July 2, 2002

MEMORANDUM TO: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

THROUGH: Farouk Eltawila, Director ***/RA by Charles E. Ader for/***
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

John Flack, Chief ***/RA/***
Regulatory Effectiveness Assessment and Human Factors Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

FROM: Harold VanderMolen, Chairman ***/RA/***
Reactor Generic Issue Review Panel
Office of Nuclear Regulatory Research

SUBJECT: RESULTS OF INITIAL SCREENING OF GENERIC ISSUE 192, "SECONDARY CONTAINMENT DRAWDOWN TIME"

In accordance with Management Directive (MD) 6.4, "Generic Issues Program," the Generic Issue Review Panel has completed the initial screening of Generic Issue(GI) 192, "Secondary Containment Drawdown Time," and has concluded that the issue does not represent a new safety concern (see Attachments 1 and 2). GI-192 addresses the concern for the adequacy of the calculations, testing, and acceptance criteria associated with the creation of a vacuum in the reactor building of a BWR, following an engineered safeguards actuation signal. The panel found that existing regulations are adequate to address the concern and recommends that the issue be excluded from further analysis. Your approval of the panel's recommendations is required so that RES can proceed to the next step of the MD 6.4 process.

Attachments:
1. Minutes of GI-192 Review Panel
2. GSI-192 Evaluation

Approved: ***/RA/*** Date: 7/03/02
Ashok C. Thadani, Director, RES

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PANEL MEETING TO SCREEN
GSI-192, "SECONDARY CONTAINMENT DRAWDOWN TIME"
WEDNESDAY, JUNE 5, 2002

Venue: T10-C01

Attendees (5): *Harold VanderMolen (REAHFB/DSARE/RES), Chairman*
Ronald Emrit (REAHFB/DSARE/RES)
John Lane (PRAB/DRAA/RES)
Richard Lobel (SPLB/DSSA/NRR)
John Ridgely (PRAB/DRAA/RES
Absentee (1): *John Flack, (REAHFB/DSARE/RES)*

MINUTES

The meeting was called to order at 8:13 a.m. by Chairman *Harold VanderMolen* who gave a brief explanation of the MD 6.4 process which was being implemented with the convening of the panel.

Rich Lobel then gave a brief history of the costly experience of other licensees in meeting the existing requirements for secondary containment drawdown.

Harold Vandermolen then began a step-by-step explanation of his analysis of the issue and invited questions as he proceeded.

John Ridgely inquired about the reason for referring to only two plants (Brunswick and Cooper) in the identification of the issue. *Lobel* responded that NRR could not proceed to study the issue unless a specific plant was identified. No special significance should be given to these plants.

John Lane believed that the concern was a valid one, given that some licensees did not appear to be accurately measuring secondary containment vacuum.

John Ridgely questioned the value of the stack worth used in the analysis. He stated that, based on NUREG-1169, the value used could be reduced by a factor of 2. The scenario used in NUREG-1169 was a large-break LOCA with different release pathways. Some were ground level releases, and one was a stack release without any filtration. The average reactor power for operating BWR/2, /3, and /4 plants is significantly less than the 3200 MWt assumed in WASH-1400. The PWR-9 release is overstated in the analysis because there would be no scrubbing of the release while, in a BWR, a large portion of the LOCA release would enter and be scrubbed by the suppression pool. He believed that the 100 man-rem assumed in the analysis for the fuel handling accident was an overestimate, based on the scenario where a spent fuel cask was dropped over the spent fuel pool, one core of the fuel was assumed to be damaged, and the consequence was calculated to be 0.1 man-rem. Additionally, a revised analysis using the lower consequences suggested above would lower the total risk associated with the issue.

Lane requested clarification on Concern 3. *Lobel* responded that the 1/4" water gauge criterion only accounts for wind speed up to approximately 30 mph. It does not reflect the difference in pressure between drawing a vacuum in the secondary containment during normal conditions and the pressure and temperature in different compartments during a postulated design basis LOCA.

Conclusion: After exchanging views on the options available, the panel unanimously agreed that, based on the analysis by *Harold VanderMolen*, the issue does not represent a safety enhancement, but addresses the concern for licensee **compliance** with existing plant Technical Specifications.

The meeting was adjourned at 9:55 a.m.

Prepared by: Ronald C. Emrit

ISSUE 192 : SECONDARY CONTAINMENT DRAWDOWN TIME

DESCRIPTION

Historical Background

This issue was raised[1789] by NRR and addresses the adequacy of the calculations, testing, and acceptance criteria related to the creation of a vacuum in the reactor building of a BWR, following an engineered safeguards actuation signal.

The time required to attain a vacuum in the reactor building is commonly referred to as the "drawdown time." The vacuum is necessary to ensure that any air leakage flows into the building so that any radiological contamination in the building air is processed by the appropriate safety systems, before being released to the environment. Guidelines for including the drawdown time in offsite and control room dose calculations are specified in BTP CSB 6.3 of SRP[11] 6.2.3, "Secondary Containment Functional Design." Independent calculations showed that plants could potentially exceed the limits for offsite and control room doses.

Safety Significance

The Standby Gas Treatment System (SBGT or SGTS) provides a means for minimizing the release of radioactive material from the containment to the environment by filtering and exhausting the atmosphere from any or all zones of the reactor building during containment isolation conditions. The SBGT system is classified as an Engineered Safety System. The design basis for the system is to prevent any uncontrolled release due to a design basis LOCA (during power operation), or due to a fuel handling accident (during refueling conditions, when the primary containment is open and the secondary containment is the only containment).

During normal operation, the reactor building is heated, cooled, and ventilated by a circulating air system, which generally exhausts from the reactor building roof with minimal or no filtration. This reactor building HVAC system is shut down and isolated when the secondary containment is isolated and connected to the Standby Gas Treatment System. The SBGT system will initiate automatically on reactor zone high radiation, refueling zone high radiation, low reactor water level, or high drywell pressure.

In addition to this primary function, the SBGT system also has other uses of interest:

- The SBGT system is used to process exhaust gases from the gland seal condenser of the HPCI turbine. Normally, the signals which initiate HPCI (e.g., low-low reactor water level) will also initiate the SBGT system.

- The SBGT system is used to test secondary containment integrity.

- The SBGT system is used to purge air from the drywell and suppression pool air space when necessary (e.g., prior to personnel entry).

The size and location of the SBGT system is site-specific. However, the SBGT system will normally consist of more than one train, and be capable of performing its function with one train out of service. (A single-unit site will generally have two trains, but a multiple-unit site, where there may be one reactor building housing two primary containments, may have a three-train shared system.)

Each SBGT train generally consists of a (shared) suction duct system, a moisture separator and heater to keep humidity within limits, then a set of particulate filters and charcoal adsorbers plus a blower. The train will discharge to the plant stack, to provide an elevated release. The system is design to remove particulates and iodine. Unlike the offgas system, there is no holdup pipe to allow the noble gases to decay before release.

If the SBGT system is used to exhaust just a few individually-isolated zones, it is possible to draw a significant vacuum. In order to prevent structural damage, there is also a standby gas treatment vacuum relief system, which will bleed outside air into each zone of the reactor building to prevent the outside pressure from exceeding the inside pressure by more than a certain amount (e.g., ½ inch water gauge).

There are several limiting conditions for operation and surveillance requirements in the technical specifications regarding SBGT system operability.[706] These generally include running the system monthly, plus verifying each refueling cycle (or every 18 months) that the flow rate through the system and pressure drop across the various filters are within specification.

In addition, there is generally a surveillance requirement that secondary containment integrity be demonstrated every 24 hours by verifying that the secondary containment interior pressure is at least 1/4 inch water gauge vacuum. This is checked daily, and the negative pressure is maintained by the normal HVAC system. Once per 18 months, it is verified that one train of the SBGT system can draw down the secondary containment to 1/4 inch water gauge within a set time (120 seconds in the Standard Technical Specifications). This test verifies both SBGT system efficacy and leak tightness of the secondary containment with the normal HVAC isolated. Some older Technical Specifications did not include a time limit. [See Appendix A to Facility Operating License DPR-33, Technical Specifications and Bases for Browns Ferry Nuclear Plant Unit 1, Limestone County, Alabama, Tennessee Valley Authority, Docket No. 50-259, Amendment 50, September 15, 1981.]

The three specific safety concerns raised[1789]by NRR were:

CONCERN 1: Calculations for reactor facilities (primarily Brunswick, Cooper, and BWR/4 plants and earlier) are performed using a single volume to represent the secondary containment. This doesn't account for the compartmentalization of the building and different heat sources in different compartments. Some compartments, perhaps compartments with sources of radioactivity, may not depressurize as fast as others and may be potential leakage paths.

CONCERN 2: Reactor facilities (primarily Brunswick, Cooper, and BWR/4 plants and earlier) measure the vacuum in only one location in the secondary containment. This location may not be in the most conservative location (the last area to reach the desired vacuum).

CONCERN 3: The criterion used, 0.25 in-water vacuum, only accounts for pressure distribution around the building due to wind. It does not account for the difference in inside temperature during a cold test and during a LOCA accident.

Possible Solutions

The documentation presented with the issue does not give an explicit recommended solution. Presumably, the solution would be to re-calculate the drawdown time with more accurate models. The objective is to ensure that, if an appropriate pressure reduction is achieved at the point where the pressure is being measured, an appropriate pressure reduction is being achieved throughout the entire secondary containment.

If the drawdown time were then shown to be too long, a modification to the SBGT system would be required. This could involve the installation of more ducting to provide multiple suction points for the SBGT system, and/or actions to minimize inleakage to the secondary containment, especially at points near the SBGT intake(s), where a leak could prevent more remote areas from being drawn down.

In theory, a fix might involve an upgrade to the flow capacity of the SBGT system itself. It should be noted, however, that the SBGT flow is normally adjusted to be within a certain range, under the assumption that, if design SBGT flow does not achieve the desired pressure reduction, there are leaks to be fixed. Simply increasing SBGT flow might reduce pressure in areas near the SBGT intake, but might not achieve the required pressure reduction everywhere.

ASSESSMENT

Frequency Estimate

Large-Break LOCA: The first event of interest is the large break LOCA. A large-break LOCA will cause widespread failure of the cladding integrity due to departure from nucleate boiling, resulting in the release of gap activity to the primary coolant. The accident will also release this primary coolant to the containment atmosphere. Smaller break LOCAs will also release primary coolant to the primary containment atmosphere, but will not necessarily release gap activity from the fuel rods, and thus are not included here. The “classic” large LOCA frequency of 10^{-4} event/RY will be assumed.[16]

Fuel Handling Accident: The other design basis accident for the SBGT system is a fuel handling accident, where a fuel assembly is mechanically damaged and gap activity is released. This accident is not normally modeled in modern PRAs, because it is generally not a significant contributor to a plant’s total risk profile. It was addressed many years ago in the Reactor Safety Study, which estimated a frequency of 10^{-4} event/RY (WASH-1400,[16] Appendix I, p. I-100). (This is the frequency of events in which gap activity is actually released; it is not the frequency of all events in which a fuel assembly is dropped or otherwise mishandled.)

Consequence Estimate

Large-Break LOCA: The essence of concerns 1 and 2 is that the calculations and measurement techniques used to measure the efficacy of the system may be too primitive. The secondary containment is not a single volume, but the calculations may model the secondary containment as one large volume, and the vacuum may be measured at just one point. The practical effect of this is that it may take longer than expected for the SBGT system to draw the secondary containment down to the required vacuum, and some compartments (e.g., a compartment containing a leak from the primary containment but located such that the pathway to the SBGT system intake is long) may never achieve the required vacuum at all.

The essence of the third concern is that the 0.25 inch water gauge vacuum criterion may not be sufficient, since it is based only on overcoming external wind conditions.

In either case, to evaluate the risk significance of the issue, it will be necessary to estimate the "worth" of the SBGT system in terms of averted public dose, given that a design-basis LOCA has occurred.

Ideally, the risk worth of the SBGT system could be calculated by using the source term for a successfully mitigated LOCA with the containment losing inventory at the design leakage rate. The calculation would then be done with and without the SBGT system. The SBGT system should greatly reduce the public dose because of its filtration of the air flow and because the SBGT discharge air is routed to the plant stack, resulting in an elevated release.

Unfortunately, few modern PRAs model a mitigated LOCA, since such events, which do not result in a severely damaged core, are not risk significant. The only readily-available probabilistic analysis which includes mitigated LOCA sequences is the original WASH-1400[16] calculation. The release category of interest is:

BWR-5 This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containment leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass through the elevated stack.

If there were a similar BWR release category in which the SBGT was not functioning, a simple comparison would give the risk worth of the SBGT system. Such a release category was not used in WASH-1400.[16] However, there is an analogous category for the PWR analysis:

PWR-9 This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt.

Can this category, which applies to a completely different reactor design, be used for comparison? The radioactive inventories of the two designs are very similar, and it is reasonable to assume that the gap activity releases are similar. A comparison of some containment parameters[1790] is summarized in Table 3.192-1.

Although the PWR containment is much larger, having a free volume almost a factor of six greater than the combined BWR drywell and suppression chamber free volume, the design leak rate for the PWR is one-fifth of the BWR rate. The two effects almost cancel, and the leak rate for the PWR containment is only 20% larger than that of the BWR primary containment.

The man-rem associated with the two release categories of interest have been calculated for the usual generic issue screening assumptions, and are listed in the Introduction to NUREG-0933. The results are:

BWR-5	20 man-rem
PWR-9	120 man-rem.

Table 3.192-1

WASH-1400[16] BWR		WASH-1400[16] PWR	
Core thermal power	3293 MWt	Core thermal power	2441 MWt
Drywell free volume	175,000 cubic feet	Free volume	1.8 x 10^6 cubic feet
Suppression chamber air volume	127,700 cubic feet		
Drywell design pressure	56 psig	Design pressure	60 psig
Suppression chamber design pressure	52 psig		
Design leak rate	0.5% per day	Design leak rate	0.1% per day
Calculated leak rate at design pressure	1500 cubic feet/day	Calculated leak rate at design pressure	1800 cubic feet/day

The WASH-1400[16] source terms were calculated using a “representative” reactor core power of 3200 MWt (Appendix VI, Section 3.2, p. 3-1), and the man-rem figures above assumed this core power. Thus, it was not necessary to adjust for core power differences.

Other than a minor difference in leak rate, the difference in consequences between these two release categories is presumably due primarily to the presence of the secondary containment and SBGT system. (Some other effects will be addressed below in the discussion of uncertainties.) Thus, this reduction is estimated to be :

$$(120 \text{ man-rem})[(1500 \text{ ft}^3/\text{day})/(1800 \text{ ft}^3/\text{day})] - 20 \text{ man-rem} = 80 \text{ man-rem}$$

It will be conservatively assumed that this entire reduction is due to the SBGT system. In actuality, if the SBGT system were not functioning, there would still be some retention of radioactive material in the secondary containment structure. This estimate of 80 man-rem is somewhat of an overestimate of the worth of the SBGT system by itself.

This factor of five reduction (from 100 to 20 man-rem) is significant. The 20 man-rem is likely due primarily to noble gases, since they, unlike particulates and iodine, will not be removed by the SBGT system. Even the noble gases will have fewer health effects, because the SBGT system will release them at an elevated location.

Fuel Handling Accident: A fuel handling accident releases gap activity due to mechanical damage to the fuel pins, such as could happen if a fuel assembly were dropped during refueling or during normal fuel pool operations. Although the release would be modest, the accident would not take place within a closed primary containment, and the secondary containment would be the only containment. The SBGT system would then preclude an “uncontrolled” release to the environment.

Unfortunately, there is no readily-available calculation of the consequences of such a release. However, it is possible to put an upper bound by adapting the PWR-8 release, which is a mitigated large-break LOCA where the containment fails to isolate. This is extremely conservative, in that the PWR-8 release category deals with fuel which was in full power operation just moments earlier, and also includes the release of all the primary coolant activity in the primary system. The fuel involved in a fuel handling accident would have had time for some radionuclides to decay away, and would not involve any primary coolant activity. Moreover, a fuel handling accident would be expected to occur with the fuel assemblies fully submerged, and any activity released would be scrubbed by the overlaying water. Thus, this upper bound may be conservative by two or more orders of magnitude.

In the Introduction to NUREG-0933, the consequences calculated for a PWR-8 release are 75,000 man-rem; this is for an entire reactor core. The reactor of interest here consists of 764 fuel assemblies (NUREG/CR-5640,[1790] Table 8.3-1, page 8-17). Thus, the normalized release for one fuel assembly is about 100 man-rem. It will be assumed that the SBGT is capable of reducing this to zero.

Cost Estimate

A cost estimate was not performed. The results of the value assessment are such that a cost estimate is not necessary.

Screening Assessment

The screening criteria of core damage frequency and large early release frequency are not applicable to this issue. The SBGT system is a mitigative system, and does not affect CDF or LERF. The applicable criterion is averted offsite man-rem/year.

For generic issue screening purposes, the criterion is based on the absolute value of the system's risk worth. The risk reduction for the large break LOCA is a simple product of the frequency of large-break LOCAs multiplied by this risk worth:

$$(10^{-4}\ \text{LOCAs/RY})(80\ \text{man-rem/LOCA}) = 0.008\ \text{man-rem/RY}$$

Similarly, the risk reduction associated with the fuel handling accident is given by:

$$(10^{-4}\ \text{event/RY})(100\ \text{man-rem/event}) = 0.01\ \text{man-rem/RY}$$

The two risk reductions add up to 0.018 man-rem/RY. Adding up the BWR/2, BWR/3, and BWR/4 plants, there are 27 reactors affected by this issue. This implies a total value of 0.5 man-rem/year. Comparing with Figure C6 of Management Directive Handbook 6.4, this issue should be excluded from further consideration, regardless of cost.

Uncertainties

(1) There is some ongoing research regarding whether the large-break LOCA frequency of 10^{-4} per RY is too high. Lowering this frequency would not change the conclusion.

(2) It is possible that some intermediate-LOCA events could also result in significant gap activity being released into the primary containment atmosphere. However, even including

the small (“S1") LOCAs would increase the frequency by a factor of ten, which would not be enough to change the conclusion.

(3) As was discussed above, the estimate of change in man-rem estimated above includes more than just the effect of the SBGT system. In addition to this, the first two concerns raised in the issue have to do with transient conditions - can the SBGT system draw a vacuum in the secondary containment in time? Presumably, once the secondary containment reaches the desired subatmospheric pressure, the SBGT system would still be efficacious, and the above estimate of change in man-rem would be conservative. This also would have no effect on the conclusion.

(4) The analysis approximates the consequences of a mitigated LOCA in a BWR with no credit for the SBGT system by using a PWR-9 release, although the assumed leakage from primary containment differs by about 20%. In addition to this difference, an actual release pathway in such circumstances in a BWR large LOCA sequence would at least partially be from the suppression pool airspace, where the release would have been scrubbed by the suppression pool water. Also, some of the particulates and aerosols would be removed by plateout in the reactor building. Finally, unlike a BWR release, a PWR release would be entirely at ground level. Thus, the approximation of a BWR release with no SBGT system by a PWR-9 release is conservative. This also will have no effect on the conclusion.

(5) The calculations above assume a core power of 3200 MWt. In actuality, most of the affected reactors are not licensed to this high a power, so the consequences will be somewhat overestimated.

(6) The estimate of the fuel handling accident consequences are extremely conservative. This remains appropriate for a generic issue screening, where conservatism is normally included, and does not affect the conclusion of this study. However, the value estimated here is not appropriate for use as a best estimate for other purposes.

Other Considerations

(1) This issue appears to have no effect on the ability of the SBGT system to accommodate the exhaust from the HPCI gland seal. Moreover, the HPCI system is not used to mitigate a large LOCA.

(2) Similarly, this issue does not affect the ability of the SBGT system to test secondary containment integrity.

(3) The analysis above deals with a design basis LOCA event, not a severe (i.e., core melt) accident. In theory, the presence of the SBGT system could help mitigate the radiological effects of a core melt. In practice, the SBGT system will not have the capacity to make a significant difference in the release associated with a core melt, even if upgraded. Moreover, in most severe accident sequences, the core would not actually melt immediately. Generally, severe core damage would not occur until the existing SBGT systems were able to achieve drawdown in any case.

Moreover, if a core melt event were to occur in a BWR, one strategy for dealing with the situation would be to intentionally vent the suppression pool air space to the outside, thereby preventing containment failure due to overpressure, but also using the suppression pool to scrub the release. In such an event, SBGT system drawdown time is unlikely to be of much concern.

(4) Another effect of the presence of the SBGT system is to reduce personnel exposure within the plant area, allowing more freedom of movement for plant personnel to take mitigative actions. This is rather limited in that there is no effect inside the secondary containment. In addition, the control room has its own separate ventilation system with a filtered intake. Thus, the effect of the SBGT system is limited to other areas, such as outdoors near the reactor building. There is no simple way to quantify this effect, but it is unlikely to be major.

Discussion

The low potential risk reduction associated with this issue implies that it is very unlikely that a backfit requirement could be imposed under 10 CFR 50.109. However, the originator of the issue has made a valid point in that SRP[11] 6.2.3 does state in acceptance Criterion 3a that "The secondary containment depressurization and filtration systems should ... be capable of maintaining a uniform negative pressure throughout the secondary containment, as well as other areas served by the systems." In addition, Criterion 1g states that "heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered." In hindsight, it appears that these acceptance criteria were not fully used in the reviews of individual BWR technical specifications.

CONCLUSION

Based on the above estimates of averted offsite man-rem, this generic issue should be excluded from further consideration.

REFERENCES

11. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

16. WASH-1400 (NUREG-75/014), “Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” U.S. Atomic Energy Commission, October 1975.

706. NUREG-0123, “Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5),” U.S. Nuclear Regulatory Commission, (Rev. 3) December 1980.

1789. Memorandum to A. Thadani from S. Collins, “Proposed Generic Safety Issue Related to Secondary Containment Drawdown Time,” December 3, 2001. [ML013330114]

1790. NUREG/CR-5640, “Overview and Comparison of U.S. Commercial Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, September 1990.