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TECHNICAL EVALUATION REPORT OF THE OVERPRESSURE PROTECTION SYSTEM FOR OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

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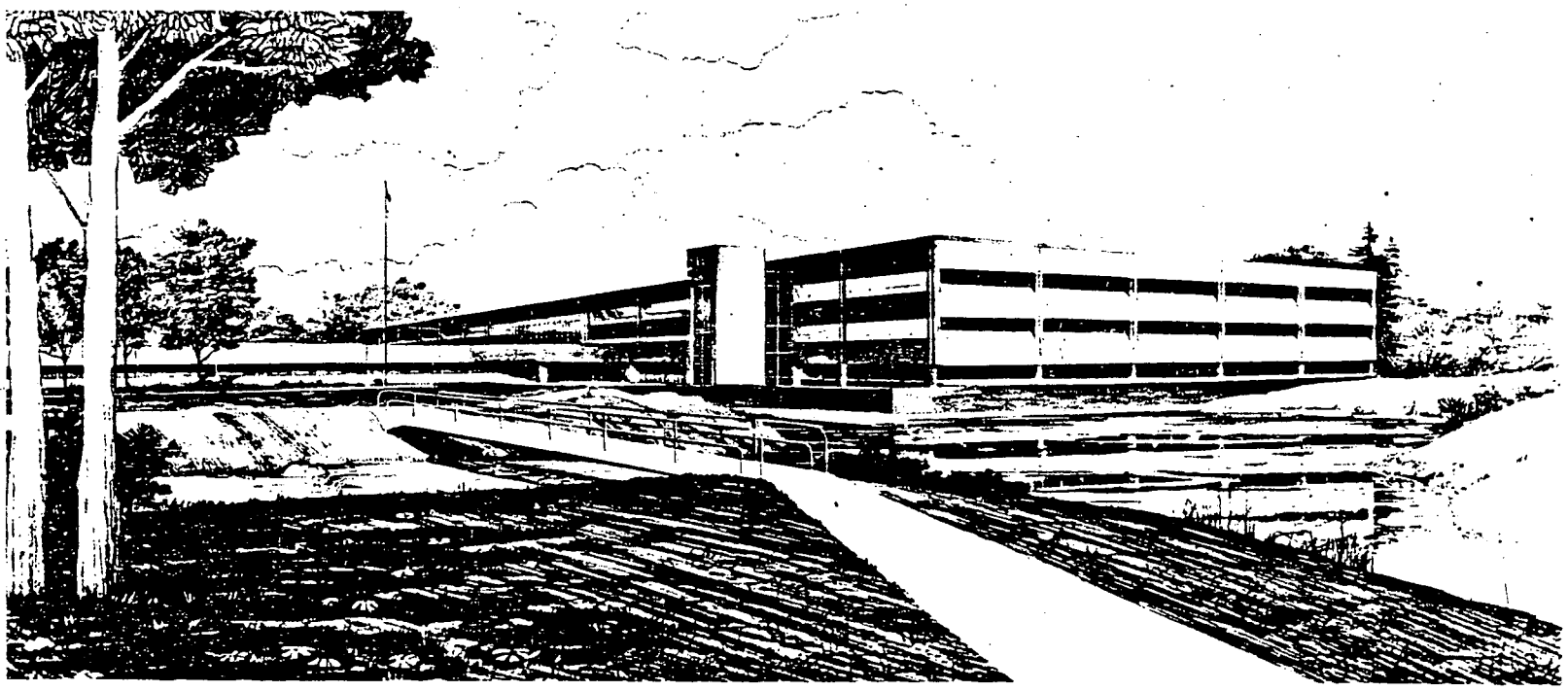

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TECHNICAL EVALUATION REPORT OF THE OVERPRESSURE PROTECTION SYSTEM FOR OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

October 1982

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ABSTRACT

This report documents the technical evaluation of the Low-temperature Overpressure Protection System of the Oconee Nuclear Station, Units 1, 2 and 3. The criteria used to evaluate the acceptability of the system are those criteria contained in NUREG-0224 as appended by the Branch Technical Position (RSB 5-2).

FOREWORD

This report is supplied as part of the "Steam Generator Transients and Operating Reactors Evaluation for Reactor Systems Branch" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Integration, by EG&G Idaho, Inc., Reliability and Statistics Branch.

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1. INTRODUCTION

Several instances of reactor vessel overpressurization have occurred in pressurized water reactors in which the technical specifications implementing Appendix G to 10 CFR Part 50 have been exceeded. The majority of cases have occurred during cold shutdown while the primary system was in a water-solid condition. By letter to Duke Power Company, owner and operator of Oconee Nuclear Station, Units 1, 2, and 3 dated August 11, 1976 (Reference 1), the U.S. Nuclear Regulatory Commission (NRC) requested an evaluation of Oconee Units 1, 2, and 3 to determine susceptibility to overpressurization events and an analysis of these possible events, and required Duke Power Company to propose interim and permanent modifications to the systems and procedures to reduce the likelihood and consequences of such events.

By letter dated October 14, 1976 (Reference 2), Duke Power submitted to the NRC the interim measures that they had taken to minimize the probability of a low-temperature overpressure transient at Oconee 1, 2, and 3. In this letter Duke Power also submitted their final hardware change proposal along with the B&W Generic Analysis. The final hardware change involved the installation of a dual setpoint on the pressurizer pilot actuated relief valve (PORV). This dual setpoint feature will enable the setpoint of the PORV to be reduced to 550 psig upon reducing the reactor coolant system temperature to 325°F. For plants where Babcock & Wilcox (B&W) is the Nuclear Steam System Supplier (NSSS), a primary factor concerning overpressure protection is that they always (except during hydro tests) maintain a steam or gas volume in the pressurizer which retards the pressure increase and allows time for operators to take action to terminate the pressure increase prior to exceeding any limits. The NRC asked Duke Power to answer further questions on their proposed Overpressure Protection System (References 3, 5, and 7),

and Duke Power responded to these questions and proposed additional modifications in their subsequent submittals (References 4, 6, and 8).

This is a report of the evaluation of the compliance of the licensee's Overpressure Protection System with the design criteria established by the NRC.

2. DESIGN CRITERIA

The NRC formally addressed reactor vessel overpressurization in August 1976, and requested that the utilities provide a solution to the problem. The design criteria were subsequently identified through meetings and correspondence with utility representatives. NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" with appended Branch Technical Position (RSB 5-2) formalizes the staff requirements for the overpressure mitigating system. This NUREG also includes a thorough discussion of the background of this problem and technical discussions pertaining to vessel stresses and other aspects of vessel overpressurization.

3. SYSTEM DESCRIPTION AND EVALUATION

The Oconee Overpressure Protection System consists of both an active and a passive subsystem. The active subsystem utilizes the pressurizer pilot actuated relief valve (PORV) which provides high pressure protection during normal plant operation. The PORV actuation circuitry has been modified to provide a second setpoint (550 psig) that is used during low-temperature operations. The low setpoint is manually enabled at 325°F by positioning a key-operated switch in the Reactor Control Room. The passive subsystem is based on the plant design and operating philosophy that precludes the plant from being in a water solid condition (except for system hydrotests). The Oconee 1, 2, and 3 Reactor Coolant System always operates with a steam or gas space in the pressurizer; the steam bubble is replaced with nitrogen during plant cooldown when system pressure is reduced. The vapor space in the pressurizer greatly retards the increase in RCS pressure, as compared to a water solid system, for all mass and heat

input transients. Retarding the rate of pressure increase during transients provides the operator with time to recognize that a pressure transient is in progress and take action to mitigate the transient.

3.1 Pilot Actuated Relief Valve

The pilot actuated relief valve is an electromatic valve that uses system pressure, controlled by an electric solenoid valve, to a pilot mechanism to open the valve and a spring to close it. Characteristics of this valve at the lower setpoint are:

Open setpoint	550 psig
Close setpoint	500 psig
Steam capacity at 550 psig	25,985 lb/hr
Equivalent liquid insurge rate	2,650 gpm
Liquid capacity at 550 psig	550 gpm
Nitrogen capacity at 550 psig	32,420 lb/hr
Equivalent liquid insurge rate	2,350 gpm

3.2 Electrical Controls

The electrical, instrumentation, and control system aspects of the Oconee Nuclear Station, Units 1, 2, and 3 Low-Temperature Overpressure Protection System have been reviewed and reported in a separate technical evaluation (Reference 10).

3.3 Testability

The staff position requires that a test be performed to assure operability of the system electronics prior to each shutdown and that a test for valve operability, as a minimum, be conducted as specified in the ASME Code Section XI. The Oconee 1, 2, and 3 pilot actuated relief valves are tested during every plant startup to demonstrate their operational capability. This is done by positioning the key operated switch to "open" and then verifying that the valve opens by observing various parameters and then closing the valve. The PORV setpoint is verified annually.

We conclude that the Oconee OPS does not meet the staff testability criteria. The sytem is not tested prior to placing it in service during plant cooldown. The testing that is performed does not provide a reasonable assurance that the system will function on demand to protect the vessel from overpressure conditions.

3.4 Single Failure Criteria

The specified single failure criteria for the overpressure mitigating system is that it should be designed to protect the vessel given a single failure in addition to the failure that initiated the pressure transient. The Oconee OPS meets this criteria for all events that were evaluated except for an inadvertent actuation of the High Pressure Injection System (HPI) and for the makeup valve failing full open. For an inadvertent HPI actuation, with the failure of the PORV being the single active failure, the 550 psig setpoint limit would be exceeded in approximately 5 minutes. For a makeup valve failing full open, with the failure of the PORV being the single active failure, the limit would be exceeded in approximately 10.1 minutes. The analysis provided indicates that during a makeup valve full open transient, the first alarms (makeup tank low level and pressurizer high level) are not recieved until approximately 3.5 minutes into the transient. This would leave only 6.6 minutes between the alarms and when the pressure setpoint would be exceeded. The staff position is that no credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress. Therefore, these events do not provide sufficient time for operator action to terminate the transient prior to exceeding the pressure limit.

The two subsystems (or methods) of the Oconee Overpressure Protection System are sufficiently independent and diverse so that there is no known failure mode which could defeat both subsystems. A loss of offsite power will not affect the pressurizer steam bubble or the operator's action ability. A loss of off-site power also will not affect operation of the pilot actuated relieve valve. Power for the instrumentation which controls the pilot actuated relief valve and other parameter indications and alarms will be supplied either by the diesel generator or storage batteries during

a loss of offsite power. A seismic event will not affect the pressurizer steam bubble or the operator's action ability. A seismic event also should not affect operation of the pilot actuated relief valve (refer to Section 3.5 for further discussion).

The most limiting failure for the Oconee OPS is failure of the single PORV. Given this failure, the steam or gas volume in the pressurizer provides a time delay before the pressure increases to the 550 psig limit. This time period would be greater than 10 minutes for all events analyzed except for the inadvertent HPI actuation and the makeup valve failing full open, as discussed above. Duke Power states that they have procedural and administrative controls that make an inadvertent HPI actuation an incredible event at Oconee. These include: (a) bypassing the Engineered Safeguard Actuation of the HPI System at 1750 psig, (b) locking out and tagging the circuit breakers for the four HPI motor operated valves with the valves in the closed position prior to going below 325°F, and (c) securing the operating makeup pump, as soon as the final reactor coolant pump is stopped, to reduce the time the makeup pump is running while the plant is less than 325°F.

We conclude that the Oconee OPS meets the staff single failure criteria for all events except for inadvertent HPI actuation and failure full open of the makeup control valve. We also conclude that the criteria will be satisfied for inadvertant HPI actuation if acceptable Technical Specifications are submitted that require that the three steps mentioned in the previous paragraph are implemented. The makeup control valve failing full open event has not yet been resolved in compliance with the staff's single failure criteria.

3.5 Seismic Design

The specified seismic criteria is that the Overpressure Protection System should be designed to function during an Operating Basis Earthquake (OBE). Detailed stress analyses have been performed for the pilot actuated relief valve in accordance with ASME Section III, Class 1 requirements. The valve design has been found to be adequate for Class 1 application.

Stresses are shown to be within the allowables as specified in ASME Section III, 1971 Edition. Through conservative calculations, the natural frequency is shown to be greater than 500 Hz, well above seismic excitation frequencies, and the maximum axial plus bending stress in the pilot assembly connection pipe due to seismic motion of 3.0 g horizontal and 3.0 g vertical is significantly lower than the allowable. Testing with simulated seismic loadings has not been performed as this was not a requirement at the time this plant was designed and constructed.

We conclude that the Oconee Overpressure Protection System meets the seismic criteria. Even if it is assumed that the relief valve, connection pipe, or actuation circuitry failed due to a seismic event, the steam or nitrogen blanket in the pressurizer and the control room operator would provide protection for postulated low-temperature overpressure events.

3.6 Analysis Results

The analyses provided by Duke Power for the Oconee OPS used a PORV setpoint of 550 psig. In a more recent submittal (Reference 12), Duke indicated that the low temperature PORV setpoint would be less than or equal to 500 psig. Neither the 550 psig setpoint nor the proposed 500 psig setpoint provide protection from exceeding the current Oconee Technical Specification pressure-temperature limits for all RCS temperatures below the minimum pressurization temperature (approximately 408°F). In their letter dated July 6, 1982 (Reference 14), Duke provided a pressure-temperature curve for use with the Oconee OPS that removes some of the safety factors included in the 10 CFR Part 50 Appendix G curve calculations. This new curve and the basis for it are presently being evaluated by the Materials Engineering Branch of the NRC. Pending completion of this evaluation, it is concluded that the present Oconee OPS does not meet the established acceptance criteria. In the event of a negative determination by the Materials Engineering Branch, it would be necessary for Duke Power to reanalyze and modify the Oconee OPS to bring the system into compliance.

The analysis submitted by Duke Power using the 550 psig setpoint have been reviewed and are discussed below.

The analyses are divided into two general categories of pressure transients: mass input from sources such as charging pumps, safety injection pumps, and core flood tanks; and heat input, which causes thermal expansion, from sources such as steam generators and decay heat. All events involving insurge to the pressurizer were evaluated with the pressurizer and makeup tank initially at high water levels. For the pressurizer, an initial water level at the high level alarm setpoint was used for initial pressures above 100 psig and an initial water level at the high high alarm setpoint was used for an initial pressure of 100 psig or below. The relationship of these levels to the other pressurizer water level setpoints is:

0-400 in.	Level indicating range
315 in.	High high level alarm
260 in.	High level alarm
220 in.	Normal level
200 in.	Low level alarm
80 in.	Low level interlock (heater cut-out) and alarm

For the makeup tank, which is the normal suction source for the makeup/HPI pump, a water level at the high level alarm setpoint was used. The relationship of this level to the other makeup tank level setpoints is:

0-100 in.	Level indicating range
86 in.	High level alarm
73 in.	Normal level
55 in.	Low level alarm

3.6.1 Mass Input Cases

The mass input events analyzed in the B&W Generic Analysis are:

a. Makeup control valve (makeup to the RCS) fails full open.

b. Erroneous opening of the core flood tank discharge valve.

c. Erroneous actuation of the High Pressure Injection (HPI) System.

d. Erroneous addition of nitrogen to the pressurizer.

Duke Power subsequently re-evaluated some of these events using plant specific parameters and more realistic conditions.

According to the Duke Power submittals, the most limiting credible mass input transient results from failure full open of the makeup control valve. The makeup control valve is used to regulate the makeup flow rate to the RCS and is normally automatically controlled by the pressurizer level controller. If this valve failed full open, the makeup flow would exceed the letdown flow which would result in an increase in the pressurizer level and RCS pressure. The pressure response for this event has been evaluated using the computer code DYSID with the following assumptions and initial conditions.

a. 260 in. pressurizer water level for 250 psig initial pressure.

b. 315 in. pressurizer water level for 100 psig initial pressure.

c. 86 in. makeup tank water level.

d. 32 gpm total seal injection flow to RC pumps.

e. 45 gpm letdown flow from the RCS to the makeup tank.

f. No spray into the pressurizer (normally there would be spray during cooldown).

The DYSID Code analysis showed that, assuming no operator action, the RCS pressure would increase to 550 psig in approximately 10.1 minutes at which time the pressurizer pilot actuated relief valve would lift to reduce the system pressure. System pressure overshoot, the pressure increase after reaching the PORV setpoint of 550 psig, is almost nonexistent due to the rapid action of the electromatic PORV and the relatively slow rate of pressure increase due to the steam or nitrogen volume in the pressurizer.

For the transient that would result from the makeup valve failing full open, the most limiting single failure is a failure of the single PORV. Given this failure, the steam or nitrogen volume in the pressurizer will allow approximately 6.6 minutes (refer to Section 3.4) after the operator is alerted to the problem by an alarm before the pressure reaches the 550 psig setpoint limit. As stated in Section 3.4, this is a problem that has not yet been resolved.

Duke Power also evaluated the remaining three mass input mechanisms. They stated that inadvertent opening of a core flood tank discharge valve and inadvertent actuation of the HPI system are not credible events due to the administrative controls established and the low probability chain of events required to initiate the transients. The administrative controls in use to prevent an inadvertent HPI actuation are discussed in Sections 3.4 and 4.0. Duke Power considers inadvertent opening of a core flood tank discharge valve not credible because the valves are closed and the circuit breakers for the motor operators are "racked out" during the plant cooldown before the RCS pressure is decreased to 600 psig. The analysis shows that the final RCS pressure for this event is 450 psig which is less than the 550 psig setpoint. The erroneous addition of nitrogen to the pressurizer does not pose a threat due to the 125 psig regulator used in the system and the relief valve, set at 150 psig, that provides protection in the event of a regulator failure.

We conclude that the Oconee Overpressure Protection System will adequately mitigate all mass input cases except the inadvertent actuation of the HPI system and failure full open of the makeup control valve as discussed in Section 3.4 of this report. The Oconee system may be judged acceptable for the inadvertent HPI actuation event if adequate technical specifications are submitted and approved implementing the necessary controls that minimize the probability of an event.

3.6.2 Heat Input Cases

The three events analyzed that involve heat input into the primary coolant system are:

a. All pressurizer heaters erroneously energized.

b. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS.

c. Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator.

Of these three events, only one, temporary loss of the Decay Heat Removal System's capability to remove decay heat, could credibly result in exceeding a pressure limit. All pressurizer heaters erroneously energized is not a significant hazard because of the slow rate of pressure increase. Even with the worst case initial conditions such as a pressurizer level of 90 inches, the rate of pressure increase is so slow that 550 psig is not reached until 47 minutes after energizing the heaters. This time period should be adequate to allow the control room operator to recognize that a pressure transient is occurring and terminate it by de-energizing the heaters.

In the analysis of thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator, Duke Power reported on the evaluation of two specific conditions:

a. Filling of the once through steam generator (OTSG) secondary side with hot water with subsequent start of an RC pump, and

b. Restart of an RC pump during heatup following a period of stagnant (no flow) conditions.

The results of these analyses using conservative initial conditions are a maximum pressure of 430 psig for case "a" and 530 psig for case "b". These values are below the PORV setpoint of 550 psig. Other conditions of primary and secondary temperatures which may exist prior to starting an RC pump have been evaluated and are bounded by the above analyses.

A loss of the Decay Heat Removal (DHR) System capability could be caused by loss of flow in the DHR System or in the cooling water system serving the DHR System. A loss of DHR System capability was analyzed using the following conditions:

a. Event occurs during plant cooldown after shutdown of steam generators.

b. Pressurizer level at 260 inches (high level alarm).

c. All decay heat absorbed by reactor coolant, no heat absorbed by the metal components or by the steam generators.

d. 32 gpm total seal injection to RC pumps.

e. 45 gpm initial letdown from RCS to makeup tank.

f. No spray into the pressurizer.

g. Cooldown rate of 100°F/hr until DHR system "cut-in" temperature, this produces the maximum decay heat generation rate.

The analysis determined that if no operator action were taken the RCS pressure would increase to the PORV setpoint in approximately 29 minutes. At that point, the PORV should open and limit the RCS pressure to 550 psig. Given a failure of the pressurizer PORV, the 29 minutes should be sufficient time to allow the operator to detect the problem and take action to correct it. The operator should be alerted to the loss of DHR capability by a flow alarm indicating either a loss of flow in the DHR System or a loss of flow in the cooling water system serving the DHR System.

In the heat input analysis, the 550 psig limit is not exceeded; therefore, the performance of the Oconee Overpressure Protection System is judged to be adequate for heat induced transients.

4. ADMINISTRATIVE CONTROLS

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, procedural and administrative controls should be provided by the licensee. Those specific conditions required for the plant to be operated within the bounds of the analysis and requirements for enabling and testing the OPS should be spelled out in the plant Technical Specifications.

4.1 Procedures

A number of provisions for the prevention of pressure transients are incorporated in the plant operating procedures. Some examples of these provisions are given below:

a. The Oconee Overpressure Protection System is to be manually enabled prior to the reactor coolant system temperature dropping below 325°F during plant cooldown.

b. The plant is to be operated with a steam or nitrogen blanket in the pressurizer at all times except for system hydrostatic tests. The pressurizer water level is maintained at or below the high level alarm at system pressures above 100 psig and less than the high high level alarm for pressures less than or equal to 100 psig.

c. The makeup tank water level is to be less than the high level alarm.

d. The core flood tank discharge valves are closed and the circuit breakers for the motor operators are "racked out" before the RCS pressure is decreased to 600 psig.

e. During a plant cooldown, the Engineered Safeguard Actuation of the HPI System is bypassed at 1750 psig. Prior to going below 325°F, the circuit breakers for the four HPI motor operated valves are "locked out" with the valves in the closed position.

4.2 Technical Specifications

Duke Power has not yet submitted technical specifications that relate specifically to the operation of the Oconee OPS or establish requirements that decrease the probability of occurrence of an initiating event that could result in a pressure transient.

We consider it necessary for Duke Power to submit technical specifications that cover this system in order for the Oconee OPS to be acceptable. These technical specifications should, as a minimum, contain requirements that cover each of the items in Section 4.1 of this report. In addition, the technical specifications should include requirements for testing the OPS, setting system setpoints, and enabling the system alarms. We will consider the administrative controls acceptable for Oconee if adequate technical specifications are submitted and approved.

5. CONCLUSIONS

The administrative controls and plant modifications proposed by Duke Power Company provide a degree of protection for Oconee Nuclear Station, Unit 1, 2, and 3 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient. The system will not prevent the RCS pressure from exceeding the limits set by 10 CFR 50 Appendix G for all of the transients analyzed and, therefore, does not meet the current staff criteria. In order for the Oconee Overpressure Protection System to meet the current acceptance criteria, resolution is required on the open items identified in Sections 3.3, 3.4, 3.6, 3.6.1, and 4.2 of this report. Pending resolution of these items, the Oconee Nuclear Station, Units 1, 2, and 3 Overpressure Protection Systems are judged as adequate interim solutions to the problem of low-temperature overpressure transients.

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