Outside Design Basis Because of Construction Error," what are the key assumptions and engineering analysis/calculations performed in late October 1993 which led to the initial determination that the turbine and reactor building panels would blow out at 60 and 53 psf, respectively, to relieve internal pressure.

   [Provide engineering calculations which support the above stated blowout pressures.]

3. Two design internal pressures of 40 and 45 psf are shown in different locations of the FSAR for Nine Mile Point 1 pressure relief panels (PRPs) for the reactor and turbine buildings.

   Please clarify the ambiguity about the two pressure values and indicate the correct licensing basis design pressure for the PRPs?

4. Has an assessment been made on the error in the design assumptions for load distribution which was identified during the March 1995 refueling outage?

   Also explain what assumptions for load distribution were used in conjunction with the consideration of the 1/4" bolts with higher ultimate strength (78 ksi) which led to the determination of the revised panel blowout pressures of 92 and 88 psf for the reactor and turbine buildings, respectively (pages 11 and 13 of DER No. 1-93-2526).

5. With respect to the above referenced DER, provide a detailed discussion of the key assumptions, panel/bolt configurations (including pertinent drawings) and bolt ultimate strength used in concluding that the revised panels would blowout at about 45 psf with the use of the 1/4" diameter bolts and the removal of every other bolt from the existing panels.

6. The Reactor Building Blowout panel revised calculations are based on certain assumptions, the conservatism of which cannot be determined unless compared to a more rigorous analysis or test.

It is not clear how NMPC demonstrated conformance with FSAR commitments by reevaluating the panel pressure capacity using a more exact methodology, or revising the analysis and stating clearly the conservatism of each assumption used in the analysis.

The staff has identified the following effects which appear to have not been considered in the analysis or are not clearly stated:

a. The calculation of the pressure capacity of the top and bottom connections do not consider the membrane effect of the panel in the longitudinal (vertical) direction and the effect of the flexibility of the side connections between the flutes and the columns. In addition, the effect of friction between the bolts and the sheetmetal surfaces, due to bolt pre-loading, has not been considered.

b. The calculation of the pressure capacity of the side connections are based on the assumption of a simply supported beam. This is not valid if the panel is rigidly bolted to the angle members, in which case the analysis should be based on a beam with elastically built-in ends.

c. The calculation of the pressure capacity of the side connections was determined from the analysis of a typical flute, without considering the in-plane membrane forces acting on the flutes.

d. The calculation of the sheet-metal shear capacity is based on one shear area. It should be based on two shear areas.

e. The effect of the panel dead-weight on the connections has not been considered.

What are results of these effects on the calculation of the panel pressure capacities, as demonstrated by detailed calculations, or by the conservatism of the existing calculations?

Also, a weakness of the 1995 modification evaluation for the blowout panel interim corrective actions appears to be the implied assumption of imminence of failure of the 1/4 " diameter shear bolt with a computed "unity" value for the "shear-tension interaction" equation.
What is the pertinent analytical or test-supported basis for such an assumption?

[As appropriate, a more realistic, non-linear finite element analysis of the PRPs with rigorous modeling of the PRP elements including proper representation of the combined shear/tension stiffness of the bolts and the supporting steel frame may be performed to demonstrate the adequacy of the current PRP evaluation.]

8. With respect to Nine Mile Point 1 Calculation Nos. S7-RX340-W01 Revisions 0 and 1, in support of bolt strength:

a. What is the basis for selecting the revised bolt ultimate tensile strength of 78 ksi in the latest panel blowout capacity calculation?

b. What is the test verified ultimate shear strength of the same set of bolts tested?

c. Are the strengths (both the ultimate tensile and shear strengths) based on ultimate strength tests of an adequate sample size of the 1/4" bolts?

d. Is the 78 ksi a mean ultimate tensile strength?

e. What is the corresponding mean ultimate shear strength used in the assessment?

f. If they represent mean ultimate strengths, what are their corresponding standard deviations?

g. If the values represent nominal lower-bound strengths and they were used in your latest calculation which confirmed the revised blowout capacity of approximately 45 psf, how can one be sure that the panels would blowout approximately at 45 psf and not at a higher value?

h. Given the above mentioned uncertainties in ultimate tensile and shear strengths and load distribution assumptions used in the analysis, has NMPC established a conservatively determined upper-bound panel blowout pressure and demonstrated that the computed pressure capacity is lower than the 45 psf pressure stipulated in the licensing basis document?

9. With respect to the same calculations noted above, discuss the appropriateness of using conservative engineering assumptions including conservative modeling (e.g., one way horizontal action for Robertson's panels) and use of mean or non-upper-bound ultimate bolt tensile and shear capacities to determine a realistic upper-bound internal pressure which will cause failure of the PRPs.

Such an approach could underestimate the real panel blowout pressure, thus, resulting in a non-conservative conclusion. Specifically, use of a lower-bound or mean bolt ultimate tensile strength and an assumed ultimate bolt shear strength of 0.6 F_u (instead of a test verified shear strength) would lead to a unrealistic PRP failure pressure and potentially unsafe conclusion.

Discuss the safety implications of such a practice in light of the objective of the PRP evaluation and the need to modify the evaluation and demonstrate that the physical panel disposition proposed is still valid.

factors in arriving at the appropriate severity level will be dependent on the circumstances of the violation. However, if a licensee refuses to correct a minor violation within a reasonable time such that it willfully continues, the violation should be categorized at least at a Severity Level IV.

D. Violations of Reporting Requirements

The NRC expects licensees to provide complete, accurate, and timely information and reports. Accordingly, unless otherwise categorized in the Supplements, the severity level of a violation involving the failure to make a required report to the NRC will be based upon the significance of and the circumstances surrounding the matter that should have been reported. However, the severity level of an untimely report, in contrast to no report, may be reduced depending on the circumstances surrounding the matter. A licensee will not normally be cited for a failure to report a condition or event unless the licensee was actually aware of the condition or event that it failed to report. A licensee will, on the other hand, normally be cited for a failure to report a condition or event if the licensee knew of the information to be reported, but did not recognize that it was required to make a report.

V. Predecisional Enforcement Conferences

Whenever the NRC has learned of the existence of a potential violation for which escalated enforcement action appears to be warranted, or recurring nonconformance on the part of a vendor, the NRC may provide an opportunity for a predecisional enforcement conference with the licensee, vendor, or other person before taking enforcement action. The purpose of the conference is to obtain information that will assist the NRC in determining the appropriate enforcement action, such as: (1) A common understanding of facts, root causes and missed opportunities associated with the apparent violations, (2) a common understanding of corrective action taken or planned, and (3) a common understanding of the significance of issues and the need for lasting comprehensive corrective action.

If the NRC concludes that it has sufficient information to make an informed enforcement decision, a conference will not normally be held unless the licensee requests it. However, an opportunity for a conference will normally be provided before issuing an order based on a violation of the rule on Deliberate Misconduct or a civil penalty to an unlicensed person. If a conference is not held, the licensee will normally be requested to provide a written response to an inspection report, if issued, as to the licensee's views on the apparent violations and their root causes and a description of planned or implemented corrective action.

During the predecisional enforcement conference, the licensee, vendor, or other persons will be given an opportunity to provide information consistent with the purpose of the conference, including an explanation to the NRC of the immediate corrective actions (if any) that were taken following identification of the potential violation or nonconformance and the long-term comprehensive actions that were taken or will be taken to prevent recurrence. Licensees, vendors, or other persons will be told when a meeting is a predecisional enforcement conference.

A predecisional enforcement conference is a meeting between the NRC and the licensee. Conferences are normally held in the regional offices and are not normally open to public observation. However, a trial program is being conducted to open approximately 25 percent of all eligible conferences for public observation, i.e., every fourth eligible conference involving one of three categories of licensees (reactor, hospital, and other materials licensees) will be open to the public. Conferences will not normally be open to the public *if the enforcement action being* contemplated:

(1) Would be taken against an individual, or if the action, though not taken against an individual, turns on whether an individual has committed wrongdoing;

(2) Involves significant personnel failures where the NRC has requested that the individual(s) involved be present at the conference;

(3) Is based on the findings of an NRC Office of Investigations report; or

(4) Involves safeguards information, Privacy Act information, or information which could be considered proprietary;

In addition, conferences will not normally be open to the public if:

(5) The conference involves medical misadministrations or overexposures and the conference cannot be conducted without disclosing the exposed individual's name; or

(6) The conference will be conducted by telephone or the conference will be conducted at a relatively small licensee's facility.

Notwithstanding meeting any of these criteria, a conference may still be open if the conference involves issues related to an ongoing adjudicatory proceeding with one or more intervenors or where the evidentiary basis for the conference is a matter of public record, such as an adjudicatory decision by the Department of Labor. In addition, with the approval of the Executive Director for Operations, conferences will not be open to the public where good cause has been shown after balancing the benefit of the public observation against the potential impact on the agency's enforcement action in a particular case.

As soon as it is determined that a conference will be open to public observation, the NRC will notify the licensee that the conference will be open to public observation as part of the agency's trial program. Consistent with the agency's policy on open meetings, "Staff Meetings Open to Public," published September 20, 1994 (59 FR 48340), the NRC intends to announce open conferences normally at least 10 working days in advance of conferences through (1) notices posted in the Public Document Room, (2) a toll-free telephone recording at 800–952–9674, and (3) a toll-free electronic bulletin board at 800–952–9676. In addition, the NRC will also issue a press release and notify appropriate State liaison officers that a predecisional enforcement conference has been scheduled and that it is open to public observation.

The public attending open conferences under the trial program may observe but not participate in the conference. It is noted that the purpose of conducting open conferences under the trial program is not to maximize public attendance, but rather to determine whether providing the public with opportunities to be informed of NRC activities is compatible with the NRC's ability to exercise its regulatory and safety responsibilities. Therefore, members of the public will be allowed access to the NRC regional offices to attend open enforcement conferences in accordance with the "Standard Operating Procedures For Providing Security Support For NRC Hearings And Meetings," published November 1, 1991 (56 FR 56251). These procedures provide that visitors may be subject to personnel screening, that signs, banners, posters, etc., not larger than 18" be permitted, and that disruptive persons may be removed.

Members of the public attending open conferences will be reminded that (1) the apparent violations discussed at predecisional enforcement conferences are subject to further review and may be subject to change prior to any resulting enforcement action and (2) the statements of views or expressions of opinion made by NRC employees at predecisional enforcement conferences, or the lack thereof, are not intended to represent final determinations or beliefs.

Persons attending open conferences will be provided an opportunity to submit written comments concerning the trial program anonymously to the regional office. These comments will be subsequently forwarded to the Director of the Office of Enforcement for review and consideration.

When needed to protect the public health and safety or common defense and security, escalated enforcement action, such as the issuance of an immediately effective order, will be taken before the conference. In these cases, a conference may be held after the escalated enforcement action is taken.

VI. Enforcement Actions

This section describes the enforcement sanctions available to the NRC and specifies the conditions under which each may be used. The basic enforcement sanctions are Notices of Violation, civil penalties, and orders of various types. As discussed further in Section VI.D, related administrative actions such as Notices of Nonconformance, Notices of Deviation, Confirmatory Action Letters, Letters of Reprimand, and Demands for Information are used to supplement the enforcement program. In selecting the enforcement sanctions or administrative actions, the NRC will consider enforcement actions taken by other Federal or State regulatory bodies having concurrent jurisdiction, such as in transportation matters. Usually, whenever a violation of NRC requirements of more than a minor concern is identified, enforcement action is taken. The nature and extent of the enforcement action is intended to reflect the seriousness of the violation involved. For the vast majority of violations, a Notice of Violation or a Notice of Nonconformance is the normal action.

A. Notice of Violation

A Notice of Violation is a written notice setting forth one or more violations of a legally binding requirement. The Notice of Violation normally requires the recipient to provide a written statement describing (1) the reasons for the violation or, if contested, the basis for disputing the violation; (2) corrective steps that have been taken and the results achieved; (3) corrective steps that will be taken to prevent recurrence; and (4) the date when full compliance will be achieved. The NRC may waive all or portions of a written response to the extent relevant information has already been provided to the NRC in writing or documented in an NRC inspection report. The NRC may require responses to Notices of Violation to be under oath. Normally, responses under oath will be required only in connection with Severity Level I, II, or III violations or orders.

The NRC uses the Notice of Violation as the usual method for formalizing the existence of a violation. Issuance of a Notice of Violation is normally the only enforcement action taken, except in cases where the criteria for issuance of civil penalties and orders, as set forth in Sections VI.B and VI.C, respectively, are met. However, special circumstances regarding the violation findings may warrant discretion being exercised such that the NRC refrains from issuing a Notice of Violation. (See Section VII.B, "Mitigation of Enforcement Sanctions.") In addition, licensees are not ordinarily cited for violations resulting from matters not within their control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls. Generally, however, licensees are held responsible for the acts of their employees. Accordingly, this policy should not be construed to excuse personnel errors.

B. Civil Penalty

A civil penalty is a monetary penalty that may be imposed for violation of (1) certain specified licensing provisions of the Atomic Energy Act or supplementary NRC rules or orders; (2) any requirement for which a license may be revoked; or (3) reporting requirements under section 206 of the Energy Reorganization Act. Civil penalties are designed to deter future violations both by the involved licensee as well as by other licensees conducting similar activities and to emphasize the need for licensees to identify violations and take prompt comprehensive corrective action.

Civil penalties are considered for Severity Level III violations. In addition, civil penalties will normally be assessed for Severity Level I and II violations and knowing and conscious violations of the reporting requirements of section 206 of the Energy Reorganization Act.

Civil penalties are used to encourage prompt identification and prompt and comprehensive correction of violations, to emphasize compliance in a manner that deters future violations, and to serve to focus licensees' attention on violations of significant regulatory concern.

Although management involvement, direct or indirect, in a violation may lead to an increase in the civil penalty, the lack of management involvement may not be used to mitigate a civil penalty. Allowing mitigation in the latter case could encourage the lack of management involvement in licensed activities and a decrease in protection of the public health and safety.

1. Base Civil Penalty

The NRC imposes different levels of penalties for different severity level violations and different classes of licensees, vendors, and other persons. Tables 1A and 1B show the base civil penalties for various reactor, fuel cycle, materials, and vendor programs. (Civil penalties issued to individuals are determined on a case-by-case basis.) The structure of these tables generally takes into account the gravity of the violation as a primary consideration and the ability to pay as a secondary consideration. Generally, operations involving greater nuclear material inventories and greater potential consequences to the public and licensee employees receive higher civil penalties. Regarding the secondary factor of ability of various classes of licensees to pay the civil penalties, it is not the NRC's intention that the economic impact of a civil penalty be so severe that it puts a licensee out of business (orders, rather than civil penalties, are used when the intent is to suspend or terminate licensed activities) or adversely affects a licensee's ability to safely conduct licensed activities. The deterrent effect of civil penalties is best served when the amounts of the penalties take into account a licensee's ability to pay. In determining the amount of civil penalties for licensees for whom the tables do not reflect the ability to pay or the gravity of the violation, the NRC will consider as necessary an increase or decrease on a case-by-case basis. Normally, if a licensee can demonstrate financial hardship, the NRC will consider payments over time, including interest, rather than reducing the amount of the civil penalty. However, where a licensee claims financial hardship, the licensee will normally be required to address why it has sufficient resources to safely conduct licensed activities and pay license and inspection fees.

2. Civil Penalty Assessment

In an effort to (1) emphasize the importance of adherence to requirements and (2) reinforce prompt self-identification of problems and root causes and prompt and comprehensive correction of violations, the NRC reviews each proposed civil penalty on its own merits and, after considering all relevant circumstances, may adjust the base civil penalties shown in Table 1A and 1B for Severity Level I, II, and III violations as described below.