



Figure 4- 24 Vertical displacements of Cold Leg and Hot Leg pipes (Units: Inches)

5.0 CONCLUSIONS

Steam explosion is a remaining risk-significant issue in nuclear power plant due to the threatening of integrity of the defense-in-depth barriers by explosive dynamic loadings that could lead to release radioactive fission product to public.

Therefore, it is needed to evaluate steam explosion risk when a new design reactor like APR1400 is considered. The steam explosion risk can be categorized into two groups in terms of the locations of steam explosion initiation, that is, in-vessel and ex-vessel.

In this report, comprehensive analyses on both in-vessel and ex-vessel steam explosion were conducted and the most updated technical information on the phenomena, risk evaluation, and analysis methodology were collected and utilized for the assessment of steam explosion risk in the APR1400 design.

The report describes the analysis efforts by selecting the base case adequate to the APR1400 design and examining the ranges of key parameters and their uncertainties. The analysis results can be used for the evaluation of structure integrity relevant to the steam explosion issue in the APR1400 design, especially in the case of ex-vessel severe accident progression.

Table 5-1 summarizes the results of the analysis for the IVSE and EVSE energetics in the APR1400 plant. This analysis concluded that IVSE provided no threat to fail RPV. For the EVSE analysis, reactor cavity structure with EVSE pressure loading is evaluated using LS-DYNA and the maximum displacements, concrete cracks, liner plate stresses, reinforcement re-bar stresses, RPV column support anchor bolt stresses and strains for EVSE loading to reactor cavity are summarized in Table 4-17.

Table 5- 1 Summary of the IVSE and EVSE Analysis for the APR1400

TS

6.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, 1975, "Reactor Safety Study; An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Report NUREG-75/0114 (WASH-1400), (Oct.).
2. U.S. Nuclear Regulatory Commission, Steam Explosion Review Group, 1985, "A Review of Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosion," NUREG-1116, U.S. Nuclear Regulatory Commission.
3. U.S. Nuclear Regulatory Commission, Steam Explosion Review Group-2, 1995, "A reassessment of the potential for an Alpha-Mode containment failure and a review of the current understanding of broader fuel-coolant interaction (FCI) issues," NUREG-1524, U.S. Nuclear Regulatory Commission.
4. OECD, 1997, OECD/CSNI Specialist Meeting on Fuel Coolant Interactions, Summary and Conclusions, NEA/CSNI/R(97)30, JAERI, Tokai-Mura, Japan
5. Magallon, D., 2006, "OECD Research Programme on Fuel-Coolant Interaction: Steam Explosion Resolution for Nuclear Applications (SERENA)," Final Report, NEA/CSNI/R(2007)11, OECD/NEA, France.
6. Hong, S.W., et al., 2010, "Status of SERENA-2," presentation slide at the Cooperative Severe Accident Research Program (CSARP) meeting, September 14-16, Nuclear Regulatory Commission, Residence Inn Bethesda, Bethesda, Maryland, USA.
7. Chen, R. H., Corradini, M. L., and Su, G. H., 2012, "Simulation of FARO Corium Coolant Interaction Experiment with TEXAS-VI," Proc. American Nuclear Society Meeting, Chicago, USA, Paper No. 1029, USA
8. Theofanous, T. G., Najafi, B., and Rumble, E., 1989 "An Assessment of Steam-Explosion-Induced Containment Failure, Part I-IV," Nuclear Science and Engineering, 97, pp. 259~326, NUREG/CR-5030.
9. Turland, B. D., Fletcher, D. F., Hodges, K. I., Attwood, G. J., 1993, "Quantification of the Probability of Containment failure Caused by an In-Vessel Steam explosion For the Sizewell B PWR," Proc. Of the CSNI Specialists Meeting on Fuel-Coolant Interactions, Santa Barbara, California, USA, NUREG/CP-0127, NEA/CSNI/R(93)8, pp. 309-321
10. Theofanous, T. G. and Yuen W. W., 1993, "The Probability of ALPHA-Mode Containment Failure Updated," Proceedings of the CSNI Specialists Meeting on Fuel-Coolant Interactions, NUREG/CP-0127, Santa Barbara, CA, USA, pp. 330~342.
11. Amarasooriya, W. H. and Theofanous, T. G., 1987, "Scaling Considerations in Steam Explosions," ANS Proceedings of National Heat Transfer Conference, Vol. 2, pp. 58~67.
12. Theofanous, T.G., et al., 1999, "Lower head integrity under steam explosion loads," Nuclear Engineering and Design 189, p. 7~57.

13. Chu, C. C., Sienicki, J. J., Spencer, B. W., Frid, W., Lowenhielm, G., 1995, "Ex-Vessel Melt-Coolant Interactions in Deep Water Pool; Studies and Accident Management for Swedish BWRs, Nuclear Engineering and Design, Vol. 155, Issue 1-2, pp. 159-213.
14. Zuchuat, O., Schmocker, U., Esmaili, H., et al., 1997, "Steam explosions-induced containment failure studies for Swiss nuclear power plants," Proc. OECD/CSNI Specialists Meeting on Fuel-Coolant Interactions, Tokai-mura, Japan, JAERI-Conf 97-011, NEA/CSNI/R(97)26 (Part I), pp. 36-61.
15. Moriyama, K., et al., 2006, "Evaluation of Containment Failure Probability by Ex-Vessel Steam Explosion in Japanese LWR Plants," J. of Nuclear Science and Technology, Vol. 43, No. 7, p. 774-784.
16. Cizelj, L., Koncar, B., and Leskovic, M., 2006, "Vulnerability of a Partially Flooded PWR Reactor Cavity to a Steam Explosion," Nuclear Engineering and Design, 236, pp1617-1627.
17. Murphy, J. and Corradini, M. L., 1997, "An Assessment of Ex-Vessel FCI Energetics for Advanced Light Water Reactors," Nuclear Technology, Vol. 117, p 49-63.
18. U.S. Nuclear Regulatory Commission, 1998, "Final Safety Evaluation Report; Related to the Certification of the AP600 Standard Design, Vol.2," U.S. Nuclear Regulatory Commission.
19. Esmaili, H., et. al., 2004, "Analysis of In-Vessel Retention and Ex-Vessel Fuel Coolant Interaction for AP1000," NUREG/CR-6849, Energy Research Inc., U.S. NRC, Washington, USA.
20. Aya K. D., and Corradini, M. L., 2011, "Ex-Vessel Vapor Explosion Simulations for Postulated Severe Accident Conditions," Proc. Of the ASME 2011 Pressure Vessels and Piping Division Conference, PVP2011, July 17-21, Baltimore, Maryland, USA.
21. Moriyama, K., and Nakamura, H., 2006, "A Strategy for the Application of Steam Explosion Codes to Reactor Analysis," Technical Meeting on Severe Accident and Accident Management, Toranomon Pastoral, Minato-ku, Tokyo, Japan, March 14-16, 2006.
22. U.S. Nuclear Regulatory Commission, 1994, "Final Safety Evaluation Report; Related to the Certification of the Advanced Boiling Water reactor Design, Main Report," U.S. Nuclear Regulatory Commission.
23. U.S. Nuclear Regulatory Commission, 2009, "Design Control Document for US-APWR, Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation," MUAP-DC0019, Rev. 2, Mitsubishi Heavy Industries, Ltd.
24. Hwang, M. K., et al., 1999, "Assessment of Steam Explosion Impact on KNGR Plant," KAERI/TR-1303/99, Korea Atomic Energy Research Institute, Daejeon, Korea, (In Korean).
25. Bang, K. H., Cho, Jong-Rae, and Park, S. Y., 2000, "An Assessment of Reactor Vessel Integrity under In-Vessel Vapor Explosion Loads," J. of the Korean Nuclear Society, Vol. 32, No. 4, pp. 299-308.

26. Berman, M., Swenson, D. V., and Eickett, A. J., 1984, "An Uncertainty Study of PWR Steam Explosions," SAND83-1438, NUREG/CR-3369, Sandia National Laboratories, USA.
27. Bohl, W. R. and Bulter, T. A., 1985, "Comments on Proposed Research Contributing to the Relation of Residual Steam Explosion Issues," Letter Report in "Review of Current Understanding Failure Arising from In-Vessel Steam Explosion," NUREG-1116, U.S. Nuclear Regulatory Commission.
28. Shockey, D. A., Seaman, L., Dao, K. C., and Curran, D. R., 1980, "Kinetics of Void Development in Fracturing A533B Tensile Bars," J. Pressure Vessel Technology, 102, pp14-21.
29. Bang, K.H., Park, I.G., and Park, G.C., 1997, "TRACER-II: A Complete Computational Model for Mixing and Propagation of Vapor Explosions," Proc. Of OECD/CSNI FCI Specialists Meeting, Tokia-Miura, Japan, pp.804-816, May.
30. Corradini, M. L., Blanchard, J. P., and Martin, C. J., 2012, "Evaluation of Dynamic Pressures from Steam Explosions applied to Advanced LWRs," submitted to Nuclear Science and Engineering.
31. Kim S. B., et al., 2004, "Development of Optimal Severe Accident Management Strategy and Engineering Safety Features: Optimization of the Severe Accident Management for Domestic Plants and Validation Experiments," KAERI/RR-2528/2004, Korea Atomic Energy Research Institute, Daejeon, Korea (In Korean).
32. Paik, C. Y., 2012, "Corium Flow Characteristics delivered from a Failed Vessel for Steam Explosion Calculations," Draft Report No. FAI/12-0395, Fauske Associates, LLC, USA.
33. Chu, C. C., 1986, "One-Dimensional Transient Fluid Model for Fuel-Coolant Interactions," Ph.D. Thesis, University of Wisconsin-Madison, Madison, Wisconsin.
34. Tang, J., 1993, "Modeling of the Complete Process of One-Dimensional Steam explosions," Ph.D. Thesis, University of Wisconsin-Madison, Madison, Wisconsin, USA.
35. Young, M. F., 1982, "The TEXAS Code for Fuel-Coolant Interaction Analysis," Proc. ANS/ENS Fast Reactor Safety Conference, July, Lyon, France.
36. Chu, C. C., Corradini, M. L., 1989, "One-Dimensional Transient Fluid Model for Fuel/Coolant Interaction Analysis," Nuclear Science and Engineering, V 101, No 1, p 48-72 (Jan.).
37. Pilch, M., 1981, "Acceleration Induced Fragmentation of Liquid Drops," Ph.D. Thesis of University of Virginia.
38. Kim, B. J., 1985, "Heat Transfer and Fluid Flow Aspects of Small-Scale Single Droplet Fuel-Coolant Interactions," Ph.D. Thesis, University of Wisconsin-Madison, Madison, Wisconsin.
39. Huhtiniemi, I., Magallon, D., and Hohmann, H., 1999, "Results of recent KROTOS FCI tests: Alumina versus Corium melts," *Nuclear Engineering and Design*, 189:379-389.

40. Nelson, L. S. and Duda, P., M., 1982, "Steam Explosions of Molten Iron Oxide Drops: easier Initiation at Small Pressurizations, *Nature*, 296, pp. 844-846.
41. Corradini, M., et al, 2000, A Users' Manual for TEXAS-V: A One-Dimensional Transient Fluid Model for Fuel-Coolant Interaction Analysis, UW Nuclear Engineering and Engineering Physics (Aug.).
42. ASME Boiler & Pressure Vessel Design Code Section III, ASME, 2004.
43. Knudson, D.L., Rempea, J.L., Condiea, K.G., Suh, K.Y., Cheung, F.-B., Kim, S.-B., 2004, "Late-phase melt conditions affecting the potential for in-vessel retention in high power reactors," *Nuclear Engineering and Design*, Vol. 230, Issues 1-3, pp.133-150.
44. Humphries, L.L., et al., 2002, OECD Lower Head Failure Project Final Report, Sandia National Laboratories, Albuquerque, NM.
45. Sehgal, B. R. et al., 2003, "Assessment of Reactor Vessel Integrity (ARVI)," *Nuclear Engineering and Design*, 221, pp. 23-53.
46. Cole, R. H., 1948, *Underwater Explosions*, Princeton-New Jersey, Princeton University Press.
47. Khatib-Rahbar, M., Cassoli, E., Lee, M., Nourbakhsh, R. Davis, and Schmidt E., 1989, "A probabilistic approach to quantifying uncertainties in the progression of severe accidents," *Nuclear Science and Engineering*, 102, 219
48. Iman, R. L., Davenport, J. M., and Zeigler, D. K., 1980, "Latin Hypercube Sampling (A Program User's Guide)," Technical Report SAND79-1473, Sandia Laboratories, Albuquerque (1980).
49. Corradini, M. L. and Swenson, D. V., 1981, "Probability of Containment Failure Due to Steam Explosions Following a Postulated Core meltdown in an LWR," Report SAND 80-2132.