Secondary System Pressure Boundary as an Extension of Containment Liner

ULD-0-TOP-14, Containment Isolation and Containment Leak Rate Testing (Ref. 1), states that, "ANO has taken the position that secondary system penetrations are not subject to GDC 57 since the containment barrier integrity is not breached during DBA LOCA conditions. The containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes the outer surface of the lines emanating from the steam generator and the outer surface of the steam generator between the tube sheets." This position is based on the concept of treating the secondary system pressure boundary as an extension of the containment liner. Though not well documented prior to the writing of this ULD, this position has always been the understanding of plant personnel familiar with the original design and licensing basis. Over time, this original design and licensing basis has become obscured by mixing the concepts of the secondary system pressure boundary being an extension of the containment liner with the concept of a GDC 57 closed system. The fact that the two concepts may appear to be equivalent without careful consideration has contributed to the confusion. Before the issuance of the GDCs, no attempt to distinguish between the concept of "extension of containment" and "reactor building isolation valve" was made nor was it necessary. In 1969 (Ref. 2), Bechtel indicated for ANO-1 that "in the PSAR, the main steam lines are defined as an extension of containment"[1]. About that time an impending ASME code requirement pertaining to piping and valves forming an "extension of containment" prompted an internal Bechtel memo (Ref. 3) that recommended all projects provide single isolation valves on main steam and feedwater lines. At that time the ANO-1 design did not include main steam isolation valves. With the issuance of ASME Special Rulings #1425 and #1427 regarding "piping which forms an extension of the pressure boundary of containment vessels", Bechtel recommended adding main steam isolation valves to the ANO-1 design (Ref. 4). Bechtel later insisted on this addition despite the reluctance of AP&L, again citing Code Case 1427 and using the phrase "extension of containment" (Ref. 5). This was followed up by Ref. 14 providing additional information advising the addition of main steam isolation valves. This recommendation was accepted by AP&L in Ref. 15 with the careful insistence that the valves not be called "main steam containment isolation valves" but be called "main steam block valves". That the concept of extension of containment was not distinguished from containment isolation valve was evident in these communications and was unnecessary since the GDCs had not been issued, defining what a containment isolation valve was and what the design requirements for containment isolation valves were. Once this occurred, the distinction began to appear but only evolved as the full implications of the related GDCs (54, 55, 56 and 57) evolved over the next two or three decades.

The earliest documented distinction between the two concepts appears to be WCAP-7451 (Ref. 6), dated September 1972, which explicitly treated the distinction approximately a year after the issuance of GDC 57. Other examples of documentation of an explicit distinction between the two include NUREG-0830 (Ref. 7), page 6-18 (approving for Callaway the wording of the SNUPPS FSAR, see pages 6.2.4-5 and 6.2.4-6 and figures 6.2.4-1 pages 1 and 5, and 6.2.4-2); NUREG-0881 (Ref. 8), page 6-1 (endorsing the approvals of NUREG-0830 for use at Wolf Creek) and NUREG-0857 (Ref. 9) (approving the Palo Verde FSAR, see page 6.2.4-25). Even with the more explicit wording in the more modern SNUPPS and Palo Verde FSARs, none of the three SERs referenced above explicitly addressed this distinction; they just gave general approval based on the applicable FSAR discussion. These facts do not establish a licensing basis for ANO but do establish the practice and understanding that formed the context within which ANO-1 and ANO-2 were licensed.

By early 1972, when Amendment 23 to the ANO-1 FSAR was issued, the response to AEC question 10.1[2] indicated that the main steam block valves were designed to serve as containment isolation valves and that they met GDC57. There was no statement that acknowledged the applicability of GDC57 to these valves, just that the valves met GDC57. However, later that same year, when Amendment 30 to the ANO-1 FSAR was issued, ANO indicated that they had begun to comprehend the distinction by carefully wording a response to AEC question 5.83 such that there was no acknowledgement that the main feedwater isolation valves were governed by GDC57 but that they were "designed to perform the function of the reactor building isolation valve in accordance with General Design Criterion 57, 10 CFR 50 Appendix A", after having indicated in the preceding paragraph that the new (at that time) GDC 57 did not apply because ANO-1 was designed under older, previously existing requirements. When responding to the TMI-2 event in early 1980 in Ref. 10, none of the valves associated with the secondary side of the steam generators were listed in the category of "automatically actuated valves which provide penetration isolation". For ANO-1, the confusion appears to have peaked with the insertion of the GDC designations into SAR Table 5-1 in Amendment 11 in July 1993 along with a note that read as follows:

> These penetrations are associated with the secondary side of the steam generators and are not subject to GDC-57 since the containment barrier integrity is not breached during DBA LOCA CONDITIONS. The containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes the outer surface of the lines emanating from the steam generator and the outer surface of the steam generator between the tube sheets.

For ANO-1, the following building penetrations have been listed in SAR Table 5-1 as governed by GDC57 at one time or another. For reference, this table correlates the penetration numbers with the system in which the piping which goes through the penetration is involved.

Bldg. Penetration #	System	Bldg. Penetration #	System
P1, P2	Main Steam	P58, P64	SG Blowdown
P3, P4	Main Feedwater	P17, P65	EFW to SGs
P21, P22, P55, P63	Service Water		

No other licensing basis document has indicated that P1, P2, P3, P4, P58, P64, P17 or P65 were governed by GDC57 and only SAR amendments 11 and/or 12 and/or 13 included these eight penetrations. Amendment 11 did not include P21, P22, P58 or P64 in this category and Amendment 12 eliminated the above note that Amendment 11 had added and added P58 and P64 to the GDC57 category. Amendment 13 added P21 and P22 to the GDC57 category and Amendment 14 added back the above note that Amendment 11 had added plus the following sentence at the end of the note: "Valves associated with these penetrations are not reactor building isolation valves." Amendment 14 also changed P58 and P64 back to GDC56 and applied the note to P1, P2, P3, P4, P10, P17 and P65, effectively removing the valves associated with these penetrations from the table. Notably, P10 has never been shown as governed by GDC57.

DCD project discrepancy CI-3 (Ref. 13) highlighted the lack of documentation of the design basis for the piping penetrations of the reactor building. This lack of documentation was causing confusion which was exacerbated by personnel with experience at other nuclear power plants that had ownership

of the Appendix J testing program. The lack of design documentation caused design basis questions to be addressed to the Appendix J testing program personnel by default and the answers that were generated were frequently colored by their experience at other sites in lieu of the absent design documentation. Resolution of CI-3 was assigned to design engineering and eventually addressed by the issuance of Engineering Report 93-R-0007-01 (Ref. 11) in March 1995. One of the many issues that the developers of this engineering report had to deal with was the confusion regarding the role of the secondary system in containment isolation that had compounded over the years. This report restated the original position that the penetrations "associated with the secondary side of the steam generators . . . are not subject to GDC 57 since the containment barrier integrity is not breached during DBA LOCA conditions" in its configuration note C-57-2. That note also stated that, "Although not directly applicable, the penetration arrangement most closely matches the requirements of GDC 57, which has been conservatively applied." This last part of the configuration note explains the appearance of the "57" in the column of the attachments entitled "Applicable GDC" for valves for which GDC 57 is not really "directly applicable". The implementation of this engineering report included nearly 90 changes to SAR table 5-1 (plus one to SAR §5.2.2.4.1) including several to make the table consistent with configuration note C-57-2 and internally consistent on the issue of penetrations "associated with the secondary side of the steam generators."

Of the nearly 90 changes to SAR table 5-1, all but 25 were specifically exempted from evaluations by Attachment 1 to procedure 1000.131 (Ref. 12) including the changes (other than the addition of note 8) related to penetrations "associated with the secondary side of the steam generators". The addition of note 8 was treated as a change back to a previous version of the SAR and, therefore, not "unreviewed" by definition since it had been previously submitted to NRC as part of the amendment 11 SAR update. The other changes related to penetrations "associated with the secondary side of the steam generators" were exempted under F1 (rearranging information to be more easily understood) as an effort to make the individual penetration listings more consistent with the restored note 8 and, therefore, more easily understood.

In summary, the effort to eliminate inconsistencies and confusion regarding treatment and identification of containment isolation valves is an ongoing one. That effort has been aided greatly by the issuance of the Engineering Report 93-R-0007-01. One of the areas of confusion that the engineering report has been of value toward addressing is that of the application of the concept that the secondary system pressure boundary inside containment is to be treated as an extension of the containment liner and is itself the containment boundary. The treatment of the changes to the SAR table was consistent with the requirements and guidance that existed at the time and subsequently. Ultimately, the concept of the secondary pressure boundary being an extension of containment boundary was explicitly approved by NRC in the Safety Evaluation Report approving the renewal of the ANO-1 operating license for another 20 years (Ref. 16) as follows:

> In drawing LRA-M-237, sheet 1, the redundant isolation valves (SS-1017B, SS-1018B) for the test connections of the sampling system are not highlighted as being within the scope of license renewal. However, containment isolation provisions require double isolation at the test connections for greater assurance of containment integrity. The staff asked why the second isolation valve on each test connection were not identified as being subject to an AMR. In its response to the NRC, the applicant states that this penetration is associated with the secondary side of the steam generator, and is not required to meet

GDC 57 of Appendix A to 10 CFR Part 50. The reactor building boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the line emanating from the steam generator, and the outer surface of the steam generator below the lower and above the upper tube sheet. Valves SS-1017B and SS-1018B are not within the scoping of license renewal because they do not meet any of the scoping criteria in 10 CFR 54.4(a). The staff found the applicant's response acceptable.

In addition, Amendment 215 to the ANO-1 Technical Specifications included the following words in the basis for Technical Specification 3.6.3. "The service water system is the only closed system within the reactor building to which Specification 3.6.3 Condition C applies." Since Specification 3.6.3 is intended to apply to all closed systems, i.e. those with only one reactor building isolation valve, the service water system is the only GDC57 system according to the ANO-1 Technical Specification bases.

References:

1. ULD-0-TOP-14, Containment Isolation and Containment Leak Rate Testing
2. Bechtel letter BL-283, dated 2/17/69, subject: Main Steam Line Design Criteria
3. Internal Bechtel memo dated 7/11/69, subject: Main Steam Line Isolation Valves
4. Bechtel letter BL-468, dated 8/1/69, subject: Main Steam Containment Isolation Valves
5. Bechtel letter BL-602, dated 10/27/69, subject: Main Steam Containment Isolation Valves
6. WCAP-7451, Rev. 1, September 1971, Steam Systems Design Manual
7. NUREG-0830, Safety Evaluation Report related to the operation of Callaway Plant, Unit No.1, Docket STN 50-483
8. NUREG-0881, Safety Evaluation Report related to the operation of Wolf Creek Generating Station, Unit No.1, Docket STN 50-482
9. NUREG-0857, Safety Evaluation Report related to the operation of Palo Verde Nuclear Generating Station, Units 1, 2 and 3
10. 0CAN018014
11. 93-R-0007-01, Containment Penetration Design Summary
12. 1000.131, 10CFR50.59 Review Program
13. DCD Discrepancy CI-3, Detailed Design Implementation

14. Bechtel letter BL-750, dated 1/21/70, subject: Main Steam Containment Isolation Valves

15. Letter Harlan Holmes to Burt Lex, dated 3/20/70

16. 1CNA010106

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[1] Though no words in the PSAR explicitly corroborating this statement have been found, §5.6 of the PSAR does imply this by the way that the steam line and feedwater line building penetrations are differentiated from other building penetrations. That section appears to equate leakage from the steam line and feedwater line building penetrations with leakage "through the containment vessel itself" by pointing out that other building penetrations are grouped and are in penetration areas and leakage from these groups of penetrations will be collected and exhausted in a manner that isolates it "from leakage which might occur through the containment vessel itself".

[2] This question referred to the turbine stop valves as containment isolation valves even though they couldn't serve that function under GDC57 because neither the line nor the valves were seismic. Therefore, the AEC did not necessarily imply compliance with the new GDCs when they referred to containment isolation valves.

UNIT 2

Secondary System Pressure Boundary as an Extension of Containment Liner

ULD-0-TOP-14, Containment Isolation and Containment Leak Rate Testing (Ref. 1), states that, "ANO has taken the position that secondary system penetrations are not subject to GDC 57 since the containment barrier integrity is not breached during DBA LOCA conditions. The containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes the outer surface of the lines emanating from the steam generator and the outer surface of the steam generator between the tube sheets." This position is based on the concept of treating the secondary system pressure boundary as an extension of the containment liner. Though not well documented prior to the writing of this ULD, this position has always been the understanding of plant personnel familiar with the original design and licensing basis. Over time, this original design and licensing basis has become obscured by mixing the concepts of the secondary system pressure boundary being an extension of the containment liner with the concept of a GDC 57 closed system. The fact that the two concepts may appear to be equivalent without careful consideration has contributed to the confusion. The NRC did tacitly acknowledge, just prior to granting the ANO-2 operating license, that no post-LOCA isolation of the secondary system flow paths was anticipated by requesting (and getting) a commitment that ANO-2 raise the steam generator level after a LOCA high enough to cover the steam generator tubes (Ref. 2). This commitment was to provide a second barrier to containment leakage if the steam generator tubes were to leak. If there was already expected to be a second barrier at the valves outside of the containment building there would have been no need for this commitment.

The earliest documented distinction between the two concepts appears to be WCAP-7451 (Ref. 3), dated September 1972, which explicitly treated the distinction approximately a year after the issuance of GDC 57. Other examples of documentation of an explicit distinction between the two include NUREG-0830 (Ref. 4), page 6-18 (approving for Callaway the wording of the SNUPPS FSAR, see pages 6.2.4-5 and 6.2.4-6 and figures 6.2.4-1 pages 1 and 5, and 6.2.4-2); NUREG-0881 (Ref. 5), page 6-1 (endorsing the approvals of NUREG-0830 for use at Wolf Creek) and NUREG-0857 (Ref. 6) (approving the Palo Verde FSAR, see page 6.2.4-25). Even with the more explicit wording in the more modern SNUPPS and Palo Verde FSARs, none of the three SERs referenced above explicitly addressed this distinction; they just gave general approval based on the applicable FSAR discussion. These facts do not establish a licensing basis for ANO but do establish the practice and understanding that formed the context within which ANO-1 and ANO-2 were licensed.

For ANO-2, the following building penetrations have all been listed as governed by GDC57 at one time or another. For reference, this table correlates the penetration numbers with the system in which the piping which goes through the penetration is involved.

Bldg. Penetration #	System	Bldg. Penetration #	System
2P1, 2P2	Main Steam	2P32, 2P64	SG Blowdown
2P3, 2P4	Main Feedwater	2P35, 2P65	EFW to SGs
2P7	SG Sample	2P42, 2P48	Plant Heating
2P20, 2P21, 2P55, 2P63	Service Water	2P51, 2P59	Chilled Water

From Kathy Weaver

For ANO-2, the confusion appears to have begun with the insertion of the old table 3.6-1 of the Technical Specifications into NUREG-0336 (Ref. 7) when the ANO-2 operating license was issued in 1978. The originally proposed Technical Specifications did not include this table of containment isolation valves. The table in NUREG-0336 was based upon a hand written table submitted in Reference 8 in response to NRC question 042.32. Besides 2P7, 2P32 and 2P64, that table also listed building penetrations 2P42, 2P48, 2P51 and 2P59 as GDC57. However, Reference 9 describes local leak rate tests for both the inside and outside valves for 2P51 and 2P59 and the system drained and vented inside and outside the containment building for all four (2P42, 2P48, 2P51 and 2P59) building penetrations. The table also listed both the inside and outside valves for 2P7 as GDC57 even though GDC57 can only apply to outside valves. The inside valves for this building penetration are also not operable with a loss of offsite power. Furthermore, the table did not list any of the valves associated with 2P1, 2P2, 2P3, 2P4, 2P35 or 2P65 even though their absence from this list would exclude them from the more restrictive AOT for containment isolation valves than would apply to six of the eight valves that are otherwise covered by technical specifications. The rest of the valves associated with the secondary system building penetrations are not otherwise covered at all by technical specifications. The six valves that were listed in the table that are in the secondary system are 2CV-5852-2, 2CV-5859-2, 2CV-5850, 2CV-5858, 2CV-1015 and 2CV-1065. Of those, only 2CV-5852-2 and 2CV-5859-2 received a CIAS. The ANO-2 FSAR had a table, 6.2-26, which was added in Amendment 23 in 1974, entitled Containment Penetration Barriers that listed all of the secondary system valves and assigned them to GDC 57. The absence of the other secondary system valves in ANO-2 FSAR Table 6.2-26 from NUREG-0336 table 3.6-1 is further indication of confusion regarding what constituted a containment isolation valve. There is no apparent logic to explain why the six valves appeared in the Technical Specification table while the other secondary system valves in the ANO-2 FSAR table did not. A reasonable explanation might be that the individual that drafted the Reference 8 table had less than a thorough understanding of the function (or even the configuration) of these six valves.

Similar inconsistencies to those demonstrated above also appeared in the ANO-2 FSAR. Though table 6.2-26 listed building penetrations 2P1, 2P2, 2P3, 2P4, 2P7, 2P32, 2P35, 2P64 and 2P65 as GDC57, Figure 10.2-3 shows steam traps 2F211 and 2F197 with open lineups to the main steam line upstream of the MSIV. This path is open to the piping that passes through 2P1 and 2P2 and opens through the steam generators to the piping that passes through 2P3, 2P4, 2P7, 2P32, 2P35, 2P64 and 2P65. Without remotely operable isolation valves on these steam traps, none of these penetrations could meet the requirements of GDC57[i]. Furthermore, Section 6.2.4.2 of the ANO-2 FSAR stated "A means of leak testing all barriers in fluid systems that serve a containment isolation function has been provided." There were no such means provided for the valves in the lines that pass through 2P1, 2P2, 2P3, 2P4, 2P32, 2P35, 2P64 and 2P65. This was apparent from Figure 10.2-3. In addition, Table 6.2-26 lists building penetrations 2P42, 2P48, 2P51 and 2P59 as GDC57. However, it is apparent from Figures 3.2-2, sht. 1 and 3.2-4, sht. 1 that these systems are not seismically qualified beyond the inside the containment building isolation valve. Since GDC57 relies on the inside piping as a barrier to the release of radioactive materials that are generated by a LOCA, the piping is considered to be a component used to mitigate an accident. As such it is required by 10CFR100 to be seismically qualified. Therefore, these four building penetrations can also not meet the requirements of GDC57 though they are listed as such. The ANO-2 FSAR was internally inconsistent. (Note that 2P42, 2P48, 2P51 and 2P59 are now recognized as GDC56 penetrations of containment boundary, not GDC57 and not extensions of containment liner plate.)

It should be pointed out that there are several references in the ANO-2 FSAR to double barriers and being single failure proof but the implications of these depend entirely upon whether the word "penetration" refers to a penetration of the containment building or a penetration of the containment boundary. Indeed, ANO-2 FSAR section 6.2.4.2 refers to "each fluid penetration through the containment liner plate". This expression is more consistent with the understanding that "penetration" refers to a penetration of the containment boundary. Because of the steam traps shown in ANO-2 FSAR Figure 10.2-3, the claim (in ANO-2 FSAR section 6.2.4.1) that the containment isolation systems are designed to withstand "the failure of any single active or passive component without loss of isolating capability" and similar assertions made elsewhere can only be true if "penetration" refers to a penetration of the containment boundary. It should also be noted that the two times that the NRC asked for a match of penetration with GDC (PSAR question 5.38 and FSAR question 042.37.4) no such match was provided in direct answer to the question.

By 1980, when responses to the TMI-2 event were being generated, further confusion was created when two of these six valves were included in Reference 10 in a list of valves to which an SIAS signal would be added. No other secondary system valves were included in this response. Again, there is no apparent logic to explain why these two valves were included while none of the other secondary system valves were. A reasonable explanation might be that these were the only two secondary system valves that received a CIAS, a fact that, in itself, is inconsistent with the requirements unless it was just considered to be the conservative thing to do, a practice (going beyond the requirements) that was encouraged as long as it was reasonable.

By this time, the drawbacks of having detailed component lists in the Technical Specifications began to become obvious enough across the industry that efforts to correct the situation were initiated. The first results of these efforts were Generic Letter 84-13 which facilitated the removal of lists of snubbers from the Technical Specifications. Lists of fire protection barriers and equipment subsequently found their exit from the Technical Specifications and, eventually, Generic Letter 91-08 facilitated the removal of lists of containment isolation valves from the Technical Specifications. For ANO-2, this was accomplished in License Amendment 154 dated December 22, 1993, which was anticipated to eliminate some of the problems with Table 3.6-1 that had been seen (e.g. License Amendments 108 and 112). The SER for Amendment 154 indicated that the list would be moved to Procedure 2203.005 (Ref. 27). This list was subsequently relocated to procedure 1015.034 (Ref. 11). The SER for Amendment 154 also stated, "Overall, these changes will allow licensees to make corrections and updates to the list of components for which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TSs."

Unfortunately, although the processes were in place for these controls, the documentation of the technical bases that was needed to support these processes was not available. This fact had been identified by the DCD project more than a year earlier in discrepancy CI-3 (Ref. 26). This lack of documentation was causing confusion which was exacerbated by personnel with experience at other nuclear power plants that had ownership of the Appendix J testing program. The lack of design documentation caused design basis questions to be addressed to the Appendix J testing program personnel by default and the answers that were generated were frequently colored by their experience at other sites in lieu of the absent design documentation. Resolution of CI-3 was assigned to design engineering and eventually addressed by the issuance of Engineering Report 93-R-0007-01 (Ref. 13) in March 1995. One of the many issues that the developers of this engineering report had to deal with was

the confusion regarding the role of the secondary system in containment isolation that had compounded over the years. This report restated the original position that the penetrations "associated with the secondary side of the steam generators . . . are not subject to GDC 57 since the containment barrier integrity is not breached during DBA LOCA conditions" in its configuration note C-57-2. That note also stated that, "Although not directly applicable, the penetration arrangement most closely matches the requirements of GDC 57, which has been conservatively applied." This last part of the configuration note explains the appearance of the "57" in the column of the attachments entitled "Applicable GDC" for valves for which GDC 57 is not really "directly applicable". The implementation of this engineering report included nearly 100 changes to SAR table 6.2-26 including several to make the table consistent with configuration note C-57-2 and internally consistent on the issue of penetrations "associated with the secondary side of the steam generators."

Of the nearly 100 changes to the SAR table, all but 3 were specifically exempted from evaluations by Attachment 1 to procedure 1000.131 (Ref. 14) including the changes related to penetrations "associated with the secondary side of the steam generators". The addition of note 9 was exempted under F2 of Attachment 1 to 1000.131 (increase in the level of detail without changing the intent) and the other changes related to penetrations "associated with the secondary side of the steam generators" were exempted under F1 (rearranging information to be more easily understood) as an effort to make the individual penetration listings more consistent with note 9 and, therefore, more easily understood. NEI 96-07, Rev. 1 (Ref. 15), section 4.1.3 states that "10 CFR 50.59 need not be applied to . . . corrections of inconsistencies within the UFSAR (e.g., between sections)" and this action was consistent with that guidance. However, the usual practice of producing a discretionary 50.59 evaluation should have been applied in order to document that understanding since the understanding was based on safety issues that were reasonably likely to be raised in the future outside the context of 10CFR50.59.

Later, in 2002, when this was questioned by the resident NRC inspector, a discretionary evaluation of the changes related to penetrations "associated with the secondary side of the steam generators" was prepared. This evaluation was prepared under the 2002 version of 10CFR50.59 and the 2002 Entergy 10CFR50.59 procedure (LI-101). The evaluation showed that no unreviewed safety existed. Since the acceptance criteria for dose under the 2002 version of 10CFR50.59 is no "more than a minimal increase", that is what the evaluation showed. However, the evaluation could also have easily shown that there was not even a minimal increase if that had been the criteria. Under the 50.59 program that was in place when the changes to the SAR table were made, the criteria was a shift from one consequence category to the next higher or a significant shift within a consequence category.

The steam generator tube rupture dose analysis for the ANO-2 SAR prior to steam generator replacement was performed in CE calculation 6370-111240-SQ-TR-001 (Ref. 16) approved on 5/24/77. This calculation was performed using the base ANO-2 CESEC model and modifying it to simulate a steam generator tube rupture with a concurrent loss of AC power. The changes made to the base ANO-2 CESEC model are listed on pages 12 through 16 of the calculation. An explicit assumption that the steam generator sample lines are open is made to justify the timing assumed for detection of the tube rupture and the subsequent operator actions to lower RCS pressure to a point that terminates the tube leak at ½ hour. 100% of the Xe in the steam generators is assumed to be released to the atmosphere during this time. In order to cool down to shutdown cooling, the intact steam generator has pressure lowered and a pre-existing 100 gpd tube leak is assumed to continue in that steam generator until shutdown cooling conditions are reached at 3.03 hours. There is no change listed in the CESEC model

that would simulate the closing of the steam generator sample lines at any time during this 3.03 hours. This could only be the result of one thing since all doses are calculated on the basis of radioactive materials being "transported to the site environs by steam released during the transient" (see page 2 of the calculation). That is that releases via the sample lines are negligible within the context of the calculation. The results of the steam generator tube rupture dose analysis would, therefore, be unaffected by whether the sample lines were isolated at any time during the transient. This not only establishes that there is not even a minimal increase in offsite accident dose but, in conjunction with additional material in the belated 50.59 evaluation mentioned above, establishes a basis for concluding that the sample line valves are not equipment important to safety with respect to their close function.

A PIF to make procedure 1015.034 consistent with the engineering report was also issued but the configuration checklist failed to identify a need to change procedure 2305.005 (Ref. 12) or the IST controlling document (HES-18) (Ref. 17) or the Appendix J testing controlling document (HES-02) (Ref. 18). Engineering Standards were not on the configuration checklist at the time. The success of the engineering report in helping with the confusion was almost immediately evident with its use to justify the administrative closure of CR-2-95-0146 (Ref. 19) and CR-2-95-0151 (Ref. 20) as documented in ANO-95-2-00114 (Ref. 21). However, the incomplete configuration checklist led to the continuation of some inconsistencies in the treatment of the steam generator blowdown valves and the steam generator sample valves in 2305.05, Supplement 1, and HES-18 and its successor document, CEP-IST-1 (Ref. 22).

INPO SEN 97 (Ref. 23) addressed an SGTR event at Palo Verde Unit 2 in which the functional recovery procedure was entered because the tube rupture went undiagnosed, due in part to the steam generator blowdown flow isolation on low pressurizer pressure. This lead to a reevaluation of the commitment previously made in 1980 to add an SIAS to 2CV-5852-2 and 2CV-5859-2. During the reevaluation which led to LCP 93-6026 (Ref. 24) and the ultimate removal of the SIAS from these valves, the engineer developing the LCP contacted the ANO safety analysis group about the acceptability of removing the SIAS. The engineer was told that no credit was taken anywhere for the SIAS other than to meet the 1980 commitment and that any isolation requirements for these two valves, ***if any existed***, would surely still be covered by the remaining CIAS. This conversation was documented in the LCP by the statement that, "Discussions with the safety analysis group in Design Engineering indicated that only the CIAS signal is credited for the isolation function." In fact, the issue of whether there was any containment isolation function at all was not addressed in the conversation since the question regarding the SIAS did not require it to be addressed. This occurred prior to the issuance of the engineering report and might have been more accurately worded had the engineering report been available at the time.

The wording in LCP 93-6026 is in a one-time use document not subject to revision. Supplement 1 to 2305.005 and page 138 of the ANO-2 Appendix to CEP-IST-1 are revisable and have been subsequently revised to remove implications that either the blowdown or SG sample valves are containment isolation valves or have a containment isolation function. These are the types of inconsistencies that the engineering report was intended to eliminate. Additional inconsistencies are being identified and addressed under CR-ANO-2-2002-02053 (Ref. 28).

Some success in the area of clearing up confusion in this area is evident in the design specification for the replacement steam generator. ANO-M-2557 (Ref. 25), Section 304.8.8, states:

> Because the current ANO Unit-2 steam generators are considered an extension of the containment boundary per SAR Table 6.2-26, note 9), the design of the replacement

steam generators, considering the limiting Loss of Coolant Accident (LOCA) break, shall clearly meet this fission product barrier criteria. Accordingly, the Contractor shall provide an assessment supporting that the RSG design considers the inside surface of the steam generator tubes, and the outer surface of the steam generator above the bottom of the tube sheet (including secondary nozzles) as containment boundary, in addition to the Class 1 vessel requirements imposed elsewhere in this specification. The function is to be a barrier against fission product leakage to the environment as the result of a LOCA and the RSG shall meet the following design criteria, to be added to the Westinghouse Certified Design Specification (ANO-M-2564):

For revision to the Westinghouse CDS:

Add to Section 4.5:

"The RSG design Test Conditions shall consider a one time containment structural integrity test as follows:

68 psig inside containment atmosphere at 60 to 110°F, Steam Generators at 0 psig secondary pressure and primary system vented to containment.

The RSG design Test Conditions shall consider 40 cycles of containment Integrated Leak Rate Testing (ILRT) as follows:

59 psig inside containment atmosphere at 60 to 110°F, Steam Generators at 0 psig secondary pressure and primary system vented to containment."

Add to Section 4.6:

"RSG design criteria per the containment boundary requirements of SAR Table 6.2-26, note 9) shall be as follows:

For LOCA conditions (Level D, Faulted Condition), the containment boundary or barrier against fission product leakage to the environment is the inside surface of the steam generator tubes, the outer surface of the lines emanating from the steam generator (main steam, feedwater, sample, and blowdown line nozzles), and the outer surface of the steam generator above the bottom of the tubesheet. Accordingly, the design of these subcomponents shall consider the transition from normal operating differential pressure, to LOCA differential pressure. Peak Containment pressure for this scenario is 59 psig (300 °F)."

In summary, the effort to eliminate inconsistencies and confusion regarding treatment and identification of containment isolation valves is an ongoing one. That effort has been aided greatly by the issuance of the Engineering Report 93-R-0007-01. One of the areas of confusion that the engineering report has been of value toward addressing is that of the application of the concept that the secondary system pressure boundary inside containment is to be treated as an extension of the containment liner and is itself the containment boundary. The treatment of the changes to the SAR table was consistent with the requirements and guidance that existed at the time and subsequently. However, the decision to not prepare an evaluation as part of the 10CFR50.59 process to document the disposition of the potential safety issues was neither prudent nor consistent with normal practice. Had one been prepared, its

existence could have prevented even further confusion in this area in which so much effort has been invested to eliminate confusion.

References:

1. ULD-0-TOP-14, Containment Isolation and Containment Leak Rate Testing
2. 2CAN067837
3. WCAP-7451, Rev. 1, September 1971, Steam Systems Design Manual
4. NUREG-0830, Safety Evaluation Report related to the operation of Callaway Plant, Unit No.1, Docket STN 50-483
5. NUREG-0881, Safety Evaluation Report related to the operation of Wolf Creek Generating Station, Unit No.1, Docket STN 50-482
6. NUREG-0857, Safety Evaluation Report related to the operation of Palo Verde Nuclear Generating Station, Units 1, 2 and 3
7. NUREG-0336, Arkansas Nuclear One, Unit 2 Technical Specifications, Appendix "A" to License No. NPF-6, July 18, 1978
8. 2CAN017708
9. 2CAN027806
10. 0CAN018014
11. procedure 1015.034, Containment Penetration Administrative Control
12. procedure 2305.005, Valve Stroke and Position Verification
13. 93-R-0007-01, Containment Penetration Design Summary
14. 1000.131, 10CFR50.59 Review Program
15. NEI-96-07, Rev. 1, Guidelines for 10 CFR 50.59 Implementation
16. 6370-111240-SQ-TR-001, Steam Generator Tube Rupture With Concurrent Loss of AC
17. HES-18, ANO-2 IST Program Bases Document
18. HES-02, Containment Leak Rate Testing Program

19. CR-2-95-0146, involving EFW sample lines

20. CR-2-95-0151, involving EFW steam supply line drain lines

21. ANO-95-2-00114

22. CEP-IST-1, Inservice Testing Bases Document

23. INPO SEN 97, Steam Generator Tube Rupture

24. LCP 93-6026, Removal of SIAS#2 Signal to 2CV-5852-2 & 2CV-5859-2

25. ANO-M-2557, Replace SG Bid Specification

26. DCD Discrepancy CI-3, Detailed Design Implementation

27. procedure 2203.005, Loss of CNTMT Integrity

28. CR-ANO-2-2002-02053

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[i] NRC staff reviewers tended to pay more attention to FSAR drawings than to FSAR statements which they might have called FSAR "assertions". For example, item 15 in the November 29, 1977, letter from the NRC's J. F. Stolz pointed out that the information provided in a previous response differed from FSAR Figure 7.2-22 and the cable routing provided in FSAR Figure 5.1-3 differed from that provided previously in design drawing B6370-413-105, Rev. 4. It turned out that the FSAR drawing updates lagged the design drawing updates and the FSAR drawings reviewed were out of date. This practice actually led to enforcement action for out of date FSAR drawings in inspection report 50-368/77-26 (2CNA127714). Larger versions of FSAR P&IDs were provided and updated periodically to facilitate NRC staff reviews (e.g. 2CAN067507, 2CAN017604, 2CAN057610, 2CAN077610, 2CAN097809). These submittals were explicitly made part of the FSAR in response to staff reliance on them as part of the basis for the issuance of the operating license (e.g. 2CAN067805).